
Safety Evaluation Report

related to the operation of
Perry Nuclear Power Plant,
Units 1 and 2

Docket Nos. 50-440 and 50-441

Cleveland Electric Illuminating Company

**U.S. Nuclear Regulatory
Commission**

Office of Nuclear Reactor Regulation

September 1986



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ABSTRACT

Supplement No. 10 to the Safety Evaluation Report (NUREG-0887) on the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or CAPCO), as applicants and owners for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This supplement reports the status of certain issues and action items that had not been resolved or completed at the time of publication of the Safety Evaluation Report and Supplements Nos. 1 through 9 to that report.

CONTENTS

	<u>Page</u>
ABSTRACT.....	iii
ABBREVIATIONS.....	ix
1 INTRODUCTION AND GENERAL DISCUSSION.....	1-1
1.1 Introduction.....	1-1
1.9 Outstanding Issues.....	1-3
1.10 Confirmatory Issues.....	1-6
1.11 License Conditions.....	1-13
2 SITE CHARACTERISTICS.....	2-1
2.4 Hydrologic Engineering.....	2-1
2.4.8 Accidental Releases of Liquid Effluents in Ground and Surface Waters.....	2-1
2.5 Geology and Seismology.....	2-1
3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS.....	3-1
3.7 Seismic Design.....	3-1
3.7.2 Seismic System and Subsystem Analysis.....	3-1
3.7.3 Seismic Instrumentation Program.....	3-4
3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment.....	3-6
3.10.1 Seismic and Dynamic Qualification.....	3-6
6 ENGINEERED SAFETY FEATURES.....	6-1
6.2 Containment Systems.....	6-1
6.2.7 TMI-2 Requirements.....	6-1
6.3 Emergency Core Cooling System.....	6-2
6.3.3 Performance Evaluation.....	6-2
6.4 Control Room Habitability.....	6-5
7 INSTRUMENTATION AND CONTROL.....	7-1
7.2 Reactor Protection System.....	7-1

CONTENTS (Continued)

		<u>Page</u>
	7.2.2 Specific Findings.....	7-1
	7.2.2.3 Anticipated Transients Without Scram (ATWS).....	7-1
9	AUXILIARY SYSTEMS.....	9-1
	9.6 Other Auxiliary Systems.....	9-1
	9.6.3 Emergency Diesel Engine Fuel Oil Storage and Transfer System.....	9-1
	9.6.3.3 Conclusion.....	9-1
11	RADIOACTIVE WASTE MANAGEMENT.....	11-1
	11.5 Effluent Monitoring.....	11-1
13	CONDUCT OF OPERATIONS.....	13-1
	13.3 Emergency Plans.....	13-1
	13.3.5 Atomic Safety and Licensing Board Hearings on Emergency Preparedness.....	13-1
	13.5 Plant Procedures.....	13-1
	13.5.2 Operating and Maintenance Procedures.....	13-1
	13.5.2.2 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures.....	13-1
	13.5.2.7 Hydrogen Igniter Emergency Procedures....	13-4
16	TECHNICAL SPECIFICATIONS.....	16-1
	16.1 Introduction.....	16-1
	16.2 Specific Technical Changes for Full-Power Operation.....	16-1
	16.2.1 Operation in the Maximum Extended Operating Domain.....	16-1
	16.2.2 Definition 1.7, Core Alteration.....	16-2
	16.2.3 Table 3.3.7.1-1, Radiation Monitoring Instrumentation.....	16-3
	16.2.4 Table 3.3.7.5-1, Accident Monitoring Instrumentation and Basis 3.3.7.5.....	16-3
	16.2.5 Table 3.3.7.10-1, Radioactive Gaseous Effluent Monitoring Instrumentation.....	16-4

CONTENTS (Continued)

	<u>Page</u>
16.2.6 Table 4.3.7.10-1, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements.....	16-4
16.2.7 3.4.9.2, Residual Heat Removal, Cold Shutdown.....	16-4
16.2.8 4.5.1, Emergency Core Cooling Systems - Operating..	16-4
16.2.9 3/4.6.1.2, Primary Containment Leakage.....	16-5
16.2.10 3.6.1.3, Primary Containment Air Locks.....	16-5
16.2.11 Table 3.6.4-1, Containment and Drywell Isolation Valves.....	16-5
16.2.12 3.6.5.2, Containment Humidity Control.....	16-5
16.2.13 4.7.4.e.1, Snubbers.....	16-5
16.2.14 3/4.7.6, Main Turbine Bypass System Surveillance Requirements.....	16-6
16.2.15 3.8.2.1 and 3.8.3.3, D.C. Sources.....	16-6
16.2.16 3.8.4.1, Containment Penetration Conductor Overcurrent Protective Devices.....	16-6
16.2.17 Table 3.8.4.1-1, Containment Penetration Conductor Overcurrent Protective Devices.....	16-7
16.2.18 3.8.4.2, Reactor Protection System Electric Power Monitoring.....	16-7
16.2.19 3.9.11.2, Residual Heat Removal and Coolant Circulation, Low Water Level.....	16-8
16.2.20 3.10.1, Primary Containment Integrity/Drywell Integrity.....	16-8
16.2.21 3/4.11.1, Liquid Effluents.....	16-8
16.2.22 Basis 3/4.10.3, Shutdown Margin Demonstration.....	16-8
16.2.23 6, Administrative Controls.....	16-9
16.2.24 6.9.4, Special Reports.....	16-10
16.3 Technical Specifications Not Applicable Above 5% Power.....	16-11
16.4 Correction of Typographical Errors or Editorial Clarification.....	16-11
18 CONTROL ROOM DESIGN REVIEW.....	18-1
18.2 Evaluation of DCRDR Program Plan Report.....	18-1
18.2.1 Establishment of a Qualified Multidisciplinary Review Team.....	18-1
18.2.2 Use of Function and Task Analyses.....	18-1
18.2.3 Comparison of Display and Control Requirements With Control Room Inventory.....	18-2
18.2.4 A Control Room Survey.....	18-2
18.2.5 Assessment of HEDs.....	18-2
18.2.6 Selection of Design Improvements.....	18-2
18.2.7 Verification That Selected Improvements Provide the Necessary Correction and Do Not Introduce New HEDs.....	18-3
18.2.8 Coordination of Control Room Improvements With Other Programs.....	18-3

CONTENTS (Continued)

	<u>Page</u>
18.2.9 Other.....	18-3
18.2.9.1 HED Corrections for Unit 1.....	18-3
18.2.9.2 Completion of DRCR for Unit 2.....	18-4
18.3 Summary Report Requirements.....	18-4
18.4 Conclusion.....	18-5
18.4.1 HED Correction and DCRDR Elements.....	18-5
18.4.2 Documentation and Reporting.....	18-6

APPENDICES

- A CONTINUATION OF CHRONOLOGY, PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2
- B REFERENCES
- E NRC STAFF CONTRIBUTORS AND CONSULTANTS
- F ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS REPORT ON THE PERRY NUCLEAR POWER PLANT, UNIT 1, DISCUSSING OHIO EARTHQUAKE ON JANUARY 31, 1986
- S FEDERAL EMERGENCY MANAGEMENT AGENCY SUPPLEMENTAL FINDINGS
- T REVIEW OF THE EFFECT OF THE JANUARY 31, 1986 EARTHQUAKE ON THE PERRY NUCLEAR POWER PLANT

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
APRM	average power range monitor
ASLAB	Atomic Safety and Licensing Appeal Board
ASLB	Atomic Safety and Licensing Board
ATWS	anticipated transient(s) without scram
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling water reactor
BWROG	BWR Owners Group
CAPCO	Central Area Power Coordination (Group)
CDI	cumulative damage index
CEI	Cleveland Electric Illuminating Company
CFR	<u>Code of Federal Regulations</u>
CRARM	control room area radiation monitor
CRD	control rod drive
DBA	design-basis accident
DCRDR	detailed control room design review
DR/QR	design review/quality revalidation
ECCS	emergency core cooling system
ELLR	extended load line region
EPG	emergency procedure guideline
ESF	engineered safety feature(s)
FaAA	Failure Analysis Associates, Inc.
FEMA	Federal Emergency Management Agency
FPCC	fuel pool cooling and cleaning
FSAR	Final Safety Analysis Report
G/C	Gilbert/Commonwealth Associates
GDC	General Design Criterion(a)
GE	General Electric Company
HCOG	Hydrogen Control Owners Group
HCU	hydraulic control unit
HED	human engineering discrepancy
HEO	human engineering observation
HEPA	high efficiency particulate air
HIS	hydrogen ignition system
HPCS	high-pressure core spray
ICFR	increased core flow region
IE	Office of Inspection and Enforcement

IFTS	inclined fuel transfer system
IRM	intermediate range monitor
JIO	justification for interim operation
LCO	limiting condition for operation
LFHW	loss of feedwater heating
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCI	low-pressure core injection
LPCS	low-pressure core spray
LPRM	linear power range monitor
MAPLHGR	maximum average planar linear heat-generation rate
MCPR	minimum critical power ratio
MEOD	maximum extended operating domain
MSLBA	main steamline break accident
mya	million years ago
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OCRE	Ohio Citizens for Responsible Energy
PEI	Perry Emergency Instruction
PGCC	power generation control complex
PNL	Pacific Northwest Laboratory
PORC	Power Operations Review Committee
PSI	preservice inspection
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RG	regulatory guide
RHR	residual heat removal
RRS	required response spectrum(a)
RV	reactor vessel
RV	relief valve
SDV	scram discharge volume
SER	Safety Evaluation Report
SNPP	Salem Nuclear Power Plant
SOE	sequence of events
SPDS	safety parameter display system
SRM	source range mointor
SRP	Standard Review Plan
SSE	safe shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SV	safety valve
TDI	Transamerica Delaval, Inc.
TIP	traversing incore probe
TMI	Three Mile Island

TMI-2 Three Mile Island, Unit 2
TRS test response spectrum(a)
USGS U.S. Geological Survey (U.S. Department of the Interior)
ZPA zero period acceleration

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER) (NUREG-0887) on the application of the Cleveland Electric Illuminating Company (CEI or the applicant or licensee) for a license to operate the Perry Nuclear Power Plant (Perry), Units 1 and 2, was issued in May 1982. Supplements to the Safety Evaluation Report (SSERs) were issued as follows:

- SSER No. 1 in August 1982
- SSER No. 2 in January 1983
- SSER No. 3 in April 1983
- SSER No. 4 in February 1984
- SSER No. 5 in February 1985
- SSER No. 6 in April 1985
- SSER No. 7 in November 1985
- SSER No. 8 in January 1986
- SSER No. 9 in March 1986

The purpose of this supplemental report (SSER No. 10) is to further update the SER by providing the results of the staff's review of information submitted by the licensee, by letter, in response to Confirmatory Issue (65) and those license conditions required to be completed for authorization of Perry Unit 1 to operate above 5% of rated thermal power levels, described in SSER No. 8. This supplement also presents the results of the staff's review of confirmatory work items relative to the earthquake that occurred near the plant site on January 31, 1986, documented in SSER No. 9.*

Each section or appendix of this supplement is designated and titled so that it corresponds to the section or appendix of the SER** that has been affected by the staff's additional insertions and, except where specifically noted, does not replace the corresponding SER section or appendix. Appendix A is a continuation of the chronology of correspondence between the NRC technical review staff and

* SSER No. 9 reported exclusively on the staff's evaluation of the effect of the January 31, 1986, earthquake on the plant's seismic design. Confirmatory work items identified as a result of the staff's evaluations in that supplement are required to be completed by the licensee in order to obtain licensing authorization to operate Perry Unit 1 at power levels above 5% of rated thermal power. That confirmatory work is being added as Confirmatory Issue (66) in Section 1.10 of this supplement, and the staff's evaluation findings discussed in this supplement.

**SSER No. 1 added Appendices F and G to the SER; SSER No. 4 added Appendix H; SSER No. 5 added Appendices I, J, K, and L; SSER No. 6 added Appendix M; SSER No. 7 added Appendices N, O, P, and Q; and SSER No. 8 added Appendix R.

the licensee that updates issues and items listed in the SER and SSER Nos. 1 through 9. Appendix B is a list of references cited in this supplement.* Appendix E is a list of the principal contributors to this supplement. Appendix F is updated to incorporate the Advisory Committee on Reactor Safeguards letter report to the Chairman of the NRC, dated March 17, 1986, relative to the staff's findings pertaining to the January 31, 1986, earthquake documented in SSER No. 9. Appendices S and T are being added to the SER by this supplement. Appendix S contains the Federal Emergency Management Agency letter report relative to the Atomic Safety and Licensing Board offsite emergency preparedness items listed as a part of License Condition (31), added to the SER by SSER No. 7. Appendix T contains the staff consultant's engineering report on the January 31, 1986, earthquake discussed in Section 3.7.2 of this supplement. No changes were made to Appendices C, D, G, H, I, J, K, L, M, N, O, P, Q, and R in this supplement.

In addition to updating the status of the above-noted confirmatory items and license conditions, this supplement documents:

- the staff's evaluation of the confirmatory work items documented in SSER No. 9 relative to the earthquake that occurred near the Perry plant site on January 31, 1986, picked up in Section 1.10 of this supplement as Confirmatory Issue (66) and satisfactorily resolved in this supplement for Perry Unit 1 operation above 5% of rated thermal power levels (see Sections 2.5, 3.7, and 3.10 of this supplement)
- the staff's favorable evaluation findings relative to the Perry plant's compliance with the required action items in NRC Generic Letter 83-28, and specifically action items 1.2, 3.1.1., 3.1.2., 3.2.1, 3.2.2., and 4.5.1 (see Section 7.2.2.3 of this supplement)
- the staff's evaluation of the analytical basis on which the licensee developed containment venting procedures committed to by the licensee in Appendix 1B of the Final Safety Analysis Report (FSAR) (see Section 13.5.2.2 of this supplement)
- the staff's evaluation of the hydrogen control equipment survivability analysis and ignition system procedures required by the Atomic Safety and Licensing Board for full-power licensing and listed as License Condition 2.C.(10) in the low-power operating license for Perry Unit 1 (see Sections 6.2.7 and 13.5.2.7 of this supplement)
- the staff's evaluation of the licensee's response to the detailed control room design review license condition stated in Paragraph 2.C(7) in the Perry Unit 1 low-power operating license (see Section 18 of this supplement)
- the staff's evaluation of Technical Specification changes that permit plant operation in the maximum extended operating domain; i.e., operation at rated power at lower-than-rated core flow rates and operation at core flows up to 105% of rated flow (see Sections 6.3.3 and 16.2 of this supplement)

*Availability of all material cited is described on the inside front cover of this supplement.

- the staff's evaluation of the TDI Owners Group DR/QR Program for validating and upgrading the design and manufacturing quality of the Perry TDI diesel generators for nuclear emergency standby service (see Section 9.6.3 of this supplement)
- the staff's acceptance of Revision 1 to the Perry Offsite Dose Calculations Manual that uses the approved methodology and guidelines delineated in NUREG-0133 for calculating offsite radioactive effluent doses (see Section 11.5 of this supplement)
- Technical Specification changes for the full-power licensing of Perry Unit 1 (see Section 16 of this supplement).

As a result of the staff's evaluations, documented in SSER Nos. 8 and 9, an operating license was issued for Perry Unit 1 on March 18, 1986. The license (NPF-45) permitted CEI to load fuel and conduct initial startup tests up to power levels not to exceed 5% of rated thermal power (178 Mwt). This supplement reports on the staff's evaluation findings and conclusions on confirmatory items listed in SSER Nos. 8 and 9 and license conditions listed in Operating License NPF-45, that were required to be completed to permit Perry Unit 1 to operate at power levels above 5% of rated thermal power.

Copies of this SSER are available for public inspection in the NRC Public Document Room at 1717 H Street NW, Washington, DC, and at the Perry Public Library, 3735 Main Street, Perry, Ohio. Copies of this SSER are also available for purchase from the sources indicated on the inside front cover of this report.

The NRC Project Manager is John J. Stefano. Mr. Stefano may be contacted by calling (301) 492-9473, or by writing to the following address:

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 U.S. Nuclear Regulatory Commission
 Washington, DC 20555

1.9 Outstanding Issues

In Section 1.9 of the SER, the staff identified 19 outstanding issues that had not been resolved at the time the SER was issued in May 1982. New issues were added, were reported as being resolved, or were redefined in SSER Nos. 1 through 8. As of the issuance of SSER No. 8 in January 1986, all of the initial and subsequently added outstanding issues were fully resolved for Perry Unit 1 licensing. A composite of these issues and the history of their resolution follows:

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(1) Turbine missile protection	Resolved and added as License Condition (19) in SSER No. 3; license condition deleted in SSER No. 8.	----
(2) Seismic system and subsystem analysis	Resolved in SSER No. 1	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(3) Reactor internals vibration prototype (BWR/6-238 in.) test program	Changed to Confirmatory Issue (53) in SSER No. 2; resolved in SSER No. 4.	----
(4) Environmental qualifications of equipment importance to safety:	Seismic/dynamic qualification of equipment changed to License Conditions (27) and (28) in SSER No. 5; environmental qualification redefined in SSER No. 5 as shown under <u>Issue</u> column; redefined equipment qualification items resolved in SSER No. 7 (i.e., items 4a and 4b)	----
(a) Notification that all electrical equipment is qualified or submittal of justification for interim operation (JIO) per 10 CFR 50.49(i) for all unqualified equipment		
(b) Certification that all mechanical equipment is qualified and submittal of three qualification files for staff review, or provide JIO		
(5) Inservice testing of pumps and valves	Resolved and changed to License Condition (26) in SSER No. 5	----
(6) Transient and accident analysis for ECCS, overpressure and operating MCPR	Resolved in SSER No. 1	----
(7) Control room design	Report of April 1985 in progress audit of Perry Unit 1 Detailed Control Room Design Review (DCRDR) issued. Followup audit conducted in September 1985 and discussed in SSER No. 7, changing the issue to License Condition (30). License Condition (30) modified in SSER No. 8 on the basis of the staff's update of its DCRDR review	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(8) Mark III containment system (Humphrey Issues)	Changed to Confirmatory Issue (64) in SSER No. 7; resolved in SSER No. 8	----
(9) Pool dynamic loads	Partially resolved in SSER No. 4; fully resolved in SSER No 7	----
(10) Containment purge	Changed to Confirmatory Issue (56) and License Condition (24) in SSER No. 4; resolved in SSER No. 8	----
(11) Periodic testing of ADS actuation systems during plant operation	Resolved in SSER No. 2	----
(12) Manual initiation/termination of ESF systems	Changed to Confirmatory Issue (55) in SSER No. 4; resolved in SSER No. 5	----
(13) IE Bulletin 79-27	Resolved in SSER No. 4	----
(14) Control system failures	Resolved in SSER No. 5	----
(15) Fire protection - safe shutdown	Resolved in SSER No. 2; detailed basis for resolution addressed in SSER No. 3	----
(16) Fire protection - PGCC system (CO ₂ vs halon) fire suppressant in control room	Resolved in SSER No. 2; detailed basis for resolution addressed in SSER No. 3	----
(17) HPCS skid piping	Resolved in SSER No. 1	----
(18) Interim shift staffing for two-unit operation	Deferred; applies to Unit 2 operation only	----
(19) Emergency plans (onsite)	Changed to Confirmatory Issue (61) in SSER No. 5	----
(20) Standby liquid control system design	Added in SSER No. 1; resolved in SSER No. 3	----
(21) Reanalysis of transients and accidents; development of emergency operating procedures per TMI Action Plan Item I.C.1	Added in SSER No. 2; resolved in SSER No. 8	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(22) Fire protection items		
(a) Fire brigade staffing	Resolved in SSER No. 7	----
(b) Gypsum board wall design	Resolved in SSER No. 7	----
(c) Unlabeled fire door and multipurpose wall information	Resolved in SSER No. 7	----
(d) Automatic suppression additions	Resolved in SSER No. 7	----
(e) Fire alarm conformance with NFPA 72D	Resolved in SSER No. 7	----
(f) Switchgear room drainage	Resolved in SSER No. 7	----
(g) PGCC CO ₂ automatic initiation	Resolved in SSER No. 7	----
(h) Fire damper failure evaluation	Resolved in SSER No. 7	----
(i) Further deviations to 10 CFR 50, Appendix R	Resolved in SSER No. 7	----
(j) Penetrations per BTP CMEB 9.5-1	Resolved in SSER No. 7	----
(k) Fire Protection Evaluation Report (Rev.4)	Resolved in SSER No. 7	----
(23) FSAR Table 3.2-1 safety-related items list	Added in SSER No. 3; resolved in SSER No. 4	----
(24) TDI diesel generator reliability	Changed to Confirmatory Issue (63) in SSER No. 6	----

1.10 Confirmatory Issues

In Section 1.10 of the SER, the staff identified 49 confirmatory issues that were not fully resolved when the SER was issued. Issues resolved, added, and/or redefined in SSER Nos. 1 through 9 follow:

SSER No. 1 - Five issues were reported resolved, and Issue (50) was added. (Issue (50) was initially cited as License Condition (8) in SER Section 1.11.)

- SSER No. 2 - Twenty-two issues were reported resolved; Issue (6) was deleted; and Issues (51), (52), and (53) were added.
- SSER No. 3 - Six issues were reported resolved.
- SSER No. 4 - Nine issues were reported resolved; Issue (35) was reopened; and Issues (54), (55), and (56) were added. Additionally, the staff's acceptable findings relative to the containment annulus concrete design modification and flaws detected in the steel shell weld radiographs were reported (Issue 3)).
- SSER No. 5 - Issues (8), (11), (12), and (55) were reported resolved; Issue (52) was deleted; and Issues (57) through (61) were added.
- SSER No. 6 - Issue (61) was partially resolved, and Issues (62) and (63) were added.
- SSER No. 7 - Confirmatory Issues (1), (7), (23), (28), (31), (35), (56), (57), (59), (60), (61), and (62) were reported resolved; Confirmatory Issues (41) and (42) were reopened; and Confirmatory Issue (64) was added.
- SSER No. 8 - Confirmatory Issues (41), (42), (51), (54), (63), and (64) were resolved, and Confirmatory Issue (65), "Control Room Habitability," was added.

This supplement adds Confirmatory Issue (66), "1986 Ohio Earthquake Items," introduced in SSER No. 9 (which solely documented the staff's evaluation of the effects of the 1986 Ohio earthquake on the Perry plant's seismic design). The staff's evaluation findings relative to these two open confirmatory issues are documented in this supplement and conclude that the issues/items involved have been sufficiently resolved or completed to permit operation of Perry Unit 1 at power levels above 5% of rated thermal power. A composite history of the 49 issues in the SER and those added in SSER Nos. 1 through 9 follows:

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(1) Piping final stress analysis	Resolved in SSER No. 7	----
(2) Containment buckling analysis	Resolved in SSER No. 1	----
(3) Containment ultimate capacity analysis	Resolved in SSER No. 1; staff's acceptance of the concrete "annulus fix" design modification and flaws detected in steel shell weld radiographs addressed and resolved in SSER No. 4	----
(4) Emergency service water tunnel structure analysis	Resolved in SSER No. 1	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(5) Vibration monitoring program for BOP systems	Resolved in SSER No. 1	----
(6) Mark III containment hydrodynamic loads	Resolved in SSER No. 2	----
(7) Testing safety-relief valves per TMI Action Plan Item II.D.1	Resolved in SSER No. 7	----
(8) IE Bulletin 79-02	Resolved in SSER No. 5	----
(9) Dual function pipe whip/support restraints	Resolved in SSER No. 2	----
(10) Hydrodynamic effect on CRD/HCU	Resolved in SSER No. 4	----
(11) Fuel mechanical fracturing	Resolved in SSER No. 5	----
(12) Fuel assembly damage from external sources	Resolved in SSER No. 5	----
(13) Fuel rod bowing	Resolved in SSER No. 3	----
(14) Overheating of gadolinia fuel pellets	Resolved in SSER No. 4	----
(15) Preservice/in-service inspection programs	PSI program partially resolved in SSER No. 5; fully resolved in SSER No. 7.	----
(16) Material surveillance program - reactor vessel (RV) beltline materials	Resolved in SSER No. 2	----
(17) Fracture toughness RCPB materials	Partially resolved in SSER No. 2; added to Technical Specifications and fully resolved in SSER No. 7	----
(18) HPCS and RCIC initiation per TMI Action Plan Item II.K.3.13	Resolved in SSER No. 7	----
(19) Isolation of HPCS and RCIC per TMI Action Plan Item II.K.3.15	Resolved in SSER No. 7	----
(20) Subcompartment pressure analysis	Resolved in SSER No. 3	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(21) Suppression pool temperature limits	Resolved in SSER No. 4	----
(22) Secondary containment penetration leakage	Resolved in SSER No. 2	----
(23) Containment isolation dependability per TMI Action Plan Item II.E.4.2(f)	Partially resolved in SSER No. 2; partially resolved in SSER No. 5; fully resolved in SSER No. 7	
(24) Type C test of all ECCS injection valves	Resolved in SSER No. 2	----
(25) ADS logic modification per TMI Action Plan Item II.K.3.18	Resolved in SSER No. 4	----
(26) ATWS recirculation pump trip	Resolved in SSER No. 4	----
(27) Modified SDV level monitoring system	Resolved in SSER No. 2	----
(28) HPCS initiation circuitry final design	Resolved in SSER No. 2; site confirmatory audit conducted in April 1985; resolved in SSER No. 7	----
(29) Remote shutdown panel non-safety-grade readouts	Resolved in SSER No. 2	----
(30) RCIC testing procedures	Resolved in SSER No. 2	----
(31) Calibration of relief valve/safety valve (RV/SV) pressure switches (TMI Action Plan Item II.D.3)	Resolved in SSER No. 2; site confirmatory audit conducted in April 1985; resolved in SSER No. 7	----
(32) Accident monitoring per TMI Action Plan Items II.F.1.4, II.F.1.5, and II.F.1.6	Resolved in SSER No. 2	----
(33) Failures in vessel level sensing lines common to control and reactor protection systems	Resolved in SSER No. 2	----
(34) Final valve design setpoint and analysis	Resolved in SSER No. 2	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(35) Physical separation of redundant electrical systems	Initially resolved in SSER No. 2; reopened in SSER No. 4 on basis of staff onsite audit; finally resolved in SSER No. 7	----
(36) Documentation or test of 3-hour-fire resistance of gypsum board walls	Resolved in SSER No. 4	----
(37) Light and communication fire protection features	Resolved in SSER No. 3	----
(38) Revision of fire protection standpipe and hose locations	Resolved in SSER No. 3	----
(39) Portable fire extinguisher locations	Resolved in SSER No. 3	----
(40) Watertight curbs in switchgear/diesel generator rooms	Resolved in SSER No. 3	----
(41) Design for noble gas effluent monitors per TMI Action Plan Item II.F.1.1	Initially resolved in SSER No. 4; reopened in SSER No. 7; resolved in SSER No. 8	----
(42) Design for sampling and analysis of plant effluents per TMI Action Plan Item II.F.1.2	Initially resolved in SSER No. 4; reopened in SSER No. 7; resolved in SSER No. 8	----
(43) Leakage surveillance preventive maintenance program per TMI Action Plan Item III.D.1.1	Changed to License Condition (16) in SSER No. 1; resolved in SSER No. 8	----
(44) Radiation/shielding design of IFTS tube	Resolved in SSER No. 2	----
(45) Location of plant area radiation monitoring per TMI Action Plan Item II.F.1.3	Resolved in SSER No. 2	----
(46) Training program per TMI Action Plan Item II.B.4	Resolved in SSER No. 2	----
(47) Nuclear section training program	Resolved in SSER No. 2	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(48) Shift supervisor training per TMI Action Plan Item I.C.3	Resolved in SSER No. 2	----
(49) Verify implementation of equipment control measures in radiation areas per TMI Action Plan Item I.C.6	Resolved in SSER No. 2	----
(50) No load, light load, and test loading of the diesel generators	Resolved in SSER No. 2	----
(51) NSSS vendor review of low-power ascension and emergency operating procedures per TMI Action Plan Item I.C.7	Resolved in SSER No. 8	----
(52) Pilot monitoring of selected emergency operating procedures per TMI Action Plan Item I.C.8	Deleted in SSER No. 5	----
(53) Reactor internals vibration prototype (BWR/6-238 in.) test program	Made a confirmatory issue in SSER No. 2; resolved in SSER No. 4	----
(54) Preoperational and periodic testing plans for the two subsystems in the permanent dewatering system	Added in SSER No. 4; resolved in SSER No. 8	----
(55) Labeling LPCI/LPCS injection valve switches to warn operator that inadvertent operation could cause overpressurization	Added in SSER No. 4; resolved in SSER No. 5	----
(56) Containment purge system - details of program to be used to determine purge criteria for life of plant after first refueling cycle, to be submitted 6 months before initial fuel load	Added in SSER No. 4; resolved in SSER No. 7	----
(57) SPDS design requirements per NRC Generic Letter 82-33	Added in SSER No. 5; resolved in SSER No. 7; postimplementation audit required	----

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(58) Confirmation that the lowest temperature will be experienced by the limiting materials of the reactor containment pressure boundary under conditions cited in General Design Criterion 51	Added in SSER No. 5; resolved in SSER No. 6	----
(59) Clarification of changes made in Chapter 17 of FSAR Amendments 13, 14, and 15	Added in SSER No. 5; resolved in SSER No. 7	----
(60) Use of meteorology as a part of plant emergency response capability	Added in SSER No. 5; resolved in SSER No. 7	----
(61) Emergency plans	Added in SSER No. 5; resolution of onsite emergency plan issues reported in SSER No. 6; FEMA findings on offsite emergency preparedness documented and this issue changed to License Condition (31) in SSER No. 7	----
(62) Resolution of questions related to documentation of preoperational/startup test in Amendments 15 and 16 of the FSAR, Chapter 14 per RG 1.68	Added in SSER No. 6; resolved in SSER No. 7	----
(63) TDI diesel generator reliability	Added as a confirmatory issue in SSER No. 6 and resolved for low-power operation of Unit 1; resolved for full-power operation predicated on License Condition (32) stated in Section 1.11 of SSER No. 8	----
(64) Mark III containment (Humphrey) issues	Added as a confirmatory issue in SSER No. 7; resolved in SSER No. 8	----
(65) Control room habitability - detailed review of thyroid doses to control room personnel and any required system modification	Added in SSER No. 8; resolved in this supplement	6.4

<u>Issue</u>	<u>Status</u>	<u>Section</u>
(66) January 31, 1986, Ohio earthquake	Added in SSER No. 9; resolved in this supplement	
(a) Fault plain solu- tions of the earth- quake and its after- shocks; identifica- tion of a possible source structure		2.5
(b) Possible impact of injection wells in the vicinity of the earthquake epicenter		2.5
(c) Assessment of geologi- cal faults at the plant site		2.5
(d) Consideration of the impact of enriched high- frequency content earthquakes		2.5
(e) Further generic evaluations of energy content and potential safety signifi- cance of high- frequency, short- duration earth- quakes		3.7.2, 3.10.1
(f) Relocation of plant seismic instrumen- tation		3.7.3
(g) Modification of specific plant procedures		3.7.3
(h) Additional assess- ments of seismic qualification of plant equipment		3.10.1

1.11 License Conditions

In Section 1.11 of the SER, the staff identified 15 license conditions. These included several issues that had to be settled before the licensing of the

Perry plant and other longer term issues (noted by asterisk) that were to be cited in the licenses issued for Perry Units 1 and 2, to ensure that NRC requirements would continue to be met. Conditions added, redefined, and deleted in SSER Nos. 1 through 8 follow:

- SSER No. 1 - License Condition (8) was deleted and added to the list of issues in Section 1.10 of the SER as Confirmatory Issue (50), and Confirmatory Issue (43) was made License Condition (16).
- SSER No. 2 - License Condition (17) was added (also listed as Confirmatory Issue (25) in SER Section 1.10) as was License Condition (18). The results of the staff's generic evaluation of License Condition (2) were also presented.
- SSER No. 3 - License Condition (19) was added, the partial findings of the staff's review of information received pertaining to License Condition (4) were reported, and License Condition (4) was redefined.
- SSER No. 4 - License Conditions (20) through (25) were added, License Condition (7) was resolved, and License Condition (1) was rephrased to make it consistent with the statement in Section 2.5.5 of the SER text.
- SSER No. 5 - License Conditions (4) and (18) were deleted; License Condition (13) was redefined; and License Conditions (26), (27), (28), and (29) were added.
- SSER No. 6 - License Condition (10) was modified based on the staff's updated review discussed in Section 13.1.2.3 of SSER No. 6, License Conditions (5) and (15) were deleted, and License Condition (29) was redefined.
- SSER No. 7 - License Conditions (2), (3), (6), (9), (11), (12), (16), (17), (20), (22), (23), (26), (27), and (28) were deleted; License Conditions (30) and (31) were added; and all remaining license conditions as they are to be documented in the operating license issued for Perry Unit 1 were redefined.
- SSER No. 8 - License Conditions (1), (10), (19), (24), and (29) were deleted; conditions listed in SSER No. 7 that were cited in the operating license issued for Perry Unit 1 (NPF-45) on March 18, 1986, were further clarified; and License Conditions (32) and (33) - also included in the NPF-45 license - were added. All conditions confirmed the staff's evaluation of FSAR amendments through Amendment 24 and other supplemental information contained in correspondence from the licensee. The numerical assignment of each condition was retained in SSER No. 8 for reference to previous SSERs and the SER.

This supplement documents the deletion of those conditions, in part or in their entirety, where the conditions specified in the NPF-45 operating license had to be completed before plant operation above 5% of rated thermal power. A composite of those conditions, including those that will be documented in the license authorizing Perry Unit 1 operation above 5% of rated thermal power, follows. As in SSER No. 8, the numerical assignment of each condition is retained for reference in future SER supplements.

(13) Plant Security (SSER No. 7, Section 13.6)

CEI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans, including all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p), which are part of the license. The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Perry Nuclear Power Plant Security Plan," with revisions submitted through May 15, 1986; "Perry Nuclear Power Plant Security Force Training and Qualification Plan," with revisions submitted through August 12, 1986; and "Perry Nuclear Power Plant Safeguards Contingency Plan" (Chapter 8 of the Security Plan), with revisions submitted through May 15, 1986.

(14) Post-Fuel Loading Initial Test Program (SSER No. 3, Section 14, TMI Action Plan Item I.G.1)

Any changes to the Initial Test Program described in Section 14 of the FSAR shall be made in accordance with the provisions of 10 CFR 50.59 and shall be reported in accordance with 10 CFR 50.59(b) within 1 month of the change.

(25) Inservice Inspection Program (SSER No. 4, Section 6.6.3)

CEI shall submit the initial inservice inspection program required by 10 CFR 50.55(a) for NRC staff review and approval within 6 months after exceeding 5% of rated thermal power.

(30) Detailed Control Room Design Review (SSER No. 10, Section 18.2)

Before start of the 100-hour warranty run, CEI shall implement corrections to human engineering discrepancies per commitments in Supplement 2 to the Detailed Control Room Design Review Summary Report, dated August 26, 1986, and in a letter from M. R. Edelman to W. R. Butler, dated August 26, 1986.

Before startup following the first refueling outage, CEI shall implement corrections to human engineering discrepancies per commitments in

- (a) the Detailed Control Room Design Review Summary Report, dated January 10, 1985
- (b) Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 14, 1985
- (c) Revision 1 to Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 21, 1985
- (d) Supplement 2 to the Detailed Control Room Design Review Summary Report, dated May 28, 1986
- (e) the Control Room Validation Summary Report, dated July 11, 1986

- (f) errata sheets to Supplement 2 to the Detailed Control Room Design Review Summary Report, attached to letter PY-CEI/NRR-0510 L, dated July 29, 1986

Before startup following the first refueling outage, CEI shall also provide results of the final sound surveys in the control room and at the remote shutdown facilities for NRC review per the commitment in Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 14, 1985.

Before startup following the second refueling outage, CEI shall complete the augmented verification of human engineering discrepancy corrections implemented after full-power licensing per the commitment in Supplement 2 to the Detailed Control Room Design Review Summary Report, dated May 28, 1986. CEI shall also correct any problems identified by the augmented verification before startup following the second refueling outage per the commitment in a letter from M. R. Edelman to W. R. Butler, dated August 26, 1986.

(31) Emergency Plans (SSER No. 7, Section 13.3)

CEI shall comply with the following emergency preparedness requirements:

- (a) In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule (44 CFR 350) is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR 50.54(s)(2) will apply.

(32) TDI Diesel Generator Reliability (SSER No. 8, Section 9.6.3)

CEI shall comply with the following requirements related to the TDI diesel generator engines:

- (a) Changes to the maintenance and surveillance program for the TDI diesel engines, as identified and approved by the NRC staff in the supplemental safety evaluation report transmitted by NRC letter dated November 5, 1985 (W. R. Butler to M. R. Edelman), shall be subject to the provisions of 10 CFR 50.59.
- (b) Crankshafts shall be inspected as follows: The oil holes and fillets of the three main bearing journals subject to the highest torsional stresses (Nos. 4, 6, and 8) shall be examined with fluorescent liquid penetrant and, as necessary, eddy current during the one-time 5-year and each 10-year major disassembly. The same inspection on oil holes and fillets shall be performed on at least three crankpin journals between journals 3 and 8.
- (c) Cylinder blocks shall be inspected at intervals calculated using the cumulative damage index (CDI) model and using inspection methodologies described by Failure Analysis Associates, Inc. (FaAA), report entitled "Design Review of TDI R-4 Series Emergency Diesel Generator Cylinder Blocks" (FaAA-84-9-11), dated December 1984. Liquid penetrant inspection of the cylinder liner landing area shall be performed any time liners are removed.

- (d) The diesel engine shall be rolled over with the airstart system and the cylinder stopcocks open before any planned starts, unless the start occurs within 4 hours of a shutdown. The diesel engines shall also be rolled over with the airstart system and the cylinder stopcocks open after 4 hours, but no more than 8 hours, after engine shutdown and then rolled over once again approximately 24 hours after each shutdown. In the event a diesel engine is removed from service for any reason other than the rolling over procedure before expiration of the 8- or 24-hour periods noted above, that diesel engine need not be rolled over while it is out of service. CEI shall air roll the diesel engine over with the stopcocks open at the time it is returned to service. The origin of any water detected in the cylinders must be determined, and any cylinder head that leaks as a result of a crack shall be replaced. No cylinder heads that contain a throughwall weld repair, where the repair was performed from one side only, shall be used on the diesel engines.
- (e) If inspections of either TDI generator engine reveal cracks in the crankshaft or in the cylinder block between stud holes of adjacent cylinders, this condition shall be reported promptly to the NRC staff and the affected engine(s) shall be considered inoperable. The engine(s) shall not be restored to "operable" status until the proposed disposition and/or corrective actions have been approved by the NRC staff.

(33) Fire Protection (SER, SSER Nos. 1, 2, 3, 4, 7, and 8, Section 9.5)

CEI shall comply with the following requirement of the fire protection program:

- (a) CEI shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the Perry Nuclear Power Plant, as amended and as approved in the Safety Evaluation Report (NUREG-0887) dated May 1982 and SER Supplement Nos. 1-8 thereto, subject to the following provisions:
 - (i) CEI may not make changes for the approved fire protection program which would adversely affect the ability to achieve and maintain safe shutdown in the event of a fire without prior approval of the Commission.

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.8 Accidental Releases of Liquid Effluents in Ground and Surface Waters

In FSAR Amendment 25 submitted by letter dated June 27, 1986 (M. R. Edelman to W. R. Butler), the licensee stated that during preoperational testing, groundwater inflow to the plant underdrain system was measured at 11.7 gallons per minute (gpm). During plant construction, the groundwater inflow was estimated to be about 4 gpm.

The licensee's analysis of an accidental spill of radioactive effluents was initially dependent on the assumed groundwater inflow into the plant underdrain system. In the original licensee analysis, the estimated inflow rate was 4 gpm. The licensee revised the analysis to consider a higher groundwater inflow of 15 gpm. This resulted in less radionuclide decay within the underdrain system, but the concentrations of all released radionuclides were still estimated to be below the maximum permissible concentrations listed in Part 20 of Title 10 of the Code of Federal Regulations (10 CFR 20), Appendix B, Table 2.

The staff's independent analysis was not directly related to measured inflows to the underdrain system. Instead, the staff used estimated long-term inflow rates that were based on the permeability and gradient in the lacustrine soil at the site. This estimate remains unchanged by the preoperational testing measurements, and the staff's conclusion in the SER is unchanged; that is, the plant meets the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluents.

2.5 Geology and Seismology

On January 31, 1986, at 11:46 a.m. EST, an earthquake of magnitude 5.0 (m_{blg}) occurred about 16 km (10 mi) south of the Perry plant in northeastern Ohio. The maximum intensity was VI (Modified Mercalli). The staff's consultant, the U.S. Geological Survey (USGS), and the licensee's consultants have submitted the following reports on their evaluations of the January 31 earthquake and its aftershocks:

- (1) R. D. Borchardt, ed., "Preliminary Report on Aftershock Sequence for the Earthquake of January 31, 1986, near Painesville, Ohio," U.S. Geological Survey Open-File Report 86-181, 1986
- (2) R. L. Wesson and C. Nicholson, eds., "Studies of the January 31, 1986, Northeastern Ohio Earthquake," U.S. Geological Survey Open-File Report 86-336, 1986
- (3) P. Talwani and S. Acree, "Deep Well Injection at the Calhio Wells and the Leroy, Ohio Earthquake of January 31, 1986," submitted by Cleveland Electric Illuminating Company letter dated June 17, 1986 (M. R. Edelman to W. R. Butler)

- (4) Weston Geophysical Corporation, "Investigations of Confirmatory Seismological & Geological Issues, Northeastern Ohio Earthquake of January 31, 1986," submitted by Cleveland Electric Illuminating Company letter dated June 24, 1986 (M. R. Edelman to W. R. Butler)

SSER No. 9 discussed the following geological and seismological confirmatory issues that resulted from the occurrence of the January 31, 1986, earthquake: fault plane solutions of the main shock and aftershocks, the search for a possible source structure, the possible effect of injection wells, assessment of faults at the plant site, and consideration of the effect of enriched high-frequency content. These confirmatory issues are discussed in the following sections and considered resolved. The engineering significance of the in-plant seismic recordings is discussed in Section 3.7 of this report.

Fault Plane Solutions and the Search for a Possible Source Structure

The Perry site is located in the Central Stable Region tectonic province. In the site vicinity, Paleozoic sedimentary rock formations, about 1524 m (5000 ft) thick, overlie a Precambrian crystalline basement. There are no known capable faults in the site region. Within 321 km (200 mi) of the site, the only historical or instrumental seismicity that has been associated with a tectonic structure is near Attica, New York, about 257 km (160 mi) from the Perry site. Earthquake activity around the vicinity of the Perry site is not substantially different from that of the Central Stable Region.

The January 31, 1986, earthquake and its aftershocks occurred in a cluster about 16 km (10 mi) south of the Perry plant. At least five research teams deployed portable seismometers and accelerometers to record aftershocks. As of mid-July, 15 aftershocks were detected. The aftershocks range in magnitude from 2.5 to less than 0. The cluster of epicenters has a diameter of about 1 km (0.6 mi) and the focal depths range from 2 to 7 km (1.2 to 4.3 mi). The main shock is assumed to have occurred in this depth range. Thus, the events are occurring in the Precambrian basement. There is some suggestion of an alignment in the aftershock locations with an orientation of 15° to 20° east of north.

Fault plane solutions are available for the main shock and the aftershocks. The fault plane solutions for the main shock and the largest aftershocks indicate predominantly strike-slip motion, which is right lateral if the north-northeast plane is assumed to be the fault plane. The rupture plane is almost vertical. The other aftershocks have different mechanisms, with a larger component of dip slip motion. The different orientations of slip suggest that more than one fault plane moved. The P-axis (maximum compressive stress direction) for the main shock and larger aftershocks is almost horizontal and close to east-west, similar to the average stress direction observed in earlier studies for the Eastern United States.

Since no association was established with a known geological structure, the licensee examined geological and geophysical data in the area for any previously unidentified structures. No earthquake-related structures were found in outcrops in the epicentral area. The staff accompanied licensee personnel to examine geological features discovered as part of the geological reconnaissance of the epicentral area. The features observed were folds, thrust faults, and popups in the Devonian (410 to 360 million years ago (mya)) shale bedrock. All of these features have limited lateral extent and are underlain by undeformed

strata. These features are due to either tectonic compression related to the formation of the Appalachian Mountain Belt 240 mya or Pleistocene glacial activity (2.5 million to 10,000 years ago) and are not capable.

The licensee constructed structural contour maps of two subsurface Paleozoic (570 to 240 mya) horizons from gas well data. The structural contour map on top of the Packer Shell, which is at a depth of about 732 m (2400 ft) in the epicentral area, shows a gentle south trending dip. Secondary features have a relief up to about 12.2 m (40 ft). The closest feature of interest is a north-northeast trending high, which starts about 1.6 km (1 mi) west of the epicentral area and continues to about 6.4 km (4 mi) north-northeast of the epicentral area. This feature has the same trend as one of the fault plane directions found in the fault plane solutions. Although the Packer Shell does not appear offset, the north-northeast trending feature could be associated with a deeper structure where the 1986 earthquake occurred.

The other structural contour map is on top of the Delaware Limestone, which is at a depth of about 244 m (800 ft) in the epicentral area. The Delaware Limestone dips at about the same rate as the Packer Shell; however, the dip of the Delaware is toward the southeast. The feature described in the Packer Shell does not continue upward to the Delaware. The closest feature of interest is a high that starts just south of the epicentral area and trends south-southeast for about 8 km (5 mi).

The features revealed by the contour maps could be depositional or erosional features or Paleozoic folds or faults. Unfortunately, similar maps could not be constructed for the Precambrian (before 570 mya) basement, where the earthquakes are occurring. To assess the geological characteristics at hypocentral depths of 2 to 7 km, the licensee conducted aeromagnetic and ground gravity surveys in the epicentral area. The magnetic and gravity anomalies originate largely in the Precambrian basement rocks.

The aeromagnetic data for Ohio show a change in magnetic texture from complex anomalies in central Ohio to smooth anomalies in eastern Ohio. The lineament that marks the change in magnetic character is assumed to mark a change in the physical characteristics of the Precambrian basement. Near the epicentral area this boundary trends northeast and lies about 5 mi to the southeast of the recent epicenters. The map resulting from the detailed aeromagnetic survey conducted for the licensee shows a northeast trending magnetic pattern with a magnetic low passing through the epicentral area.

The gravity anomaly map shows the 1986 epicentral area is on the east side of a positive gravity anomaly near a deflection in contour lines. This gravity high is centered near a magnetic high. When the data are adjusted to the same elevation level as the magnetic survey and filtered, the residual gravity map indicates a northeast regional trend.

The gravity and magnetic data were modeled to determine potential sources of the anomalies. The anomalies were related to changes in basement lithologies. The licensee concluded that the geophysical evidence does not localize a unique structure in the basement and there is no evidence of strike slip motion along a north-northeast trend.

The staff concludes that the geological and geophysical studies have found a northeast regional trend in the gravity and magnetic data, but no obvious geological structure associated with the January 31, 1986, earthquake. Hence, there is no discernible capable fault associated with this earthquake. The magnitude, depth, and maximum compressive stress direction for the 1986 earthquake were similar to those of other events that have occurred in the Eastern United States.

The concept of tectonic province was developed to provide an appropriate design basis for earthquakes, such as the January 31 event, whose cause is currently indeterminate. The NRC staff interprets tectonic provinces to be large regions of similar geology and uniform earthquake potential. The most important factors for the determination of tectonic provinces are (1) the development and characteristics of the current tectonic regime of a region, which is most likely reflected in the neotectonics (about 5 million years and younger geological history) and (2) the pattern and level of historical seismicity. For the Perry site the controlling earthquake for the seismic design basis was the largest event not associated with geological structure in the Central Stable Region tectonic province - a magnitude 5.3 event. The consideration of the largest event not associated with geological structure ensures consideration of as yet undefined structures that might cause earthquakes in the vicinity of a site, such as the January 31 event near the Perry site.

Possible Effect of Injection Wells

The USGS and the licensee's consultants explored the possibility that the recent seismicity may be related to injection of chemical wastes in two Calhio wells about 4.8 km (3 mi) south of the Perry plant and about 11.3 km (7 mi) north of the 1986 earthquakes. The large volume of waste that has been injected and past experience with seismicity associated with deep well injection in Colorado and at some oil and gas fields led to speculation that this fluid injection might have triggered the January 31, 1986, earthquake.

The two Calhio wells have been in operation since 1974 and 1981, respectively. Both wells are about 1858 m (6100 ft) deep, extending a short distance into the Precambrian basement. Since operation of the wells began, in addition to the January 31, 1986, earthquake and its aftershocks, three small earthquakes were detected in the vicinity of the wells - a January 22, 1983, magnitude 2.7 earthquake about 3 mi northeast of the injection wells; a November 19, 1983, magnitude 2.5 event at the same location; and a March 12, 1986, magnitude 0.3 earthquake 3.2 km (2 mi) south-southwest of the wells. These three events were relocated by Weston Geophysical as part of the confirmatory studies. The historical seismicity in northeastern Ohio has a diffuse pattern. The largest event in the vicinity of the wells before 1986 was a magnitude 4.5 event in 1943 about 10 mi southwest of the wells.

The licensee concluded that although it is possible for the January 31, 1986, earthquake to have been induced, it is highly unlikely that it was. The distance between the injection wells and the hypocenters is greater than in cases where seismicity has been correlated with injections. The time delay between the start of injection at the Calhio wells and the onset of seismicity is much larger. Injection-induced seismicity is also characteristically more temporally and spatially diffuse and is characterized by many more small events than the

1986 sequence. The microearthquakes closer to the wells in 1983 and on March 12, 1986, may have been induced by the wells or could be minor tectonic events; however, the licensee concluded that these events are not spatially or temporally associated with the January 31, 1986, event.

The USGS has performed calculations based on the estimated state of crustal stress in the epicentral area and the measured injection pressure to determine whether the theoretical threshold for the occurrence of an earthquake is met. Without fluid injection, it appears that the conditions are near but do not exceed failure at the bottom of the wells. Fluid injection could have brought at least the region near the bottom of the well into a critical stress state. The absence of any known earthquakes in the immediate vicinity of the well suggests there are no favorably oriented weak fractures near the well. The shear stress is maximum only for near-vertical faults, which was observed in the fault plane solution and aftershock distribution of the January 31 event. The predominant dip of fractures observed in a core taken from Calhio #2 is 20°; such fractures would not be favorably oriented for failure.

The USGS has estimated the state of stress at the hypocenter of the January 31 event. Here also the analysis indicates a near-critical stress state for favorably oriented pre-existing fractures. Using several models, the USGS estimated fluid pressure changes in the hypocentral area resulting from fluid injection. The preferred model yields an estimate of about 2 bars for the increase in fluid pressure 7 mi from the well; this increase is small compared with the estimated fluid pressure of 590 bars at the hypocentral depth of 5 km (3 mi). The USGS concluded:

In light of the fact that the mainshock and most of the aftershocks occurred at considerable distance from the active wells, the pressure fall-off with distance from the wells, the occurrence of small to moderate earthquakes in this region prior to initiation of injection, the lack of large numbers of small earthquakes (commonly observed in cases of induced seismicity) and the lack of earthquakes immediately below the wells all argue for a "natural" origin for the earthquake on January 31st. Thus, although triggering remains a possibility, the probability based on existing data that the injection wells played a significant role in causing the earthquake sequence is considered low.

The staff agrees with the licensee's and USGS conclusions. Past experience with induced seismicity has shown seismicity beginning near the wells and later spreading to surrounding areas. In the case of the January 31, 1986, earthquake, no seismicity had been reported before this event near the wells, and the recent earthquakes are about 7 mi from the wells. In addition, previous seismicity, such as the 1943 magnitude 4.5 earthquake, occurred in the vicinity before the construction of the wells. As a result, the staff considers it unlikely that the 1986 seismic event was induced by these wells.

Before the installation of portable seismometers following the January 31 earthquake, the detection threshold for northeastern Ohio was about magnitude 2 to 2½ (NUREG/CR-1649). Consequently, it is conceivable that small earthquakes could have occurred close to the wells between the initiation of injection and the January 31 earthquake. Although the licensee concluded it is highly unlikely that the January 31, 1986, earthquake was induced by injection at the Calhio

wells, the licensee also recognized the potential for induced seismicity and is developing a seismic monitoring network around the injection wells. This network would permit detection of small events (as low as magnitude 0). The staff believes this network would provide data to assess any possible connection between deep well injection and future seismicity and to possibly identify any causative structures for earthquakes in this region. The licensee's network will be operated through 1988, at which time the licensee and the staff will reevaluate its continued operation.

Assessment of Faults at the Plant Site

The Perry reactor building foundation is Devonian shale bedrock. During the plant site excavations, faults were mapped in the foundation excavation and intake and discharge tunnels under Lake Erie. These faults were discussed in the SER and judged to be noncapable. One of the bases for reaching this conclusion was that the faults were not properly oriented to fail in the present stress regime. Stress directions from fault plane solutions of the January 31, 1986, earthquake and its aftershocks are generally consistent with those previously assumed. The staff sees no reason to change its judgment as to the noncapability of the foundation and tunnel faults, as a result of the January 31, 1986, earthquake.

Consideration of the Effect of Enriched High-Frequency Content

The January 31, 1986, earthquake activated the in-plant seismic monitoring instruments. Some of the recorded ground motions exceeded the operating-basis earthquake and safe shutdown earthquake design spectra at high frequencies (above 15 Hz). The earthquake motion recorded at the reactor building foundation was of short duration (about 1 sec) and predominantly at high frequencies. However, the earthquake was not recorded in the free field outside the plant. The licensee and the USGS assessed all available ground motion recordings from the main shock and aftershocks to determine whether the high-frequency exceedance recorded at the Perry plant was due to the earthquake source, path effects, local site conditions, or building response effects. The possible effect of the building response is discussed in Section 3.7; the in-plant recordings were judged to be similar in frequency content to the free-field ground motion.

The USGS deployed analog and broad-band digital instrumentation (GEOS) to record aftershocks. The GEOS time histories and corresponding spectra are shown in USGS Open-File Report 86-181 (Borcherdt, 1986). The high sampling rate of 400 samples per second used to record the GEOS time histories resulted in accurate resolution of peak amplitudes and spectra plots up to 200 Hz. The recorded aftershock time histories are relatively rich in high-frequency content (up to 30 to 70 Hz and even some recorded ground motions above 100 Hz). Spectra computed for the aftershocks show amplified 20-Hz ground motion at a GEOS site near the Perry plant compared with sites closer to the hypocenters. Spectra computed for the main shock recorded in the plant also show amplified 20-Hz shaking. The observation of amplified 20-Hz motion outside the plant suggests that some combination of earthquake source, travel path, or site effects may be responsible for the high-frequency exceedance recorded in the plant.

The staff examined the spectra computed from the GEOS aftershock data. In general there was little attenuation in the recorded ground motions out to a

distance of 18 km (11.2 mi) from the epicenters. Also, as expected, the horizontal components are generally larger than the vertical components. There are also characteristic spectral shapes and predominant frequency peaks at some sites. For example, the GEOS aftershock recordings near the plant have a predominant frequency of about 20 Hz, the GEOS recordings 9 km (5.6 mi) north of the epicenters have a predominant frequency of 10 to 40 Hz, and the GEOS recordings about 1 km (0.6 mi) from the epicenters have a characteristic frequency of about 10 Hz. The licensee has suggested that differences in the predominant frequencies at each site depend on the type of fault plane solution. However, the staff observes that differences in spectra between sites are more pronounced than the effect of fault plane solutions.

Local soil conditions may amplify ground motion, and certain geological conditions can show predominant ground frequencies. USGS modeling studies of the soil column under the GEOS station near the plant suggest that vertical peaks in the spectra near 20 Hz may be attributable to resonances of the soil layers. However, the Perry plant structures, which are founded on rock, recorded 20-Hz spectral peaks in the horizontal direction. In addition, energy at 20 Hz was recorded by most of the GEOS instruments and was certainly part of the earthquake source.

The staff conclusion is that the 20-Hz ground motion observed in the Perry plant was due at least in part to a seismic source possessing high frequencies. The ground motion recorded in the plant was due to some combination of source, path, and site effects.

Improvements in recording capability have contributed to investigations of high-frequency ground motion. High-amplitude, high-frequency accelerations of limited duration were recorded in previous studies (for example, New Brunswick and Monticello Reservoir). The Regulatory Guide 1.60 spectral shape does not envelop these recordings at high frequencies; however, as at Perry, these earthquakes did not result in any significant damage. Considerable effort has and is being expended, including NRC Office of Nuclear Regulatory Research contracts, in an attempt to determine the cause of high-frequency energy. As is apparent from the Perry case, this issue is extremely complex, and it is likely to be some time before the cause of the recorded high frequencies is fully understood. The question of conservatism in the plant design with regard to high-frequency free-field ground motions is discussed in Section 3.7 of this supplemental report.

St. Mary's, Ohio, Earthquake

On July 12, 1986, a magnitude 4.2 earthquake occurred near St. Mary's, Ohio, about 322 km (200 mi) southwest of the Perry plant. The maximum intensity was VI. This event was not felt at the Perry plant and did not trigger the in-plant seismic monitoring instruments. The event occurred near a cluster of seismicity at Anna, Ohio.

Conclusions

On the basis of geological and geophysical studies, no obvious geological structures have been associated with the January 31, 1986, earthquake. The aftershocks occur in a cluster, with some suggestion of a north-northeast alignment.

Fault plane solutions for the main shock and the largest aftershocks indicate predominantly strike-slip motion, which is right lateral if the north-northeast plane is assumed to be the fault plane. The magnitude, depth, and maximum compressive stress direction for the main shock and larger aftershocks were similar to those for other events that have occurred in the Eastern United States.

It is unlikely that injection of chemical wastes in two wells about 7 mi from the 1986 epicenter were related to the earthquake. However, the licensee will continue to seismically monitor the area around the injection wells.

The 20-Hz ground motion recorded in the plant is due to some combination of source mechanism, path, and site effects. Similar ground motions of short duration and high frequencies have been recorded during other events and did not result in significant damage. An assessment of the engineering significance of the high-frequency ground motion is provided in Section 3.7 of this supplement.

The staff, therefore, concludes that there are no remaining outstanding or confirmatory issues related to the January 31, 1986, earthquake.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.7 Seismic Design

3.7.2 Seismic System and Subsystem Analysis

In Section 3.7.2 of SSER No. 9, the staff concluded that, on the basis of the preliminary information available at that time, the Ohio earthquake of 1986 represented a negligible effect on the safe operation of the Perry plant and that the staff's conclusion as stated in SER Section 3.7 regarding the adequacy of the structural seismic design remained valid. Since then, the licensee has continued its effort in the prediction of building responses using the vibratory motion of the 1986 Ohio earthquake. In addition, the licensee, as requested by the staff in Section 3.10 of SSER No. 9, has performed a generic evaluation of the effect of a high-frequency, short-duration earthquake with regard to its potential safety significance for equipment and structures at Perry.

Reanalysis of Perry Structures

In a final report prepared by the licensee's consultant, Gilbert/Commonwealth Associates (G/C), dated June 16, 1986, the licensee has documented all the confirmatory efforts that have been made since SSER No. 9 was published. All seismic Category I structures (except the off-gas building, which does not house safety-related equipment) were reanalyzed to generate in-structure response spectra. For rock-founded structures, the recorded reactor building foundation time histories were used as input. Fixed-base analyses were performed. Structural damping corresponding to Regulatory Guide 1.61 was used and justified on the basis of the levels of predicted response in the containment vessel using these values.

For the diesel generator building, which is founded on fill, soil-structure interaction analyses were performed in a manner identical to those performed for design - a finite element approach. For the present reevaluation, the input motion was assumed to be the time histories recorded on the reactor building foundation; that is, the motion was assumed to be the free-field ground motion. These motions were applied assuming they existed in the soil column rather than on a hypothetical rock outcrop, which would have been more appropriate. Because of the soil column characteristics, the effect of applying the motion within the soil column is likely to add conservatism to the calculated response. The shear modulus value used corresponds to the shear strain value of 0.05% and is equivalent to 0.45 times the low strain shear modulus. The in-structure response spectra that were generated indicated a predominant mode at around 4 Hz for the horizontal direction and at around 20 Hz for the vertical direction.

The approach for generating in-structure response spectra is found to be acceptable.

The applicant has calculated elastic loads in the concrete shield building and the containment vessel. The stress levels in the containment vessel at the critical section were very low (see also Section 3.7.2, SSER No. 9). The force levels in the shield building were significantly less than the design loads. Hence, from an elastic analysis standpoint, the Ohio earthquake of 1986 induced very low levels of stress in the structures. The load levels in the diesel generator building were not evaluated by the applicant, since in-structure peak accelerations were below the corresponding design levels and, hence, would induce lower forces.

The staff has found the response assessment and load levels determined by the applicant to be acceptable.

Response Comparison

Comparisons were made, by the staff's consultant, J. Johnson (see Appendix T to this supplement for the report entitled "Review of the Effect of the January 31, 1986 Earthquake on the Perry Nuclear Power Plant," dated August 1986), between calculated and measured 2% damped response spectra on the containment vessel at elevation 688 ft. Structural damping was a constant 4% for all modes. Vertically incident waves were the wave propagation mechanism. The structural model used was a modified one with respect to the original dynamic model of the Perry reactor building. The modifications included are:

- (1) The soil springs located at the base of the model were deleted to obtain a fixed-base model.
- (2) The upper portion of the containment vessel was modeled to treat the polar crane as it was positioned during the earthquake (this dynamically couples torsion and east-west translation and increases rotational inertia about the east-west axis).
- (3) A massless node was added at the actual location of the containment vessel instrument to include the effects of torsion and rocking on the predicted response.

The response spectra comparisons are separated into three portions: low-frequency response (less than 10 Hz), response near 20 Hz, and zero period acceleration (ZPA). For the north-south component, there is no specific low-frequency peak, but there exists a predominant peak near 20 Hz. Near 20 Hz and at the ZPA, the measured responses are underpredicted by 20% to 30% by the analysis. For the east-west component, the measured response has a low-frequency peak at about 4 Hz, which is not reproduced in the analysis. The analysis amplifies motion near 7 Hz. Near 20 Hz and at the ZPA, the calculated responses overpredict the measured values by about 35%. For the vertical component, the low-frequency behavior matches well in amplitude and frequency content. Near 20 Hz and at the ZPA, the calculated responses overpredict the amplitude of response by about 30%.

An inspection of the calculated and measured acceleration time histories indicates that the strong motion portion of the recorded time histories is greater than that of the calculated motions. Also, a beat-type phenomenon is observed in the north-south and vertical components, which led the applicant to hypothesize that a portion of this motion was induced by secondary causes, such as

the polar crane. Although data are not available to resolve this issue, its resolution is not essential for this evaluation.

An attempt was made to understand the causes of the discrepancies in the measured and calculated responses. These discrepancies were considered to possibly be due to unrecorded foundation rocking and torsion induced by nonvertically propagating waves. However, comparison of the response spectra due to nonvertically propagating waves of various angles of incidence with those of the recorded motion shows little improvement in agreement for nonvertically incident waves. Therefore, it appears that vertically incident waves, as was originally assumed, lead to the best estimate of containment vessel response, especially near the 20-Hz spectral acceleration.

The effect of structural damping on response of the containment vessel was also studied. The only structural model modification incorporated in this study was a fixed-base structural mode, which does not include the effect of polar crane eccentricity or the massless node simulating the actual instrument location. Qualitatively, it was observed that a reduction of structural damping from 4% to 3% does not increase containment vessel response near 20-Hz enough to match the recorded motion in the north-south direction. On the other hand, in the east-west direction, the 5% damped case still would slightly overpredict the response. Finally, vertical response at the shell centerline is independent of structural damping.

Characteristics of the Ohio Earthquake

Methods of investigating the low damage potential of earthquakes of short duration and high frequencies, as utilized in a recent study by Kennedy et al. (NUREG/CR-3805), have been followed in a similar study for the Ohio earthquake of 1986 (Johnson, 1986). The following characteristics of the earthquake time histories were considered: Fourier energy, strong motion duration, and root-mean-square acceleration, etc. The study utilized ground motion recordings at Mitchell Lake Road during the March 31, 1982, New Brunswick, Canada, earthquake, as well as the Perry 1986 recorded time histories and the Perry design motions. A comparison of the energy in the earthquake indicates that the 1986 Ohio earthquake had less energy than those of any of the other records considered. Specifically, the Ohio earthquake had only 3.25%, 1.75%, and 1.94% of the energy in the Perry design motions for north-south, east-west, and vertical components, respectively.

Nonlinear Response

Nonlinear analyses were performed by the staff's consultant (Johnson, 1986) on simple single-degree-of-freedom models representing the fundamental horizontal frequency of the drywell (5.4 Hz) and the auxiliary building (8.9 Hz). Six records were considered as input motions: the two horizontal components of the Ohio earthquake of 1986; the 1982 Mitchell Lake Road, New Brunswick, records; and the Perry foundation design motion.

The inelastic behavior of the models was described by a shear wall model exhibiting stiffness degradation after yield and pinching of the hysteresis loop during loading direction reversal. The stiffness of the shear wall model beyond yield was taken to be 10% of the elastic stiffness. Scale factors were calculated which when applied to the input motions would achieve a nonlinear

deformation in the simple model of 1.85 times the yield deformation. This level of nonlinearity is expected when typical concrete shear walls, which were designed to the American Concrete Institute (ACI)-349 code ultimate capacity, are loaded to the force level corresponding to the acceleration of the design ground response spectra (Regulatory Guide 1.60) at the fundamental frequency of the structure. Here a damping value of 7% was considered. The scale factors resulting from the Ohio earthquake were 5.3 and 5.5, respectively, for the north-south and east-west components and the 5.4-Hz model, and were 6.7 and 4.3, respectively, for the north-south and east-west components and the 8.9-Hz model. These factors are somewhat higher than those calculated for the 1982 Mitchell Lake Road, New Brunswick records, and are much higher than those corresponding to the Perry foundation design motion. This result provides additional confirmation of the applicant's finding (see Section 3.10) that the Ohio earthquake of 1986 possessed much lower energy content and ductility demand than the Perry safe shutdown earthquake.

Conclusions

On the basis of the above evaluation of the applicant's confirmatory studies as well as the independent studies performed by the staff's consultant, the Ohio earthquake of 1986 is judged to have had an insignificant effect on the Perry plant structures. Further, it is judged that the Perry seismic analysis models would adequately predict the behavior of the reactor building when subjected to this event. Although it is recognized that a portion of the high-frequency motion recorded on the containment vessel may have been due to secondary effects, such as polar crane vibration or impact, data are not available to resolve the question. Nevertheless, the plant's seismic design for the structures is judged to remain acceptable and unaffected by the event. This concludes the staff's evaluation of the application's confirmatory actions on plant seismic design.

3.7.3 Seismic Instrumentation Program

SSER No. 9 reported on the effects of the January 31, 1986, earthquake that was felt and recorded at the power plant. The report identified a deficiency in the location of an earthquake instrument and in the operating procedures. By a letter dated March 3, 1986 (M. R. Edelman to W. R. Butler), the licensee agreed to relocate the instrument and enhance the operating procedure associated with the earthquake instrumentation. The licensee has completed the physical relocation of the response spectra recorder and revised the operating procedures to satisfy the NRC staff concerns. The licensee submitted the details of the instrument relocation and procedure enhancement by letter dated April 25, 1986 (M. R. Edelman to W. R. Butler).

The triaxial response spectra recorder (D51R170) was located on a structural steel platform that is cantilevered from the biological shield wall in the reactor building and recorded the motion of the platform during the earthquake. However, the platform was also used to anchor the supports for several pipes in the area and the motion recorded was an indeterminable combination of the platform structural motion and the motion induced by the pipe supports. Because the data from this location were not useable, the licensee agreed to move the

instrument to a location that would provide more meaningful information for evaluating the power plant response to a future earthquake. The instrument was relocated approximately 15 ft from the platform location to the biological shield wall at azimuth 201° and elevation 636.6 ft. The bracket that supports the instrument is anchored directly to the wall. The bracket is similar in design to the brackets used in other seismic instrument mountings at other locations in the plant. The licensee reports that the bracket has a natural frequency greater than 33-Hz.

The operating procedure, ONI-D51, in place at the time of the January earthquake was not explicitly clear on the actions required by the operators should an earthquake occur. The licensee has revised two procedures (EPI-A1 and ONI-D51) that deal with the sensing of an earthquake and operator actions required following the occurrence of an earthquake. Procedure EPI-A1 relates to the definition of Unusual Events, Alert, and Site Area Emergency conditions. This procedure contains the definition of the seismic instrument status necessary for the three plant conditions. Procedure ONI-D51 relates to the detailed information required by the operators if an earthquake should occur. The procedure outlines the items that should be checked by the operator and the limits for defining whether or not the plant has experienced an event that might have challenged a design limit. The procedure also contains the information required by the operators for earthquake instrumentation calibration following the earthquake.

The triaxial response spectra recorder (D51R170) was relocated to a position on the biological shield wall. This wall is massive reinforced concrete and stiff enough so that the seismic equipment measurements should be free from equipment feedback and record only the structural responses to the earthquake. The natural frequency of the supporting bracket is reported by the licensee to be larger than the maximum frequency of interest and, therefore, should not influence the recorded motion. The staff finds that the new location of the D51R170 response spectra recorder is acceptable.

Operating procedure ONI-D51 was modified by the licensee to clarify the interpretation of the response spectra recorder annunciator D51-R215 located just off the control room on panel H13-P696. The procedure was modified to clarify the seismic switch annunciator located on the accelerometer control and recording panels, H51-P021, located in the loose parts monitoring room one floor below the control room in the auxiliary building. A comparison was made by the staff of the seismic instrument information between the revised operating manual and the manual in place at the time of the January 31, 1986, earthquake. The staff concludes that the revised procedure is clearly understandable by a trained operator and is, therefore, acceptable.

Conclusions

The staff concludes that the concerns about the seismic instrumentation location and the confusion that may have been present in the procedures in place at the time of the January 31, 1986, earthquake have been resolved. No further action by the licensee is required.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

In SSER No. 9, the staff found that the previous conclusions, as presented in SSER Nos. 5 and 7, regarding the adequacy of the licensee's seismic qualification program remained valid for the vibratory motion produced by the Ohio earthquake of January 31, 1986. The conclusion was based on the results of detailed plant walkdowns which found no apparent equipment or structural damage that could be attributed to the Ohio earthquake, and on the licensee's reassessment of the seismic capability of a limited sample of equipment types. On the basis of the information available at that time, it was the staff's opinion that the earthquake did not have any significant effect from an engineering viewpoint on the equipment at the Perry plant. In other words, although the design-basis earthquake was exceeded for short durations at some high-, narrow-frequency region of the corresponding response spectra, the original overall plant equipment seismic qualification was not affected.

Since that time, the licensee has provided some additional information, as documented in a final report prepared by the applicant's consultant, Gilbert/Commonwealth Associates (G/C), dated June 16, 1986, addressing the following confirmatory items that were identified in SSER No. 9:

- (1) additional quantitative assessments of the seismic qualification of a more comprehensive sample of equipment types that are located at other elevations of different buildings, and that would cover equipment that has been qualified by the test method and by the analysis method
- (2) results of a generic evaluation, based on an acceptable analytical approach, of a high-frequency, short-duration earthquake with regard to its energy content and potential safety significance for equipment and structures at Perry; using the results obtained from the analysis, an assessment of quakes of similar characteristics, but with higher magnitude and/or longer duration, would occur near the site

A comprehensive list of equipment, consisting of about 160 items, was selected by G/C for the confirmatory study based on the following criteria:

- (1) active safety class equipment required for safe shutdown of the plant
- (2) list of equipment compiled by Lawrence Livermore National Laboratory, with frequencies higher than 14 Hz and with high confidence of low probability of failure values less than 0.5g
- (3) supplied by multiple vendors
- (4) active components qualified by analyses
- (5) valves and motor operators supported by piping systems
- (6) electrical switchgear and instrument racks

- (7) vertical pumps
- (8) batteries and battery racks

The staff finds that the list as presented in the final report is an adequate representation of equipment required for the confirmatory study.

The original seismic qualification of the selected equipment was then reassessed for the 1986 Ohio earthquake using the new floor response spectra generated for all seismic Category I structures except the off-gas building, which does not house safety-related equipment (see Section 3.7). The reassessment was divided into two categories based on the method used to originally qualify the equipment, namely, by testing or by an analysis. For equipment qualified by testing, original test response spectra (TRS) were compared against the new required response spectra (RRS), which either are the newly calculated floor response spectra or, as in the case of devices in instrument racks, are in-rack response spectra derived from the floor response spectra using in situ measured transfer functions. The margin is defined as the smallest value of the ratios between the corresponding spectral values of TRS and RRS. For two cases where TRS are exceeded at some isolated frequencies, the margins are taken as the ratios of TRS to RRS at the resonance frequencies of the equipment items of concern. In all cases, the margins are found to be larger than one.

For equipment qualified by analysis, the margin is defined as the product of the spectral ratio and the stress ratio. The spectral ratio is the ratio of the original safe shutdown earthquake (SSE) floor spectral value to the newly calculated spectral value for the same damping value and at the natural frequency of the equipment. The stress ratio, on the other hand, is the ratio of the allowable stress to the originally calculated SSE stress at the most limiting location of the component. Again, in all cases, the margins (products of the ratios) are found to be larger than one.

The staff has reviewed the results of the above quantitative margin study and concurs with the licensee's conclusion that the study provides additional confirmation that the short-duration, high-frequency, low-velocity, small-displacement earthquake had no effect on the Perry plant equipment seismic qualification.

In regard to the generic evaluation of the energy content of high-frequency, short-duration earthquakes the licensee first used the computer program ADINA to perform ductility demand calculations for the earthquakes. Parametric studies and the calculation of energy dissipation for the earthquakes were then performed using a simplified program that incorporates those basic equations of the ADINA program. Since the elastic spectra of the recorded motion exceed the design spectra at around 20 Hz, an elastoplastic analysis of a single-degree-of-freedom oscillator of 20-Hz frequency was performed to calculate the relative ductility ratios required of the oscillator under the motions of the design SSE as well as the short-duration, high-frequency earthquake. The study indicated that the design SSE invariably required higher ductility ratios for the 20-Hz oscillator, under the assumption of various preload conditions. Also indicated in the study was that the energy dissipated within the same oscillator was higher for the SSE than for the 1986 Ohio earthquake, which demonstrated that the latter had much lower energy content than the SSE. This

same conclusion has been independently obtained by the staff's consultant, as stated in Section 3.7.

The effect of a hypothetical higher amplitude and longer duration version of the Ohio earthquake was also studied for ductility demand as well as for the corresponding modification of elastic response spectra. The results of the study indicated that the ductility demand would be increased by less than 10% if both amplitude and duration are doubled to what were originally recorded. It also indicated that if the S-wave duration of the recorded motion was extended to three times as long while peak amplitude was doubled, the elastic spectra would not change significantly at around the 20-Hz region, as was expected. The spectral value may increase in the lower frequency region, but its effects would be well enveloped by the design spectrum. Therefore, the design SSE remains a more demanding earthquake in relation to energy and ductility demand. Also, the potentially higher amplitude and longer duration earthquake would not in any way alter the conclusion of the adequacy of the Perry seismic qualification program, in particular, and the overall seismic design, in general.

Conclusions

The staff has reviewed the licensee's evaluation discussed above and concurs with its finding that the Perry plant's seismic design has adequate safety margins to accommodate the recorded 1986 Ohio earthquake even though the design SSE response spectra were exceeded at around 20 Hz. The staff also concurs with the licensee's finding that if a similar earthquake of somewhat higher amplitude and longer duration should occur near the Perry site, the current equipment seismic qualification program would be adequate to ensure the equipment would not be damaged. This concludes the staff's evaluation of the licensee's confirmatory actions on equipment seismic qualification.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.7 TMI-2 Requirements

In its Partial Initial Decision (September 3, 1985) with respect to the Perry hydrogen control design, the Atomic Safety and Licensing Board (ASLB) emphasized that the staff ensure that the licensee perform a further confirmatory analysis of the hydrogen ignition system (HIS) equipment in containment that has not been qualified for pressure survivability or that has a narrow margin of pressure survivability, and that confirmation of equipment survivability be ensured as a condition for authorizing plant operation above 5% of rated thermal power. This matter was specified in the Perry Unit 1 Low-Power Operating License Condition 2.C(10)(a).

In a letter dated July 17, 1986 (M. R. Edelman to W. R. Butler), the licensee referenced earlier preliminary evaluations performed to determine the pressure survivability of the HIS equipment that were submitted by letter dated March 21, 1985 (M. R. Edelman to B. J. Youngblood), and also provided the results of further confirmatory analyses in response to License Condition 2.C(10)(a). The following HIS equipment, which earlier studies indicated had a narrow margin of pressure survivability, was evaluated by the licensee:

- (1) containment vacuum relief check valves
- (2) hydrogen mixing compressors
- (3) hydrogen mixing compressor discharge valves
- (4) drywell vacuum relief check valves

The preliminary stress analysis performed by the licensee in early 1985 for the containment vacuum relief check valves considered an internal pressure loading of 17.25 psig. The more recent confirmatory analysis assumed a pressure loading of 30 psig. The stresses calculated at this higher pressure are well below the maximum allowable stress requirements, and the licensee has concluded from this that these check valves are capable of withstanding external pressures up to 30 psig without being damaged, or failing to operate as intended (the peak pressure expected during a hydrogen burn is 21.2 psig).

The hydrogen mixing compressors and motors were also reevaluated to determine their survivability at a pressure loading of 30 psig. For the compressors, this higher pressure loading was found to alter the inlet pressure ratio but does not affect compressor operability as designed. The motors were analyzed to determine the potential for bearing failure and insulation breakdown at 30 psig. Since the motor casing is designed to be open, the change in pressure would not increase the loads on the bearings. The motor and its insulation are qualified to accommodate pressure loadings up to 80 psig.

The hydrogen mixing compressor discharge check valves are identical (manufacturer, model number, and size) to check valves used in the fire protection

system at Perry. The fire protection check valves were designed for an internal pressure loading of 300 psig, which is 10 times greater than the pressure loading that plant components are expected to experience from a hydrogen burn.

The drywell vacuum relief valves were initially analyzed by the licensee for an internal pressure of 35 psig. The calculated stresses at that pressure loading were also found to be well below maximum allowable stresses.

The staff found acceptable these initial analytical results provided by the licensee in March 1985, and its evaluation report is documented in Section 6.2.7 of SSER No. 6. From its review of the confirmatory analyses provided by the licensee in July 1986, the results of which are summarized above, the staff concludes that the licensee has sufficiently demonstrated the design capability of the HIS equipment to operate as intended during a hydrogen burn, and that there is a reasonable assurance that the HIS equipment essential for hydrogen control is capable of surviving pressure loadings generated during a hydrogen burn in excess of design requirements.

Accordingly, the staff considers that the licensee's confirmatory analysis discussed above acceptably responds to the Perry Unit 1 Low-Power License Condition 2.C(10)(a) prerequisite for full-power authorization.

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

By letter dated July 18, 1986 (M. R. Edelman to H. R. Denton), the licensee requested the following changes to the Perry Unit 1 Technical Specifications:

- (1) Change the definition of core alteration.
- (2) Change the alarm setpoint of the automatic depressurization system (ADS) instrument air system.
- (3) Permit operation in the maximum extended operating domain (MEOD).
- (4) Eliminate the average power range monitor (APRM) setdown requirement.

These proposed changes involve, among other factors, the development of new power- and flow-dependent relations for the maximum average planar linear heat-generation rate (MAPLHGR) and the minimum critical power ratio (MCPR). A General Electric Company (GE) (June 1985) analysis of the consequences of operation in the MEOD was referenced in the submittal to justify the proposed changes.

The MEOD includes expansion of the normal power/flow map into two new regions. One region, which involves operation at rated power at lower-than-rated core flow rates, is called the extended load line region (ELLR). The other region, which involves operation at core flows up to 105% of rated flow, is called the increased core flow region (ICFR). Operation in the ELLR and ICFR permits greater operational flexibility and an improved unit capacity factor.

The APRM setdown requirement in the current Technical Specifications requires that the flow-biased APRM trips be reduced (set down) when the core maximum total peaking factor exceeds the design total peaking factor. This requirement was associated with a now obsolete Hench-Levy minimum critical heat flux ratio criterion. With the elimination of APRM setdown, a revision of the MCPR limit and development of new flow- and power-dependent MAPLHGR limits is provided to give fuel protection for power peaking effects. The elimination of the APRM setdown does not affect the results of the loss-of-coolant-accident (LOCA) calculations.

Change of Definition for Core Alteration

The Technical Specification definition of core alteration currently does not consider normal movement of the source range monitors (SRMs), intermediate range monitors (IRMs), traversing incore probes (TIPs), or special movable detectors to be a core alteration. This change request would provide the same consideration for linear power range monitors (LPRMs).

The Technical Specification definition of core alteration provides a specific exception for the movement of incore instrumentation. Because the LPRM strings are only removed from the core when they are to be replaced and they have no normal drive mechanism, a similar exception from the definition of core alteration is requested. Currently, Technical Specification 3.3.1, "Reactor Protection System Instrumentation," contains a footnote that does not consider replacement of an LPRM string a core alteration. This change request would clarify the present definition and modify it to include an exception currently contained in the Technical Specifications.

The definition may be changed as follows:

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

Alarm Setpoint Change for the ADS Instrument Air System

The proposed Technical Specification change to Surveillance Requirement 4.5.1.e.2.c will change the alarm setpoint for the ADS from the current value of 2475 ± 25 psig to greater than or equal to 155 psig. This change is needed to accommodate a new design for the safety-related instrument air system. The new design of the air system will reduce the air compressor pressure from 2500 psig to 160 psig. The air pressure supplied to the ADS accumulators, however, will remain the same, that is, 150 psig.

The alarm setpoint proposed for the system of greater than or equal to 155 psig on decreasing pressure is higher than the required ADS accumulator pressure (150 psig) and is, therefore, acceptable.

Evaluation of Operation in MEOD

The staff has reviewed and evaluated the MEOD operation for Perry (G. C. Lainas, December 17, 1985). The MEOD operation was approved only for startup testing because the loss-of-feedwater-heating (LFWH) transient analysis for the MEOD was not satisfactory. For the 100°F LFWH transient analysis for the MEOD, the licensee had used a generic statistical LFWH analysis described in a letter to the staff from J. S. Charnley (GE) dated July 5, 1983. The staff has not approved this generic statistical analysis. Therefore, the MEOD was not approved except for startup testing until the analysis is approved or the LFWH is reanalyzed using an approved methodology.

In FSAR Amendment 25 (July 27, 1986), the licensee submitted a plant-specific LFWH analysis for the MEOD. This analysis indicated that the LFWH transient is less severe than the feedwater controller failure event. The results show that the LFWH event is not the limiting transient in the MEOD. Because the licensee submitted the plant-specific LFWH analysis and the staff finds it acceptable, MEOD operation is acceptable for Perry.

Elimination of APRM Setdown

In the current Technical Specifications, the flow-biased APRM trips are reduced (set down) when the core maximum total peaking factor exceeds the design total peaking factor. The GE analysis (NEDM-30963) performed for Perry includes results from analyses made to determine the new initial conditions of fuel thermal limits that would be needed to satisfy the pertinent licensing criteria if APRM setdown were eliminated. The new limits should (1) prevent violation of the MCPR safety limit, (2) keep the fuel thermal-mechanical performance within the design and licensing basis, and (3) keep peak cladding temperature and maximum cladding oxidation within allowable limits. It was concluded that current MAPLHGR limits protect against a LOCA even without APRM setdown. The flow-dependent MCPR limit is also not affected by elimination of APRM setdown because the design-basis flow runout event is a slow flow/power increase not terminated by scram. The results of the analysis with approved methods are:

- (1) New power-dependent relations for MCPR and MAPLHGR limits are provided, which include both high- and low-flow relations at powers below 40% where reactor scram on turbine control valve fast closure is bypassed. The MAPLHGR relation is a factor, $MAPFAC_p$, which is multiplied by the normal full-power MAPLHGR limit to obtain the power-dependent MAPLHGR limit.
- (2) A new flow-dependent MAPLHGR factor, $MAPFAC_f$, is provided. This factor was determined from analysis of slow flow runout transients with the requirement that peak transient MAPLHGR values not exceed the fuel design-basis values.

The staff finds the elimination of APRM setdown acceptable for Perry.

6.4 Control Room Habitability

In Section 6.4 of SSER No. 8, Confirmatory Issue (65) was added to the SER. This issue concerned the need for the staff to complete its reevaluation of the control room habitability necessitated by an increase in the secondary containment bypass leakage from 4.0% to 5.04%. Specifically reviewed by the staff were the control room operation doses associated with this increased bypass leakage under design-basis accidents (DBAs) such as a loss-of-coolant accident (LOCA) and an accident involving a main steamline break outside of containment.

The staff has reevaluated the control room habitability of Perry Units 1 and 2 on the basis of its independent estimate of the secondary containment bypass leakage increase from 4.0% to 5.04%. The staff's evaluation included a review of operator doses under DBA conditions such as a LOCA and a main steamline break accident, with credit given for iodine removal by the containment spray system during a LOCA, since the system is qualified as a safety-related system. The licensee estimated the time-dependent buildup of airborne radioactivity within the control room using the methodology in Gilbert/Commonwealth Associates Topical Report GAI-TR-101-NP-A, previously approved by the NRC staff.

The Perry control room has a volume of 34,400 ft³, which is shared by the control areas for Units 1 and 2. If a radiological accident should occur, a LOCA signal or a high radiation signal in the control room air intake would isolate the control room in about 10 sec by closing the ventilation system dampers. The isolated control room would be placed on 100% recirculation, with 30,000 ft³ per minute of air recirculating through charcoal and high efficiency particulate air (HEPA) filters. The charcoal filters are credited with a 95% iodine removal efficiency. The HEPA filters are credited with a 99% particulate removal efficiency. The unfiltered inleakage is limited to 90 ft³ per minute, as verified by testing. The staff's LOCA analysis indicates that the control room doses are within the guidelines of General Design Criterion (GDC) 19 of Appendix A to 10 CFR 50 and of Section 6.4 of the Standard Review Plan (SRP, NUREG-0800).

For the main steamline break accident (MSLBA) condition, there is no fuel damage postulated. Hence, the amount of radioactivity released to the environment would be much less than that from a LOCA.

In view of the smaller amount of radioactivity released and the larger distance between the point of release (i.e., between the turbine building and the control room), the staff finds that the LOCA is a bounding design-basis accident (i.e., the MSLBA would result in lower operator doses). On the basis of its independent assessment, the staff estimates that control room doses under LOCA and MSLBA conditions are within the SRP Section 6.4 guidelines. Hence, the staff concludes that the increase of secondary containment bypass leakage from 4.0% to 5.04% is acceptable with respect to control room habitability and in accordance with GDC 19. SER Confirmatory Issue (65) is accordingly considered to be completed.

7 INSTRUMENTATION AND CONTROL

7.2 Reactor Protection System

7.2.2 Specific Findings

7.2.2.3 Anticipated Transients Without Scram (ATWS)

In Section 7.2.2.3 of SSER No. 7, the staff reported that the applicant (now the licensee) had provided information on actions taken at Perry relative to the action items required by the staff in NRC Generic Letter 83-28 (July 8, 1983) entitled "Required Actions Based on Implications of the Salem ATWS Events." A commitment to comply with those required actions, discussed in the following paragraphs, was documented by the licensee in Appendix 1B of FSAR Amendment 24.

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant (SNPP) failed to open on an automatic reactor trip signal from the reactor protection system. This incident occurred during the plant startup, and the reactor was tripped manually by the operator about 30 sec after the initiation of the automatic trip signal. It has been determined that the failure of the circuit breakers was related to the sticking of the undervoltage trip attachment. On February 22, 1983, during startup of SNPP Unit 1, an automatic trip signal occurred as the result of steam generator low-low level. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip. Following these incidents, on February 28, 1983, the NRC Executive Director for Operations directed the staff to investigate and report on the generic implications of these occurrences. The results of the staff's inquiry into these incidents are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System Reliability Improvements, each consisting of subitems (i.e., 1.1, 1.2, 2.1, etc.).

This SER supplement documents the staff's review of the licensee's response pertaining to Action Items 1.2, 3.1.1, 3.1.2, 3.2.1, 3.2.2, and 4.5.1 of Generic Letter 83-28. The staff's review findings pertaining to the remaining action items of Generic Letter 83-28, as they relate to boiling-water-reactor (BWR) plants and Perry specifically, will be documented in a future supplement.

Post-Trip Review, Action Item 1.2

The following review guidelines were developed after initial evaluation of the various utility responses to Item 1.2 of Generic Letter 83-28 and incorporate the best features of these submittals. As such, these review guidelines in

effect represent a "good practices" approach to post-trip review. The staff has reviewed the licensee's response to Item 1.2 against these guidelines:

- (1) The equipment that provides the digital sequence-of-events (SOE) record and the analog time history records of an unscheduled shutdown should provide a reliable source of the necessary information to be used in the post-trip review. Each plant variable that is necessary to determine the cause and progression of the events following a plant trip should be monitored by at least one recorder (such as an SOE recorder or a plant process computer) for digital parameters, and strip charts, a plant process computer, or analog recorder for analog (time history) variables. Performance characteristic guidelines for SOE and time history recorders are as follows:
 - (a) Each SOE recorder should be capable of detecting and recording the sequence of events with a sufficient time discrimination capability to ensure that the time responses associated with each monitored safety-related system can be ascertained, and that a determination can be made as to whether the time response is within acceptable limits based on FSAR Chapter 15 accident analyses. The recommended guideline for the SOE time discrimination is approximately 100 msec. If current SOE recorders do not have this time discrimination capability, the licensee should show that the current time discrimination capability is sufficient for an adequate reconstruction of the course of the reactor trip and post-trip events. As a minimum, this should include the ability to adequately reconstruct the transient and accident scenarios presented in FSAR Chapter 15.
 - (b) Each analog time history data recorder should have a sample interval small enough so that the incident can be accurately reconstructed following a reactor trip. As a minimum, the licensee should be able to reconstruct the course of the transient and accident sequences evaluated in the plant's FSAR Chapter 15 accident analyses. The recommended guideline for the sample interval is 10 sec. If the time history equipment does not meet this guideline, the licensee should show that the time history capability is sufficient to accurately reconstruct the transient and accident sequences presented in FSAR Chapter 15. To support the post-trip analysis of the cause of the trip and the proper functioning of involved safety-related equipment, each analog time history data recorder should be capable of updating and retaining information from approximately 5 min before the trip until at least 10 min after the trip.
 - (c) All equipment used to record SOE and time history information should be powered from a reliable and noninterruptible power source. The power source used need not be Class 1E.
- (2) The SOE and time history recording equipment should monitor sufficient digital and analog parameters, respectively, to ensure that the course of the reactor trip and post-trip events can be reconstructed. The parameters monitored should provide sufficient information to determine the root cause of the unscheduled shutdown, the progression of the reactor trip, and the response of the plant parameters and protection and safety systems to the unscheduled shutdowns. Specifically, all input parameters

associated with reactor trips, safety injections, and other safety-related systems as well as output parameters sufficient to record the proper functioning of these systems should be recorded for use in the post-trip review. The parameters deemed necessary, as a minimum, to perform a post-trip review that would determine if the plant remained within its safety limit design envelope are presented in Table 7.1 (added to the SER by this supplement). They were selected on the basis of staff engineering judgment, following a complete evaluation of utility submittals. If the licensee's SOE recorders and time history recorders do not monitor all of the parameters suggested in this table, the licensee should show that the existing set of monitored parameters is sufficient to establish that the plant would remain within the design envelope for the accident conditions analyzed in FSAR Chapter 15.

- (3) The information gathered by the SOE and time history recorders should be stored in a manner that will allow for data retrieval and analysis. The data may be retained in either hardcopy (e.g., computer printout, strip chart record) or in an accessible memory (e.g., magnetic disc or tape). This information should be presented in a readable and meaningful format, taking into consideration good human factors practices such as those outlined in NUREG-0700.
- (4) Retention of data from all unscheduled shutdowns provides a valuable reference source for the determination of the acceptability of the plant vital parameter and equipment response to subsequent unscheduled shutdowns. Information gathered during the post-trip review is to be retained for the life of the plant for post-trip review comparisons of subsequent events.

By letters dated November 2, 1983 (D. R. Davidson to A. Schwencer), April 6, 1984 (M. R. Edelman to B. J. Youngblood), and November 15, 1985 (M. R. Edelman to W. R. Butler), the licensee provided information regarding its post-trip review program data and information capabilities for Perry. The staff has evaluated the licensee's submittals against the review guidelines described above. A brief description of the licensee's responses and the staff's evaluation of the response against each of the review guidelines follows:

- (1) The licensee has described the performance characteristics of the equipment used to record the SOE and time history data needed for post-trip review. On the basis of its review of the licensee's submittals, the staff finds that the SOE recorder and time history characteristics conform to Review Guideline (1) above and, therefore, are acceptable.
- (2) The licensee has established and identified the parameters to be monitored and recorded for post-trip review. On the basis of its review, the staff finds that the parameters selected by the licensee include all of those identified in Table 7.1 and conform to Review Guideline (2) above and, therefore, are acceptable.
- (3) The licensee described the means for storage and retrieval of the information gathered by the SOE and time history recorders and for the presentation of this information for post-trip review and analysis. On the basis of its review, the staff finds that this information will be presented in a readable and meaningful format and that the storage, retrieval, and presentation conform to Review Guideline (3) above and, therefore, is acceptable.

- (4) The licensee's submittals have indicated that the data and information used during post-trip reviews will be retained in an accessible manner for the life of the plant. On the basis of this information, the staff finds that the licensee's program for data retention conforms to Review Guideline (4) above and, therefore, is acceptable.

On the basis of the above findings, the staff concludes that the licensee's post-trip review data and information capabilities for Perry satisfactorily comply with Item 1.2 of Generic Letter 83-28.

Post-Maintenance Testing (RTS Components), Action Items 3.1.1 and 3.1.2

The review criteria for these items require that the licensee submit a statement indicating that the licensee has reviewed plant test and maintenance procedures and Technical Specifications to ensure that post-maintenance operability testing of safety-related components in the reactor trip system is required. Also, the licensee's statement should contain a verification that vendor-recommended test guidance has been reviewed, evaluated, and where appropriate, included in the test and maintenance procedures or the Technical Specifications. The staff has evaluated the licensee's April 6, 1984, submittal for this item and has determined it to be adequate in content.

Post-Maintenance Testing (All Other Safety-Related Components), Action Items 3.2.1 and 3.2.2

The review criterion for this item requires that the licensee submit a statement indicating that the licensee has reviewed plant test and maintenance procedures and Technical Specifications to ensure that post-maintenance operability testing of all safety-related components is required. Also, a statement is required that contains a verification that vendor-recommended test guidance be reviewed, evaluated, and where appropriate, included in the test and maintenance procedures or the Technical Specifications. The staff has evaluated the licensee's April 6, 1984, submittal for this item and has determined it to be adequate in content.

Reactor Trip System Reliability Improvements (System Functional Test Description), Action Item 4.5.1

The review criterion for this item requires that the licensee submit a statement committing to independent, on-line functional testing of the diverse trip features. The staff has evaluated the licensee's April 6, 1984, submittal for this item, committing to on-line testing of the scram pilot valve solenoid and initiating circuitry, and has determined it to be adequate in content.

On the basis of the above review findings, the staff concludes that the proposed programs outlined in the licensee's submittals adequately address the requirements of Generic Letter 83-28 specified in Action Items 3.1.1, 3.1.2, 3.2.1, 3.2.2, and 4.5.1.

Table 7.1 - BWR parameter list

Sequence-of-events recorder	Time history recorder	Parameter/signal
x		Reactor trip
x		Safety injection
x		Containment isolation
x		Turbine trip
x		Control rod position
x (1)	x	Neutron flux, power
x (1)		Main steam radiation
(2)		Containment (drywell) radiation
x (1)	x	Drywell pressure (containment pressure)
(2)		Suppression pool temperature
x (1)	x	Primary system pressure
x (1)	x	Primary system level
x		Main steam isolation valve position
x (1)		Turbine stop valve/control valve position
x		Turbine bypass valve position
	x	Feedwater flow
	x	Steam flow
(3)		Recirculation: flow, pump status
x (1)		Scram discharge level
x (1)		Condenser vacuum
(3)		Auxiliary feedwater system: flow, pump/valve status
x		AC and DC system status (bus voltage)
x		Diesel generator status (start/stop, on/off)
x		Power-operated relief valve position

(1) Trip parameters

(2) Parameter may be monitored by either an SOE or time history recorder.

(3) Acceptable recorder options are (a) system flow recorded on an SOE recorder, (b) system flow recorded on a time history recorder, or (c) equipment status recorded on an SOE recorder.

9 AUXILIARY SYSTEMS

9.6 Other Auxiliary Systems

9.6.3 Emergency Diesel Engine Fuel Storage and Transfer System

9.6.3.3 Conclusion

In Section 9.6.3.3 of SSER No. 8, the NRC staff reported that it and its consultant (Pacific Northwest Laboratory or PNL) concluded that the generic Phase 1 review, on which the Perry Transamerica Delaval, Inc. (TDI), diesel generator design review/quality revalidation (DR/QR) is based, had progressed to the point where all significant diesel generator reliability issues warranting priority attention for issuance of a plant operating license were adequately resolved, and that the Perry TDI diesel generators would provide a reliable source of onsite power in accordance with GDC 17, subject to the license condition noted in SSER No. 8 and cited in Attachment 3 to the Perry Unit 1 low-power operating license (NPF-45).

Since the publication of SSER No. 8 (January 1986), the staff has completed its evaluation of the TDI Diesel Generator Owners Group DR/QR program for validating and upgrading the design and manufacturing quality of the TDI diesel generators for nuclear emergency standby service. The staff's SER pertaining to that program, which is to be published as a separate NUREG report, was transmitted to the licensee by letter dated July 8, 1986 (W. R. Butler to M. R. Edelman).

In summary, the staff reaffirmed, from its review of the TDI Owners Group DR/QR program, that implementation by the licensee of the TDI Owners Group and PNL recommendations concerning quality revalidation inspections, component modifications and replacement, load restrictions, operating precautions, etc., will maintain the adequacy of the TDI diesel generator for nuclear service as required by GDC 17. The staff concluded that the design programs (actions) proposed will ensure that the design and manufacturing quality of the TDI diesel engines is within the range normally assumed for diesel engines designed and manufactured in accordance with 10 CFR 50, Appendix B.

The staff also concluded that several of the Phase 1 diesel engine components merited special emphasis in the areas of load restrictions and/or maintenance and surveillance. These areas were discussed in SSER No. 8, addressed in the Perry Unit 1 Technical Specifications, and cited as a license condition in the Perry Unit 1 low-power operating license (NPF-45).

As a result of the staff's evaluation of the TDI Owners Group DR/QR program, the license condition cited in Attachment 3 of License NPF-45 will be amended in the license condition cited in the Perry Unit 1 full-power operating license as follows (reference is made to the license condition items in License NPF-45 affected; also see Section 1.11 of this supplemental report):

- Item 2 - Change the first sentence to read: "The oil holes and fillets of the three main bearing journals, subject to the highest torsional stresses (Nos. 4, 6, and 8), shall be examined with fluorescent liquid penetrant and, as necessary, eddy current, during the one-time 5-year and each 10-year major disassembly." (The change is underlined.)
- Item 6 - Delete in its entirety.

11 RADIOACTIVE WASTE MANAGEMENT

11.5 Effluent Monitoring

By letter dated August 15, 1985, the licensee submitted Revision 1 to the Off-site Dose Calculation Manual (ODCM) for Perry Unit 1. The staff has reviewed the ODCM, Revision 1, and finds that the ODCM uses documented and approved methods that are consistent with the methodology and guidelines delineated in NUREG-0133 and is therefore acceptable as a reference. The staff also finds that this revised ODCM is in compliance with the Perry Unit 1 Technical Specifications, Section 6.14.1, which states: "The ODCM shall be approved by the Commission prior to implementation." The staff had previously found the ODCM, Revision 1, acceptable during its review of the Technical Specifications for Perry Unit 1. However, the staff's acceptance had not been documented in previous SER supplements. This supplement (SSER 10) serves to provide such documentation for the record.

13 CONDUCT OF OPERATIONS

13.3 Emergency Plans

13.3.5 Atomic Safety and Licensing Board Hearings on Emergency Preparedness

In SSER No. 7, the staff reported on the Atomic Safety and Licensing Board (ASLB) hearing on emergency preparedness issues. The staff noted that final resolution of Conditions 2 and 3 of the September 3, 1985, ASLB Decision, as confirmed by the Federal Emergency Management Agency (FEMA), would be provided in a future supplement to the SER. To ensure satisfactory resolution of those ASLB-identified emergency preparedness issues, License Condition (31) was added to the SER by SSER No. 7 as follows:

- (a) Before exceeding 5% of rated thermal power, CEI shall submit letters of agreement obtained from all school districts for the supply of buses for evacuation purposes.
- (b) Before exceeding 5% of rated thermal power, CEI shall complete training of fire personnel in monitoring and decontamination procedures and ensure that all reception centers are provided with the necessary decontamination equipment.

FEMA's Supplemental Findings on ASLB Conditions

On February 4, 1986, FEMA provided its supplemental findings on the offsite emergency preparedness conditions specified in the ASLB Decision of September 3, 1985, and documented in SSER No. 7 as License Condition (31). FEMA concluded that the emergency preparedness issues as specified in the ASLB Decision have been satisfactorily resolved. The supplemental findings by FEMA are provided in Appendix S, which is being added to the SER by this supplement. Accordingly, the staff finds that the licensee has complied with License Condition (31), which is being deleted by this supplement. However, that portion of the license condition pertaining to FEMA's approval of offsite plans under 44 CFR 350 remains in effect and will be cited in the Perry Unit 1 operating license permitting operation at power levels up to 100% of rated thermal power.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.2 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

13.5.2.2.1 Plant-Specific Technical Guidelines

As indicated in Section 13.5.2.2.1 of SSER No. 8, the Perry Plant Emergency Instructions are based on the BWR Owners Group (BWROG) Emergency Procedure Guidelines (EPGs) with the exception of plant-unique deviations. The BWROG

EPGs call for emergency containment venting as the last step in a sequence of procedural steps involving operator actions designed to reduce containment pressure. In FSAR Appendix 1B, the licensee committed to provide a plant-unique analysis and resulting venting pressure value for the Perry facility before operating above 5% power. Also, as discussed in SSER No. 8, the licensee was advised to provide additional information on four items relating to the emergency venting procedure. These items are summarized as follows:

- (1) an assessment of the design capability of the suppression pool, which, in response to a rapid depressurization from venting, may result in suppression pool flashing and hydrodynamic loads
- (2) a prioritization of the selected emergency vent paths to minimize radioactive release rates
- (3) consideration of flow paths other than the dedicated purge and exhaust lines
- (4) the consequences of containment venting on ductwork failure (if used as a pathway) and subjecting equipment near the failed duct to the steam/radiation environment

By letter dated July 29, 1986 (M. R. Edelman to W. R. Butler), the licensee submitted its plant-specific analysis dealing with emergency containment venting. The staff's review of the licensee's submittal is provided below.

The Mark III primary containment at Perry is a steel vessel surrounded by a concrete secondary building. The containment has a 15-psig design pressure. Its ultimate pressure capability is limited by a penetration to about 56 psig.

The staff's SER (Crutchfield, November 23, 1983) on the BWR0G EPGs, Revision 3, established an interim limit of twice the design pressure for emergency venting with the understanding that a more precise analysis may be used to establish a venting pressure limit. The licensee has determined that the venting pressure (50 psig) is limited by safety/relief valve operability. The value is approximately three times the design pressure.

The primary function of the containment venting procedure is to provide a predetermined capability for venting the containment atmosphere at a rate that will prevent damaging the containment structure because of overpressurization. Containment pressure is directly dependent on the energy added to the containment atmosphere because of decay heat generation following reactor shutdown, and serves as the basis for determining the required relief capability of the vent paths. The Perry venting procedure incorporates a sequential use of preferred vent paths by consideration of vent path size and minimizing offsite dose rates.

As a result of the licensee's effort to assess potential vent paths, three systems were identified. The first system is the fuel pool cooling and cleaning (FPCC) system. This path utilizes the skimmers in the upper pool region, which is located on top of the drywell structure. The smallest restriction in this path has a 9.5-in. diameter. The path discharges through the FPCC surge tank vent and overflow lines, and the spent fuel pool skimmers to the fuel handling building atmosphere. The path is established by remotely opening the inboard

and outboard FPCC containment isolation valves. Therefore, the licensee estimates a quick response time to achieve venting through the FPCC system (i.e., less than 5 min).

The other two paths identified by the licensee are the containment spray systems, which consist of two redundant systems/paths (i.e., trains A and B). The venting is through the containment spray nozzles, and in this vent path a 5.75-in. orifice is the limiting restriction. These paths discharge through the residual heat removal (RHR) and FPCC supplemental cooling connection and out to the fuel handling building atmosphere from the spent fuel pool. The valve lineup using these systems will involve manual action to open the selected isolation valves. Thus, the licensee estimates a response time to achieve venting through any one of the spray lines to be about 30 min.

The licensee has stated that the actuators of the inboard containment isolation valves for the three systems are qualified to 105 psig at 340°F and were exposed to temperatures in excess of 300°F for over 9 hours and temperatures in excess of 250°F for over 4 days during qualification testing. Environmental qualification is not a limiting condition to ensure operation of the outboard motor-operated valves. Also, the three systems are designed as water systems and will withstand a pressure in excess of the 50-psig venting limit. The FPCC system is used first for venting to maintain the RHR system available for containment spray and/or core cooling as long as possible. Then either containment spray path may be used if necessary. Using the FPCC system or the spray system takes advantage of the decontamination action of the water volumes that are part of the vent path, for example, the FPCC surge tank and the spent fuel pool and also the fuel handling building ventilation system filters. Should the ventilation system fans fail, pressurization of the fuel handling building will force flow through the ventilation system filters and other leakage paths.

Also, the licensee has concluded that the hydrodynamic loads from flashing of the suppression pool water as a result of containment venting are insignificant and will not exceed the design capability of the suppression pool. This conclusion is based on the steaming rates, as determined from the licensee's thermodynamic analysis to be insignificant as compared with the design-basis considerations.

In discussions with the licensee, the staff requested that the licensee consider performing the sequential implementation of the vent paths in a timely manner so that the vent pressure limit would not be exceeded. By letter dated August 20, 1986 (M. R. Edelman to W. R. Butler), the licensee committed to integrate this aspect in the development of the Plant Emergency Instructions. Specifically, when the containment pressure exceeds 15 psig, the operator will be directed to align the FPCC vent path except for the last valve. In addition, the instructions will provide a caution statement to the operator regarding the time required to align the remaining two vent paths, should they be required.

The staff has reviewed the material provided by the licensee as justification for the selection of 50 psig as the primary containment pressure limit for venting of the Perry containment. The staff agrees that the approach taken by the licensee satisfies the requirement in the SER approving the generic

BWROG EPGs, Revision 3 (transmitted to BWROG by letter dated November 23, 1983 (D. Crutchfield to T. Dente)) for a more precise criterion for defining venting pressure and concludes that the information provided is an adequate basis for the selection of 50 psig as an upper limit, and for aligning valves when the containment pressure exceeds the design pressure of 15 psig to allow expeditious vent operation at pressures above 15 and below 50 psig. In addition, in the future the BWROG is expected to submit Revision 4 of the EPGs for staff review. Any long-term followup will be pursued generically in connection with the BWROG. The licensee has committed to incorporate applicable changes resulting from the staff's generic review (letter dated August 20, 1986, referenced previously).

Conclusion

The staff has reviewed the adequacy of the paths selected for venting and the anticipated consequences of such an action and concludes that the challenge of venting at elevated pressures to those affected systems is within its design limits. The staff further concludes that the information is an adequate basis for the selection of 50 psig as the primary containment pressure limit for Perry containment venting and is, therefore, acceptable.

13.5.2.7 Hydrogen Igniter Emergency Procedures

In its Partial Initial Decision (September 3, 1986) relative to the adequacy of the hydrogen control issue litigated at the ASLB hearing in May 1985, the ASLB indicated that before 5% of rated thermal power is exceeded, the staff shall ensure that written procedures are available for operation of the hydrogen ignition system. This item was accordingly specified as License Condition 2.C(10)(b) in the low-power operating license for Perry Unit 1 (License No. NPF-45).

By letter dated August 6, 1986 (M. R. Edelman to W. R. Butler), the licensee provided the requisite information in response to NPF-45 License Condition 2.C(10)(b). Specifically, Perry Emergency Instruction (PEI) M51/56 provides guidance to the plant operators on the use of the Perry hydrogen ignition system. The PEI follows the guidance provided in the Hydrogen Control Owners Group Emergency Procedure Guidelines (HCOG-EPGs) submitted to the staff by letter dated June 3, 1986, from J. R. Langley, and Appendix B to the HCOG-EPGs submitted by letter dated July 3, 1986, from J. R. Langley.

The generic guidelines instruct the operator to initiate purging the containment at hydrogen concentrations at the minimum detectable limit and at 4 volume percent. Although the staff has not completed its review of the HCOG-EPGs, it has concluded that venting the containment at nonthreatening hydrogen concentrations is not appropriate. Accordingly, the licensee has responded to the staff's concerns and has deleted the affected venting provisions in its plant-specific procedures.

With regard to the hydrogen ignition system, on determining that the hydrogen concentration is above 0.5 volume percent or that the reactor pressure vessel water level may be below the top of active fuel, the hydrogen igniters are to be placed into service. Also, the operator is directed to secure the hydrogen igniters if power to the igniters is lost and if the hydrogen concentrations inside containment exceed the various concentrations limits.

In conclusion, suitably written procedures for operation of the hydrogen ignition system are contained in the PEI. Accordingly, the licensee has satisfied the requirements set forth by the license condition. Therefore, the staff finds that the licensee's response satisfies the Perry Unit 1 Low-Power License Condition 2.C(10)(b) prerequisite for full-power authorization.

16 TECHNICAL SPECIFICATIONS

16.1 Introduction

This section documents the staff's approval of the Technical Specification changes proposed by the licensee for the full-power license for Perry Unit 1, submitted by letters dated June 18, July 18, and July 30, 1986 (M. R. Edelman to W. R. Butler).

16.2 Specific Technical Changes for Full-Power Operation

16.2.1 Operation in the Maximum Extended Operating Domain

The licensee proposed the following changes to the Technical Specifications to permit operation of Perry in the maximum extended operating domain (MEOD), and the elimination of the average power range monitor (APRM) setdown:

- (1) Table 2.2.1-1(2): The proposed changes increase the flow-biased APRM setpoint, and provide an allowable value of 16% for two-loop operation, to permit operation in the extended load line region (ELLR) part of the MEOD. The GE analysis, dated June 1985, shows that operation in the MEOD would not exceed design limits. This is acceptable to the staff.
- (2) Specification 3/4.2.1: The proposed change to this specification dealing with maximum average planar linear heat-generation rate (MAPLHGR) limits results from the proposed elimination of APRM setdown in Specification 3/4.2.2. The current specifications provide for a reduction in the flow-biased APRM trips when the core maximum total peaking factor exceeds the design total peaking factor. With the proposed elimination of APRM setdown, this peaking effect is covered by a revision of the MAPLHGR limits. The revised limits are presented as graphs for both a flow-dependent and power-dependent MAPLHGR factor in Figures 3.2.1-4 and 3.2.1-5. The revised limits provide equal or increased margins to fuel integrity limits relative to those obtained with APRM setdown. The staff finds the proposed change acceptable.
- (3) Specification 3/4.2.2: The proposed change is to eliminate this specification, which involves the APRM setdown. As discussed under Specification 3/4.2.1, this proposed change is acceptable to the staff.
- (4) Figures 3.2.2-1 and 3.2.2-2: The slow recirculation flow runout analysis for the proposed operation in the MEOD results in new flow-dependent MCPR limit curves. The new curves, shown in Figure 3.2.2-1, are slightly above the curves in the current Technical Specifications. The new set of power-dependent MCPR limits is shown in Figure 3.2.2-2. The new limits include the effect of operation at feedwater temperature reductions up to 170 Fahrenheit degrees. The operating limit MCPR at any power/flow condition is the larger of the new flow- and power-dependent values. The staff finds the proposed changes acceptable.

- (5) Table 3.3.1-1: The proposed change involves Note h of Table 3.3.1-1, which deals with bypassing the turbine stop valve closure and turbine control valve fast closure scram when thermal power is less than 40% of rated thermal power. The high-pressure turbine first-stage pressure is used to measure thermal power. New setpoints for the first-stage pressure are provided for feedwater temperatures between 420°F and 250°F. The proposed change clarifies the current requirement and is based on the results of the startup tests of similar plants. The staff finds the changes acceptable.
- (6) Table 3.3.4.2-1: This revision to Note b of Table 3.3.4.2-1 is a proposed change that clarifies the current requirement and is based on results of the startup tests of similar plants. This change for the end-of-cycle recirculation pump trip is identical to that for the turbine stop valve and turbine control valve fast closure scram (see Item 5) and is acceptable.
- (7) Table 3.3.6-2: The proposed changes are made to permit operation in the MEOD. Increase in the APRM flow-biased rod block setpoint is proposed to permit operation in the ELLR. However, the high-flow clamp to this setpoint value for rod block is added to maintain the clamp setpoint of 108%. These changes provide the same margin between the simulated thermal power monitor scram and rod block setpoints as the current Technical Specifications. In addition, the recirculation flow-high rod block setpoint is increased from 108% to 111% to decrease unnecessary rod block alarms when operating in the increased core flow region. The staff finds these changes acceptable.
- (8) Administrative changes made to the Technical Specifications include elimination of references to the deleted Specification 3.2.2 in Bases 2.2.1 and 3/4.2.2, Specifications 3/4.2.2 and 3/4.2.3, the index, and Figures 3.2.3-1 and 3.2.3-2.
- (9) Bases 2.2.1, 3/4.2.1, 3/4.2.2, and 3/4.2.3, Bases Table B 3.2.1-1, and Bases Figure B 3/4.2.3-1: Proposed changes to the Bases were those modifications and additions provided to reflect the changes to the Technical Specifications needed for the proposed operation in the MEOD. The staff finds these changes acceptable.

On the basis of the considerations discussed above, the staff has concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and, therefore, the Technical Specification changes proposed by the licensee to permit operation in the MEOD are acceptable.

16.2.2 Definition 1.7, Core Alteration

The Technical Specification definition of CORE ALTERATION currently does not consider normal movement of the source range monitors (SRMs), intermediate range monitors (IRMs), traversing incore probes (TIPs), or special movable detectors to be a core alteration. The licensee proposed a change that would provide the same consideration for linear power range monitors (LPRMs).

The Technical Specification definition of core alteration provides a specific exception for the movement of incore instrumentation. Because the LPRM strings are only removed from the core when they are being replaced and they have no normal drive mechanism, a similar exception from the definition of core alteration is requested. Currently, Specification 3.3.1, "Reactor Protection System Instrumentation," contains a footnote that does not consider replacement of an LPRM string a core alteration. This change request would clarify the present definition and modify it to include an exception currently in the Technical Specifications. The staff finds this proposed change acceptable.

16.2.3 Table 3.3.7.1-1, Radiation Monitoring Instrumentation

Specification 3.3.7.1-1, Action 72, currently requires a portable continuous noble gas monitor to be operable in the control room when the control room ventilation radiation monitor (noble gas) is inoperable. The licensee has proposed modifying the Technical Specification to allow the control room area radiation monitor (CRARM) to meet the action requirement because the licensee will calculate the CRARM alarm setpoint using the same criteria as it uses in determining the control room ventilation radiation monitor setpoint.

The requirement in GDC 19 will be satisfied. Therefore, the staff finds the licensee's request acceptable.

16.2.4 Table 3.3.7.5-1, Accident Monitoring Instrumentation and Basis 3.3.7.5

Action 82 is applicable to the accident monitoring instrumentation channel for the primary containment isolation valve position. Action 82, as currently worded, is excessively punitive in that it would require shutdown of the plant if primary containment isolation valve position indication is inoperable. It allows the use of alternate means of verifying valve position, but the time associated with the action is not consistent with the testing requirements of the valves in accordance with Specification 4.0.5. The licensee proposed a change that would allow the plant to remain at power provided the valve position could be verified by use of alternate indication and that the instrumentation is returned to operable status the next time the valve is required to be demonstrated operable pursuant to Specification 4.0.5. The licensee further proposed a change to the bases of Specification 3.3.7.5 to clarify what a channel check and a channel calibration for primary containment isolation valve position indication entail. The primary safety concern of containment isolation is the operability of the valves to isolate the containment.

Each containment isolation valve has the position indications in the main control room; in addition, a valve's position may be verified by alternate methods, such as flow rate, temperature, and/or pressure.

The surveillance frequency of containment isolation valves is set in accordance with ASME requirements. This frequency should be the basis on which instrumentation operability is required. Requiring a valve to change position in order to perform this position verification is unwarranted and may cause the unit to be shut down.

This change, that the valve position could be verified by use of alternate indication and that the instrumentation is returned to operable status the next time

the valve is required to be demonstrated operable pursuant to Specification 4.0.5, provides operational flexibility and adequate assurance of containment isolation valve position and is, therefore, acceptable to the staff.

The change to the bases to clarify what a channel check and channel calibration entail ensures consistent surveillance monitoring and is consistent with requirements for similar instrumentation and is, therefore, acceptable to the staff.

16.2.5 Table 3.3.7.10-1, Radioactive Gaseous Effluent Monitoring Instrumentation

This change will make Perry Unit 1 Technical Specifications consistent with the GE Standard Technical Specifications and is, therefore, acceptable.

16.2.6 Table 4.3.7.10-1, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

The brief duration required for changing the iodine cartridge and particulate filter is not considered as a basis for declaring a system inoperable. Therefore, it is not necessary to enter the applicable action statement and the proposed change is acceptable.

16.2.7 3.4.9.2, Residual Heat Removal, Cold Shutdown

Specification 3.4.9.2 is currently applicable in Operational Condition 4. The licensee proposed a change that would require this specification to be applicable during Operational Condition 4, when heat losses to the ambient are not sufficient to maintain Operational Condition 4.

The basis for this specification is to provide sufficient heat removal capability for the removal of core decay heat, and mixing to ensure accurate temperature indication. During those times when core decay heat is low enough that heat losses to the ambient are capable of preventing an increase in reactor coolant temperature, there is no need for the residual heat removal (RHR) system and coolant recirculation. This change has been approved for several recently licensed boiling water reactors (BWRs). The staff finds the proposed change acceptable.

16.2.8 4.5.1, Emergency Core Cooling Systems - Operating

Specification 4.5.1.e.2.c currently lists the low-pressure alarm system setpoint for the automatic depressurization system (ADS) as 2475 ± 25 psig. The licensee proposed a change that would change this setpoint to greater than or equal to 155 psig.

The proposed change to Specification 4.5.1.e.2.c will change the alarm setpoint for the ADS instrument air pressure from the current value of 2475 ± 25 psig to greater than or equal to 155 psig. This change is needed to accommodate a new design for the safety-related instrument air system. The new design of the air system will reduce the air compressor pressure from 2500 psig to 160 psig. The air pressure supplied to the ADS accumulators, however, will remain the same, that is, 150 psig.

The alarm setpoint proposed for the system of greater than or equal to 155 psig on decreasing pressure is higher than the required ADS accumulator pressure (150 psig) and is, therefore, acceptable to the staff.

16.2.9 3/4.6.1.2, Primary Containment Leakage

Specification 4.6.1.2.j states that the provisions of Specification 4.0.2 are not applicable to various specifications, including Specification 4.6.1.2.e. This change would delete Specification 4.6.1.2.e from the list under Specification 4.6.1.2.j. Specification 4.6.1.2.e refers to Specification 4.6.1.3, which currently contains a footnote indicating the provisions of Specification 4.0.2 do not apply. Therefore, this proposed change is a necessary enhancement to alleviate the inconsistency that would exist.

16.2.10 3.6.1.3, Primary Containment Air Locks

The proposed change provides an editorial enhancement to the limiting condition for operation (LCO) in Specification 3.6.1.3, Action a. The requested change adds "in one or both air locks" to the action statement. This confirms that one door in each air lock may be inoperable and operation may continue provided appropriate actions are taken, even though the LCO is applied to each containment air lock. The staff finds this proposed change acceptable.

16.2.11 Table 3.6.4-1, Containment and Drywell Isolation Valves

The proposed Technical Specification change was ambiguous. A change was negotiated with the licensee to allow the two containment isolation valves to be opened as necessary during Operational Condition ** to allow a water supply to the fire mains. This is acceptable because it conforms with good fire protection practices.

16.2.12 3.6.5.2, Containment Humidity Control

- (1) Regarding the LCO in Specification 3.6.5.2, the change consists of adding Specification 3.6.1.1.2 to the applicability statement. The change is necessary to cover operational conditions other than 1, 2, or 3, which are addressed in the following action statement. Therefore, the staff finds the proposed change acceptable.
- (2) Regarding the removal of Specification 4.6.5.2.b, which is a temperature instrumentation check related to the containment humidity control LCO, the requirements for the channel check are performed under Specification 4.6.1.7, "Primary Containment Average Air Temperature." Because this change would remove redundant testing requirements, the staff finds the proposed change acceptable.

16.2.13 4.7.4.e.1, Snubbers

Specification 4.7.4.e.1 currently requires functional testing of snubbers. The initial sample size requirement is 10% of the total number of subject snubbers in the plant. The licensee has proposed a change that would reduce the required number of additional snubbers to be tested from 10% to 5% for every failed snubber discovered during functional testing.

In the absence of a suitable snubber failure data base, it was required that, for every failed snubber, an additional 10% of that snubber type was to be tested. Subsequently, the ASME OM4 group developed a sampling plan that determined that only 50% of the initial sample size (10% in this case) need be tested for each failed snubber. This change has been approved for several recently licensed BWRs. The staff finds the proposed change acceptable.

16.2.14 3/4.7.6, Main Turbine Bypass System Surveillance Requirements

The licensee proposed to change the frequency for cycling the non-safety related turbine bypass valves from 7 days to 31 days. Cycling of these valves every 31 days is consistent with the valve manufacturer's recommendations and the surveillance intervals for safety-related equipment in the plant. In addition, the 31-day surveillance interval for these valves is the standard as practiced throughout the nuclear industry. For the above-mentioned reasons, the staff concludes that cycling the turbine bypass valves every 31 days is acceptable.

16.2.15 3.8.2.1 and 3.8.3.3, D.C. Sources

The proposed changes to Specifications 3.8.2.1 and 3.8.2.2 request that the licensee be permitted to take credit for the Unit 2 batteries in place of the Unit 1 dc sources to meet the LCO.

Perry's design currently includes an ability to provide inter-unit ties between the Class 1E dc buses of the two units through tie bus breakers. These breakers are normally open but may be closed under administrative control to allow maintenance or the equalizing of a battery, provided any single component failure does not degrade the Class 1E power systems of any unit below an acceptable level and provided the independence of the redundant systems is maintained. The circuit design meets these applicable criteria and is such that the Unit 2 batteries may be powered from the respective divisional unit battery chargers through the tie bus breakers. When the Unit 2 battery is allowed to connect to the respective Unit 1 dc division, the Unit 2 battery and bus are isolated from Unit 2 associated systems including the dc loads and chargers.

The proposed change to take credit for the Unit 2 batteries to meet the LCO for dc sources is acceptable to the staff.

16.2.16 3.8.4.1, Containment Penetration Conductor Overcurrent Protective Devices

The 120-V circuit breakers are used for the containment penetration conductor overcurrent protective devices. The present Technical Specifications require that an inoperable circuit breaker be removed from service by racking out the breaker. The 120-V circuit breaker mechanism is quite different from the 480-V circuit breaker that is to be physically removed from service. Therefore, the licensee proposed to remove the inoperable 120-V circuit breaker from service by tripping it under administrative control to accomplish the same purpose as that of the racking out. Therefore, the proposed change to remove the inoperable 120-V circuit breaker from service by tripping it under administrative control is acceptable to the staff.

16.2.17 Table 3.8.4.1-1, Containment Penetration Conductor Overcurrent Protective Devices

Table 3.8.4.1-1 lists the containment penetration conductor overcurrent protective devices for Perry Unit 1. The proposed change to this table deletes the circuits that feed the space heaters that were used to maintain the environmental qualification of equipment during plant construction. Some space heaters are no longer used, and the supply circuit breakers have been administratively locked open. Because the overcurrent protective devices for these circuits of the space heaters see no electrical power, the proposed changes to delete the testing requirement for these overcurrent protective devices from the list is acceptable.

16.2.18 3.8.4.2, Reactor Protection System Electric Power Monitoring

Specification 3.8.4.2 is currently applicable at all times. The licensee proposed a change that would change the applicability of Specification 3.8.4.2 to Operational Conditions 1, 2, 3, 4*, and 5.

The reactor protection system (RPS) electric power monitoring assemblies maintain the voltage and frequency of the output of the RPS motor generator sets to acceptable values. If the voltage and frequency of the power to the RPS and the containment isolation instruments are maintained within the limits of Specification 4.8.4.2, the assumptions used in the setpoint calculation will remain valid. Thus, confidence in the RPS instrumentation and the isolation actuation instrumentation to actuate at their respective setpoints is ensured.

The RPS instrumentation and the isolation actuation instrumentation are only required during specified operational conditions, not at all times. The purpose of the RPS is to

- (1) preserve the integrity of the fuel cladding
- (2) preserve the integrity of the reactor coolant system
- (3) minimize the energy that must be adsorbed following a loss-of-coolant accident
- (4) prevent inadvertent criticality

The purpose of the isolation actuation instrumentation is to mitigate the consequences of accidents by prescribing trip setpoints and response times for the isolation of the reactor systems.

Changing the applicability of Specification 3.8.4.2 will not reduce the effectiveness of either the RPS instrumentation or the isolation actuation instrumentation. This change will only make the applicability of the power supply for this instrumentation consistent with the applicability for the instrumentation itself. The staff finds this proposed change acceptable.

*Must be demonstrated operable prior to control rod withdrawal.

16.2.19 3.9.11.2, Residual Heat Removal and Coolant Circulation, Low Water Level

Specification 3.9.11.2 is currently applicable in Operational Condition 4, when irradiated fuel is in the reactor vessel and water level is less than 22 ft 10 in. above the top of the reactor pressure vessel flange. The licensee proposed a change that would require Specification 3.9.11.2 to also be applicable when heat losses to the ambient are not sufficient to maintain Operational Condition 5.

The basis for this specification is to provide sufficient heat removal capability for the removal of core decay heat, and mixing to ensure accurate temperature indication. During those times when core decay heat is low enough that heat losses to the ambient are capable of preventing an increase in reactor coolant temperature, there is no need for the RHR system and coolant recirculation. This change has been approved for several recently licensed BWRs. The staff finds this proposed change acceptable.

16.2.20 3.10.1, Primary Containment Integrity/Drywell Integrity

Specification 3.10.1 allows certain provisions of containment integrity and drywell integrity to be suspended to permit the reactor mode switch to be placed in the startup position with the reactor vessel head and the drywell head removed. The applicability of this specification is Operational Condition 2, during low-power physics tests. The licensee proposed a change that would allow suspension of these same requirements during the conduct of shutdown margin demonstrations as well.

Specification 3.10.1 currently places restrictions on reactor coolant temperature and rated thermal power, with actions required if these limits are exceeded. Shutdown margin demonstrations can easily be performed within the bounds of the present restrictions. Shutdown margin is described as a physics test in FSAR Chapter 14, but it is conducted in Operational Condition 5 in accordance with Specification 3.10.3. This change would revise the applicability of Specification 3.10.1 to allow its provisions to apply to the shutdown margin demonstrations. This change makes Specification 3.10.1 consistent with Specification 3.10.3. The staff finds this proposed change acceptable.

16.2.21 3/4.11.1, Liquid Effluents

The licensee has proposed modifying Specification 4.11.1, "Liquid Effluent Surveillance Requirements," by adding Specification 4.11.1.1.3 for situations during which continuous releases of radioactive liquid effluents occur. Currently, only batch liquid releases have a specific surveillance requirement. Because this change only clarifies that continuous releases, as described in Table 4.11.1.1-1, must meet the same surveillance requirements as batch releases, the staff finds the proposed change acceptable.

16.2.22 Basis 3/4.10.3, Shutdown Margin Demonstration

Specification 3.10.3 allows certain provisions of the Technical Specifications to be suspended in order to perform shutdown margin demonstrations. The bases indicate that shutdown margin demonstrations with the vessel head removed require additional restrictions to ensure that criticality does not occur. The

basis for this special test exception is not accurate. The licensee proposed a change that would make the basis correct by indicating that criticality is properly monitored and controlled.

A description of shutdown margin demonstration is included in FSAR Chapter 14 and the GE Startup Test Specification. Both documents indicate that once shutdown margin has been demonstrated, rods are continued to be withdrawn until criticality is achieved. This is consistent with industry practice. The change to Basis 3/4.10.3 would accurately describe that criticality is properly monitored and controlled, not prevented. The staff finds this proposed change acceptable.

16.2.23 6, Administrative Controls

Management Organization and Technical Support

The licensee has proposed to change Figures 6.2.1-1 and 6.2.2-1 to reflect a realignment of the Perry organization. As described in the licensee's letter dated May 30, 1986, the changes are intended not to create new functions but to maintain and strengthen the nuclear operations organization for plant management and technical support.

In the Plant Operations Department, the organizational changes involve the establishment of the Outage Planning Section, which had already been identified as a future section. In addition, the entire existing Instrumentation and Controls Section will report directly to the Manager, Plant Operations Department, rather than to the Technical Superintendent in the Plant Technical Department.

The changes within the Plant Technical Department involve incorporating the fuel management and analysis function into the Technical Section, allowing for closer coordination between related functions of fuel analysis and reactor engineering, and consolidating nuclear licensing and regulatory compliance activities into a new Licensing and Compliance Section. These two consolidations replace the Nuclear Licensing Fuels Management Section previously in the Nuclear Engineering Department.

In the Nuclear Engineering Department, the design and analysis activities are realigned by discipline resulting in the establishment of three sections: Electrical Design Section, Mechanical Design Section, and Engineering Project Support Section. The existing Cost and Schedules and Construction Services Sections will report to the Manager, Nuclear Engineering Department.

The Quality Assurance Department has refocused the construction quality activities into a Maintenance and Modification Quality Section and has incorporated the Quality Audit Unit into the Operational Quality Section to enhance coordination between audit and surveillance activities.

The Nuclear Construction Department is no longer shown, since the organization charts now depict the corporate organization for plant management and technical support for operations. The ancillary corporate support, previously shown under the Administrative Services Group and the Finance Group, is deleted because it does not directly reflect plant operation or technical support.

The staff finds the proposed changes acceptable because they do not reduce the functions previously provided and the resulting organization is consistent with the acceptance criteria of SRP Section 13.1.1 and Sections 13.1.2-13.1.3 (NUREG-0800).

Plant Operations Review Committee (PORC)

The licensee has proposed to increase the membership of the PORC by adding the General Supervising Engineer (licensing and compliance) and the General Supervising Engineer (outage planning) as members and a Principal Nuclear Operations Engineer as a third vice-chairman/member. The staff finds these additions acceptable because they would broaden the expertise of the PORC. However, to maintain the continuity of experience from PORC meeting to meeting, the addition of three new members to Specification 6.5.1.2 must be accompanied by an increase in the quorum requirements of Specification 6.5.1.5 to a total of at least seven (the Chairman or his designated alternate and at least six members including alternates). Similarly, the number of alternates permitted to participate as voting members may be increased to three from the present two because the ratios of alternates to total members and of alternates to the quorum requirement would be very nearly the same as those ratios for other recently licensed plants and, therefore, would provide similar flexibility in meeting the quorum requirements.

Monthly Operating Reports

The licensee has proposed to change Specification 6.9.1.8 to require that monthly operating reports be sent to the Director, Division of Automated Information Services, rather than to the Director, Office of Management and Program Analysis, the latter office having been recently disbanded. The staff has discussed this matter with the licensee's representative who has concurred in having the reports sent to the Office of Resource Management of which the Division of Automated Services is a part. This change will make Specification 6.9.1.8 consistent with the Standard Technical Specifications and with similar specifications for plants currently being licensed.

16.2.24 6.9.4, Special Reports

The Technical Specifications originally contained a section on fire protection, including requirements for reporting when the fire protection program requirements were not being met. By letter dated November 15, 1985, the licensee opted to remove fire protection from this Technical Specification and add the fire protection program to the FSAR as provided for in Generic Letter 86-10, "Implementation of Fire Protection Requirements". That action also removed the requirements for reporting degradation of the fire protection program. This proposed amendment to the Technical Specification restores those reporting requirements.

The proposed amendment to the Technical Specification specifies those requirements for reporting deficiencies in the fire protection program. This action is necessary to restore to the Technical Specification those reporting requirements that were removed as a result of a previous action by the licensee to move fire protection from this Technical Specification to the FSAR, and is necessary to satisfy Section E of Generic Letter 86-10. This proposed amendment has no effect on the safe operation of the plant.

The staff concludes that the licensee's proposed changes to the Technical Specifications comply with Section E of Generic Letter 86-10 and are acceptable.

16.3 Technical Specifications Not Applicable Above 5% Power

Changes to the following specifications delete various provisions that allowed completion of preoperational tests after initial fuel load, but before 5% power is exceeded.

<u>Specification/ table</u>	<u>Page</u>	<u>Specification/ table</u>	<u>Page</u>
3.3.3.5	3/4 3-50	3.3.8	3/4 3-96
Table 3.3.7.5-1	3/4 3-78	3.4.1.2	3/4 4-4
Table 4.3.7.5-1	3/4 3-80	3.4.1.3	3/4 4-5
3.4.3.1	3/4 4-9	3.6.1.1.2	3/4 6-2
3.11.1.3	3/4 11-6	3.6.5.3	3/4 6-46
3.11.2.5	3/4 11-15	3.6.6.1	3/4 6-47
3.11.3	3/4 11-18	3.7.3	3/4 7-6

16.4 Correction of Typographical Errors or Editorial Clarification

The proposed changes to the following specifications are acceptable because they are of an editorial nature:

<u>Specification/ table/figure</u>	<u>Page</u>	<u>Specification/ table/figure</u>	<u>Page</u>
4.6.1.2.c.1	3/4 6-5	Table 1.2	1-11
3.6.2.1	3/4 6-15	Figure 3.2.1-1	3/4 2-2
3.6.2.3	3/4 6-17	Figure 3.2.1-2	3/4 2-3
Table 3.6.4-1	3/4 6-4D, 41	Table 3.3.2-1	3/4 3-13
4.6.6.2.b	3/4 6-49	4.7.6	3/4 3-96
Table 3.12.1-1	3/4 12-7	Table 4.3.6-1	3/4 3-59, 3-60

18 CONTROL ROOM DESIGN REVIEW

18.2 Evaluation of DCRDR Program Plan Report

Before low-power licensing, the licensee submitted program plan information, a Summary Report, a Summary Report supplement, and other documents describing the organization, process, and results of the Perry detailed control room design review (DCRDR). Staff review of the documents and several onsite audits indicated that sufficient work had been completed to allow low-power licensing of Unit 1. Some DCRDR activities remained to be completed after low-power licensing. A license condition (NPF-45 - Paragraph C.(7) and Attachment 3) required that most of those activities be completed before full-power licensing of Unit 1 and that the remainder be completed before startup following Unit 1's first refueling outage. Subsequent to Unit 1 low-power licensing, the licensee submitted the following additional documents related to the DCRDR:

- (1) "Detailed Control Room Design Review, Summary Report Supplement 2," dated May 28, 1986
- (2) "Detailed Control Room Design Review, HED Status," dated July 2, 1986
- (3) "Control Room Validation, Summary Report," dated July 11, 1986
- (4) "Detailed Control Room Design Review" (errata sheets), dated July 29, 1986
- (5) "Detailed Control Room Design Review" (revised commitments), dated August 26, 1986.

Those documents indicate that the DCRDR activities the licensee had committed to perform before full-power licensing of Unit 1 have been completed. Results of those activities were provided for staff review.

This supplement updates the staff's position on the DCRDR reported in Supplement No. 8. The updated position, based on all available information and arranged in order of the DCRDR elements identified in Supplement No. 1 to NUREG-0737, is provided below.

18.2.1 Establishment of a Qualified Multidisciplinary Review Team

The licensee has established a qualified multidisciplinary review team for the Perry DCRDR. Full satisfaction of this element requires continued participation of personnel from appropriate disciplines, including human factors and operations, during the remainder of the DCRDR.

18.2.2 Use of Function and Task Analyses

This element of the DCRDR, the use of function and task analyses to identify control room tasks and information and control requirements during emergency operations, has been satisfied for Unit 1 as described in Supplement No. 8.

18.2.3 Comparison of Display and Control Requirements With Control Room Inventory

This element of the DCRDR has been satisfied for Unit 1 as described in Supplement No. 8.

18.2.4 A Control Room Survey

The licensee was required to provide the following for staff review before any authorization for operations above 5% of rated thermal power:

- (1) results of a human factors survey of communications equipment in the control room and at the remote shutdown facilities for Unit 1
- (2) results of interim sound surveys in the control room and at the remote shutdown panel for Unit 1

DCRDR Summary Report Supplement 2 indicated that the above surveys were complete. Results, in the form of human engineering observations (HEOs) identified by the surveys, were provided. In the staff's judgment, the information provided confirms that the portion of the Unit 1 low-power license condition related to control room survey activities to be completed before full-power licensing has been satisfied.

The licensee has committed to conduct a final noise survey of the Unit 1 control room and remote shutdown panel after the installation of carpets but before startup following the first refueling outage (i.e., while the plant is under normal operational conditions). In the staff's judgment, that schedule is acceptable. This element should be fully satisfied for Unit 1 on completion of the final noise survey.

18.2.5 Assessment of HEDs

Supplement No. 8 indicated that the licensee had developed an assessment process that was producing acceptable results. HEOs identified as significant are re-designated human engineering discrepancies (HEDs) by the licensee. Staff review of recent submittals indicates that the assessment process continues to produce acceptable results. This element should be fully satisfied for Unit 1 when HEOs identified by the final noise survey have been assessed.

18.2.6 Selection of Design Improvements

The licensee has developed an acceptable process for selecting design improvements. DCRDR Summary Report Supplements 1 and 2 and the Control Room Validation Summary Report indicate that the process has been used to select design improvements for all HEDs (i.e., significant HEOs, including those identified by activities that have to be completed before Unit 1 exceeds 5% of rated thermal power). This element should be fully satisfied for Unit 1 when design improvements for HEDs identified by the final noise survey have been selected.

18.2.7 Verification That Selected Improvements Provide the Necessary Correction and Do Not Introduce New HEDs

Supplement No. 8 indicated staff concern about the completeness of the licensee's process for verifying that selected improvements provide the necessary correction and do not introduce new HEDs. Because of that concern, the licensee was required to provide results of an augmented verification process for staff review. The licensee was also required to address specific verification issues identified in Supplement No. 7. Submittal of the above information was required before Unit 1 exceeds 5% of rated thermal power.

DCRDR Summary Report Supplement 2 indicated completion of the augmented verification process for all HED corrections implemented in Unit 1. The results were summarized, and specific information about selected improvements that either failed to provide the necessary correction or introduced new HEDs was provided. DCRDR Summary Report Supplement 2 also addressed the specific verification issues in Supplement No. 7. In the staff's judgment, the information provided confirms that the portion of the Unit 1 low-power license condition related to verification of HED corrections has been satisfied.

A portion of the augmented verification process remains to be completed. The licensee has committed to augmented verification of HED corrections that remain to be implemented after full-power licensing (see Table 18.1). The schedule commitment is before startup following the second refueling outage and includes correction of any problems identified by the augmented verification. In the staff's judgment, the licensee's commitment and schedule are acceptable. This element of the DCRDR should be fully satisfied for Unit 1 when HED corrections that still have to be implemented in that unit have been subjected to the augmented verification.

18.2.8 Coordination of Control Room Improvements With Other Programs

As indicated in Supplement No. 8, the licensee has an acceptable process for coordinating control room improvements with changes from other programs. The licensee committed to provide, before Unit 1 exceeds 5% of rated thermal power, the results of a control room validation as evidence that this element was successfully completed. The appropriate results were provided by letter dated July 11, 1986 (M. R. Edelman to W. R. Butler). In the staff's judgment, the information provided confirms that the portion of the Unit 1 low-power license condition related to validation of the emergency instructions has been satisfied.

18.2.9 Other

Several activities, in addition to those discussed above, are required to satisfy the DCRDR requirements in Supplement No. 1 to NUREG-0737 and to complete the licensing process. Those activities include correction of remaining Unit 1 HEDs, completion of plant-specific DCRDR activities for Unit 2, and submittal of information to supplement the Summary Report.

18.2.9.1 HED Corrections for Unit 1

The DCRDR has identified 383 HEDs for Unit 1. The resolution status of those HEDs, based on the latest information available to the staff, is provided in Table 18.1. Per a licensee commitment, a large number of those HEDs were corrected before low-power licensing of Unit 1. Other HEDs, identified in letters

dated October 14, 1985 (M. R. Edelman to B. J. Youngblood) and February 19, 1986 (M. R. Edelman to W. R. Butler) were required to be corrected before Unit 1 exceeds 5% of rated thermal power. A letter dated July 2, 1986 (M. R. Edelman to W. R. Butler) confirmed that those corrections were completed and that portion of the low-power license condition related to HED correction before full-power licensing was satisfied.

The licensee previously committed to correct some HEDs before startup following Unit 1's first refueling outage. That commitment was acceptable to the staff. DCRDR activities since low-power licensing (as discussed in Sections 18.2.4 through 18.2.8) have identified

- (1) several additional HEDs
- (2) several original HEDs for which the selected improvement either did not provide complete resolution or resulted in a new HED

Summary Report Supplement 2 provides proposed resolutions and correction schedules for those HEDs. Staff review indicates that proposed corrections and justifications for not correcting or partially correcting HEDs are acceptable. Schedules for implementing corrections include "prior to the beginning of the 100-hour warranty run" and "prior to startup following the first refueling outage." The staff finds the schedules acceptable.

18.2.9.2 Completion of DCRDR for Unit 2

The staff understands that results of the DCRDR performed on Unit 1 will be applied to Unit 2 before that unit is licensed. However, certain activities specific for Unit 2 may be necessary to satisfy the DCRDR requirements in Supplement No. 1 to NUREG-0737. Those activities cannot be fully specified at this time, but should include the following:

- (1) identification of differences between the Unit 1 and Unit 2 control rooms and remote shutdown panels
- (2) determination of which of those differences indicate a need for further DCRDR activities
- (3) performance of the needed DCRDR activities

18.3 Summary Report Requirements

The licensee submitted the Summary Report for the Perry DCRDR and Supplement 1 to that Summary Report before low-power licensing of Unit 1. In the staff's judgment, those documents, along with confirmations of HED corrections, provided the required information and confirmed that the DCRDR was sufficiently complete to allow low-power licensing of Unit 1. Since low-power licensing, the licensee has submitted Summary Report Supplement 2. In the staff's judgment, that document, along with confirmations of HED corrections and the Control Room Validation Summary Report, confirmed that the DCRDR was sufficiently complete to allow full-power licensing of Unit 1. The Summary Report requirement will be fully satisfied for Unit 1 when the licensee submits the proposed resolutions and implementation schedules for HEDs identified by the final noise survey and the

results of the augmented verification of HEDs corrected after full-power licensing.

A summary report or supplement should also be submitted to signify the licensee's completion of the DCRDR for Unit 2. That submittal should be based on the activities discussed in Section 18.2.9.2 above and should

- (1) outline proposed changes to the Unit 2 control room and remote shutdown panel that result from those activities
- (2) outline proposed schedules for implementation of HED corrections in the Unit 2 control room and at the Unit 2 remote shutdown panel
- (3) provide summary justification for HEDs with safety significance to be left uncorrected or partially corrected in the Unit 2 control room and at the Unit 2 remote shutdown panel

The licensee should submit the above information at least 6 months before the expected date for low-power licensing of Unit 2.

18.4 Conclusion

The licensee's recent submittals indicate that DCRDR activities that must be performed before full-power licensing of Unit 1 have been completed. Results of those activities were provided for staff review. In the staff's judgment, those results confirm that the Unit 1 low-power license condition has been satisfied and that the DCRDR is sufficiently complete to allow full-power licensing of that unit. Activities that still have to be completed are identified below.

18.4.1 HED Correction and DCRDR Elements

The following activities must be finished before the Unit 1 DCRDR is complete:

- (1) Implementation of HED corrections per the licensee's commitments
- (2) Completion of the final noise survey
- (3) Assessment of HEDs identified by final noise survey
- (4) Selection of design improvements for safety-significant HEDs (i.e., HEDs) identified by the final noise survey
- (5) Completion of the augmented verification process for HED corrections implemented after full-power licensing. This includes selection of new design improvements for HEDs that are not corrected by the originally selected design improvements or whose correction introduces new HEDs and implementation of those design improvements

The licensee has committed to complete the first four activities before startup following Unit 1's first refueling outage and to complete the fifth activity before startup following Unit 1's second refueling outage. All the activities are addressed in the recommended full-power license condition for Unit 1. Continued participation of personnel from appropriate disciplines will be treated as a confirmatory item.

The Unit 2 DCRDR also requires completion of several activities. Those activities are identified in Section 18.2.9.2 of this supplement and should be completed before low-power licensing of Unit 2.

18.4.2 Documentation and Reporting

The licensee is required to keep an auditable record of all activities necessary to complete the DCRDR. In addition, the licensee should satisfy certain reporting commitments. For Unit 1 those commitments are:

- (1) confirm implementation of HED corrections that the licensee has committed to complete before the beginning of the 100-hour warranty run
- (2) confirm implementation of HED corrections that the licensee has committed to complete before startup following the first refueling outage
- (3) report results of the final noise survey for staff review; include proposed corrective actions and implementation schedules or justifications for not correcting or partially correcting any HEDs identified
- (4) confirm completion of the augmented verification of HED corrections implemented after full-power licensing and correction of any problems identified by that verification

Reporting requirements for Unit 2 are described in Section 18.3 of this supplement.

The staff will continue its evaluation of the Perry DCRDR when the above information has been submitted and will report the final evaluation in a future supplement to the SER.

Table 18.1 Status of human engineering discrepancy (HED) corrections based on April 8-12 and September 10-13, 1985, staff audits and October 14, December 18, and February 19, 1985, and March 28, July 2, and July 11, 1986, submittals from the licensee (revised from Supplement No. 8)

HED status	HED no.
Acceptable justification for not correcting has been provided	
Closed (24)	11, 20, 24, 26, 58, 97, 108, 132, 192, 206, 207, 300, 315, 316, 317, 352, 369, 401, 502, 505, 520, 523, 524, 605
To be corrected before low-power licensing	
Corrected and closed (282)	1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 12, 13, 14, 15, 16, 17, 18, 19, 21, 22, 23, 25, 27, 28, 30, 31, 32, 34, 35, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 73, 75, 78, 79, 82, 83, 87, 88, 89, 90, 91, 92, 95, 98, 99, 101, 103, 104, 106, 111, 112, 113, 115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 133, 134, 135, 136, 137, 138, 139, 140, 141, 142, 143, 144, 145, 146, 147, 148, 149, 150, 151, 152, 153, 154, 155, 156, 157, 159, 160, 161, 162, 163, 164, 165, 166, 167, 168, 169, 170, 171, 172, 173, 174, 175, 176, 177, 178, 179, 180, 181, 183, 184, 185, 186, 187, 188, 189, 190, 191, 193, 194, 196, 197, 198, 199, 200, 201, 202, 203, 204, 205, 208, 209, 210, 211, 212, 213, 214, 215, 216, 301, 302, 303, 304, 305, 306, 307, 308, 309, 311, 312, 318, 319, 321, 322, 323, 324, 325, 326, 327, 329, 330, 331, 332, 333, 334, 335, 336, 337, 338, 339, 340, 341, 342, 343, 344, 345, 346, 347, 348, 349, 350, 351, 353, 355, 356, 357, 358, 359, 361, 362, 367, 368, 371, 400, 402, 403, 404, 405, 406, 407, 503, 504, 507, 510, 516, 518, 530, 1001, 1002, 1003, 1004, 1005, 1006, 1007, 1008, 1009, 1010, 1011, 1013, 1014, 1015, 1016, 1017, 1018, 1020, 1021, 1022, 1023, 1024, 1025, 1026, 1027, 1028, 1029, 1030, 1031, 1032, 1033, 1034, 1035, 1037

Table 18.1 (Continued)

HED status	HED no.
Partial or interim correction will be implemented before low-power licensing; will be completely corrected before startup following the first refueling outage	
Closed on partial or interim correction - open pending complete correction (23)	77, 80, 81, 84, 100, 102, 182, 313, 314, 328, 354, 360, 363, 364, 365, 366, 370, 501, 508, 509, 511, 1012, 1019
To be corrected before full-power licensing	
Corrected and closed (5)	93, 500, 517, 519, 617
Partial or interim corrections will be implemented before full-power licensing; will be completely corrected before startup following the first refueling outage	
Closed on partial or interim correction - open pending complete correction (2)	527, 616
To be corrected before the beginning of the 100-hour warranty run	
Open pending implementation of correction (1)	600
On the basis of results of augmented verification - open pending revised or complete correction (7)	36, 85, 86, 109, 114, 310, 522
To be corrected before startup following the first refueling outage	
Open pending implementation of correction (29)	72, 74, 96, 107, 408, 506, 512, 513, 514, 515, 521, 525, 526, 528, 529, 601, 602, 603, 604, 606, 607, 608, 609, 610, 611, 612, 613, 614, 615
On the basis of results of augmented verification - open pending revised or complete correction (10)	29, 33, 76, 94, 105, 110, 158, 195, 320, 1036

APPENDIX A

CONTINUATION OF CHRONOLOGY PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

March 7, 1986 Letter to applicant forwarding additional comments on final draft Technical Specifications to correct editorial and typographical errors and requesting certification that previously transmitted Technical Specifications are consistent with Final Safety Analysis Report (FSAR), SER, and as-built facility.

March 10, 1986 Letter from applicant certifying that final draft Technical Specifications are consistent with FSAR, SER and its supplements, and as-built plant per staff's request of November 19, 1985.

March 11, 1986 Letter from applicant forwarding additional information on seismic event evaluation report, including (1) list of equipment and criteria used for selection of equipment and (2) list of equipment for further evaluation, per commitment of March 3, 1986.

March 11, 1986 Letter to applicant forwarding SSER No. 9.

March 13, 1986 Letter from applicant forwarding information on status of pending changes to FSAR, correcting editorial and typographical errors.

March 14, 1986 Letter from applicant forwarding application to amend CPPR-148, extending construction completion date to May 15, 1986.

March 18, 1986 Letter to applicant stating that draft revised FSAR pages are acceptable for low-power licensing on the basis of preliminary review. Revisions will be formally documented as proposed FSAR Amendment 25. Review of formal amendment will be documented in supplement to SER.

March 18, 1986 Letter to applicant forwarding License NPF-45, Federal Register notice, and Amendment 1 to Indemnity Agreement B-98.

March 20, 1986 Generic Letter 86-07 to all reactor licensees and applicants regarding transmittal of NUREG-1190 concerning San Onofre Unit 1 loss-of-power and waterhammer event.

March 20, 1986 Letter to licensee forwarding for comment Advisory Committee on Reactor Safeguards (ACRS) March 17, 1986, report

on January 31, 1986, earthquake, which occurred near site and requesting information on plans and schedule regarding use of sensitive seismological instruments.

- March 20, 1986 . Memorandum and Order issued by Atomic Safety and Licensing Appeal Board (ASLAB). A hearing will be held in the Cleveland area on the January 31, 1986, earthquake, and a prehearing telephone conference will be held at 3 p.m. on April 8, 1986.
- March 23, 1986 Generic Letter 86-08 to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits regarding availability of Supplement 4 to NUREG-0933, "Prioritization of Generic Safety Issues."
- March 25, 1986 Letter from licensee transmitting Emergency Action Plan, Revision 4.
- March 27, 1986 Order issued by ASLAB granting the motion of intervenor Ohio Citizens for Responsible Energy (OCRE) for leave to reply to the responses of the applicants and the NRC staff to OCRE's motion to reopen the record in the Perry operating license proceeding.
- March 31, 1986 Letter to licensee concluding that financial information submitted in utility's letter of March 14, 1986, satisfies requirements of 10 CFR 140.21 regarding approved guarantee of payment of deferred premiums.
- March 31, 1986 Letter from licensee forwarding 1985 annual report to share owners, including certified financial statements, per 10 CFR 50.71(b).
- March 31, 1986 Generic Letter 86-09 to all licensee of operating boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) and license applicants regarding technical resolution of Generic Issue B-59 - N-1 loop operation in BWRs and PWRs.
- April 3, 1986 Letter from licensee discussing plans and schedules regarding comments in ACRS letter of March 17, 1986. Evaluation of well and geological conditions to conclude whether relationship exists between injection well and January 31, 1986, earthquake is continuing, per SSER No. 9.
- April 7, 1986 Summary of March 13, 1986, meeting with Mark III Containment Hydrogen Control Owners Group in Bethesda, Maryland, regarding heat loss analysis.
- April 8, 1986 Order extending Commission review time of Director's Decision on 10 CFR 2.206 Petitions (DD-86-4) to May 30, 1986.

April 8, 1986 Letter from licensee forwarding seven additional relief requests to utility's inservice testing program for pumps and valves.

April 8, 1986 ASLAB Memorandum and Order to allow OCRE to litigate the adequacy of facility's seismic design.

April 9, 1986 Letter from Jay Silberg, Counsel for applicants, forwarding testimonies presented at April 8, 1986, oversight hearing before Subcommittee on Energy and Environment, per ASLAB's consideration of OCRE Motion To Reopen the Record based on the January 31, 1986, earthquake.

April 10, 1986 Letter from licensee forwarding pages inadvertently omitted from supplemental information to seismic event evaluation report submitted by March 11, 1986, letter.

April 14, 1986 ASLAB Order. Applicant's request that April 8, 1986, Memorandum and Order be modified to permit Talwani's testimony on or before May 5, 1986.

April 15, 1986 Letter from licensee forwarding "Snubber Augmented Inservice Inspection/Examination and Functional Testing Program for Perry Nuclear Power Plant, Unit 1," and requesting written approval of program and relief request from American Society of Mechanical Engineers requirements for snubber test sample sizes.

April 16, 1986 Letter from licensee forwarding Weston Geophysical "Seismological and Geological Investigations of January 31, 1986, Leroy, OH, Earthquake," originally described in Attachment 1 of licensee's letter dated February 28, 1986.

April 16, 1986 Letter from licensee advising that "Statement of NRC Before Subcommittee on Energy of Committee on Interior and Insular Affairs, U.S. House of Representatives..." was inadvertently omitted from April 9, 1986, letter.

April 18, 1986 Memorandum and Order CLI-86-07 denying petition to reopen hearings regarding OCRE contentions challenging the adequacy of the facility's seismic design after the January 31, 1986, earthquake.

April 23, 1986 Letter from licensee submitting Licensee Event Report 86-001.

April 24, 1986 Letter from licensee forwarding "Perry Nuclear Power Plant Emergency Instruction Validation Summary Report," before initial criticality, per license condition described in Attachment 3 to License NPF-45. Additional reports, per detailed control room design review, will be submitted by May 1986.

April 25, 1986 Letter from licensee forwarding update to March 3, 1986, commitments, including relocation of seismic instrument and

procedure enhancements. Temporary change to Revision 2 of Off-Normal Instruction ONI-D51, "Earthquake," in operations manual enclosed.

- April 28, 1986 Letter from licensee submitting Licensee Event Report 86-002.
- April 30, 1986 Letter from licensee submitting Licensee Event Report 86-003.
- May 1, 1986 Letter from Transamerica Delaval, Inc. (TDI), forwarding Revision 2 to "TDI Owners Group Appendix II: Generic Maintenance Matrix and Justifications," which was approved by TDI Owners Group and TDI.
- May 2, 1986 Letter to licensee requesting assistance in developing information for better NRC management control to ensure implementation of generic issue solutions.
- May 5, 1986 Summary of April 18, 1986, meetings with representatives from U.S. Geological Survey (USGS) and utility regarding progress on geology/seismology confirmatory issues for plants.
- May 8, 1986 Summary of operating reactors events April 28, 1986, meeting regarding Chernobyl nuclear accident and events at U.S. reactors.
- May 12, 1986 Summary of April 30, 1986, meeting with representatives from utility, NTS Engineering, USGS, Gilbert Commonwealth, Weston Geophysical, Geoscience Services, and University of South Carolina regarding earthquake confirmatory work required in SSER No. 9.
- May 13, 1986 Letter from licensee responding to request for information, including current programs, equivalency criteria to engineering degree and future plans, regarding Generic Letter 86-04 concerning engineering expertise on shift.
- May 13, 1986 Letter from licensee informing staff of organizational changes resulting from April 29, 1986, merger between utility and Toledo Edison Co. Utility now subsidiary of Centrior Energy Corp. Management organizational changes implemented per R. M. Ginn December 20, 1985, letter. Technical Specifications and FSAR unaffected.
- May 13, 1986 Letter from licensee submitting Licensee Event Report 86-004.
- May 14, 1986 Letter from licensee attaching final report concerning high voltage power circuitry by Kaman Instrument Co.
- May 21, 1986 Letter from licensee submitting Licensee Event Reports 86-005 and 86-007.
- May 22, 1986 Letter from licensee submitting Licensee Event Report 86-006.

May 23, 1986 Letter from licensee forwarding fees for review of April 8, 1986, request for relief from inservice testing program.

May 28, 1986 Letter from licensee forwarding Supplement 2 to "DCRDR Summary Report," containing results of human factors survey of communications equipment and augmented verification review. Human engineering discrepancies will be corrected by June 27, 1986.

May 30, 1986 Letter from licensee advising staff of personnel reassignments and organizational changes made at plant. Detailed description of organizational changes enclosed.

May 30, 1986 Letter from licensee responding to May 2, 1986, request for multiplant action status per May 2, 1986, letter. Enclosed information is considered preliminary and subject to future modifications.

May 30, 1986 Letter from licensee forwarding additional changes to the FSAR. Safety evaluations indicate no safety questions exist. Changes will be included in FSAR Amendment 25, due in June 1986.

May 30, 1986 Letter from licensee submitting Licensee Event Report 86-008.

June 3, 1986 Letter from licensee submitting Licensee Event Reports 86-009 and 86-011.

June 6, 1986 Letter from licensee submitting Licensee Event Reports 86-010, 86-012, and 86-013.

June 9, 1986 Letter to licensee forwarding draft full-power License NPF-58 for review and comments to ensure that the license accurately reflects various commitments reported in the FSAR, SERs, and other documents.

June 13, 1986 Letter from licensee submitting Licensee Event Report 86-014.

June 17, 1986 Letter from licensee forwarding Design Control Procedure 2.10, "Perry Power Plant Confirmatory Program of January 31, 1986, Ohio Earthquake Effect," providing final results to confirmatory activity in SSER No. 9, Sections 3.7.2 and 3.10.1.

June 17, 1986 Letter from licensee forwarding "Deep Well Injection at Calhio Wells and Leroy, OH Earthquake of January 31, 1986," per April 31, 1986, and June 11, 1986, progress meetings regarding SSER No. 9 confirmatory work concerning injection wells. Description of proposed seismic monitoring network also enclosed.

June 18, 1986 Letter from licensee forwarding list of Technical Specifications changes to accompany full-power operating license,

representing minor editorial corrections, clarifications, and enhancements. Justifications and proposed markedup Technical Specification pages also enclosed.

- June 18, 1986 Letter from licensee submitting Licensee Event Report 86-015.
- June 24, 1986 Letter from licensee forwarding "Investigations of Confirmatory Seismological and Geological Issues, Northeastern Ohio Earthquake of January 31, 1986," closing out SSER No. 9 Confirmatory Activities 1 and 3.
- June 25, 1986 Letter from licensee submitting FSAR Amendment 25.
- June 25, 1986 Letter from licensee submitting Licensee Event Reports 86-016 and 86-017.
- June 27, 1986 Letter from licensee submitting Licensee Event Reports 86-019 and 86-020.
- June 30, 1986 Summary of June 11, 1986, meeting with the licensee to discuss progress of Perry earthquake confirmatory work.
- July 2, 1986 Letter from licensee informing staff of the completion status of human engineering deficiencies addressed in Supplement 2 of the Detailed Control Room Design Review Summary Report, provided by the licensee by letter dated May 28, 1986.
- July 2, 1986 Letter from licensee submitting Licensee Event Reports 86-016 and 86-021.
- July 8, 1986 Letter from licensee advising staff of change of addressees in Perry Service List.
- July 8, 1986 Letter from licensee advising staff of a correction in the scheduling for the chemical waste injection wells monitoring. The licensee intends to operate the monitoring system through startup following the first refueling outage for Perry Unit 1.
- July 8, 1986 Letter to licensee transmitting safety evaluation report accepting the TDI Diesel Generator Owners Group program to validate and upgrade the design and manufacturing quality of the TDI diesel generators for nuclear emergency standby service.
- July 9, 1986 Letter from licensee submitting Licensee Event Report 86-022 and 86-024.
- July 10, 1986 Letter from licensee proposing FSAR changes that change the safety-related instrument air system from a high-pressure to a low-pressure system.

July 10, 1986 Letter from licensee submitting Perry plant-unique analysis for emergency containment venting in accordance with the licensee's commitment in FSAR Appendix 1B, to the extent discussed in SSER No. 8, Section 13.5.2.2.1.

July 10, 1986 Letter from licensee submitting Licensee Event Report 86-023.

July 11, 1986 Letter from licensee requesting additional relief from the NRC-approved Perry Unit 1 Preservice Inspection Program and associated justifications for the additional relief requested.

July 11, 1986 Letter from licensee submitting Perry Unit 1 control room validation report in accordance with commitment made in Section 2.7 of the Detailed Control Room Design Review Summary Report, Supplement 1, submitted by the licensee with the letter dated October 2, 1985.

July 14, 1986 Letter from licensee submitting a summary report of the offgas system charcoal fire events that occurred on June 20 and July 6, 1986.

July 15, 1986 Letter from licensee submitting Physical Security Event Report No. 86-001.

July 16, 1986 Letter from licensee submitting Licensee Event Report 86-026.

July 17, 1986 Letter from licensee notifying the staff of the completion of actions required by License Condition 2.C.(10)(a) in NPF-45, low-power operating license for Perry Unit 1.

July 17, 1986 Letter from licensee submitting Licensee Event Report 86-025.

July 18, 1986 Letter from licensee requesting additional Technical Specifications required for the full-power operation of Perry Unit 1.

July 23, 1986 Letter from licensee submitting Licensee Event Reports 86-028 and 86-029.

July 25, 1986 Letter from licensee submitting Licensee Event Report 86-027.

July 29, 1986 Letter from licensee submitting final report of the offgas system charcoal fire events that occurred on June 20 and July 6, 1986.

July 29, 1986 Letter from licensee providing clarification and updates to information supplied in letters dated May 28 and July 18, 1986, regarding Detailed Control Room Design Review Summary Report, Supplement 2.

July 29, 1986 Letter from licensee submitting a revision to the Perry plant-unique analysis for emergency containment venting, superseding the information submitted by letter dated July 10, 1986.

July 30, 1986 Letter from licensee submitting results of the environmental qualification of silicone sealant committed to in FSAR, Appendix 1B, and discussed in SSER No. 8, Section 6.5.

July 30, 1986 Letter from licensee requesting further Technical Specification changes for Perry Unit 1 full-power operation.

July 30, 1986 Letter from licensee submitting Licensee Event Reports 86-030 and 86-031.

August 1, 1986 Letter from licensee submitting Licensee Event Report 86-032.

August 6, 1986 Letter from licensee notifying staff that emergency procedures have been developed for use of the hydrogen ignition system as required by NPF-45 License Condition 2.C.10(b).

August 8, 1986 Letter from licensee submitting Licensee Event Reports 86-033 and 86-034.

August 11, 1986 Letter from licensee submitting the plant monthly operating report for July 1986 in accordance with Technical Specification 6.9.1.8.

August 13, 1986 Letter from licensee providing additional information (contained in an earlier letter dated July 18, 1986) to support maximum extended operating range Technical Specification for full-power licensing of Perry Unit 1.

August 13, 1986 Letter to licensee transmitting the staff's supplemental evaluation report related to the January 31, 1986, Ohio earthquake, which is to be documented in SSER No. 10.

August 13, 1986 Letter from licensee submitting Licensee Event Report 86-035.

August 15, 1986 Letter from the Governor of Ohio to the NRC announcing the withdrawal of State support for the Perry offsite emergency plans, pending a study of the plan's emergency planning zone area size in view of the Chernobyl accident.

August 15, 1986 Letter from licensee submitting Licensee Event Reports 86-036, 86-037, and 86-038.

August 20, 1986 Letter from licensee providing additional information relative to the Plant Emergency Instructions for emergency containment venting.

August 26, 1986 Letter from licensee submitting revisions to the Detailed Control Room Design Review Summary Report, Supplement 2, supplemental to information provided with letters dated May 28 and July 29, 1986.

APPENDIX B

REFERENCES*

Borcherdt, R. D., ed., "Preliminary Report on Aftershock Sequence for the Earthquake of January 31, 1986, Near Painesville, Ohio," U.S. Geological Survey Open-File Report 86-181, 1986.

Charnley, J. S., GE, letter to F. J. Miraglia, "Loss of Feedwater Heating Analysis," NFM-125-83, July 5, 1983.

Cleveland Electric Illuminating Company, "Final Safety Analysis Report for the Perry Nuclear Power Plant, Units 1 and 2" (Docket Nos. 50-440 and 50-441), through Amendment 25, July 1985.

Code of Federal Regulations, Title 10, "Energy" (contains general design criteria), and Title 44, "Emergency Management and Assistance," U.S. Government Printing Office, Washington, DC.

Crutchfield, D., NRC, letter to T. Dente, BWR Owners Group, "Safety Evaluation Report of Emergency Procedure Guidelines, Rev. 3," November 23, 1983.

Failure Analysis Associates, Inc., FaAA-84-9-11, "Design Review of TDI R-4 Series Emergency Diesel Generator Cylinder Blocks," December 1984.

General Electric Company, Topical Report NEDO-24934, "Emergency Procedure Guidelines, BWR 1-6," Rev. 3, December 8, 1982.

---, NEDM-30963, "Technical Specification Operating Limits With the Elimination of the APRM Trip Setdown Requirements - Perry Nuclear Power Plants," April 1985.

---, "PNPP Maximum Extended Operating Domain Analysis," June 1985.

Gilbert/Commonwealth Associates, Topical Report GAI-TR-101-NP-A, "Computation of Radiological Consequences Using INHEC Computer Program," March 1976.

---, "The Cleveland Electric Illuminating Company, Perry Power Plant, Confirmatory Program of The January 31, 1986, Ohio Earthquake Effect," June 16, 1986.

Houston, R. W., NRC, memorandum to T. M. Novak, "Partial Feedwater Heating Operation" March 15, 1985.

Johnson, J., et al., "Review of the Effect of the January 31, 1986 Earthquake on the Perry Nuclear Power Plant," Lawrence Livermore National Laboratory, Report 8666-02-01, Rev. 1, August 1986.

*All correspondence between the applicant and the NRC staff referenced in this supplement is listed in Appendix A of the SER and its supplements on a continuing basis.

Lainas, G. C., NRC, memorandum to W. Butler, "PNPP Maximum Extended Operating Domain Analysis," December 17, 1985.

Langley, J. R., Hydrogen Control Owners Group, letter to R. M. Bernero, NRC, "Combustible Gas Control Emergency Procedure Guidelines," HGN-086, June 3, 1986.

---, letter to R. M. Bernero, NRC, "Appendix B to the Combustible Gas Control Emergency Procedure Guidelines," HGN-090, July 3, 1986.

Talwani, P., and S. Acree, "Deep Well Injection at the Calhio Wells and the Leroy, Ohio Earthquake of January 31, 1986," submitted by Cleveland Electric Illuminating Company letter dated June 17, 1986.

U.S. Nuclear Regulatory Commission, Generic Letter 82-33, "Supplement No. 1, NUREG-0737 - Requirements for Emergency Response Capability," December 17, 1982.

---, Generic Letter 83-28, "Required Actions Based on Implications of the Salem ATWS Events," July 8, 1983.

---, Generic Letter 86-10, "Implementation of Fire Protection Requirements," April 24, 1986.

---, NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," November 1978.

---, NUREG-0700, "Guidelines for Control Room Design Reviews," September 1981.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, December 1982 (also Generic Letter 82-33, December 17, 1982).

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (includes branch technical positions).

---, NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," August 1982.

---, NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," Vol. 2, August 1983.

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---, NUREG/CR-3805, "Engineering Characterization of Ground Motion, Task I: Effects of Characteristics of Free-Field Motion on Structural Response," R. P. Kennedy, et al., May 1984.

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INDUSTRY STANDARDS

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National Fire Protection Association, NFPA 72D-1975, "Proprietary Protective Signaling Systems."

APPENDIX E

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APPENDIX F

ADVISORY COMMITTEE FOR REACTOR SAFEGUARDS
REPORT ON THE PERRY NUCLEAR POWER PLANT,
UNIT 1, DISCUSSING OHIO EARTHQUAKE ON
JANUARY 31, 1986



UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 17, 1986

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE PERRY NUCLEAR POWER PLANT, UNIT 1

During its 311th meeting, March 13-15, 1986, the Advisory Committee on Reactor Safeguards discussed the Ohio earthquake which occurred on January 31, 1986 and reviewed its implications with respect to the Perry Nuclear Power Plant, Unit 1. A meeting of the Subcommittee on Extreme External Phenomena was held on March 12, 1986 to consider this matter. The Reactor Operations Subcommittee discussed this matter during a meeting on February 12, 1986 and the full Committee was briefed during the 310th ACRS meeting, February 13-15, 1986. During our review, we had the benefit of discussions with representatives of The Cleveland Electric Illuminating Company (Applicant), the NRC Staff, and the U. S. Geological Survey (USGS) Staff. We also had the benefit of the documents referenced and comments from several ACRS consultants. We commented on the application for a license to operate this plant on July 13, 1982.

This earthquake occurred near Leroy, Ohio, and was characterized by relatively low energy, low velocities, small displacements, a short duration, and a response spectrum rich in high frequencies. Except at the relatively less significant high frequencies, the excitation of the plant structures and equipment was much less than that considered in the seismic design basis.

No significant damage was observed at the Perry plant in the inspections which were performed by the Applicant and the NRC Staff. The Cleveland Electric Illuminating Company, by using analysis and comparisons with prior qualification testing, has found that all the structures and equipment analyzed thus far have substantial margins of safety relative to the loads and stresses induced by the earthquake. The NRC Staff and our consultants concur in that conclusion.

The NRC Staff has several confirmatory actions that it will require of the Applicant prior to operation above 5 percent power. These confirmatory actions include the analyses of a large sample of equipment. We support the NRC Staff's proposed confirmatory actions.

There currently exists some possibility that the January 31, 1986 earthquake is related to deep well injection activities that took place between the Perry plant site and the town of Leroy or to solution mining

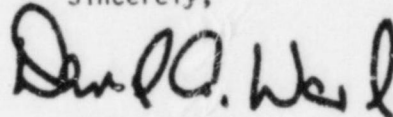
March 17, 1986

that took place in this area. The NRC Staff has engaged the services of USGS to evaluate these hypotheses to see if there really may be a causal connection, and, if so, whether there is any likelihood of substantially larger earthquakes in the future. The NRC Staff will keep the ACRS informed as to the progress of the USGS work.

One of the ACRS consultants suggested that monitoring with sensitive seismological instruments over the next few years would be helpful in assessing the possible causal connection between the deep well injection and the January 31, 1986 earthquake. The USGS representatives attending our discussions agreed that such seismic monitoring would be valuable. Therefore, unless the USGS and the NRC are able to decide that there is no causal connection or that earthquakes of a magnitude sufficient to be of concern can be ruled out from this cause, we recommend that The Cleveland Electric Illuminating Company assure that appropriately sensitive monitoring be continued over the next few years.

We agree with the NRC Staff that the January 31, 1986 earthquake is unlikely to lead to any requirements that would significantly change the design of the Perry plant's structure or its equipment. Based on the information developed in these meetings and considering the above comments, we find no reason to alter the conclusions stated in the Committee's report dated July 13, 1982 regarding operation of this nuclear plant.

Sincerely,



David A. Ward
Chairman

References:

1. The Cleveland Electric Illuminating Company, "Seismic Event Evaluation Report, Perry Nuclear Power Plant," dated February 1986
2. Letter dated March 5, 1986 from Robert M. Bernero, Director, Division of BWR Licensing, NRC, to David Ward, Chairman, ACRS, Subject: Perry Seismic Safety Evaluation, with attached Safety Evaluation Report dated March 1986

APPENDIX S
FEDERAL EMERGENCY MANAGEMENT AGENCY
SUPPLEMENTAL FINDINGS



Federal Emergency Management Agency

Washington, D.C. 20472

FEB 4 1986

MEMORANDUM FOR: Edward L. Jordan
Director, Division of Emergency
Preparedness and Engineering Response
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission

FROM: *Richard W. Krimm*
Richard W. Krimm
Assistant Associate Director
Office of Natural and Technological
Hazards Programs

SUBJECT: Perry Atomic Safety and Licensing Board
Conditions Related to Offsite Emergency
Preparedness

On October 15, 1985, the Federal Emergency Management Agency (FEMA) responded to your request of September 25, 1985, for a status report on two remaining offsite emergency preparedness licensing conditions for the Perry Nuclear Power Plant (NPP). These conditions were identified in the Perry Atomic Safety and Licensing Board's (ASLB) Concluding Partial Initial Decision issued on September 3, 1985. The conditions arose from the following two contentions: (1) there are no letters of agreement regarding the availability of school buses (Contention Q); and, (2) reception centers do not have the means or facilities for handling contaminated property (Contention U).

FEMA's October 15, status report provided you with the information available at that time from the Ohio Disaster Services Agency (ODSA). Where it was noted that certain matters had not been totally resolved, we provided you with the State's anticipated completion date for each of the unresolved conditions.

The State of Ohio recently provided FEMA with all signed letters of agreement (copies attached) from the ten (10) school districts identified in the Ashtabula County Radiological Emergency Response Plan, the eight (8) educational institutions in Geauga County, and the ten (10) school districts participating in Lake County. ODSA has also provided FEMA with written confirmation (copy attached) that 1) as of November 20, 1985, fire departments responsible for the operation of care center monitoring/decontamination stations have completed their radiological monitoring course (16 Hours) and decontamination procedures instruction (4 hours), 2) as of November 22, 1985, personnel and equipment decontamination kits have been delivered to those same fire departments, and 3) all care center monitoring/decontamination stations are equipped and capable of successful operations.

These documents provide assurance that the conditions of contentions Q and U have been satisfactorily addressed.

If you have any questions, please contact Mr. Robert S. Wilkerson, Chief Technological Hazards Division, 646-2860.

Attachments
As Stated

APPENDIX T

REVIEW OF THE EFFECT OF THE JANUARY 31, 1986,
EARTHQUAKE ON THE PERRY NUCLEAR POWER PLANT

**REVIEW OF THE EFFECT OF THE JANUARY 31, 1986
EARTHQUAKE ON THE PERRY NUCLEAR POWER PLANT**

August 1986

Prepared for:

US NRC
Division of Boiling Water Reactor Licensing
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CONTENTS

	<u>Page</u>
1. INTRODUCTION	1-1
1.1 Background	1-1
1.2 Objective and Scope	1-2
2. REVIEW AND ASSESSMENT OF CEI TECHNICAL INVESTIGATIONS	2-1
3. SOIL-STRUCTURE INTERACTION AND STRUCTURE RESPONSE	3-1
3.1 General	3-1
3.2 Response Comparison-Vertically Incident Waves	3-5
3.3 Response Comparison-Non-Vertically Incident Waves	3-10
3.4 Response Comparison--Structural Damping	3-16
4. CHARACTERISTICS OF THE JANUARY 31, 1986 EARTHQUAKE MOTION	4-1
4.1 General	4-1
4.2 Energy, Power, and Duration	4-3
4.3 Response Spectra and Power Spectral Density Functions	4-28
4.4 Nonlinear Response of Single Degree-of-Freedom Systems	4-51
5. SUMMARY AND CONCLUSIONS	5-1
6. REFERENCES	6-1

CONTENTS (CONTINUED)

TABLES

	<u>Page</u>
3-1 Summary of Soil Properties	3-12
4-1 Comparison of Duration Measures of Selected Acceleration Time Histories	4-7
4-2 Comparison of Energy, Power and RMS Acceleration of Selected Acceleration Time Histories	4-8
4-3 Comparison of Duration Measures, Peak Ground Acceleration, Peak Ground Velocity, Energy, Average Power and RMS Acceleration of Selected Input Accelerations	4-9
4-4 Scale Factors to Achieve Ductility Ratios of 1.85 and Corresponding Effective Peak Accelerations for Selected Acceleration Time Histories	4-10

FIGURES

1-1 Acceleration Response Spectra, Perry (1986) Reactor Building Foundation and Containment Vessel	1-4
1-2 Auxiliary Building Recorded Acceleration Response 1-5 Spectra	
3-1 Analytical Model of the Perry Reactor Building by CEI (Ref. 2)	3-4
3-2 Comparison of Calculated and Measured Response on the Containment Vessel - Vertically Propagating Waves	3-7
3-3 Recorded Acceleration Time Histories (ft/sec ²) on the Containment Vessel	3-8
3-4 Calculated Acceleration Time Histories (ft/sec ²) on the Containment Vessel	3-9

3-5	Foundation Input Motions Calculated for Different Angles of Incidence	3-13
3-6	Calculated Response on the Containment Shell for Different Angles of Incidence	3-15
3-7	Comparisons of Calculated Response from Fixed-Base Analyses with Recorded Response at El. 686 ft. on Containment Shell	3-17
4-1	Acceleration Time History, Perry Reactor Building Foundation (Perry, 1986), (ft/sec ²)	
	(a) North-South Component	4-10
	(b) East-West Component	4-11
	(c) Vertical Component	4-12
4-2	Acceleration Time History, Mitchell Lake Road New Brunswick (1982), (ft/sec ²)	
	(a) 18 ^o from North	4-13
	(b) 288 ^o from North	4-14
	(c) Vertical	4-15
4-3	Acceleration Time History, Perry Foundation Design Motion (ft/sec ²)	
	(a) North-South	4-16
	(b) East-West	4-17
	(c) Vertical	4-18
4-4	Cumulative Energy (Equ. 4 - 1) with Time, Perry Reactor Building Foundation (Perry, 1986), (ft ² /sec ³)	
	(a) North-South	4-19
	(b) East-West	4-20
	(c) Vertical	4-21
4-5	Cumulative Energy (Equ. 4 - 1) with Time, Mitchell Lake Road, New Brunswick (1982), (ft ² /sec ³)	
	(a) 18 ^o from North	4-22
	(b) 288 ^o from North	4-23
	(c) Vertical	4-24
4-6	Cumulative Energy (Equ. 4 - 1) with Time, Perry Foundation Design Motion (ft ² /sec ³)	
	(a) North-South	4-25
	(b) East-West	4-26
	(c) Vertical	4-27
4-7	Acceleration Response Spectra, Perry (1986) Reactor Building Foundation Motion	4-30
4-8	Acceleration Response Spectra, Mitchell Lake Road	4-31

4-9	Acceleration Response Spectra, Perry Design Reactor Building Foundation Motion	4-32
4-10	Power Spectral Density Function, Perry (1986)	
	(a) North-South	4-33
	(b) East-West	4-34
	(c) Vertical	4-35
4-11	Power Spectral Density Function, Perry (1986) Normalized to 1. g versus NRC Proposed Target	
	(a) North-South	4-36
	(b) East-West	4-37
	(c) Vertical	4-38
4-12	Power Spectral Density Function, Mitchell Lake Road	
	(a) 18 ⁰ from North	4-39
	(b) 288 ⁰ from North	4-40
	(c) Vertical	4-41
4-13	Power Spectral Density Function, Mitchell Lake Road Normalized to 1. g versus NRC Proposed Target	
	(a) North	4-42
	(b) West	4-43
	(c) Vertical	4-44
4-14	Power Spectral Density Function, Perry Design	
	(a) North-South	4-45
	(b) East-West	4-46
	(c) Vertical	4-47
4-15	Power Spectral Density Function, Perry Design Normalized to 1. g versus NRC Proposed Target	
	(a) North-South	4-48
	(b) East-West	4-49
	(c) Vertical	4-50

1. INTRODUCTION

1.1 Background

At approximately 11:47 a.m. EST on January 31, 1986, an earthquake occurred of magnitude 4.9 M_b . Its epicenter was 11 miles south of the Perry Nuclear Power Plant. The earthquake was felt at Perry and motions were recorded at several locations in the Perry structures.

Three different types of instruments recorded the event at Perry. One type of instrument is the Kinometrics Model SMA-3 strong motion time history recording accelerograph. This instrument records the three orthogonal components of acceleration over the duration of the earthquake. Two of these instruments were installed at Perry -- one on the reactor building foundation mat at approximately elevation 575' and the other is mounted on the containment shell at approximately elevation 686'. Plots of the recorded acceleration time histories and calculated response spectra are shown in Fig. 1.1. These records best describe the characteristics of the earthquake ground motion, i.e., short strong motion duration (less than 1 sec.) and high frequency motion.

The second type of instrument is the Engdahl PSR 1200-H/V response spectrum recorder. This instrument records the response of a series of single degree-of-freedom oscillators to the motion. It generates response spectral ordinates at discrete frequencies, in this case, twelve discrete frequencies ranging from 2 Hz. to 25.4 Hz., for a fixed damping value of 2% of critical. This instrument measures response spectra in three orthogonal directions. Four instruments of this type were used -- two on the auxiliary building foundation mat at an approximate elevation of 568', one on the reactor building foundation mat at an approximate elevation of 575', and one in the reactor building on the drywell platform at an approximate elevation of 630'. Response spectra on the auxiliary building foundation are shown in Fig. 1.2.

Only the North-South component is plotted because selected discrete frequency values were unreadable in the East-West and Vertical components which makes plotting difficult.

The third type of instrument is the Engdahl Par 400 peak accelerograph. It records three orthogonal components of peak local acceleration. Two instruments of this type were used and were located on the auxiliary building foundation mat at an approximate elevation of 568' and on the reactor recirculation pump at an approximate elevation of 605'. A third instrument was out of service.

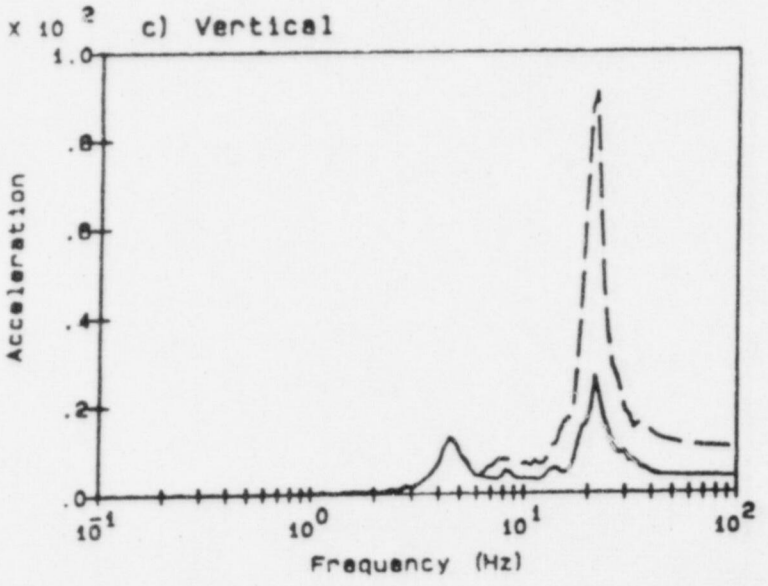
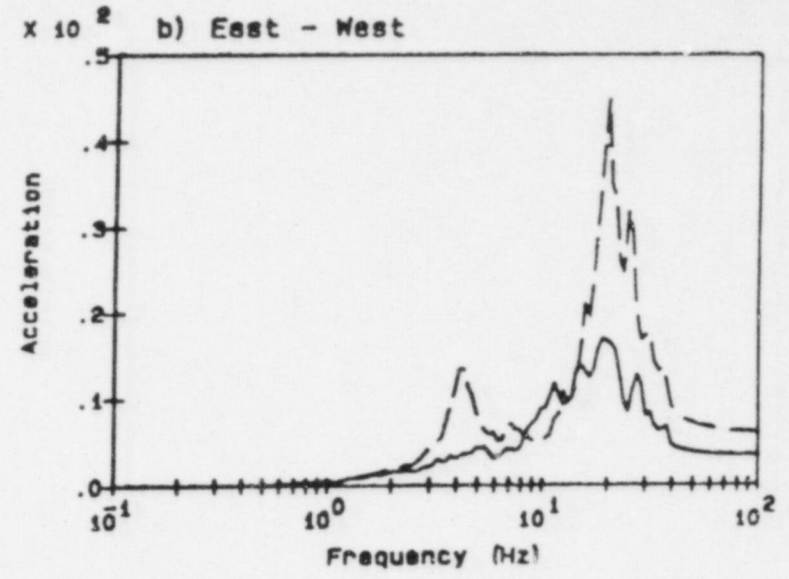
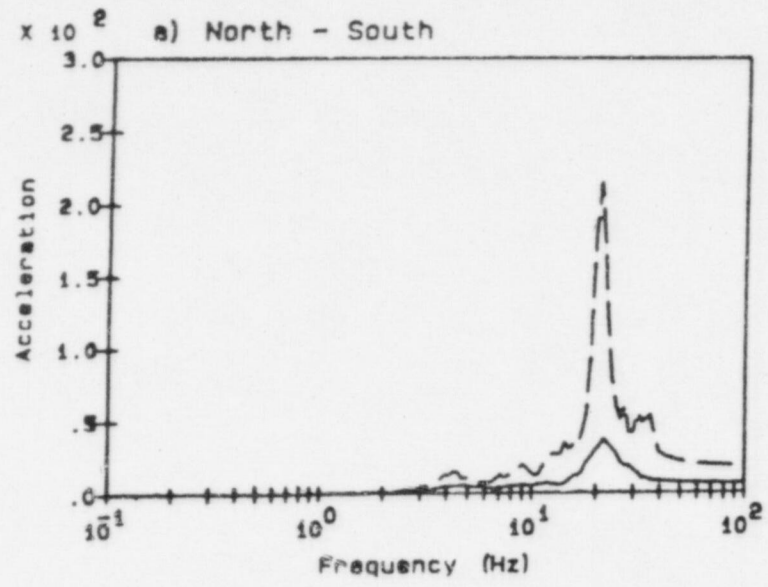
1.2 Objective and Scope

EQE Incorporated of San Ramon, CA was retained to assist Lawrence Livermore National Laboratory (LLNL) and the US Nuclear Regulatory Commission in evaluating the effect of this earthquake on the plant structures and the soil-structure interaction (SSI) and structure response aspects of the event. The scope of this evaluation was three-fold:

- Review submittals by Cleveland Electric Illuminating Co. (CEI) and its representatives and provide comments to LLNL and the US NRC. Meet with CEI personnel, NRC personnel, USGS personnel, and ACRS to discuss these submittals and the results of independent evaluations performed by EQE and discussed in subsequent sections. Perform a plant walkdown. A summary of this effort is contained in Sec. 2.
- Perform independent analyses of the Perry reactor building and compare calculated and measured response. Section 3 discusses the results of this effort.
- Investigate the characteristics of the January 31, 1986 earthquake compared to other recorded motions and the Perry Safe Shutdown Earthquake (SSE). The Fourier

energy, strong motion duration, power spectral densities (PSDs), and effective peak acceleration of the motions were calculated. Section 4 presents the results.

Section 5 presents conclusions of this evaluation.



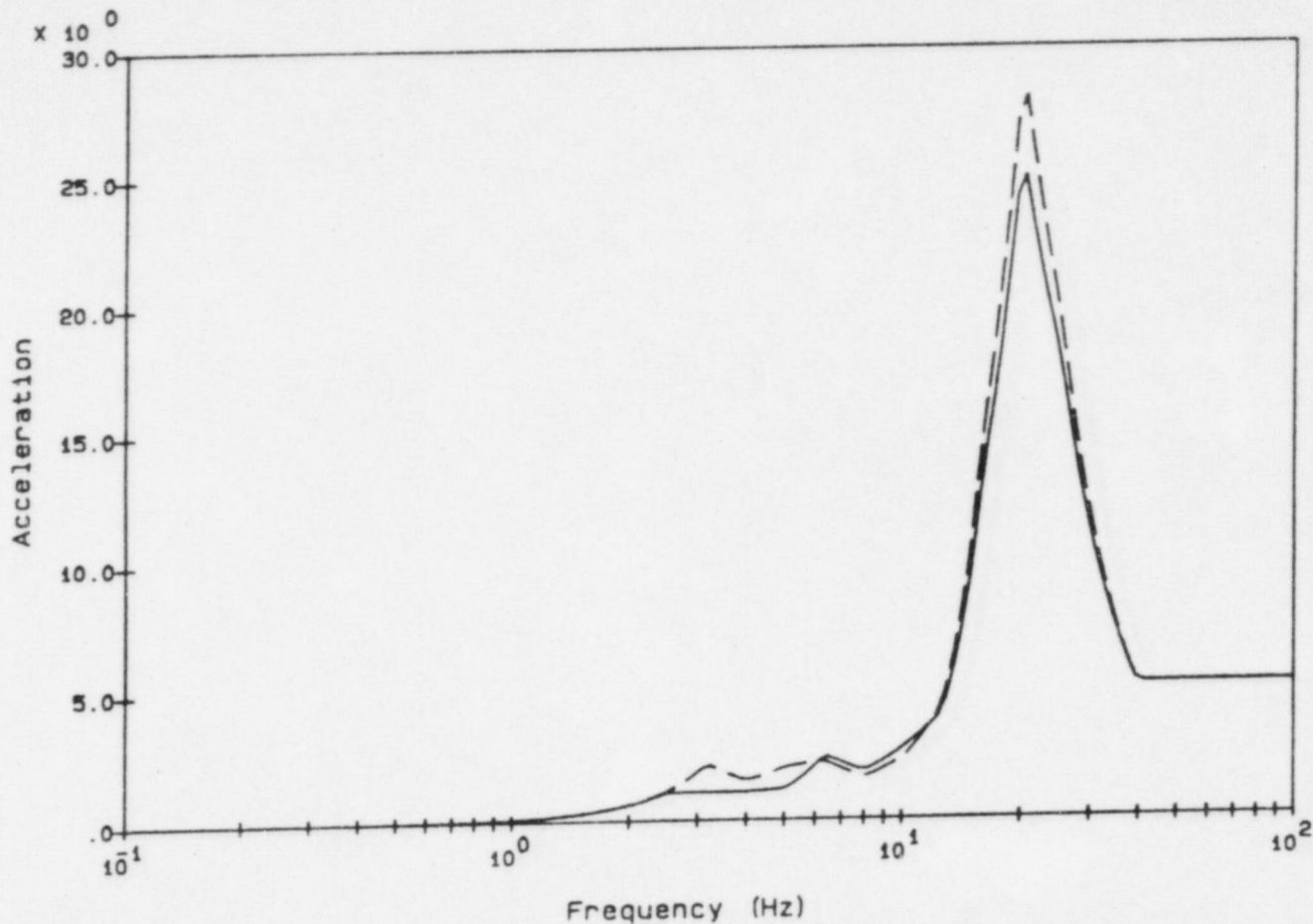
Legend:

- Reactor Bldg fnd _____
- El. 575 ft.
- Containment Vessel - - - - -
- El. 688 ft.

Notes:

All acceleration response spectra values in units of ft/(s²)
 All acceleration response spectra at 2% damping.

Fig. 1.1: Acceleration Response Spectra, Perry (1986) Reactor Building Foundation and Containment Vessel



Legend:

RCIC pump base mat _____
 HPSC pump base mat - - - - -

Notes:

North - South Translation El. 574 ft.
 All accelerations in units of ft/(s*s).
 All spectra calculated at 2% damping.

Fig. 1.2: Auxiliary Building Recorded Acceleration Response Spectra

2. REVIEW AND ASSESSMENT OF CEI TECHNICAL INVESTIGATIONS

Since the occurrence of the earthquake on January 31, 1986, we have attended several meetings with the utility (CEI), its representatives, US NRC, USGS, and the ACRS. In addition, a plant walkdown of the Perry Power Plant was conducted shortly after the earthquake. A summary of the meetings follows:

<u>Location</u>	<u>Date</u>	<u>Comments</u>
Perry Power Plant	Feb. 10-11	Plant Walkdown; open meeting
Washington, D. C.	Feb. 21	NRC Staff/CEI
Washington, D. C.	Mar. 12-13	ACRS/NRC Staff
Washington, D. C.	Apr. 30	NRC Staff/CEI
Washington, D. C.	June 11	NRC Staff/CEI

The utility and its representatives provided technical information in Refs. 1-7, 9-12, and 21 assessing the effect of the January 31, 1986 earthquake on the Perry Nuclear Power Plant structures and equipment. Numerous technical issues were discussed and Ref. 8 itemized outstanding questions and comments for CEI's response. Reference 9 responded to these questions with additional follow-up in Refs. 10 and 11 and telephone conversations.

Our review concentrated on issues related to soil-structure interaction (SSI), structure response, and the effect of the January 31, 1986 earthquake on structure capacity.

The basic purposes of CEI's evaluations were:

- Quantitative assessment of the seismic qualification of a comprehensive sample of equipment types located at various elevations in the Perry Power Plant structures.

This assessment covers equipment qualified by testing and analysis.

- Perform an evaluation of high frequency, short duration earthquake motions with regard to energy content and potential safety significance for structures and equipment at Perry.

Those aspects of these evaluations which relate to SSI, structure response, and structure capacity were reviewed.

Re-Analysis of Perry Structures

To evaluate the seismic qualification of equipment located throughout the Perry structures, the structures were re-analyzed to generate in-structure response spectra for comparison with qualification spectra or for re-analysis purposes. For rock-founded structures, the recorded reactor building foundation time histories were used as input. Fixed-base analyses were performed. Structure damping corresponding to US NRC Regulatory Guide 1.61 was used and justified based on the levels of predicted response in the containment vessel using these values (see Sec. 3 for current study analysis results). The in-structure spectra were smoothed and peak broadened.

For the diesel generator building which is founded on fill, soil-structure interaction analyses were performed by the utility in an identical manner to those performed for design--a finite element approach. For the present re-evaluation, the input motion was assumed to be the time histories recorded on the reactor building foundation, i.e., these motions were assumed to be the free-field ground motion. They were applied assuming they existed in the soil column rather than on a hypothetical rock outcrop which is more appropriate. Due to the soil column characteristics, however, the effect of applying the motion within the soil column is likely to add conservatism to the calculated response. Results from two soil profiles were calculated and presented. Reference 10 reported results using low strain soil properties, whereas,

Ref. 11 reported results using the iterated soil properties from the design SSI analysis. It is likely that the soil properties most compatible with the recorded motions lie between the two sets and closer to the low strain values. The resulting in-structure response spectra differ significantly in frequency content for the two cases. For the low strain shear modulus case, significant amplification near 20 Hz is predicted. Equipment residing in the diesel generator building was evaluated for both cases and margin was demonstrated in both cases (Refs. 10 and 11).

The approach for generating in-structure response spectra for equipment evaluation is acceptable.

Structure Loads

CEI calculated elastic loads in the concrete shield building (Ref. 21) and the containment vessel (Refs. 5 and 10). The stress levels in the containment vessel at the critical section were very low. The force levels in the shield building were significantly less than the design loads. Hence, from an elastic analysis standpoint, the Perry (1986) motions induced very low levels of stress in the structures. Section 4 discusses the inelastic response capacity of the motions for two simple structural representations. The evaluation of load levels in the diesel generator building was not explicitly made since in-structure peak accelerations were below design levels and, hence, would induce lower forces.

The response assessment and load levels determined by CEI are acceptable.

Evaluation of the Effects of High Frequency, Short Duration Earthquakes

CEI evaluated the potential effects of the Perry (1986) recorded motions (and simpler representations) on ductile structures and components. One purpose of this assessment was to investigate the effect of high frequency, short duration motions on components whose elastic frequency is near 20 Hz, the frequency range for which the Perry (1986) recorded

is near 20 Hz, the frequency range for which the Perry (1986) recorded motions exceed the SSE design spectra. The basis of this study is similar to that reported in Sec. 4--a recognition that design criteria for structures and components based on elastic analysis results specifies allowable stresses greater than yield. The behavior of the structure and component during the design level event is expected to be in the nonlinear range locally. Hence, a measure of the severity of the excitation can be its ductility demand on a particular structure or component.

CEI performed two sets of analyses (Ref. 11). One set of analyses with ADINA as described below and a second set comprised of a number of parametric studies but for simpler forms of the excitation. In the ADINA analyses, six single degree-of-freedom models were developed --one for each of the six components of Perry (1986) recorded motions. The approach was to construct the single degree-of-freedom models with frequency corresponding to the frequency of maximum spectral acceleration in the record. The six frequencies were near 20 Hz. An elastic analysis was performed for each of the six models subjected to the SSE design time histories at the corresponding locations. The loads calculated from the design time histories were assumed to be at the allowable value. The yield level was then set at one-half this value--which is representative of several ASME design cases. Nonlinear analyses were then performed on the models. Each model behaved in an elastic-perfectly plastic manner. Ductility demands were then calculated for the Perry (1986) records and the SSE design time histories and compared. The ductility demand for the SSE design time histories exceeded that for the Perry (1986) records. Hence, the SSE design time histories were judged to be more severe than the Perry (1986) records even though the elastic response spectral acceleration of the Perry (1986) records exceeded the design.

A second set of analyses was performed on a similar model but for a simpler idealization of the input motion which permitted numerous parametric studies to be performed. One objective of these analyses was

to assess the effect of a higher amplitude, longer duration high frequency event. The study showed the impact to be relatively small on nonlinear deformation--the measure considered. The effects of preload and other quantities were assessed.

CEI's conclusion was that the effect of the Perry (1986) motions on the response of ductile structures and components is less than the SSE design motion. In addition, the SSE design motion was judged to be more demanding on ductile structures and components than a longer duration, higher amplitude high frequency earthquake.

Although we differ with some elements of the approach, we concur with the assessment of the low damage potential to ductile structures and components of the Perry (1986) records and similar high frequency, short duration earthquakes.

3. SOIL-STRUCTURE INTERACTION AND STRUCTURE RESPONSE

3.1 General

The characteristics of the recorded motions on the foundations of the Perry reactor building and auxiliary building were shown in Figs. 1.1 and 1.2. They are judged to be similar in frequency content to the free-field ground motion, which was not recorded. The phenomenon which could lead to different foundation motion compared to the free-field is soil-structure interaction (SSI) and structure response. SSI can be conceptually separated into kinematic interaction and inertial interaction. Kinematic interaction is the phenomenon associated with wave scattering at the interface between the soil and the foundation/structure. Kinematic interaction leads to a different effective excitation of the structure from the free-field motion except when the foundation/structure are surface-founded and the wave propagation mechanism of the free-field motion is vertically incident waves. In all other cases, the foundation input motion differs from the free-field ground motion. Inertial interaction is the portion of SSI associated with the dynamic response of the combined soil-structure system when subjected to the foundation input motion.

All category I structures except the diesel generator building and the off-gas building are founded on very stiff rock (Chagrin shale with a shear wave velocity of 4900 ft/sec) or fill concrete of similar shear wave velocity. Even though the reactor building and auxiliary building foundations are founded approximately 50 ft. below grade, they can be treated as surface-founded at this elevation from an SSI standpoint because little or no side soil provides constraint or excitation due to the presence of adjacent structures. Hence, embedment is not considered. The very high stiffness of the rock is generally thought to preclude significant inertial interaction effects. Also, comparing the recorded motions on the reactor building and auxiliary building

foundations, shows them to be very similar. If significant inertial interaction would have occurred, the two foundation motions should differ because the structures have vastly different dynamic characteristics. Hence, it is judged that inertial interaction was not an important phenomenon. Potential kinematic interaction effects due to wave passage, i.e., non-vertically incident waves, were investigated in the present study for the reactor building.

The seismic design analysis of the Perry category I structures involved developing mathematical models of their dynamic behavior and analyzing them for the design ground motion. The ability of these models to predict response from the January 31, 1986 earthquake was investigated. Linear analysis methods, as described below, were used to calculate response on the containment vessel at the location of the recorded motion. Calculated and recorded motions were compared. The motions recorded on the reactor building foundation were the excitation. Several analyses were performed to investigate the effects of various parameters on containment vessel response.

The analyses proceeded by obtaining CEI's reactor building dynamic model (Fig. 3.1) (Ref. 2) and implementing it on the LLNL computer system. During this effort, the structure model was modified in three ways. First, the soil springs located at the base of the model were deleted so that a fixed-base model was obtained. Second, the model of the upper portion of the containment vessel was modified to treat the crane as it was positioned during the earthquake. The mass of node 11 (elev. 725') was reduced and an additional node having the mass of the polar crane was offset 50 ft. south of the containment vessel centerline and rigidly linked to node 11. This modeled the eccentric mass effects of the crane as parked during the event. Dynamically, it couples torsion and east-west translation and increases rotational inertia. Third, a massless node was added at the actual location of the containment vessel instrument and rigidly linked to node 13 which allows one to include the effects of torsion and rocking on the predicted response.

All structure response calculations were performed with the computer program CLASSI (Ref. 12).

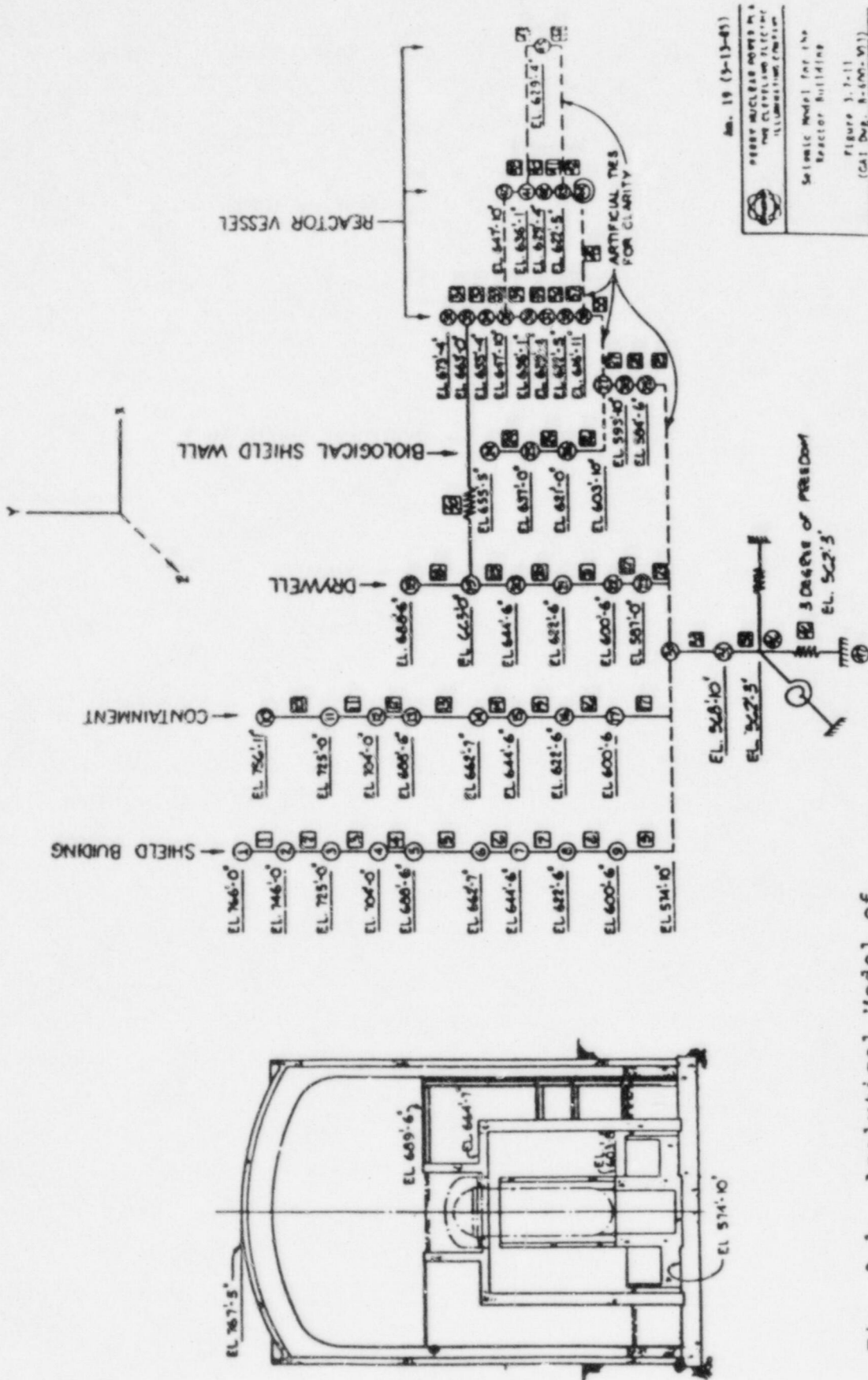


Fig. 3.1: Analytical Model of the Perry Reactor Building by CEI (Ref.2)

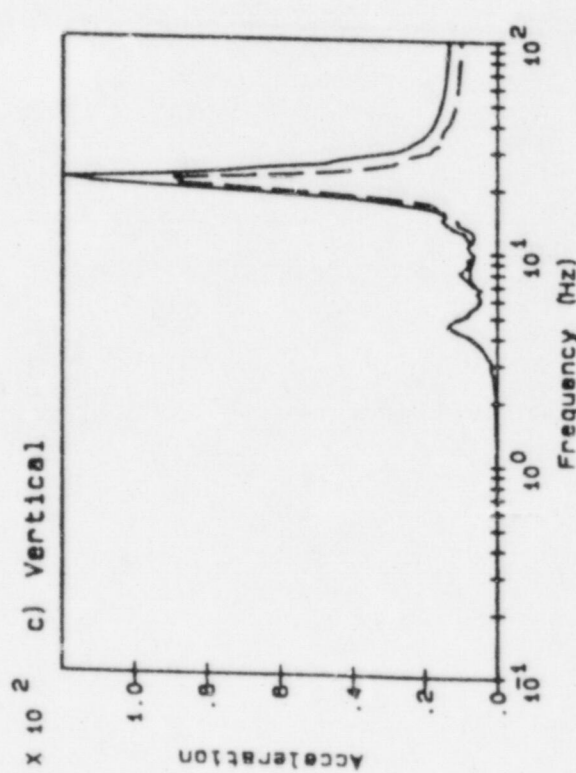
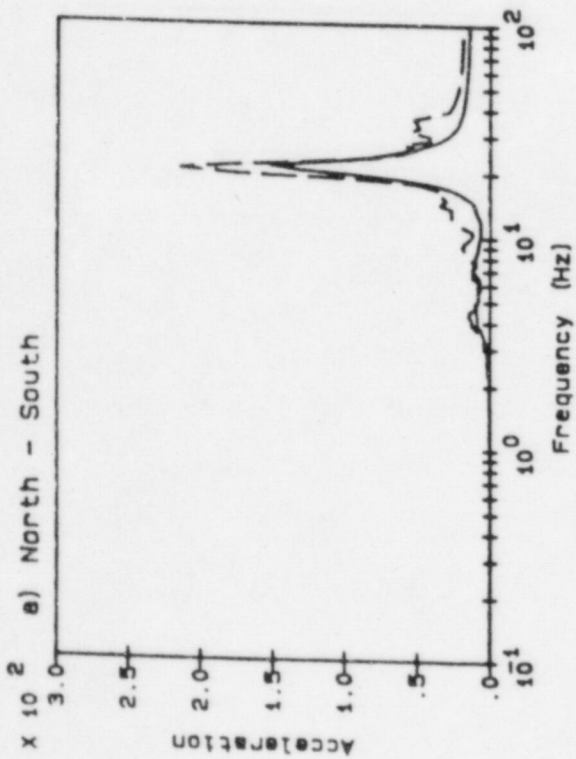
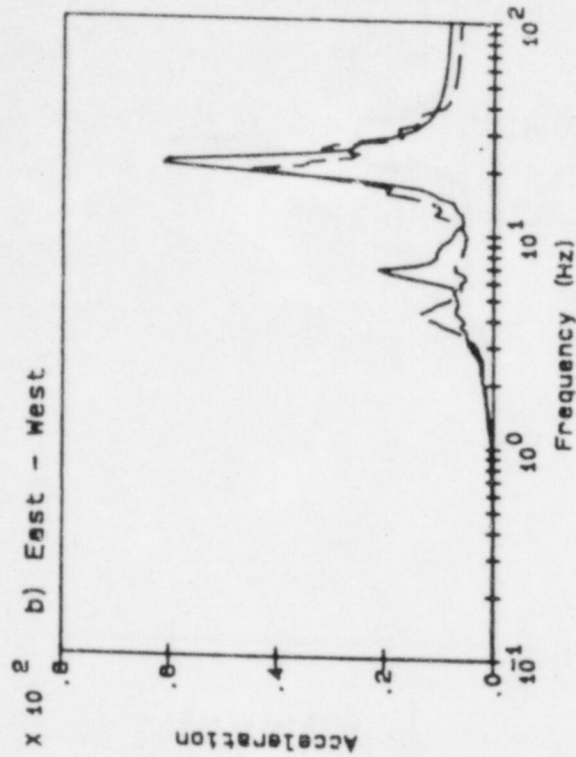
3.2 Response Comparison-Vertically Incident Waves

Figure 3.2 compares 2% damped response spectra, calculated and measured, on the containment vessel at elevation 688'. Structural damping was a constant 4% for all modes. Vertically incident waves were the wave propagation mechanism. Structural damping and non-vertically incident waves were the subject of sensitivity studies reported in subsequent sections. The fully modified structural model was used in this analysis.

The response spectra comparison can be separated into three portions: low frequency response (less than 10 Hz), response near 20 Hz, and zero period acceleration (ZPA). For the N-S component, there is no specific low frequency peak. Near 20 Hz and at the ZPA, the responses are under-predicted by 20-30% by the analysis. For the E-W component, the measured response has a low frequency peak at about 4 Hz which is not reproduced in the analysis. The analysis amplifies motion near 7 Hz. Examination of the E-W modes in this frequency range and other sensitivity studies show both peaks to be a function of the input motion. Near 20 Hz and at the ZPA, the calculated responses over-predict the measured values by about 35%. For the vertical component, the low frequency behavior matches well in amplitude and frequency content. Near 20 Hz and at the ZPA, the calculated responses over-predict the amplitudes of response by about 30%.

An inspection of the calculated and measured acceleration time histories can be done from Figs. 3.3 and 3.4 where the time histories are shown on an expanded time scale. (Note, Fig. 3.3 differs in scale from Fig. 3.4 and between Fig. 3.3 (a), (b), and (c)). The strong motion portion of the recorded time histories is greater than that of the calculated motions. Also, a beating-type phenomenon is observed in the N-S and vertical components which led CEI to hypothesize that a portion of this motion was induced by secondary causes, such as the polar crane (Refs. 4

and 11). This remains somewhat of an open issue relative to the predictive capability of the dynamic models. For evaluation purposes, CEI used the as recorded motion at elevation 688' to assess equipment qualification at this location.



Legend:

E1. 688.5 ft.

— Calculated

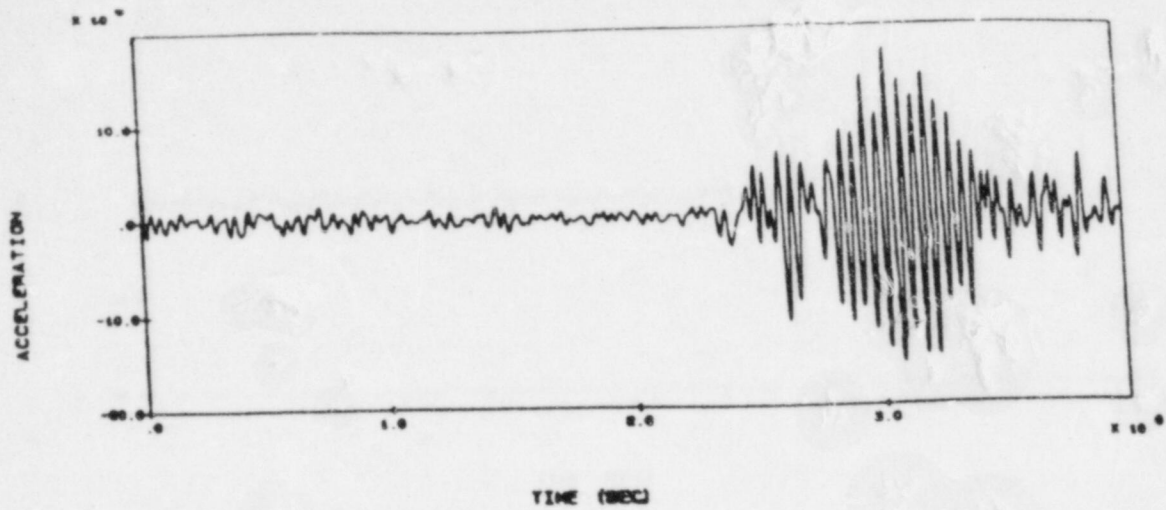
- - - Measured

Notes:

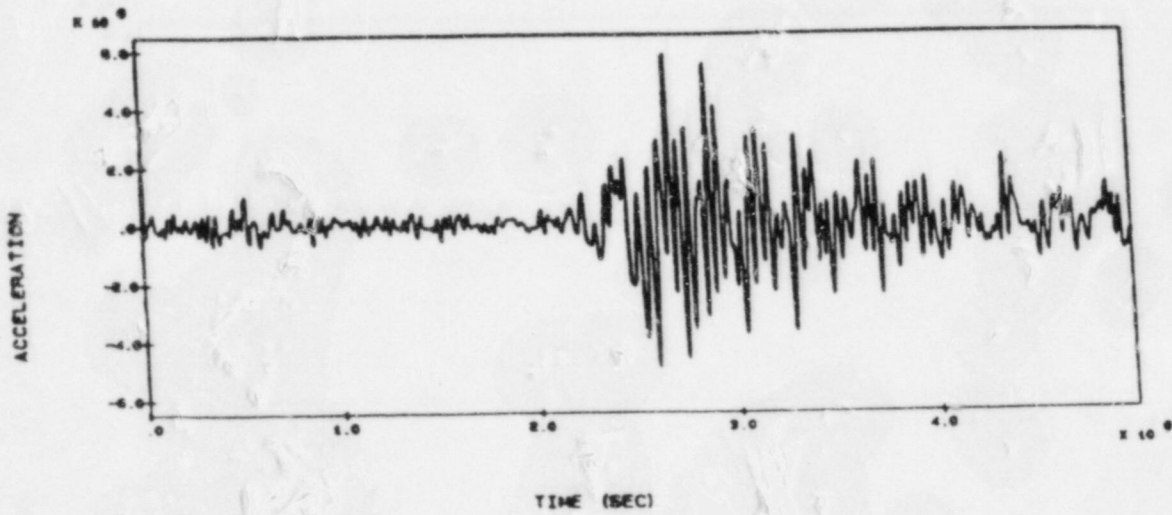
All accelerations in units of ft/(s²)

All spectra calculated at 2% damping.

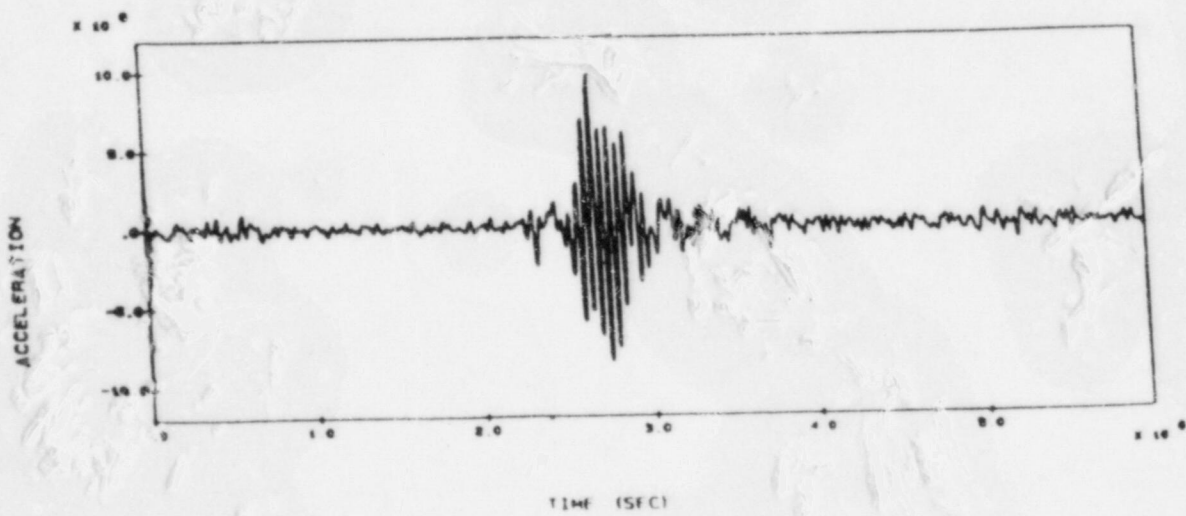
Fig. 3.2: Comparison of Calculated and Measured Response on the Containment Vessel - Vertically Propagating Waves



(a) North-South



(b) East-West



(c) Vertical

Fig. 3.3 Recorded Acceleration Time Histories (ft/sec²) on the containment vessel.

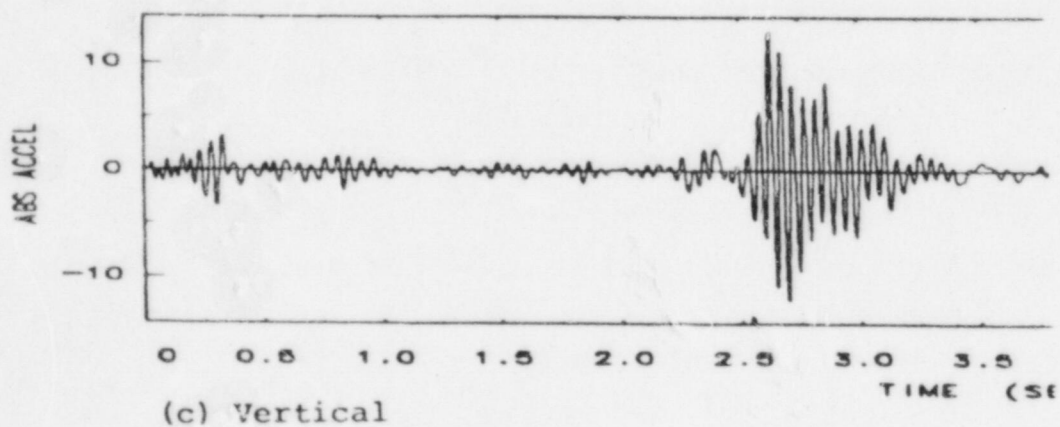
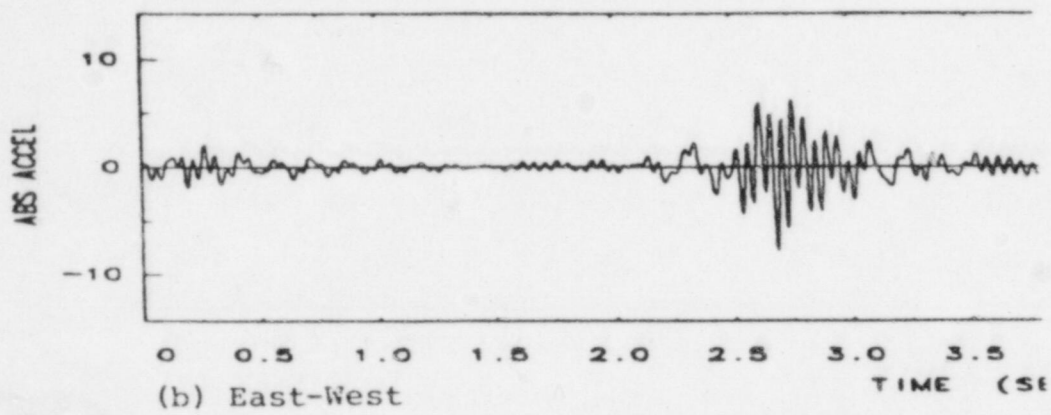
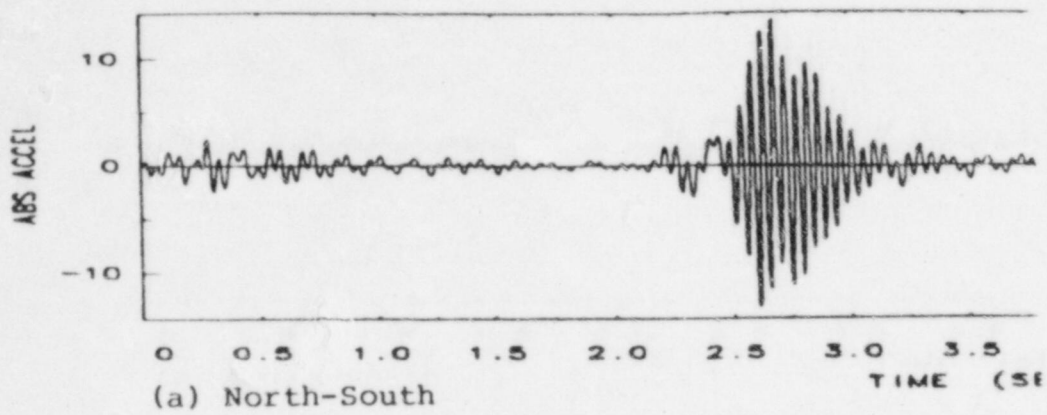


Fig. 3.4 Calculated Acceleration Time Histories (ft/sec²) on the Containment Vessel

3.3 Response Comparison- Non-Vertically Incident Waves

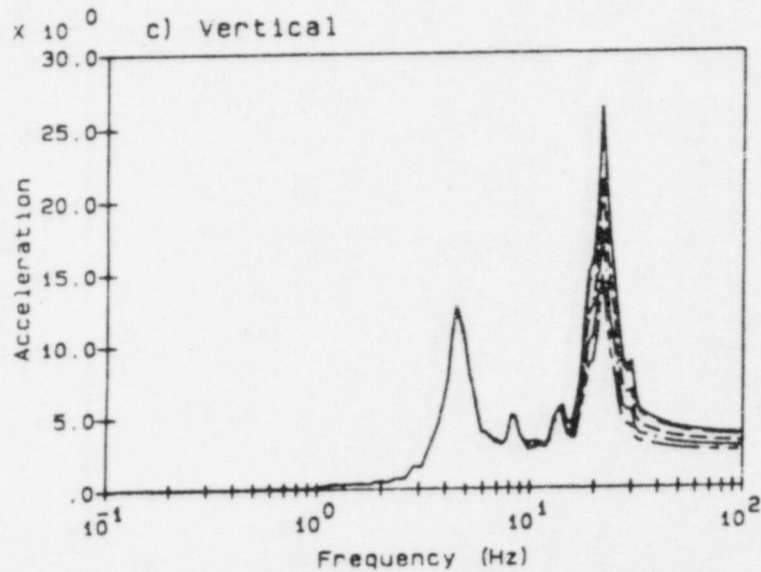
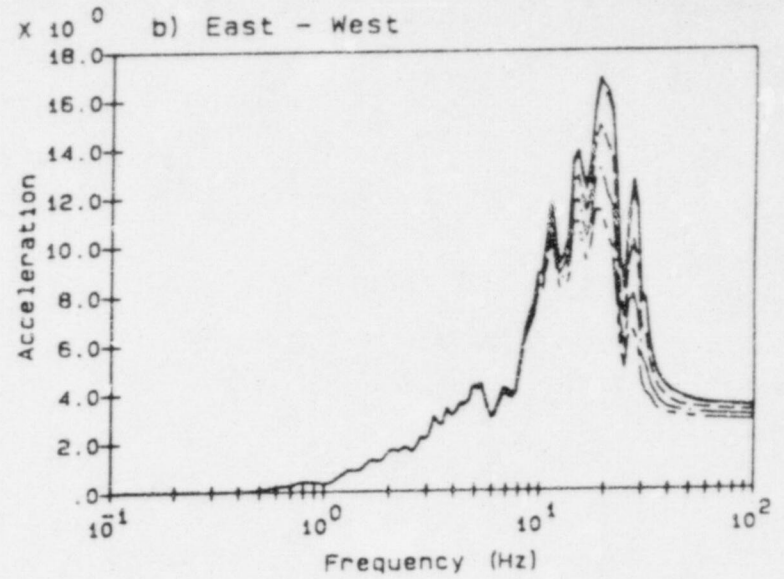
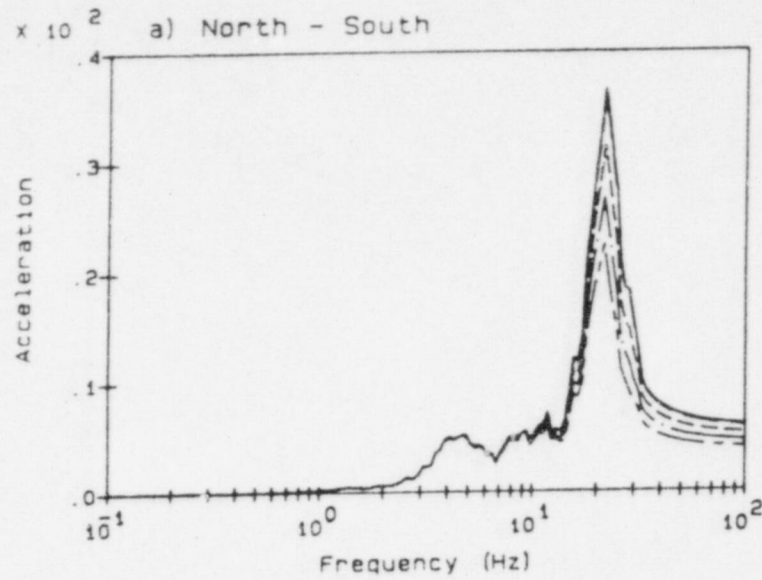
In an attempt to understand the causes of the discrepancies in measured and calculated response discussed in Sec. 3.2, the effect on response of non-vertically incident waves was assessed. In particular, we sought to investigate the difference in amplitude of the near 20 Hz spectral accelerations and the low frequency behavior in the E-W component. We considered these discrepancies to possibly be due to unrecorded foundation rocking and torsion. The effect on response of non-vertically propagating waves with their induced rocking and torsion was assessed. Two cases were considered--incoming waves from the south with varying angles of incidence and incoming waves from the east with varying angles of incidence. Only the former case is reported here. The latter provided no new information.

The effect of non-vertically incident waves generally is to reduce the horizontal foundation motion in the direction of wave propagation, increase rocking in this direction and induce torsional motion. We studied this effect on the response on the containment vessel for non-vertically incident waves of 10° , 30° , 45° , and 60° off the vertical axis and originating from the south. The method used was approximate in that foundation input motions were calculated from the recorded foundation translations assuming only their phase changed as they traversed the foundation. The foundation was assumed to be located on the surface of a uniform half space having the properties of the Chagrin shale layer directly beneath the reactor building foundation. Table 3.1 summarizes these properties. The resulting foundation input motions were used in fixed-base analyses of the fully modified structural model using 4% damping in the structure. Response spectra at 2% damping are shown in Figs. 3.5 and 3.6. Also included in these results are those for the vertically propagating wave case. Figure 3.5 shows the calculated foundation input motions. One can see from the plots that above 10 Hz the foundation translations decrease with increasing inclination angle. One can also see that rotations increase

significantly with increasing angle. Figure 3.6 shows the calculated response on the containment vessel at the location of the instrument. For all components the spectral peak near 20 Hz decreases with increasing inclination angle. Response in the N-S direction increases at 4 Hz with increasing angle, but E-W response is unaffected below about 7 Hz. Comparison of these spectra with those of the recorded motions shows little improvement in agreement for non-vertically incident waves. Thus, it does appear that vertically incident waves lead to the best estimate of containment vessel response; especially the near 20 Hz spectral acceleration.

Table 3.1 Summary of Soil Properties

Compression Wave Velocity	10400 ft/sec
Shear Wave Velocity	4900 ft/sec
Unit Weight	152 lbs/ft ³
Poisson's Ratio	0.36
Material Damping Ratio	0.01



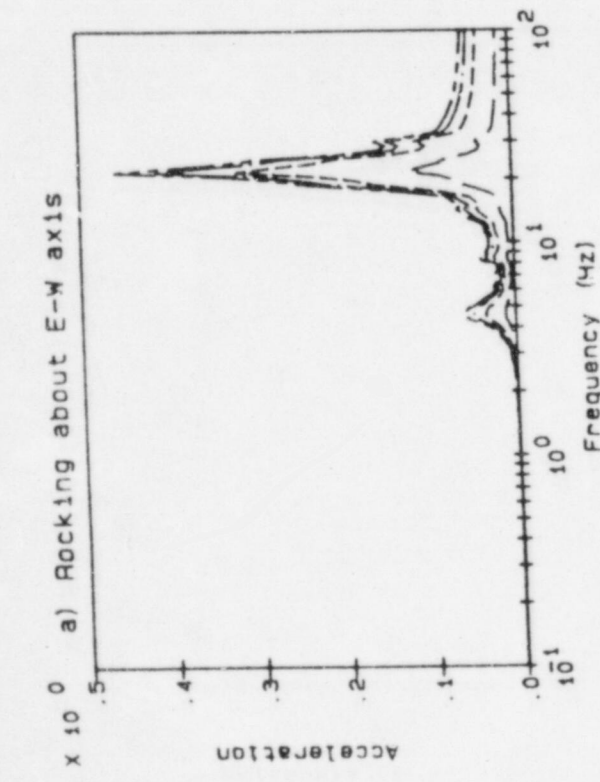
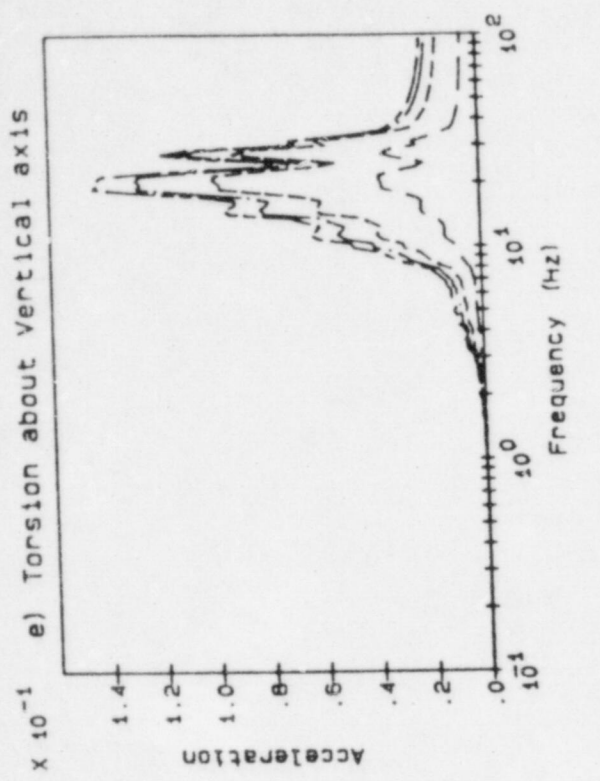
Legend:

- Angle = 0° —————
- Angle = 10° - - - - -
- Angle = 30° - . - . -
- Angle = 45° —————
- Angle = 60° - - - - -

Notes:

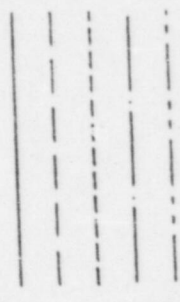
All accelerations in units of ft/(s*s)
 All spectra calculated at 2% damping.

Fig. 3.5: Foundation Input Motions Calculated for Different Angles of Incidence - Translational Components



Legend:

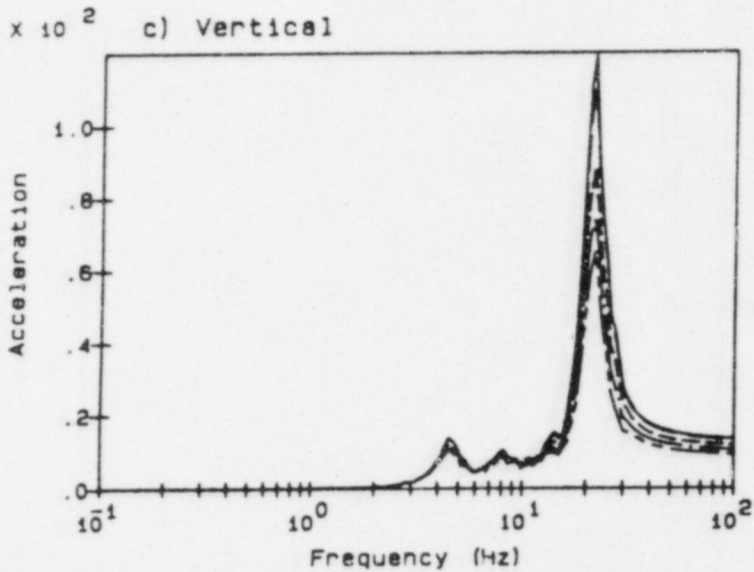
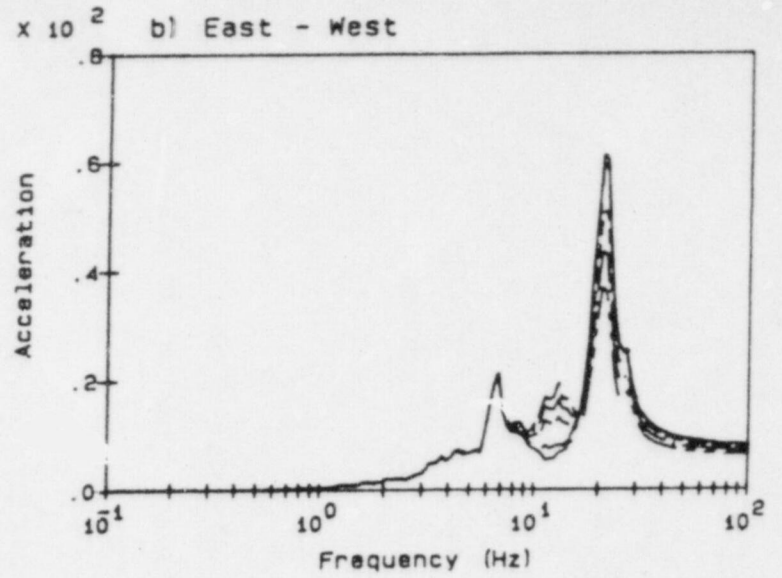
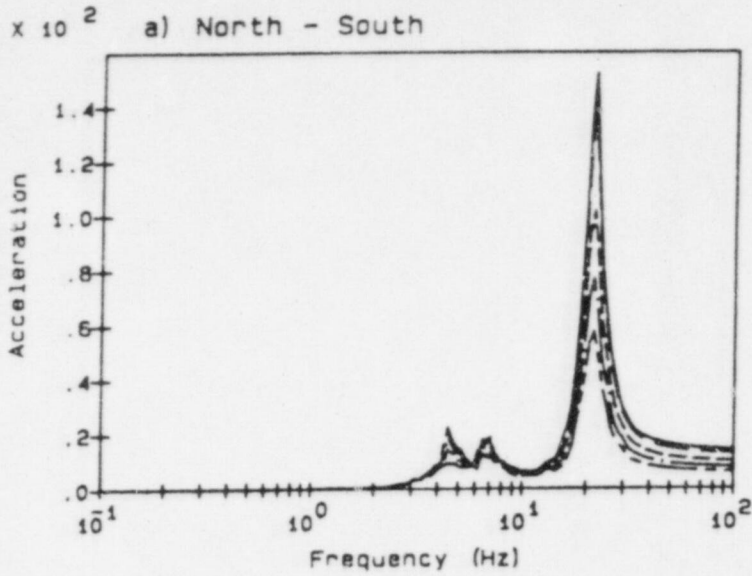
- Angle = 0°
- Angle = 10°
- Angle = 30°
- Angle = 45°
- Angle = 60°



Notes:

- All accelerations in units of rad/(s²)
- All spectra calculated at 2% damping.

Fig. 3.5: Foundation Input Motions Calculated for Different Angles of Incidence - Rotational Components



Legend:

- Angle = 0° —————
- Angle = 10° - - - - -
- Angle = 30° - - - - -
- Angle = 45° —————
- Angle = 60° - - - - -

Notes:

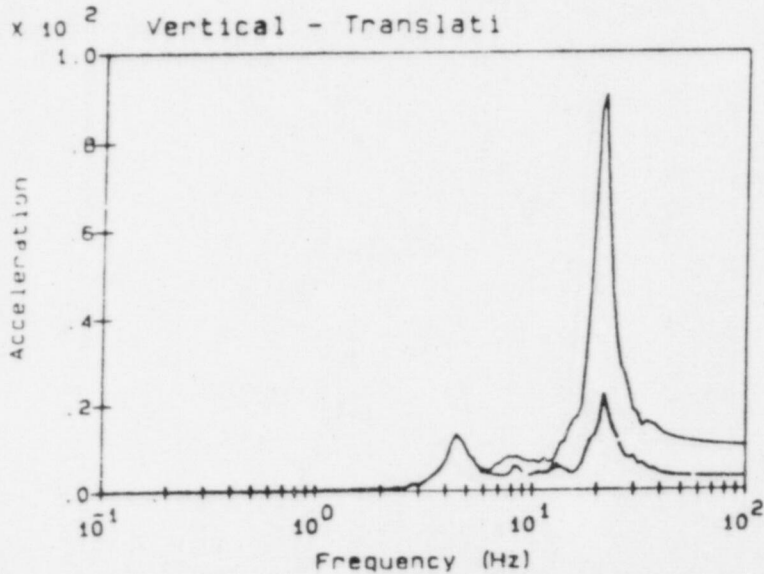
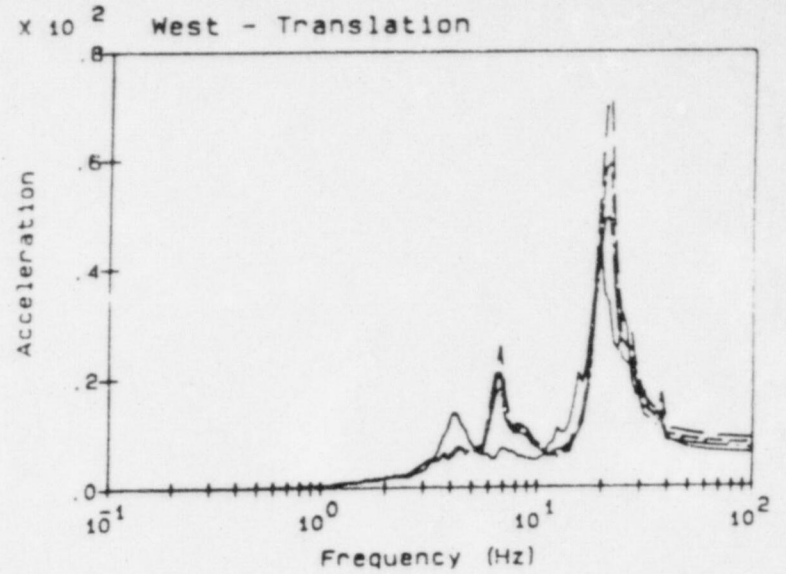
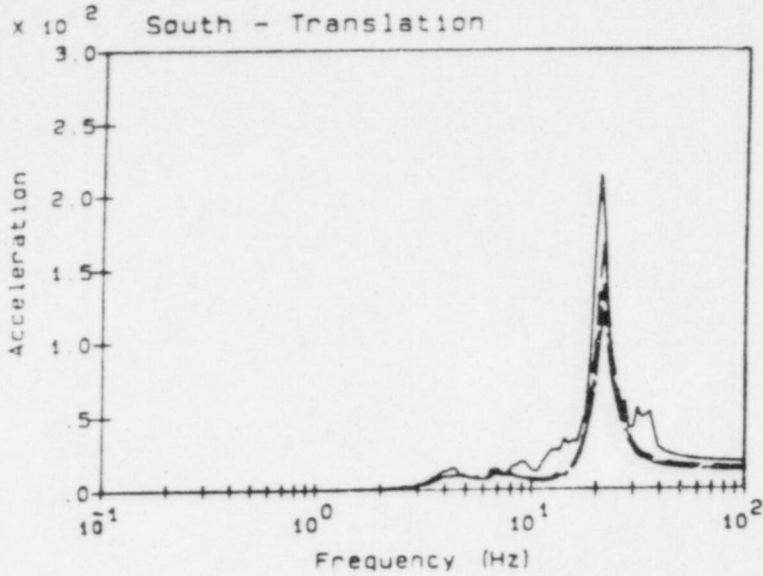
All accelerations in units of ft/(s*s)
 All spectra calculated at 2% damping.

Fig. 3.6: Calculated Response on the Containment Shell for Different Angles of Incidence

3.4 Response Comparison--Structural Damping

Early in the present investigation, we studied the effect of structural damping on response of the containment vessel. The only structural model modification incorporated in this study was considering a fixed-base structural model. Hence, the responses reported here are on the centerline of the containment vessel and do not include the crane eccentricity or a rigid link to the instrument location on the shell itself. Quantitatively, the effect of damping on response can be observed from Fig. 3.7. Of note, however, is that the 3% damped case does not increase containment vessel response near 20 Hz enough to match recorded motion in the N-S direction. In the E-W direction, the 5% damped case still slightly over-predicts response. Finally, vertical response at the shell centerline is independent of structural damping. Comparing vertical response of Fig. 3.2 and Fig. 3.7 shows the importance of rocking at elevation 688' on the vertical component.

The damping parameter variations investigated here did not produce changes in response large enough to permit predicted response to match calculated response near 20 Hz or at the ZPA.



Legend:

- El. 686 ft Measured —————
- El. 688.5 ft
Calculated —————
- 3% structure damp - - - - -
- 4% structure damp - - - - -
- 5% structure damp - - - - -

Notes:

All accelerations in units of ft/sec².
All spectra calculated at 2% damping.

Fig. 3.7 Comparisons of Calculated Response from Fixed-Base Analyses with Recorded Response at El. 686 ft. on Containment Shell

4. CHARACTERISTICS OF THE JANUARY 31, 1986 EARTHQUAKE MOTION

4.1 General

There is a vast literature which documents the low damage potential of earthquakes of short duration and high frequencies. One of the most recent was that of Kennedy et al. (Ref. 13). In their study, Kennedy et al. investigated several characteristics of recorded earthquake time histories--Fourier energy, strong motion duration, root-mean-square (RMS) acceleration, response spectra, effective peak acceleration, etc. Eleven recorded earthquake ground motions were investigated. The eleven were divided into two groups. Group 1 was comprised of records denoted Taft (1957), Olympia (1949), El Centro #12 (1979), El Centro #5 (1979), Pacoima Dam (1971), and Hollywood Storage PE Lot (1971). Group 2 was comprised of records denoted Coyote Lake (1979), Parkfield Cholame #2 (1966), Gavilan College (1974), Goleta (1978), and Melendy Ranch (1972). The precise identification of location and components is contained in Ref. 13. The earthquake records comprising Group 1 have strong motion durations greater than or equal to 3.4 seconds, are from earthquakes with local magnitudes of 6.4 and greater, and are rich in frequency content from at least 1.2 Hz to 5.5 Hz. The Group 2 records represented different earthquake characteristics. Their strong motion durations were less than or equal to 3 seconds. They are from earthquakes of local magnitude 5.7 or less. They have much narrower frequency content than the Group 1 records. One purpose of the Ref. 13 study was to investigate differences in identifying characteristics and damage potential of the Groups 1 and 2 earthquake records. In addition, an artificial acceleration time history whose response spectra approximately matched the US NRC Regulatory Guide 1.60 design ground response spectra was included. The total data set was 12 records.

The present investigation of the January 31, 1986 Ohio earthquake recorded motions proceeded in a similar fashion to Ref. 13. The recorded acceleration time histories on the reactor building foundation

were used and for brevity are denoted Perry (1986) earthquake motions. Neither the Group 1 nor the Group 2 data sets match the characteristics of the Perry records which are richer in high frequencies (near 20 Hz) and of shorter strong motion duration. The Gavilan College, Hollister, 1974 record and the Melendy Ranch Barn, Bear Valley, 1972 record of Group 2 are the most similar in terms of having a short strong motion duration and possessing somewhat higher frequency content (but still less than 10 Hz). The two horizontal components and the vertical component recorded at Mitchell Lake Road during the March 31, 1982 New Brunswick earthquake were also considered. Both sets of data are characterized by short strong motion durations and rich high frequency content, i.e. greater than or equal to 20 Hz. In one sense, these records can be considered as comprising a Group 3. Finally, the Perry Power Plant design time histories calculated on the reactor building foundation were considered.

4.2 Energy, Power, and Duration

This investigation of the Perry (1986) records parallels that of Ref. 13. This permits one to make a direct comparison of the characteristics of the Perry (1986) records, the Mitchell Lake Road, New Brunswick (1982) records, and the Perry design motions with those of the Ref. 13 data set.

Energy

The total energy E_m (or more properly the Fourier energy) of the records is defined as:

$$E(T) = \int_{t_0}^{t_0 + T_D} a^2(t) dt \quad (4-1)$$

where $E(T)$ is a measure of the total energy between times t_0 and $t_0 + T_D$, $a(t)$ is the instrumental acceleration at time t , and T_D is the duration of strong motion. This approach corresponds to Ref. 13 and this definition of $E(T)$ is attributed to Arias (Ref. 14) and Housner (Ref. 15). Cumulative energy plots as a function of time are shown later in this section.

Power

The average rate of energy input (earthquake power) is given by:

$$P = \frac{E(T)}{T_D} \quad (4-2)$$

and the root-mean-square (RMS) acceleration is defined to be:

$$a_{RMS} = \sqrt{P} \quad (4-3)$$

Duration

Energy, power and the RMS acceleration are strongly influenced by the strong motion duration T_D . In the past, many definitions of strong motion duration have been used. Two measures of strong motion duration are reported here -- two of several reported in Ref. 13.

The most common definition of strong motion duration is due to Trifunac and Brady (Ref. 16). By this definition, the strong motion duration is defined as:

$$T_D = T_{0.95} - T_{0.05} \quad (4-4)$$

where $T_{0.95}$ represents the time at which $E(T)/E_m = 0.95$ and $T_{0.05}$ represents the time at which $E(T)/E_m = 0.05$. This duration includes 90% of the energy.

A second measure of strong motion duration is presented in Ref. 13 and it is argued that it should be used when evaluating stiff structures such as those of nuclear power plants. The definition is:

$$T'_D = T_M - T_{0.05} \quad (4-5)$$

where

$$T_M = \text{Max} \left\{ \begin{array}{l} T_{0.75} \\ T_{PA} \end{array} \right\} \quad (4-6)$$

and $T_{0.75}$ represents the time at which $E(T)/E_m = 0.75$. T_{PA} represents the time associated with the first crossing of the accelerogram following the maximum positive or negative acceleration, whichever occurs later in time.

The power for this definition of strong motion duration (T'_D), can be defined by:

$$P = \frac{\Delta E}{T'_D} \quad (4-7)$$

where ΔE represents the cumulative energy between time $T_{0.05}$ and T_M . ΔE is equal to 70% of E_m except when T_M exceeds $T_{0.75}$. For that case, ΔE is greater than 70% of E_m .

Results

Figures 4.1, 4.2, and 4.3 contain plots of the acceleration time histories for Perry (1986), Mitchell Lake Road, New Brunswick (1982), and the Perry foundation design motions. Figures 4.4, 4.5, and 4.6 contain plots of cumulative energy as a function of time, again for the three sets of earthquake records, respectively. The energy, duration, power, and RMS accelerations for these records were calculated and are presented in Tables 4.1 and 4.2. For comparison purposes, Table 2-3 of Ref. 13 has been reproduced as Table 4.3.

Consider the time history plots of the Perry (1986) records shown in Fig. 4.1. Note, the two distinct phases of motion in the time histories. These two phases lead to estimates of strong motion duration T_D and T'_D which are slightly misleading. The initial phase of the motion contributes greater than 5% of the total energy; hence, estimates of strong motion duration begin in this phase and extend into the second phase when 75% or 95% of the total energy is achieved. The strong motion duration estimates T'_D for the Perry (1986) records are 2.38 sec., and 2.54 sec., and 2.40 sec. for the N-S, E-W, and vertical components.

The cumulative energy plots of Figs. 4.4, 4.5, and 4.6 show the concentration of energy over a short time period for the Perry (1986) records and the Mitchell Lake Road, New Brunswick (1982) records compared to the Perry foundation design records. In the latter case the rate of increase in energy over the time duration of the records is relatively uniform.

A comparison of the energy in the Perry (1986) records with the energy in the records: considered in Ref. 13; the Mitchell Lake Road, New

Brunswick (1982) records; and the Perry foundation design motions shows the Perry (1986) records to have less energy than any of the records considered. Comparing the Perry (1986) records to the foundation design motions shows the Perry (1986) records to have 3.25%, 1.75%, and 1.94% of the energy in the design motions for N-S, E-W, and vertical components.

The Mitchell Lake Road, New Brunswick (1982) records were included here to provide another example of recorded motions with significant high frequency content. From Tables 4.1 and 4.2, one observes that estimates of strong motion duration for the three components are less than 1 sec. for T'_D and less than 1.5 sec for T_D . The instrumental peak acceleration, the energy, and the RMS acceleration in the vertical component are substantially higher than those of the two horizontal components.

Table 4.1

Comparison of Duration Measures of Selected Acceleration Time Histories

Earthquake Record	Peak Accel (g)	T _{.05} (sec)	T _{.75} (sec)	T _{.95} (sec)	T _{PA} (sec)	T _D (sec)	T' _D (sec)
Perry, 1986 (N - S)	.18	0.24	2.62	5.85	2.55	5.61	2.38
Perry, 1986 (E - W)	.103	0.15	2.69	10.13	2.42	9.98	2.54
Perry, 1986 (Vertical)	.105	0.47	2.87	5.15	2.61	4.68	2.40
Mitchell Lake Rd New Brunswick, 1982 (18° from N)	.15	0.16	0.99	1.56	0.53	1.40	0.82
Mitchell Lake Rd New Brunswick 1982 (288° from N)	.24	0.35	1.03	1.41	0.82	1.06	0.68
Mitchell Lake Rd New Brunswick 1982 (Vert)	.58	0.10	0.78	1.23	0.35	1.13	0.68
Perry Foundation Design Motion (N - S)	.18	2.08	15.68	19.27	16.94	17.19	16.94
Perry Foundation Design Motion (E - W)	.17	2.32	16.06	19.30	12.72	16.98	13.74
Perry Foundation Design Motion (Vertical)	.19	2.90	15.52	19.07	10.20	16.17	12.62

Table 4.2
Comparison of Energy, Power and RMS Acceleration
of Selected Acceleration Time Histories

Earthquake Record	Peak Accel (g)	T'_D (sec)	ΔE (ft ² /sec ³)	P (g ² x 10 ⁻³)	RMS Accel (g)
Perry, 1986 (N - S)	.18	2.38	1.67	.678	.026
Perry, 1986 (E - W)	.103	2.54	.96	.366	.019
Perry, 1986 (Vertical)	.105	2.40	1.10	.443	.021
Mitchell Lake Rd New Brunswick 1982 (18° from N)	.15	0.82	2.90	3.39	.058
Mitchell Lake Rd New Brunswick 1982 (228°-from N)	.24	0.68	5.26	7.46	.086
Mitchell Lake Rd New Brunswick 1982 (Vert)	.58	0.68	15.3	21.7	.147
Perry Foundation Design Motion (N - S)	.18	16.94	51.45	3.65	.060
Perry Foundation Design Motion (E - W)	.17	13.74	54.9	3.85	.062
Perry Foundation Design Motion (Vertical)	.19	12.62	56.8	4.34	.066

Table 4.3

Comparison of Duration Measures, Peak Ground Acceleration,
Peak Ground Velocity, Energy, Average Power and RMS
Acceleration of Selected Input Accelerations
(Table 2-3, Ref. 13)

Earthquake Record (Component)		T'_D (sec)	T_D (sec)	a (g)	v (in/sec)	ΔE (ft ² /sec ³)	P (g ² x10 ⁻³)	RMS Acc. a_{rms} (g)
1	Olympia, WA., 1949 (N86E)	15.6	17.3	0.281	6.7	64.2	3.97	.063
2	Taft, Kern Co., 1952 (S69E)	10.3	28.1	0.180	7.0	27.4	2.57	.051
3	El Centro Array No. 12 Imperial Valley, 1979, (140)	9.6	18.6	0.142	6.9	18.6	1.88	.043
4	Artificial (R.G. 1.60)	9.4	13.0	0.200	11.3	44.2	4.54	.067
5	Pacoima Dam San Fernando, 1971 (S14W)	6.1	7.4	1.170	44.6	466.8	74.0	.272
6	Hollywood Storage PE Lot, San Fernando, 1971 (N90E)	5.4	11.7	0.211	8.3	30.0	5.37	.073
7	El Centro Array No. 5 Imperial Valley, 1979, (140)	3.4	8.2	0.530	17.3	78.1	22.2	.149
8	UCSB Goleta Santa Barbara, 1978 (180)	3.0	9.7	0.347	15.7	57.3	18.5	.136
9	Gilroy Array No. 2, Coyote Lake, 1979, (050)	2.2	7.5	0.191	4.0	13.3	5.86	.077
10	Cholame Array No. 2, Parkfield 1966 (N65E)	1.4	9.2	0.490	10.4	86.1	59.4	.244
11	Gavilan College Hollister, 1974 (S67W)	1.1	1.6	0.138	1.6	2.1	1.80	.042
12	Melendy Ranch Barn, Bear Valley 1972 (N29W)	0.8	2.6	0.520	5.4	29.8	36.0	.190

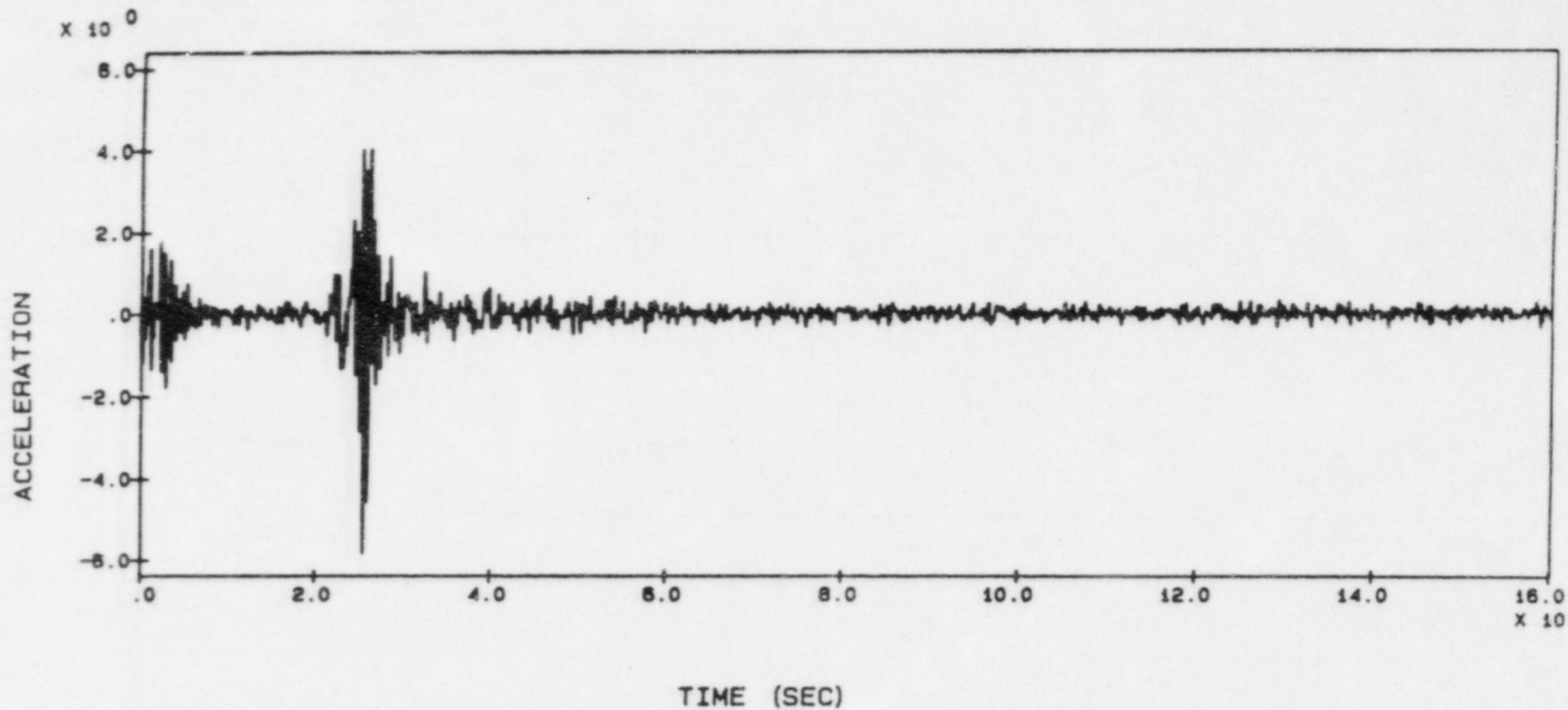


Fig. 4.1 Acceleration Time History, Perry Reactor Building Foundation (Perry, 1986), (ft/sec²)
(a) North-South Component

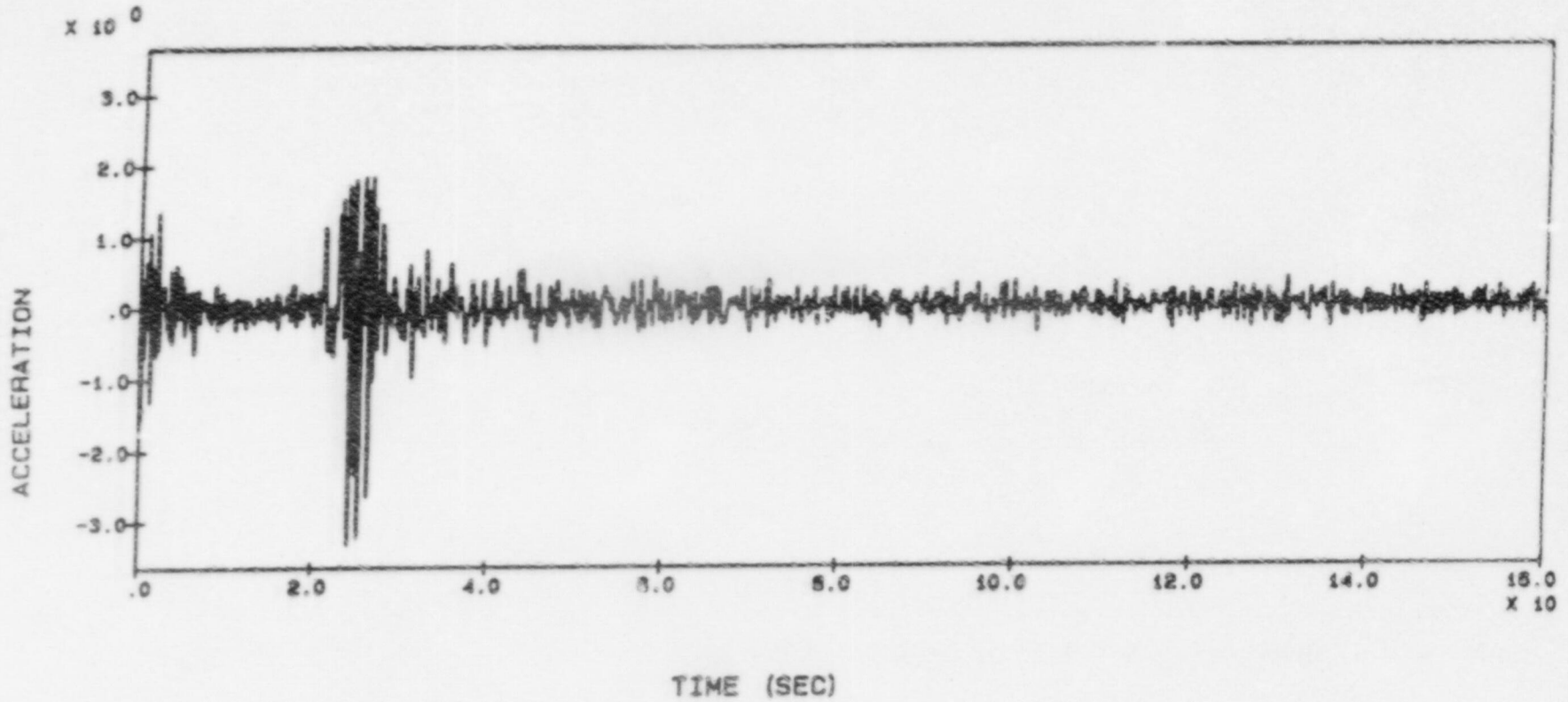


Fig. 4.1 Acceleration Time History, Perry Reactor
Building Foundation (Perry, 1986), (ft/sec²)
(b) East-West Component

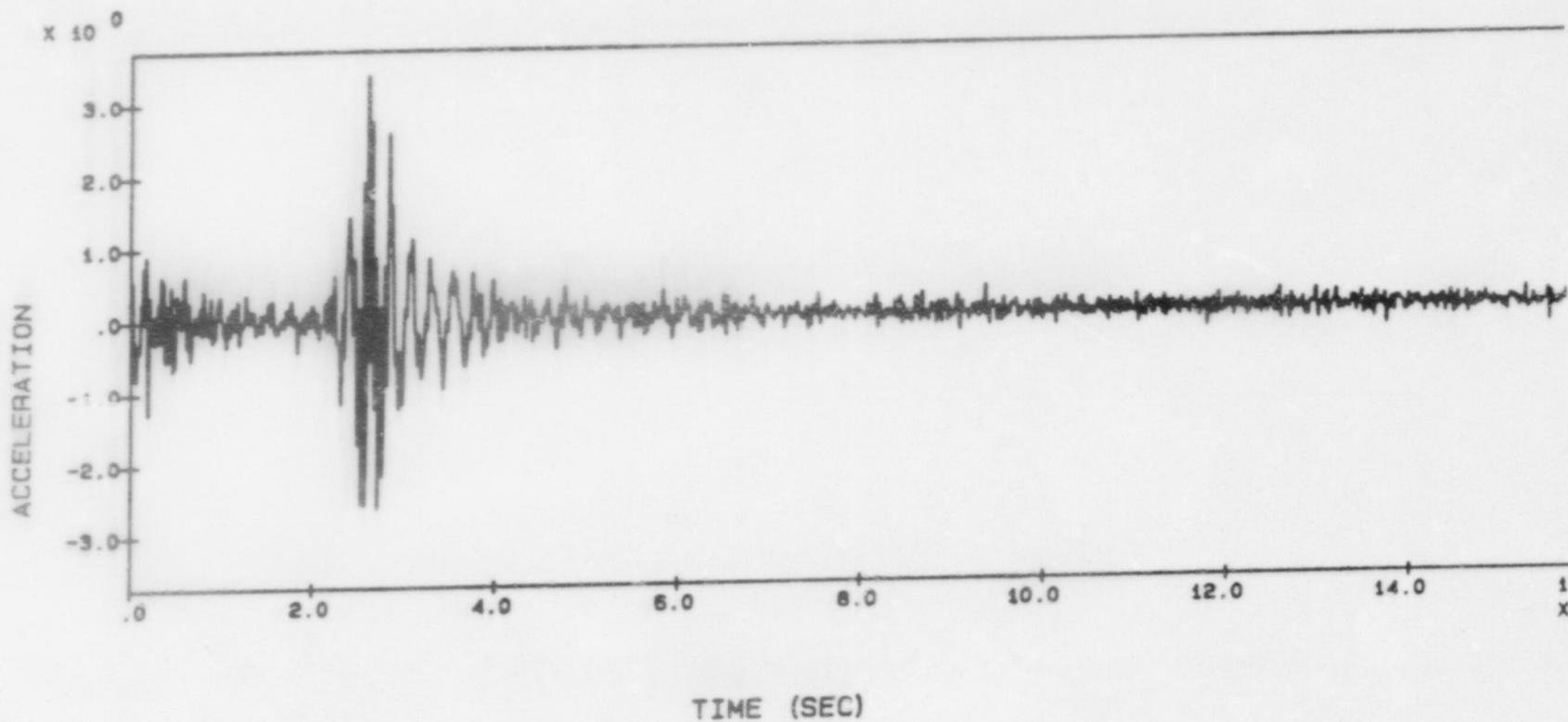


Fig. 4.1 Acceleration Time History, Perry Reactor Building Foundation (Perry, 1986), (ft/sec²)
(c) Vertical Component

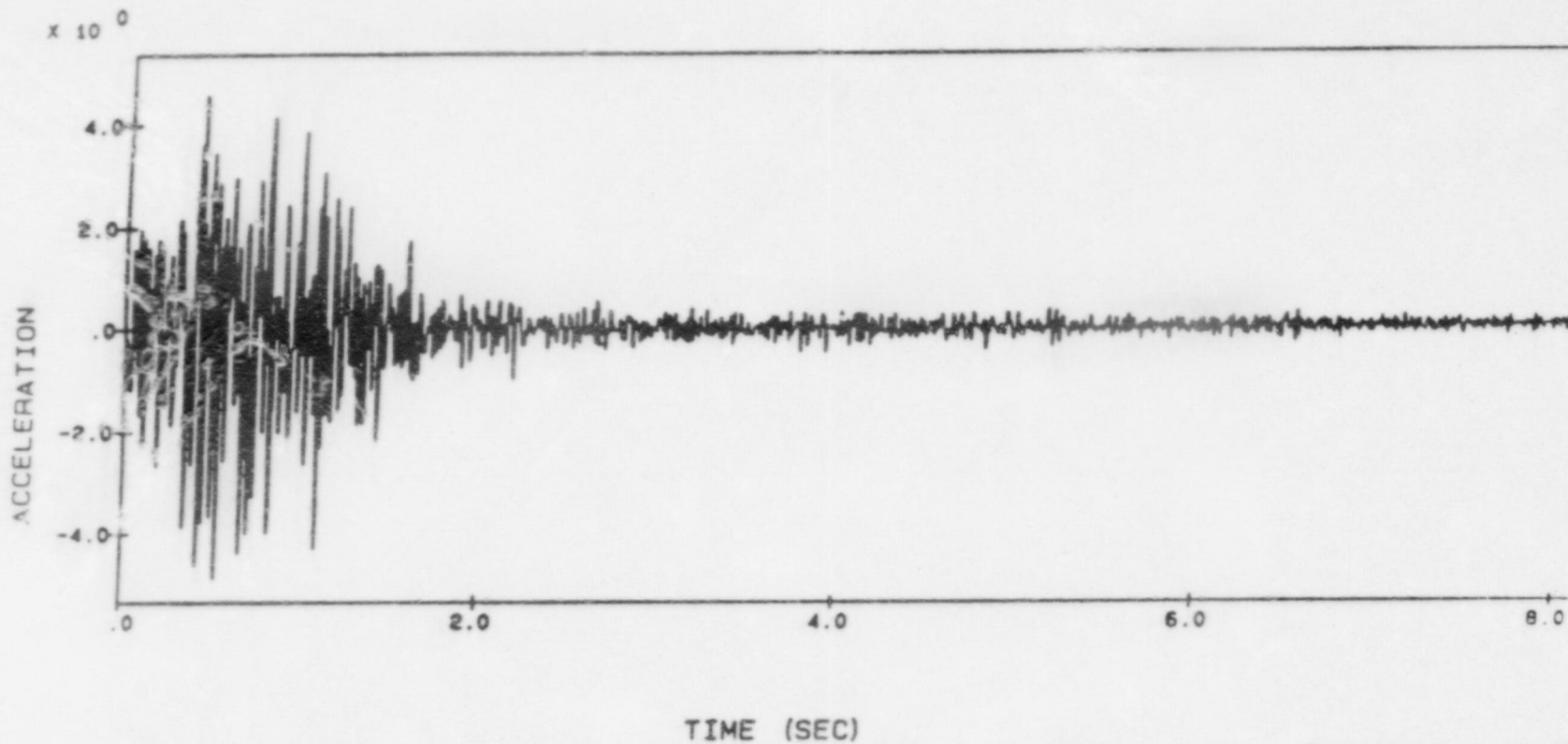


Fig. 4.2 Acceleration Time History, Mitchell Lake
Road New Brunswick (1982), (ft/sec²)
(a) 18° from North

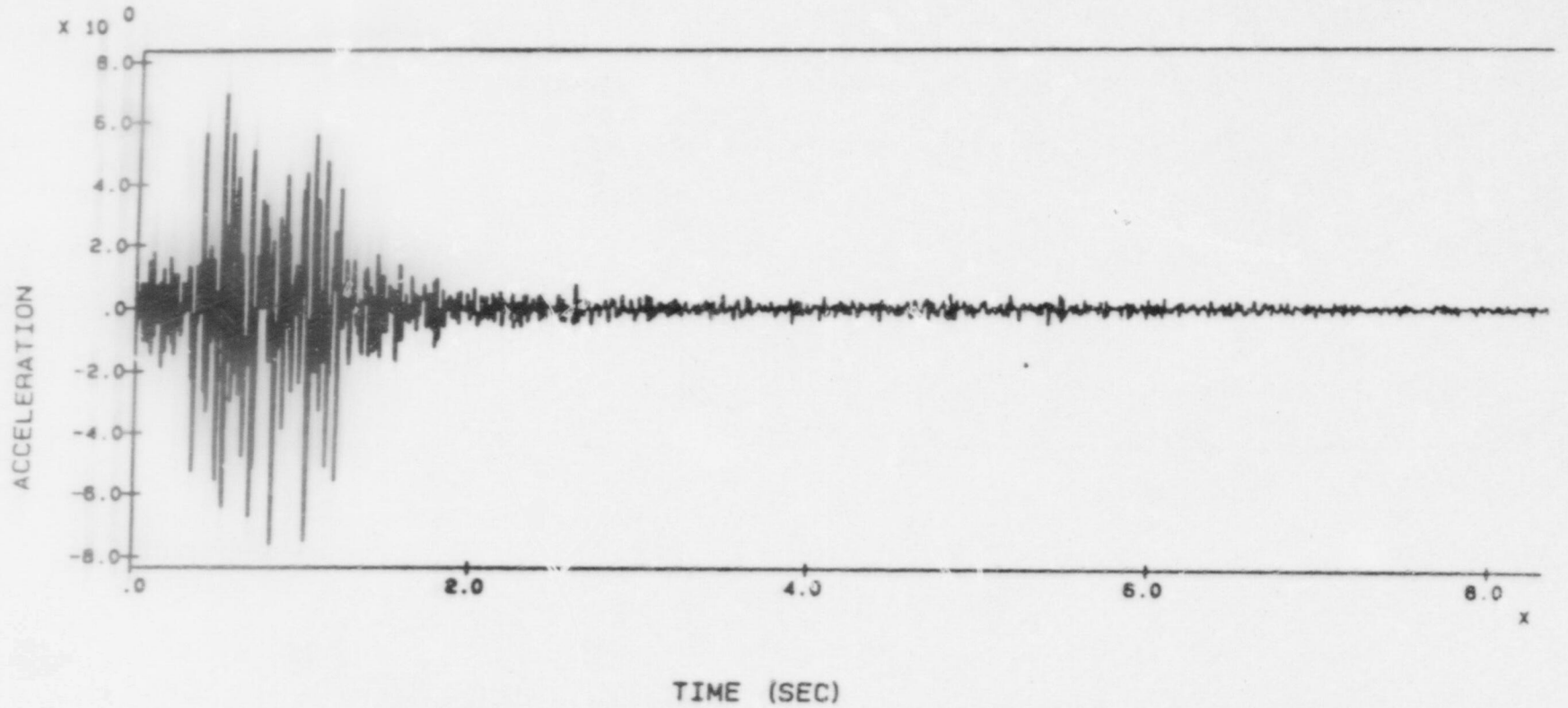


Fig. 4.2 Acceleration Time History, Mitchell Lake Road
New Brunswick (1982), (ft/sec²)
(b) 288° from North

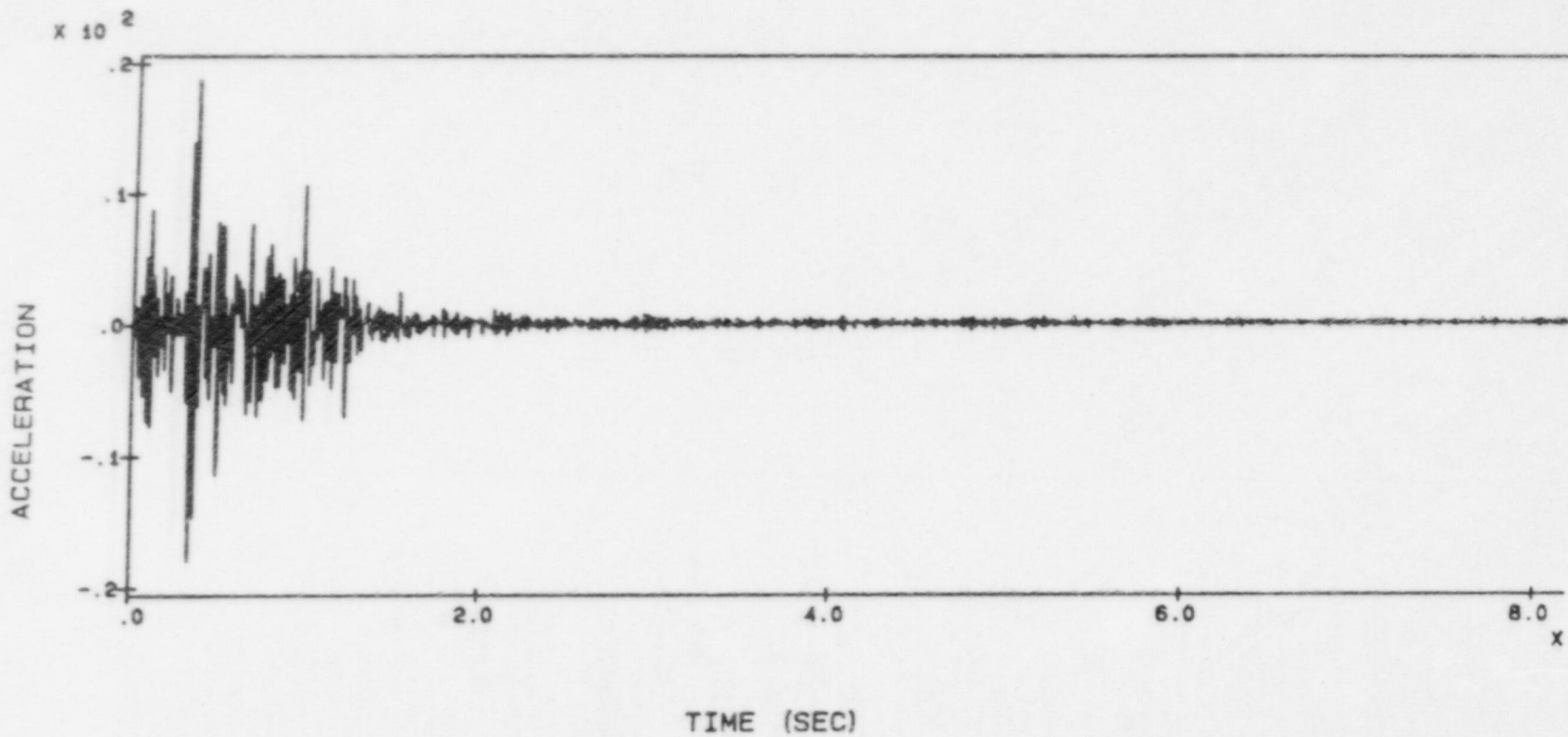


Fig. 4.2 Acceleration Time History, Mitchell Lake Road, New Brunswick (1982), (ft/sec²) (c) Vertical

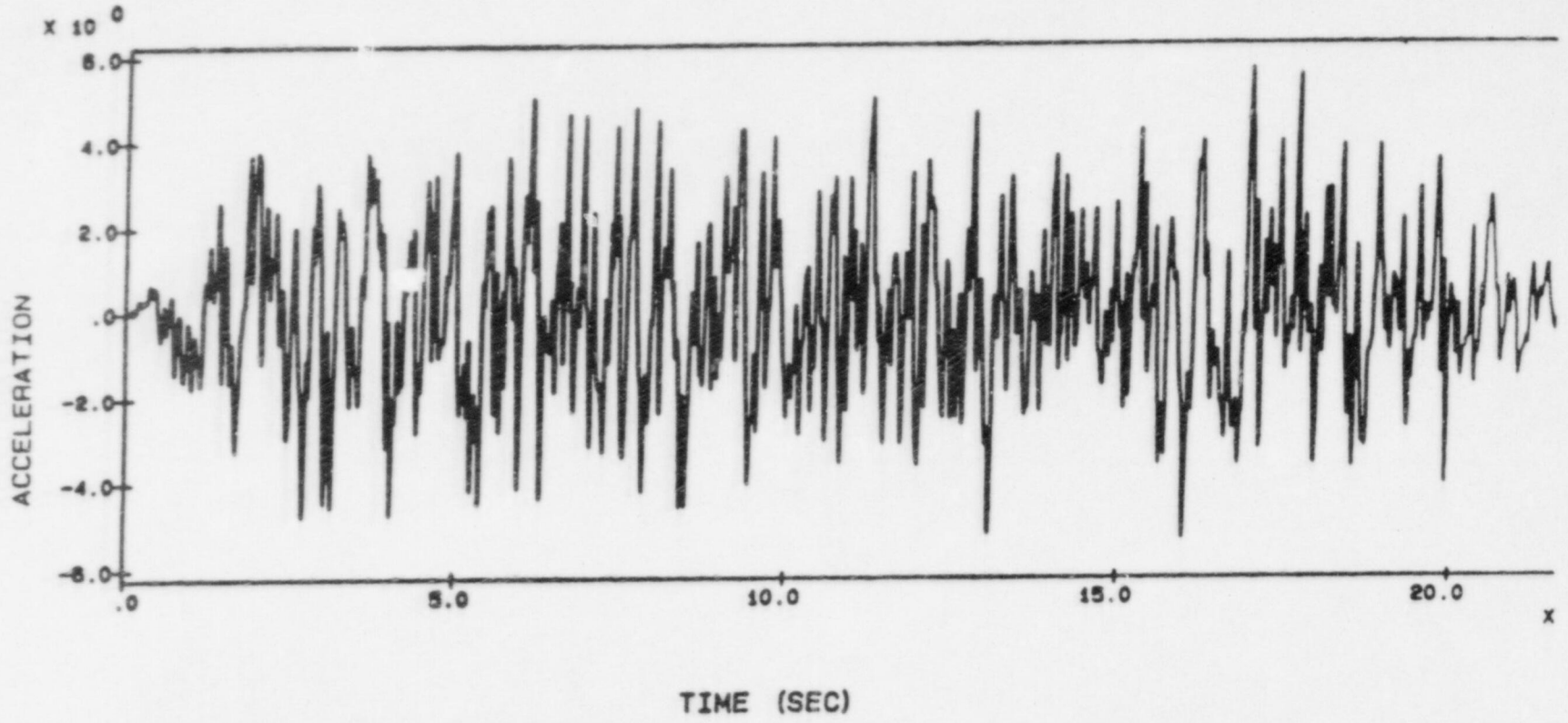


Fig. 4.3 Acceleration Time History, Perry Foundation
Design Motion (ft/sec²)
(a) North-South

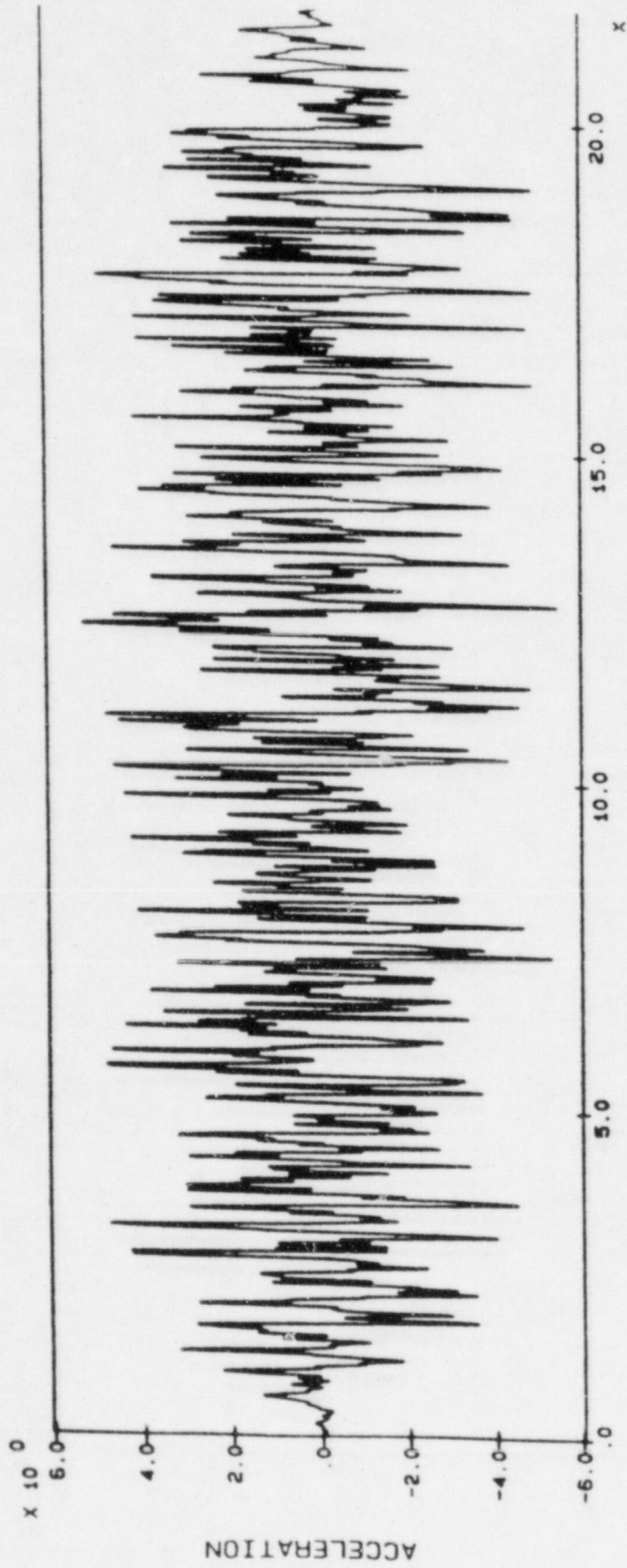


Fig. 4.3 Acceleration Time History, Perry Foundation
 Design Motion (ft/sec²)
 (b) East-West

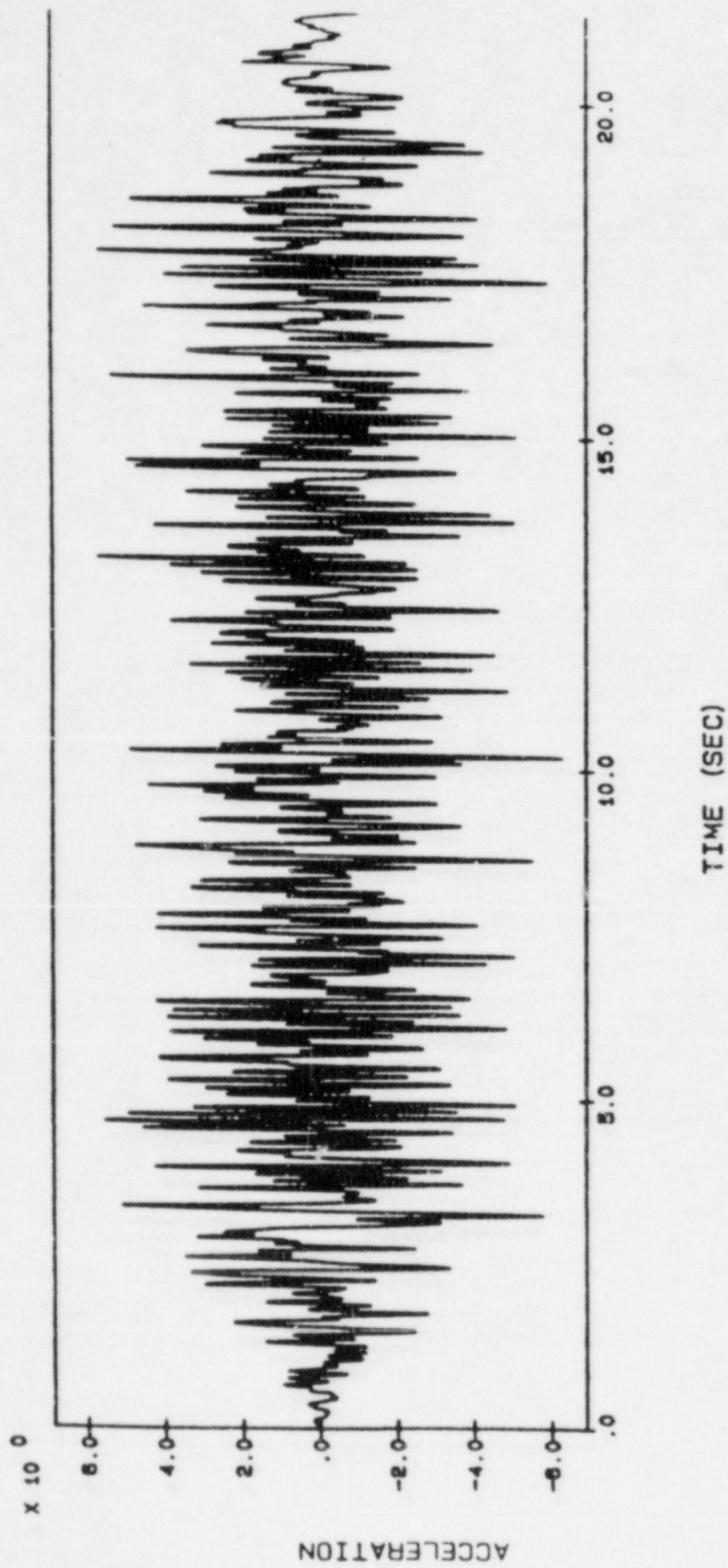


Fig. 4.3 Acceleration Time History, Perry Foundation
 Design Motion (ft/sec²)
 (c) Vertical

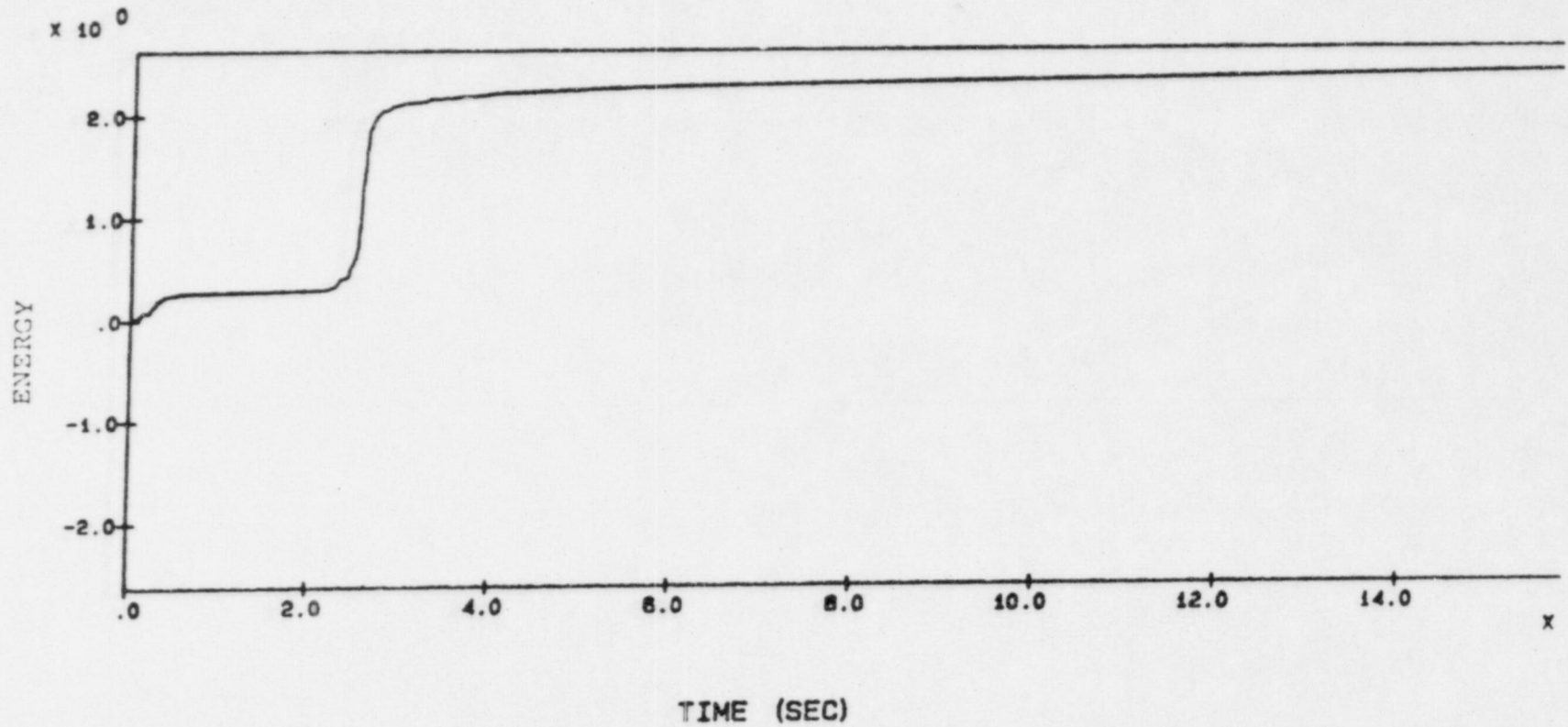


Fig. 4.4 Cumulative Energy (Equ. 4 - 1) with Time, Perry Reactor Building Foundation (Perry, 1986), (ft²/sec³)
 (a) North-South

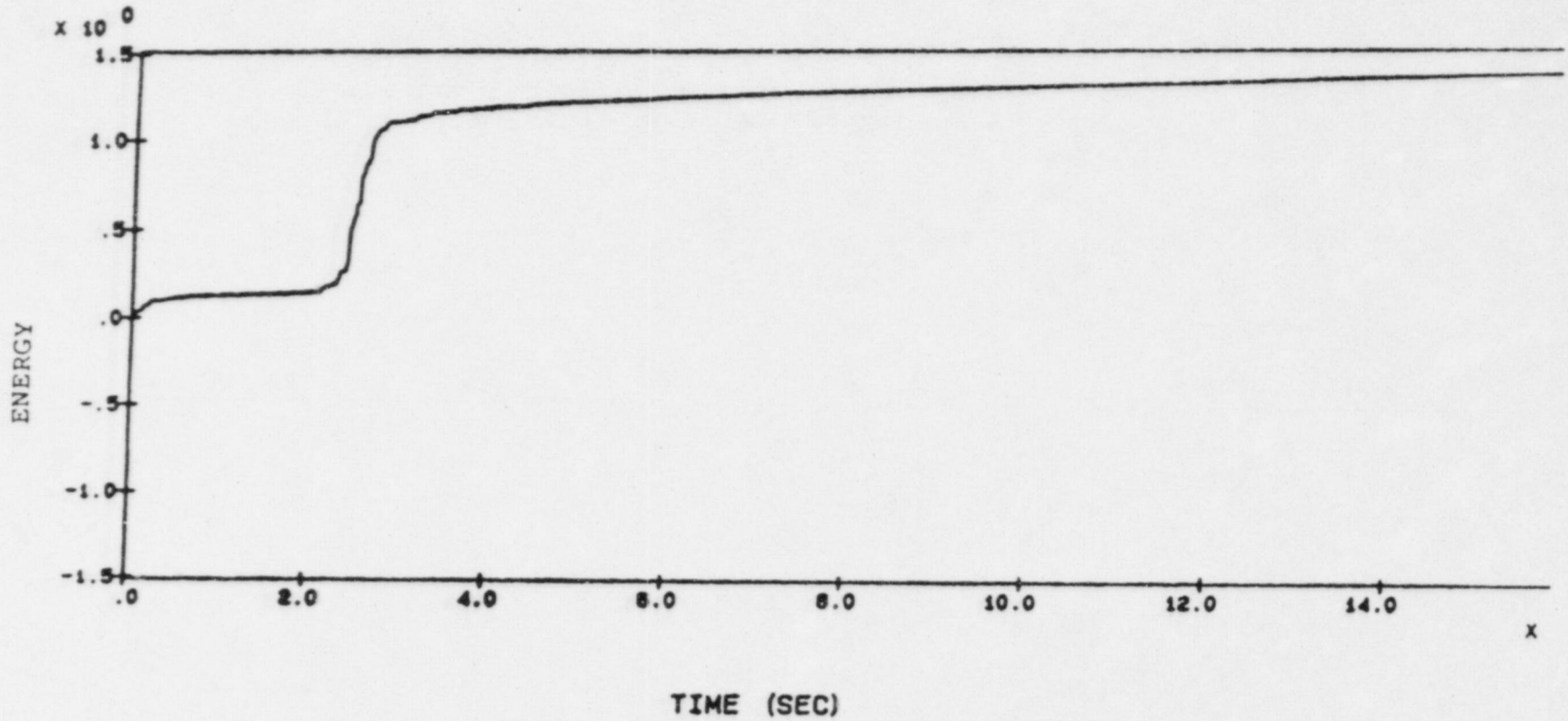


Fig. 4.4 Cumulative Energy (Equ. 4 - 1) with Time, Perry Reactor Building Foundation (Perry, 1986), (ft²/sec³)
(b) East-West

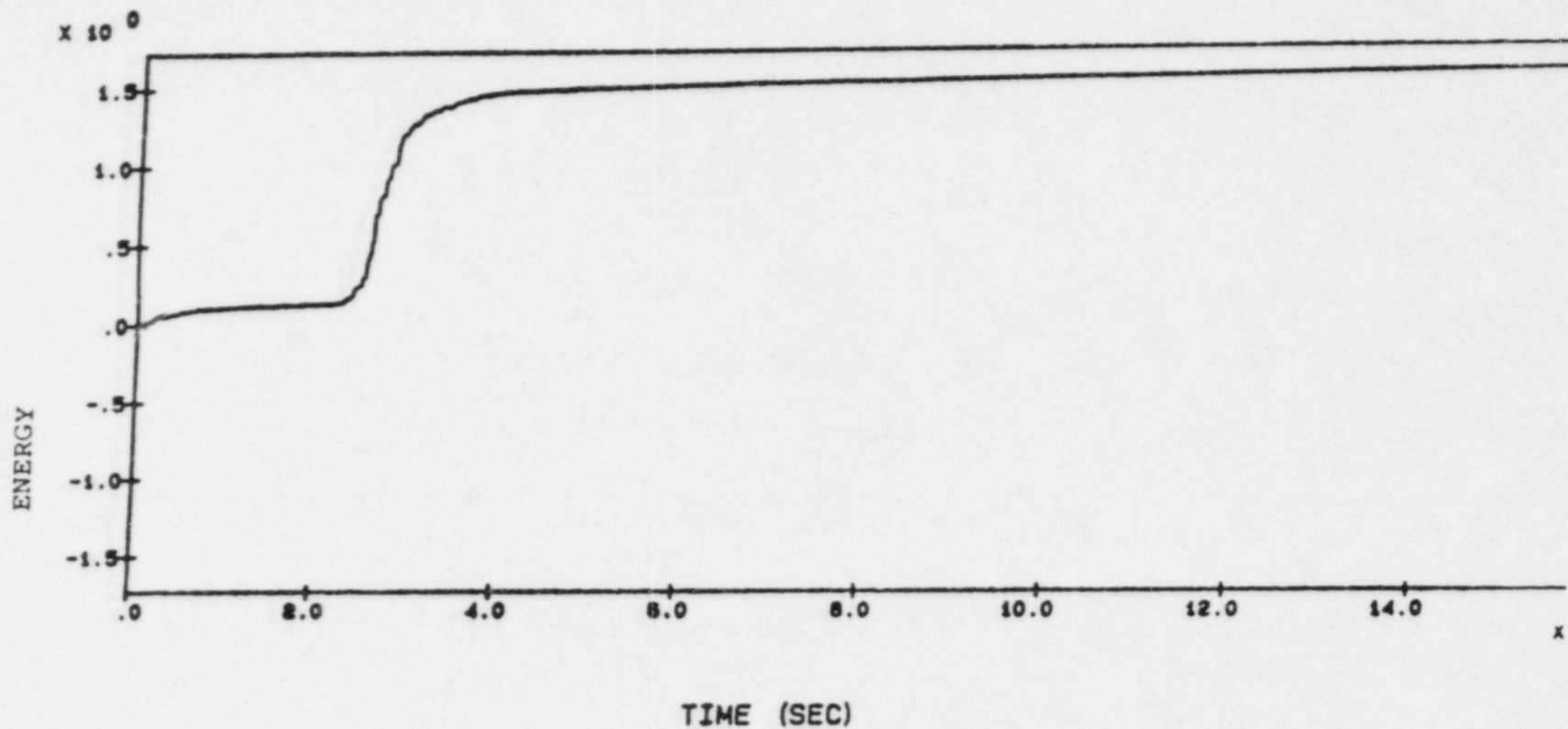


Fig. 4.4 Cumulative Energy (Equ. 4 - 1) with Time, Perry Reactor Building Foundation (Perry, 1986), (ft^2/sec^3)
(c) Vertical

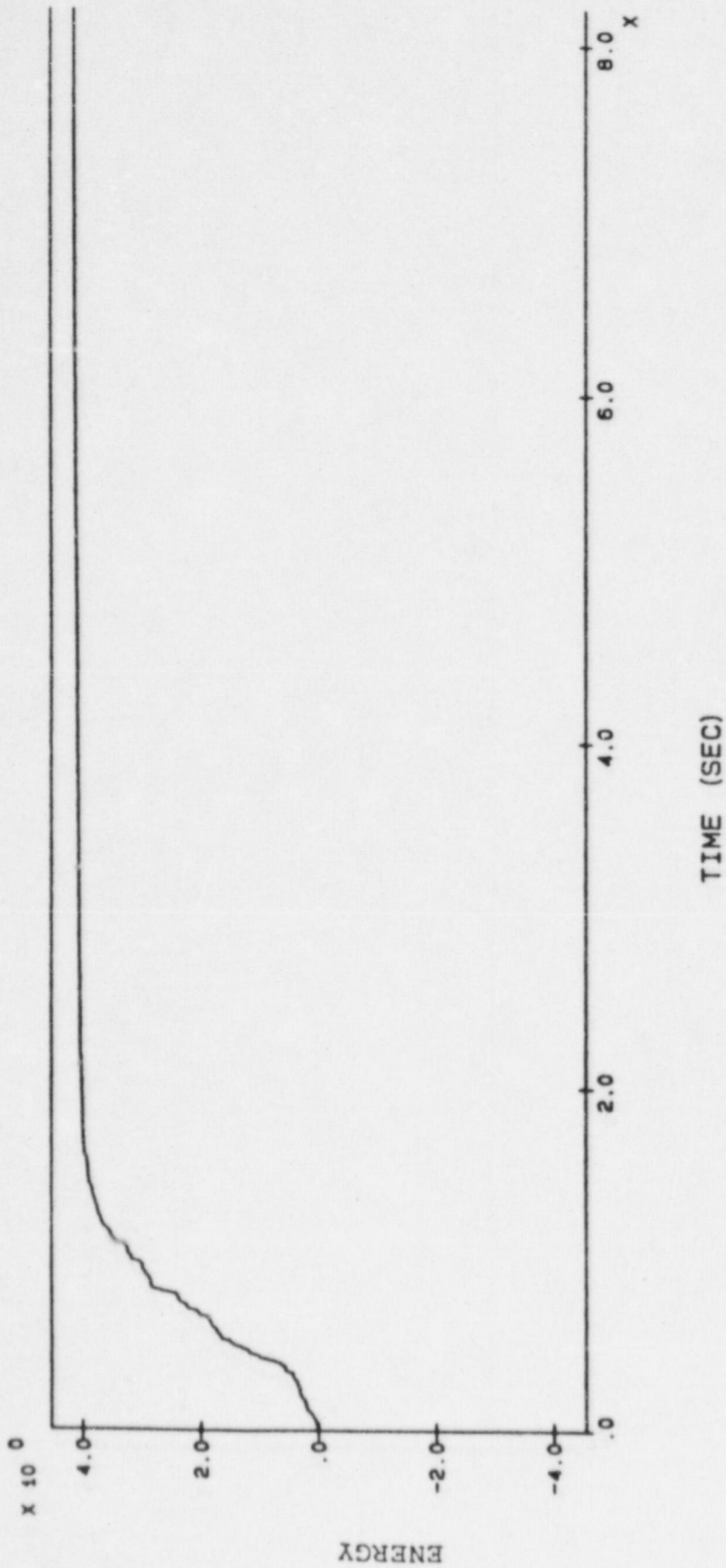


Fig. 4.5 Cumulative Energy (Equ. 4 - 1) with Time, Mitchell, Lake Road, New Brunswick (1982), (ft²/sec³) (a) 18° from North

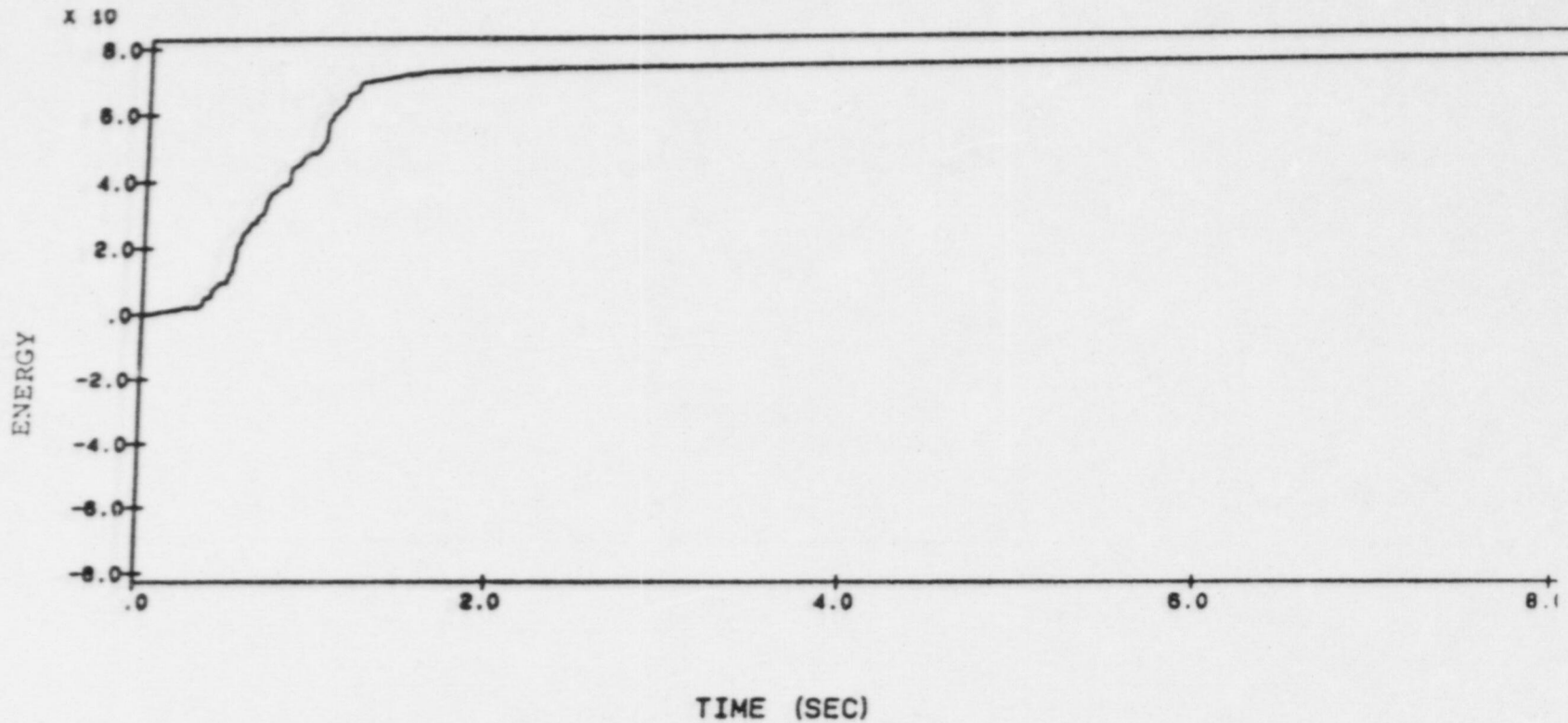


Fig. 4.5 Cumulative Energy (Equ. 4 - 1) With Time,
 Mitchell Lake Road, New Brunswick (1982),
 (ft^2/sec^3)
 (b) 288° from North

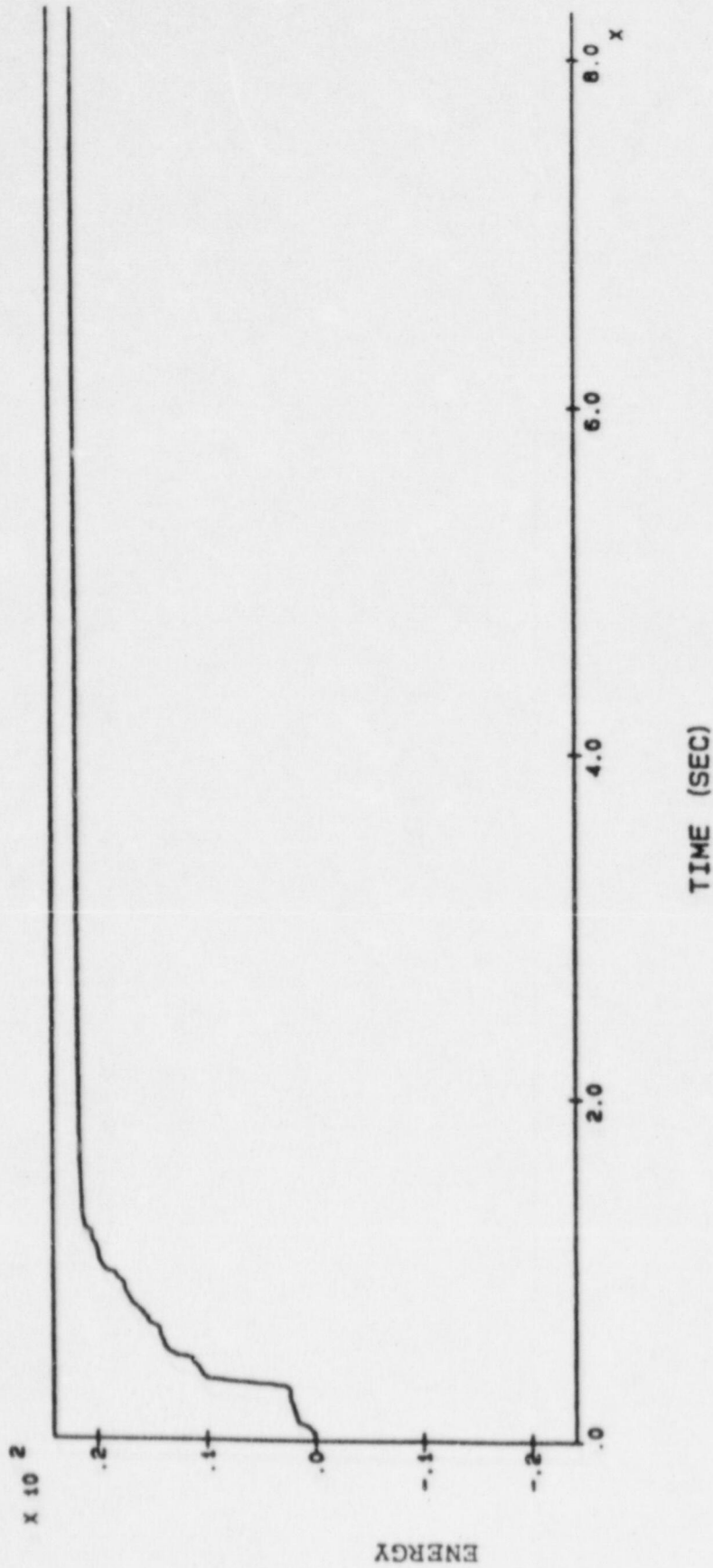


Fig. 4.5 Cumulative Energy (Equ. 4 - 1) with Time,
 Mitchell, Lake Road, New Brunswick (1982),
 (ft²/sec³)
 (c) Vertical

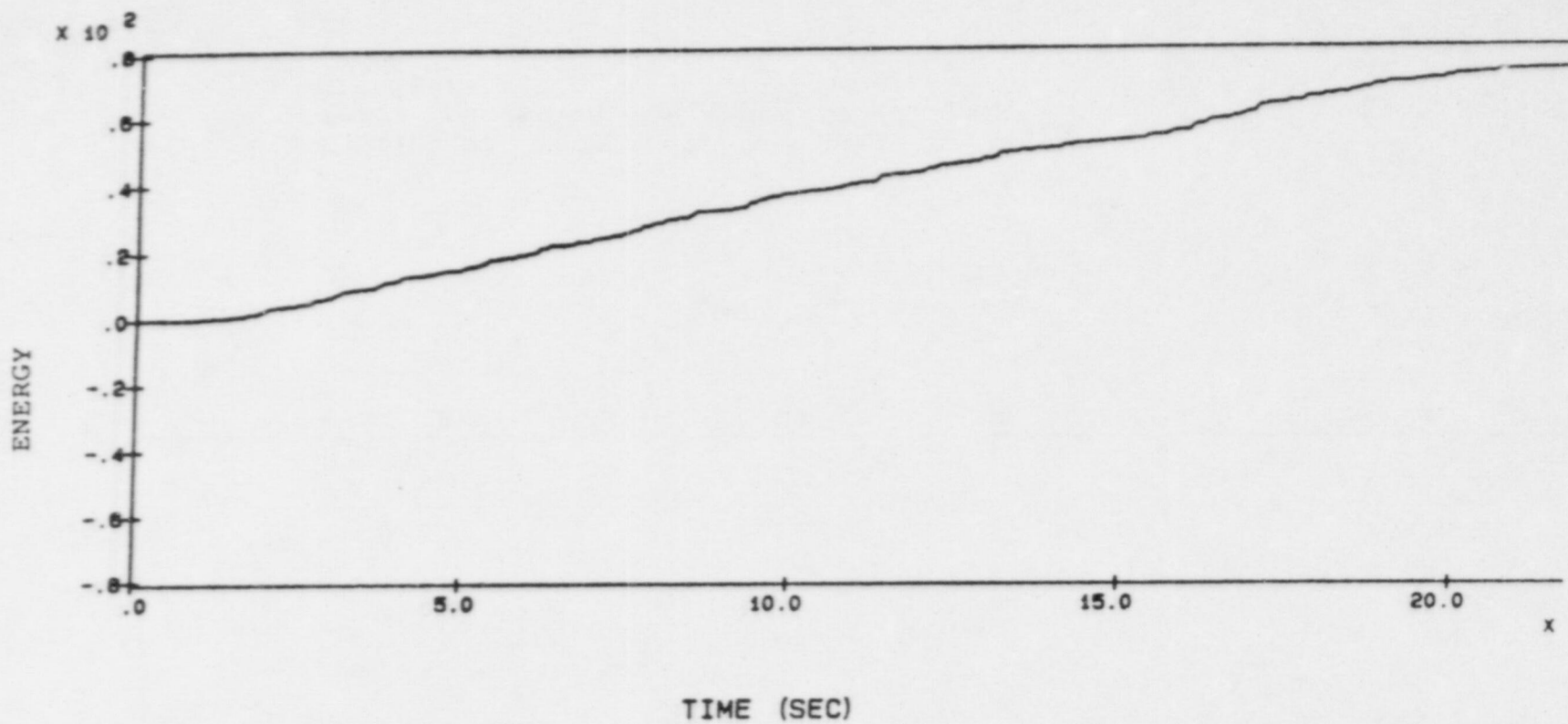


Fig. 4.6 Cumulative Energy (Equ. 4 - 1) with Time,
Perry Foundation Design Motion (ft^2/sec^3)
(a) North-South

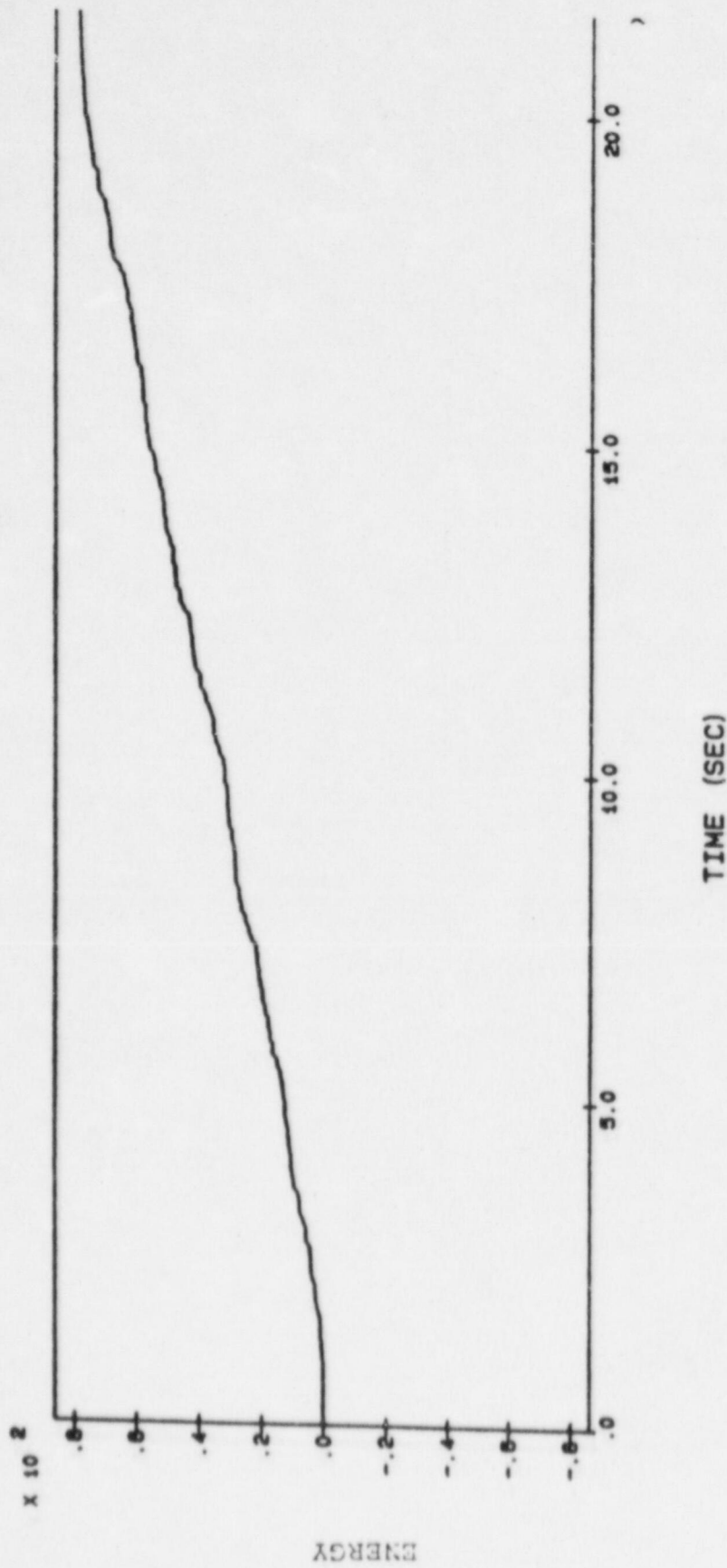


Fig. 4.6 Cumulative Energy (Equ. 4 - 1) with Time,
 Perry Foundation Design Motion (ft²/sec³)
 (b) East-West

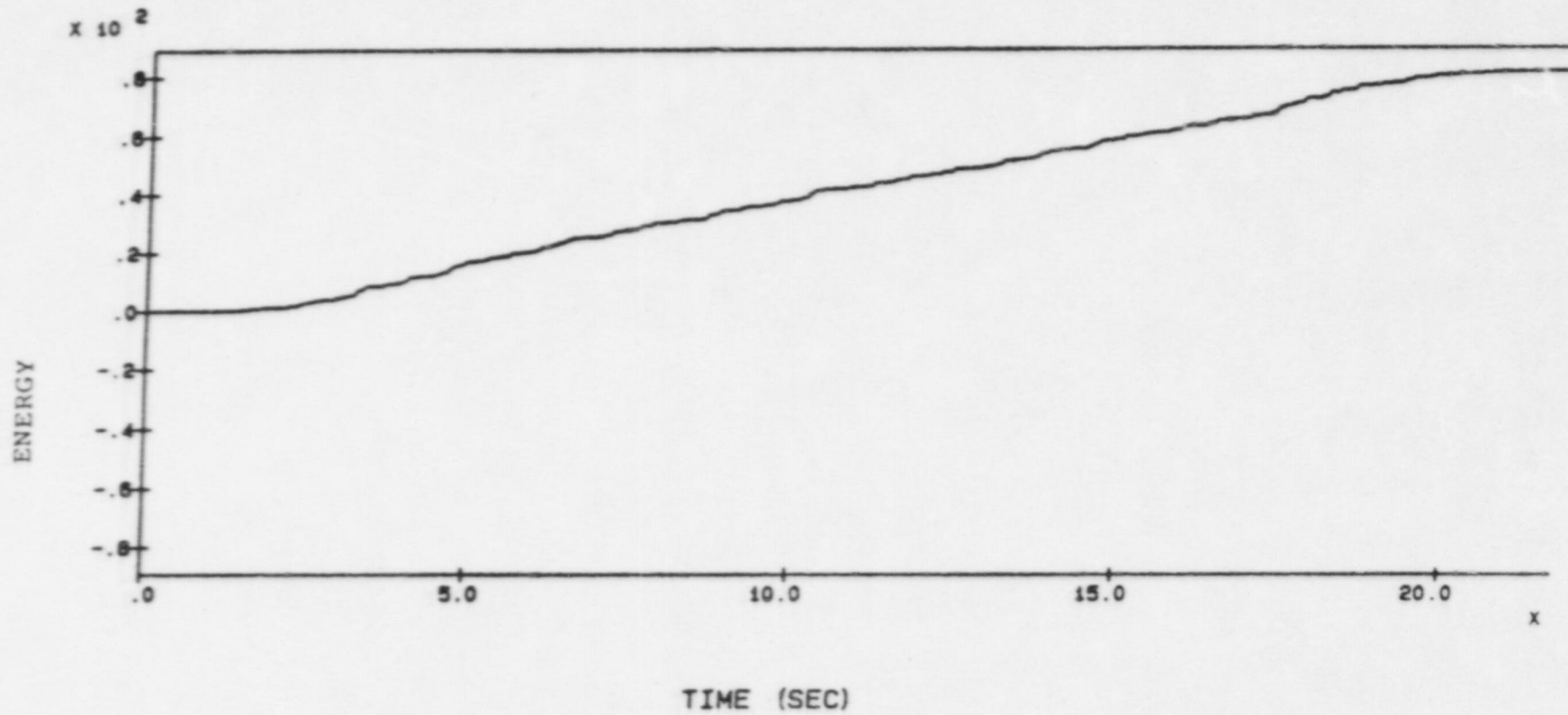


Fig. 4.6 Cumulative Energy (Equ. 4 - 1) with Time,
Perry Foundation Design Motion (ft²/sec³)
(c) Vertical

4.3 Response Spectra and Power Spectral Density Functions

Figures 4.7, 4.8, and 4.9 show the 2%, 5%, and 7% damped response spectra for the Perry (1986) records, the Mitchell Lake Road, New Brunswick (1982) records and the Perry foundation design motion. These latter components (Perry foundation design motions) do not coincide with the design ground response spectra due to the modeling of the soil by constant soil springs in the CEI reactor building dynamic model. The strong 20 Hz motion in the Perry (1986) records can be readily observed in Fig. 4.7. The broad high frequency content (above approximately 15 Hz) of the Mitchell Lake Road, New Brunswick (1982) record is observed from Fig. 4.8. Finally, Fig. 4.9 shows the relatively broad low frequency content of the Perry foundation design motion. These records are not scaled and their peak accelerations are those listed in Table 4.1.

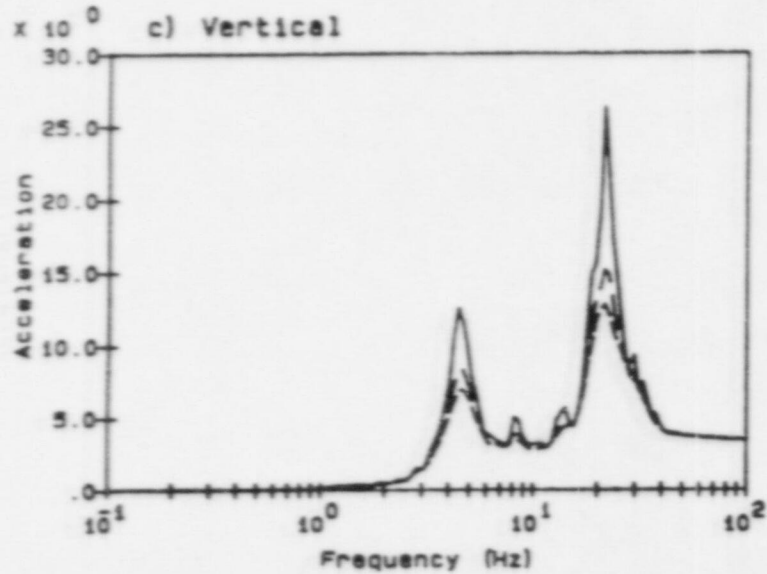
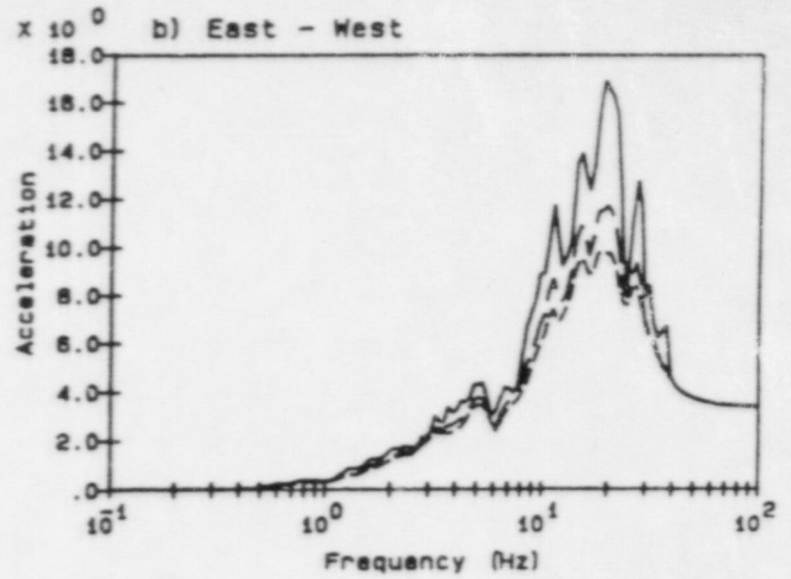
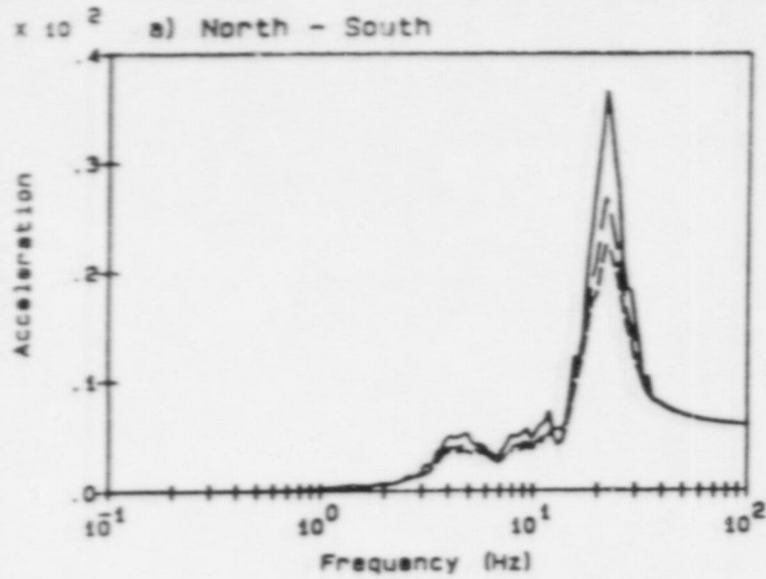
Power spectral density (PSD) functions were calculated for the nine records--the three components of the Perry (1986) motions, the three components of the Mitchell Lake Road, New Brunswick (1982) motions, and the three components of the Perry foundation design motions. The technique used was identical to that of Ref. 17. The strong motion duration T_D was used which again may be slightly misleading for the Perry (1986) records. The PSDs were smoothed by means of a three-point moving average technique which is specified in Ref. 19. The results are displayed in two ways. First, the PSDs of the un-scaled and un-normalized records are shown in Figs. 4.10, 4.12, and 4.14 for the Perry (1986) records, the Mitchell Lake Road, New Brunswick (1982) records, and the Perry foundation design motions. Second, the records were normalized to 1 g. and their PSDs re-plotted versus the US NRC target PSD (Refs. 17-19). For all plots, the units of the PSDs are ($\text{in.}^2/\text{sec.}^3$) to be consistent with the US NRC target.

Figures 4.10 and 4.11 display the distribution of energy as a function of frequency for the Perry (1986) records. Of note is the 4-5 Hz. and

20 Hz. motion especially in the N-S and vertical components. The importance of these frequency ranges was previously demonstrated in Ref. 20. The E-W component shows broader energy content over the frequency range of interest. The comparison of the normalized PSD with the US NRC target PSD shows the Perry (1986) records to generally be above the target in the high frequency range.

Figures 4.12 and 4.13 display the distribution of energy as a function of frequency for the Mitchell Lake Road, New Brunswick (1982) records. They display relatively uniform energy content in the high frequency range. Note, these PSDs are calculated and plotted up to 34 Hz as specified in Ref. 19. The comparison of the normalized PSD with the US NRC target PSD shows the Mitchell Lake Road, New Brunswick (1982) records to be above the target in the high frequency range.

Figures 4.14 and 4.15 display the PSDs for the Perry foundation design motion. They have less energy in the high frequency range and, when compared to the US NRC target PSD, drop below the target. It is important to note, however, that these records are on the foundation after SSI analysis has been performed so they vary from the free-field design ground motions for which the target was developed. The oscillation or drop below the target PSD in the high frequency range is not uncommon, however, for artificial time histories generated to match RG 1.60 design ground response spectra (Ref. 17).



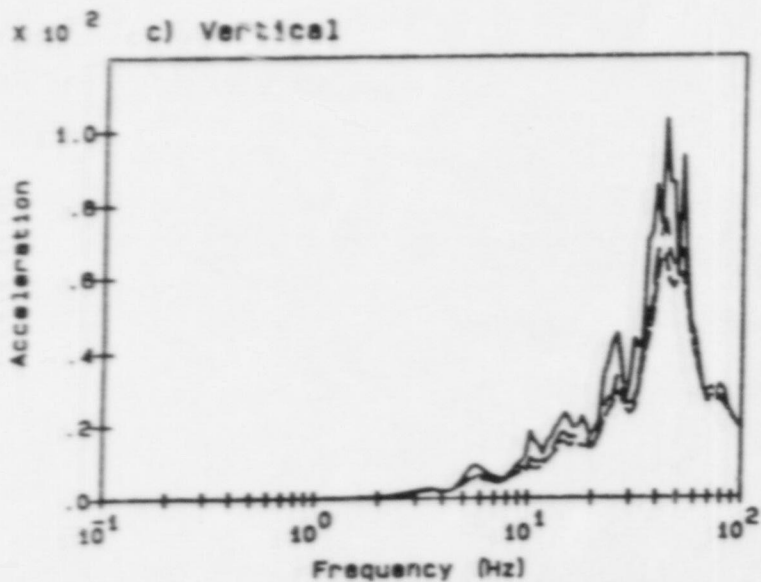
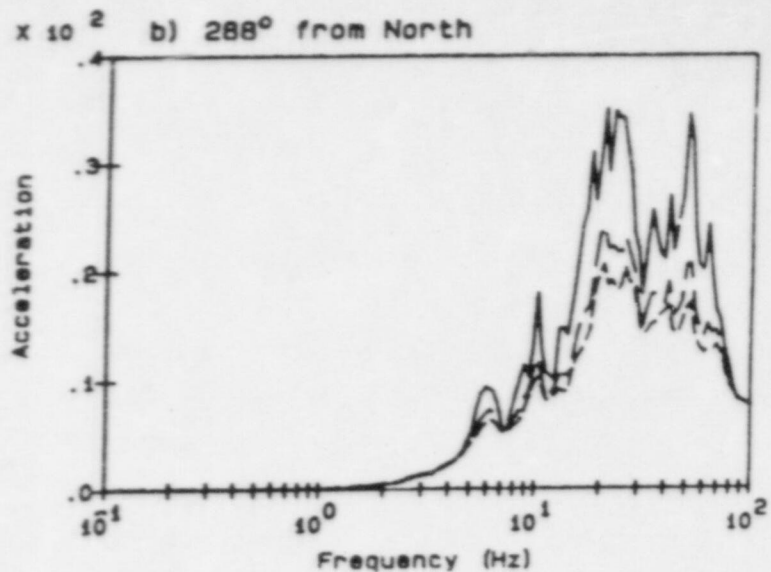
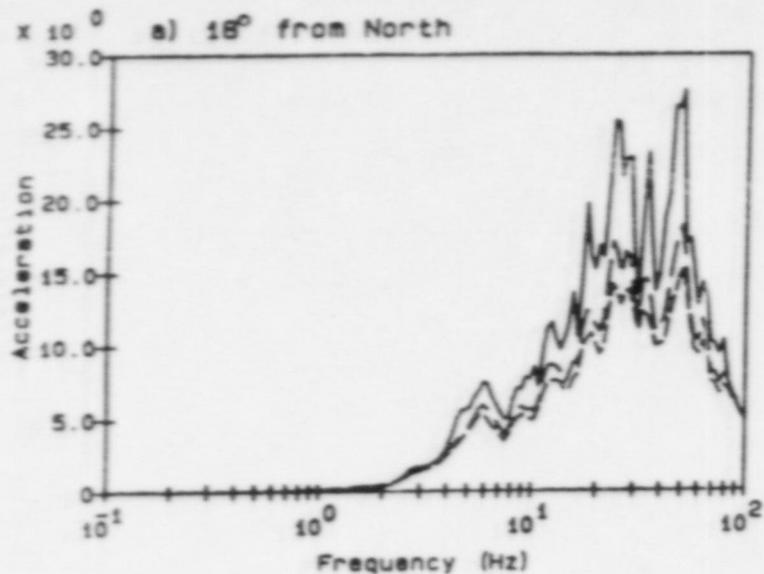
Legend:

- Perry (1986) _____
- 2% damping - - - - -
- 5% damping - - - - -
- 7% damping - - - - -

Notes:

All acceleration response spectrum values in units of ft/(s*s)

Fig. 4.7: Acceleration Response Spectra. Perry (1986) Reactor Building Foundation Motion.



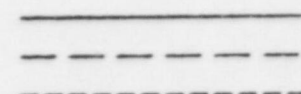
Legend:

Mitchell Lake Rd.

2% damping

5% damping

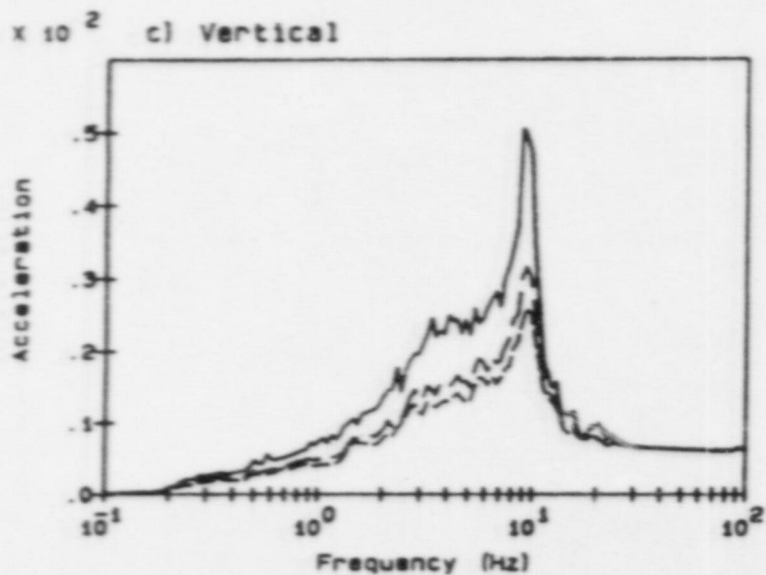
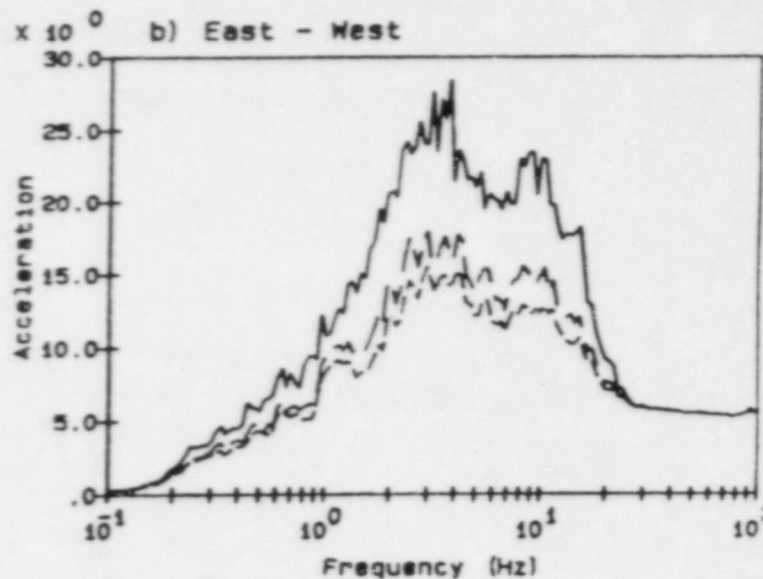
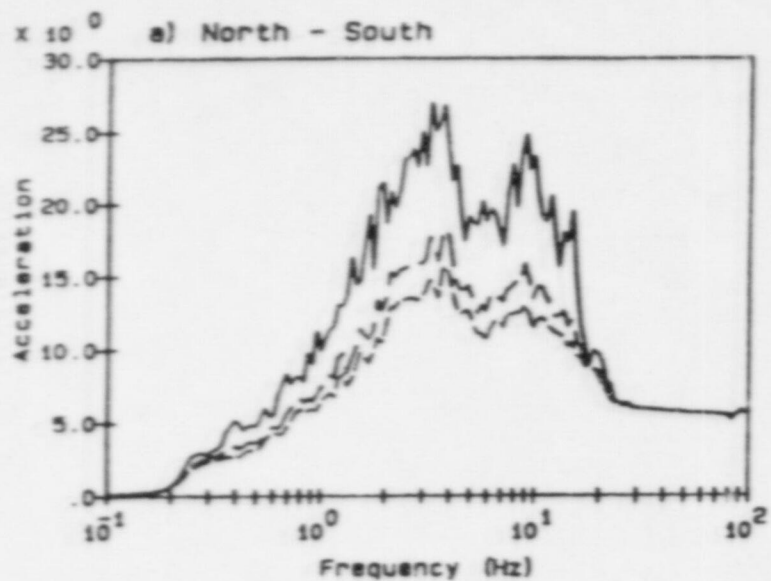
7% damping



Notes:

All acceleration response spectrum values in units of ft/(s²s)

Fig. 4.8: Acceleration Response Spectra, Mitchell Lake Road



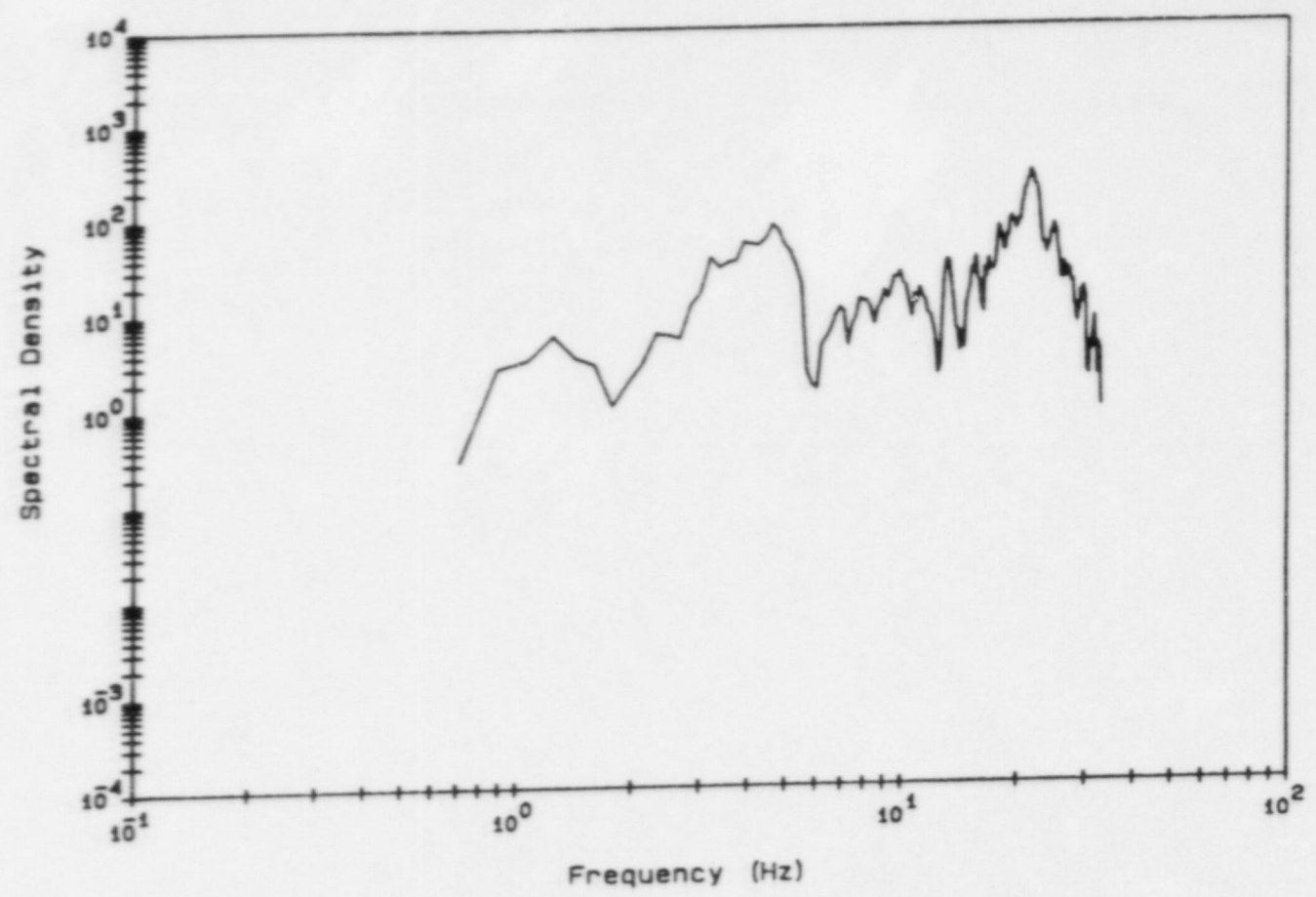
Legend:

- Perry Design _____
- 2% damping -----
- 5% damping - - - - -
- 7% damping - . - . -

Notes:

All acceleration response spectra values in units of ft/(s²)

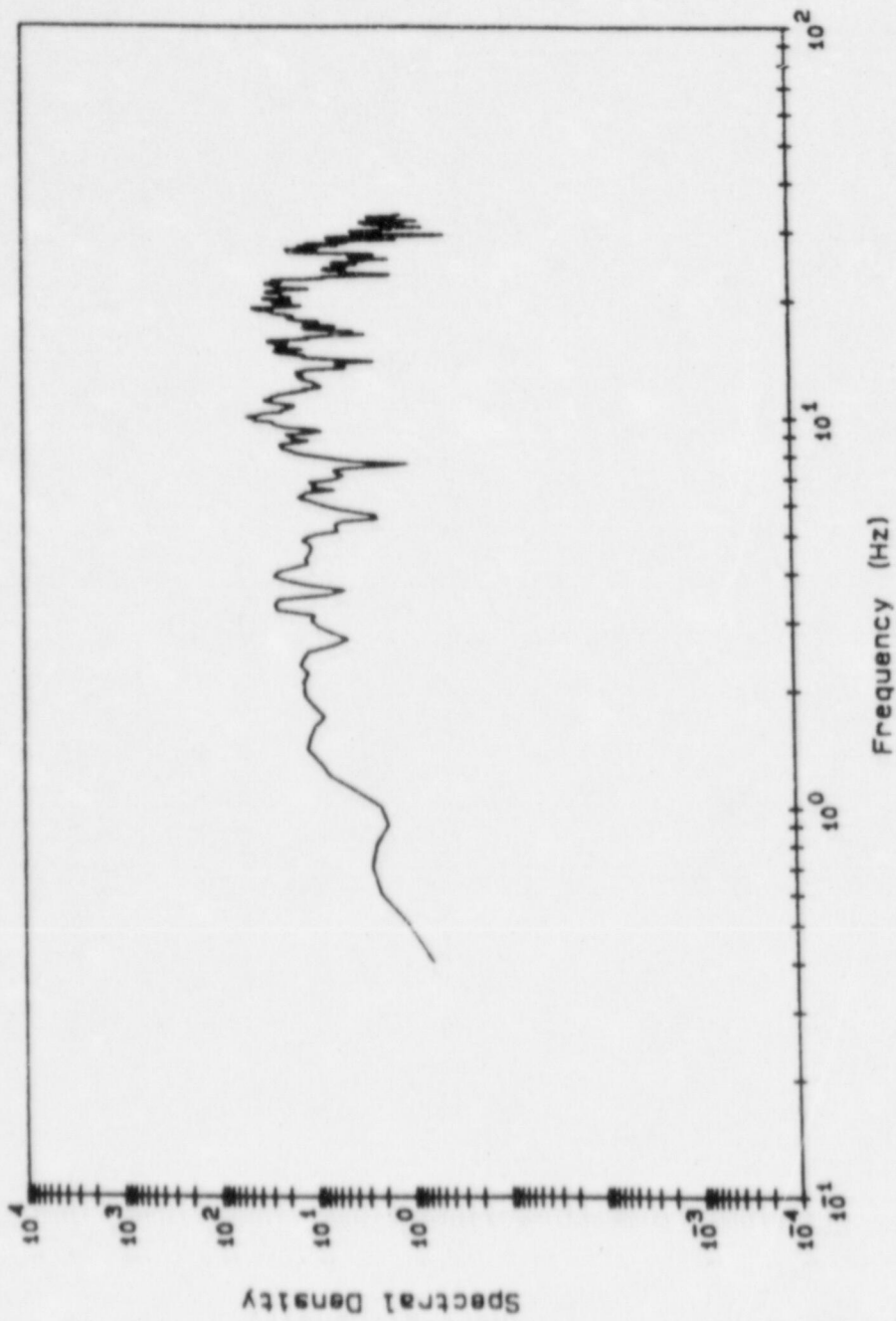
Fig. 4.9: Acceleration Response Spectra, Perry Design Reactor Building Foundation Motion.



Notes:

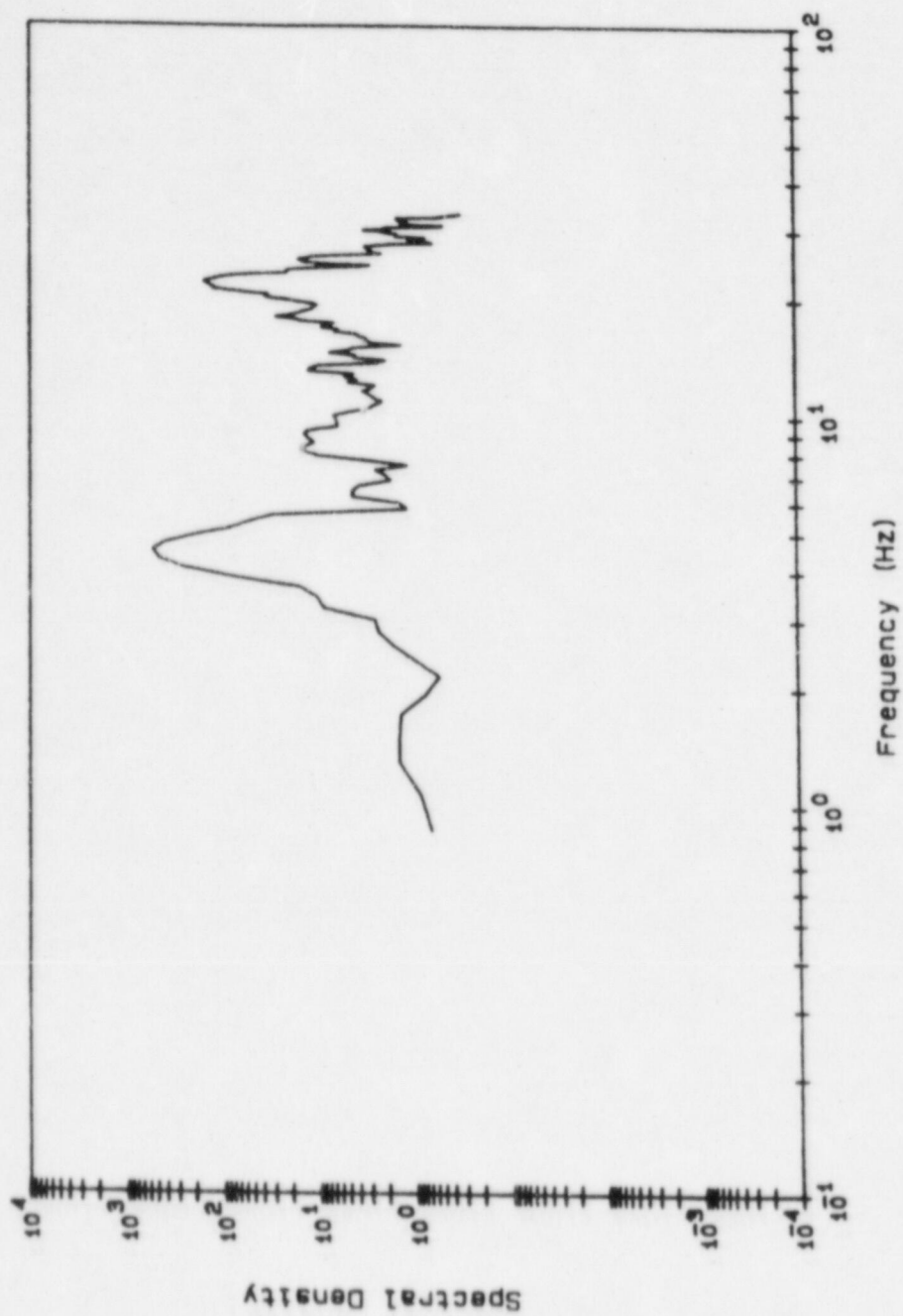
All spectral density values in units of
(in*in) / (s*s*s)

Fig. 4.10: Power Spectral Density Function, Perry (1986)
(a) North - South



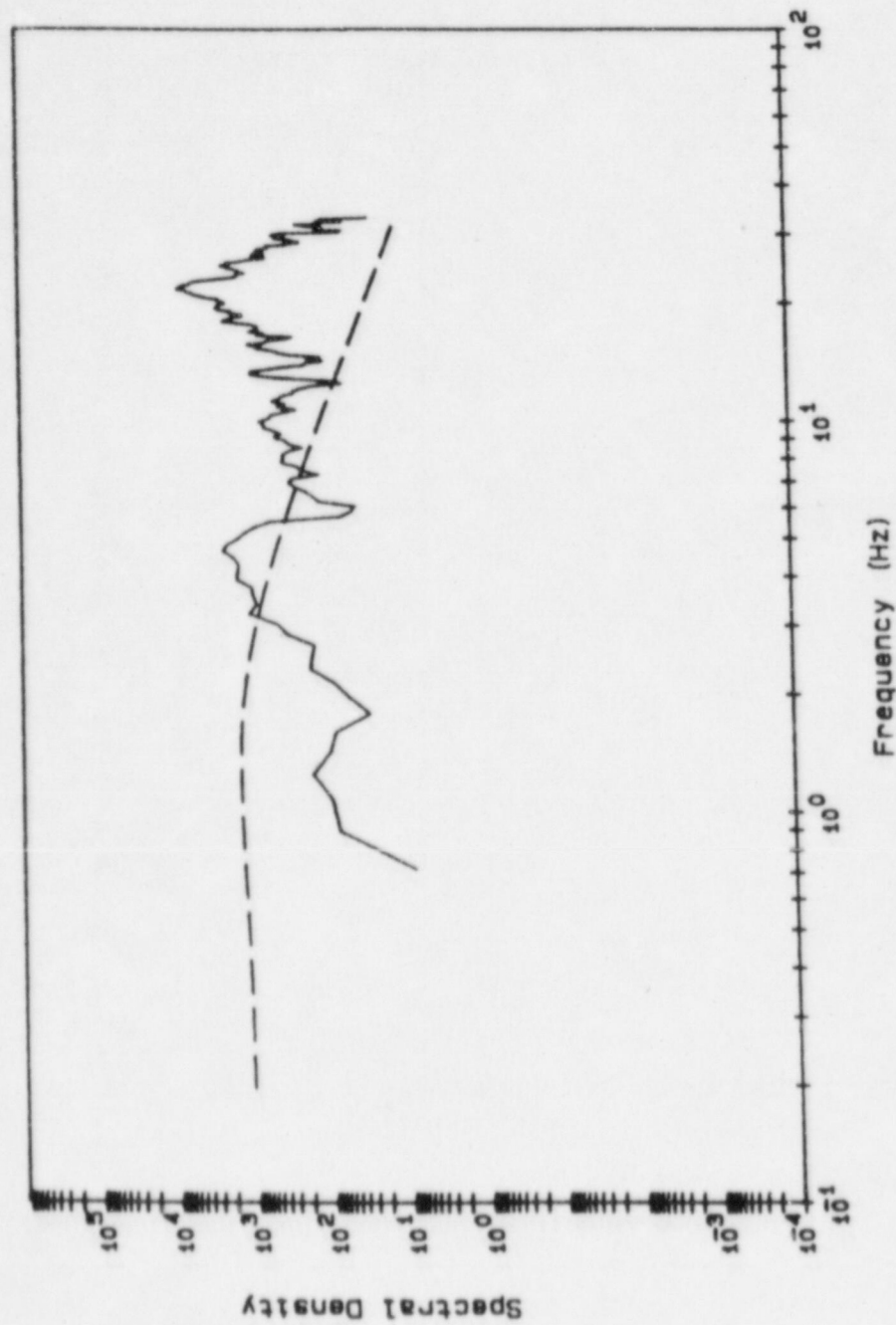
Notes:
 All spectral density values in units of
 (in/min) / (awsms)

Fig. 4.10: Power Spectral Density Function, Perry (1986)
 (b) East - West



Notes:
 All spectral density values in units of
 (in/in) / (s/s)

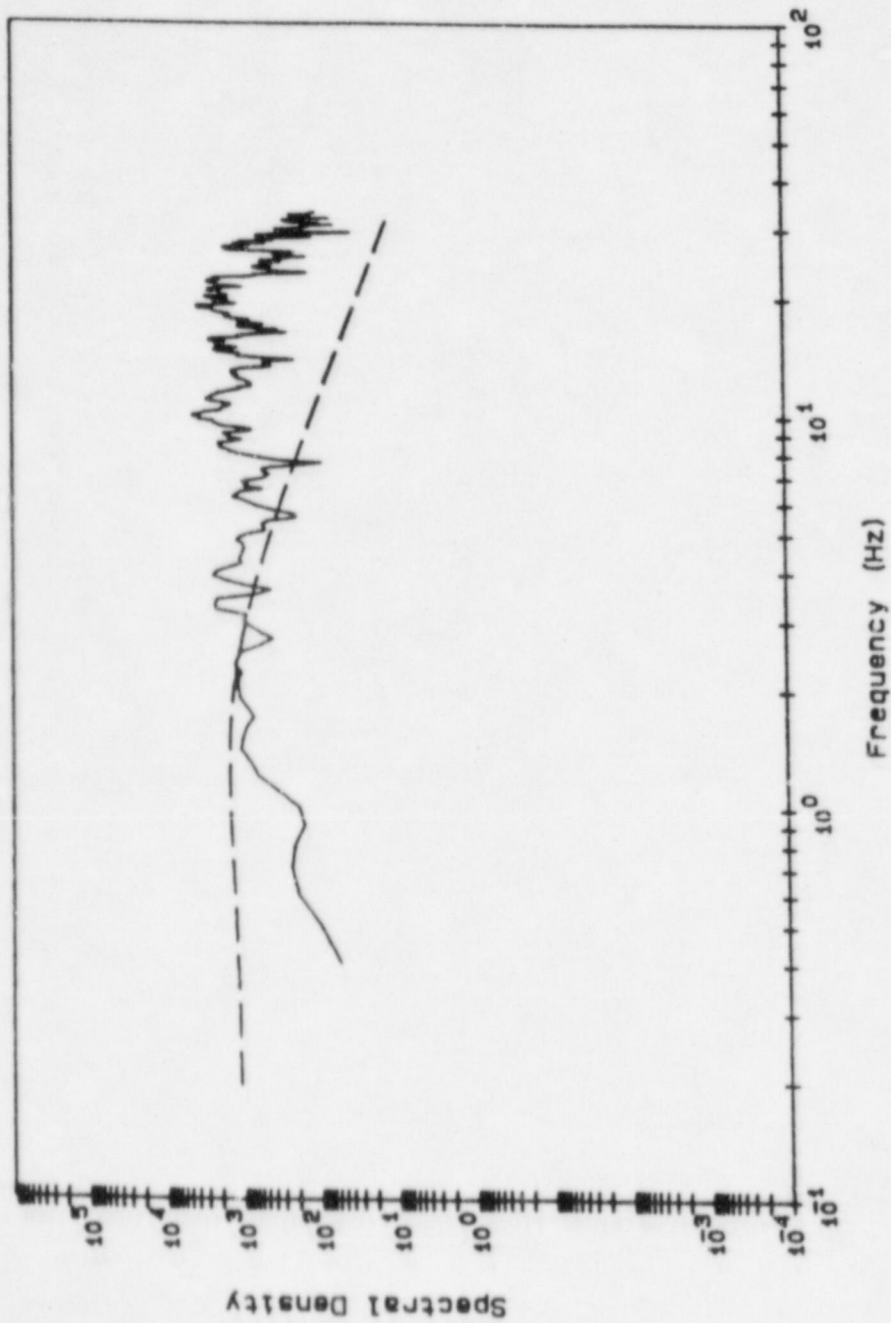
Fig. 4.10: Power Spectral Density Function, Perry (1986)
 (c) Vertical



Legend:
Perry (1986) N - S _____
NRC Proposed Target - - - - -

Notes:
All spectral density values in units of
(in*in) / (s*ms)

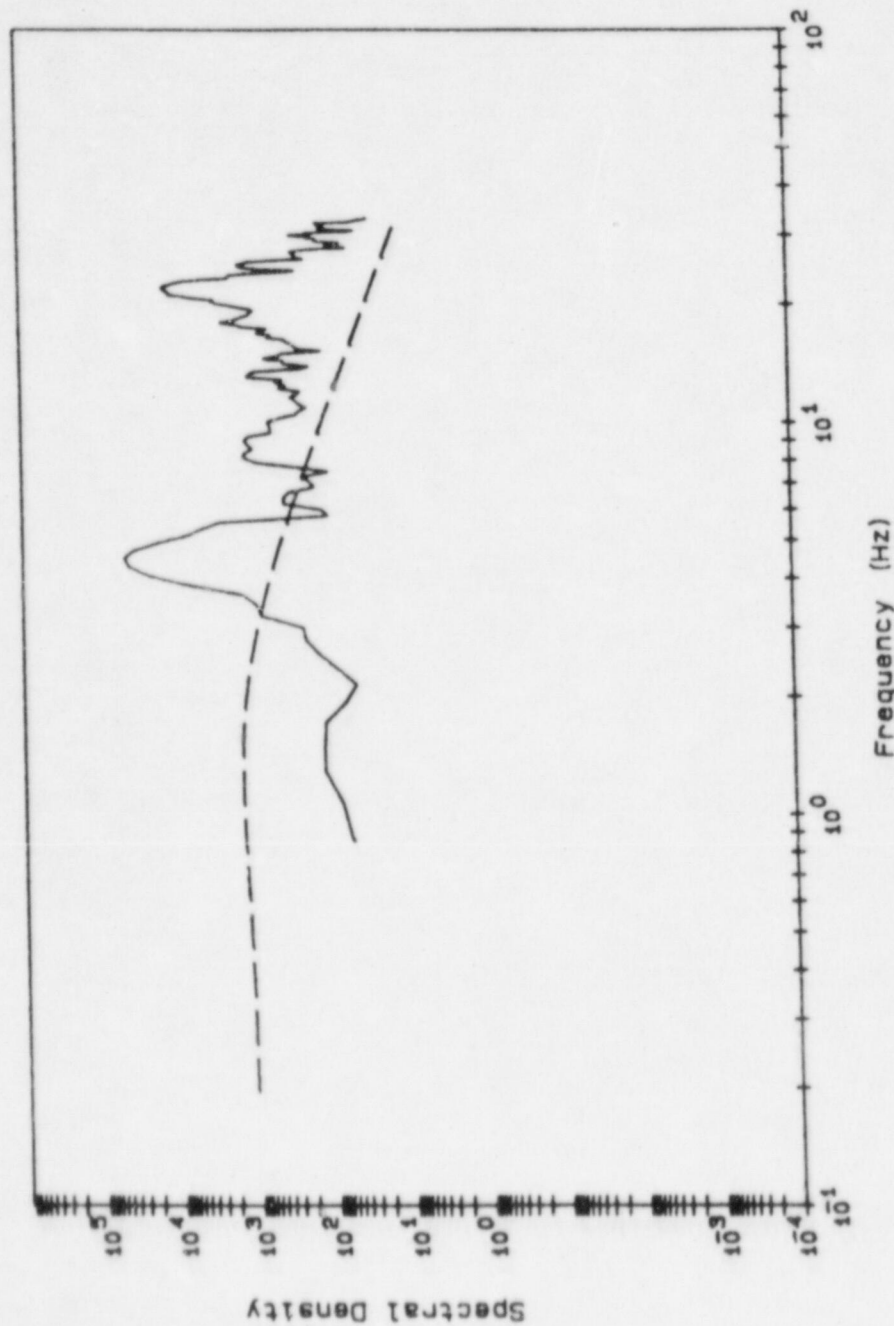
Fig. 4.11: Power Spectral Density Function, Perry (1986) Normalized to 1. g versus NRC Proposed Target (a) North - South



Legend:
 Perry (1986) E - W ———
 NRC Proposed Target - - - - -

Notes:
 All spectral density values in units of
 (in ω n) / (s ω s ω s)

Fig. 4.11: Power Spectral Density Function, Perry (1986) Normalized to 1. g versus NRC Proposed Target (b) East - West

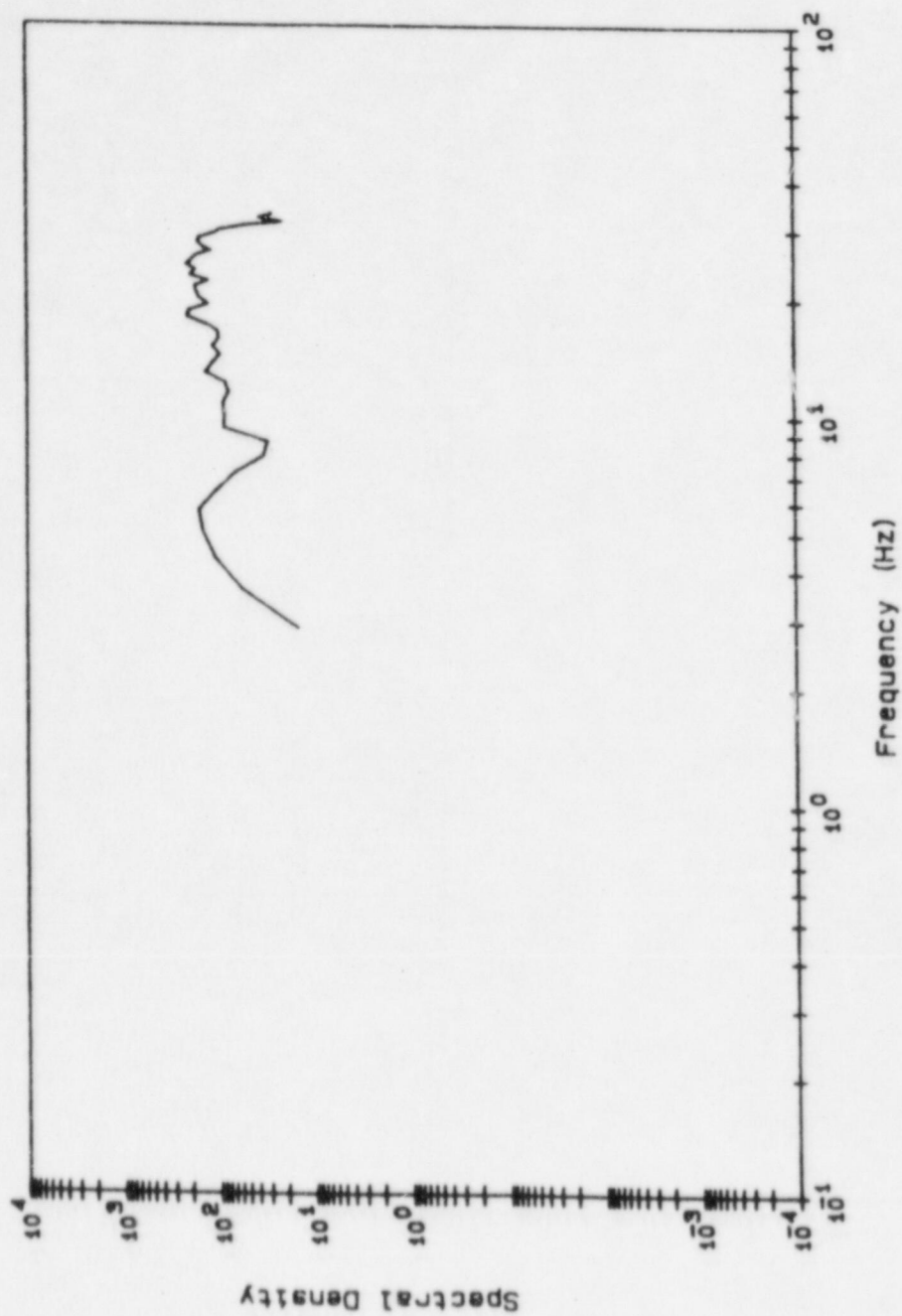


Legend:
Perry (1986)
Vertical _____
NRC Proposed Target - - - - -

Notes:

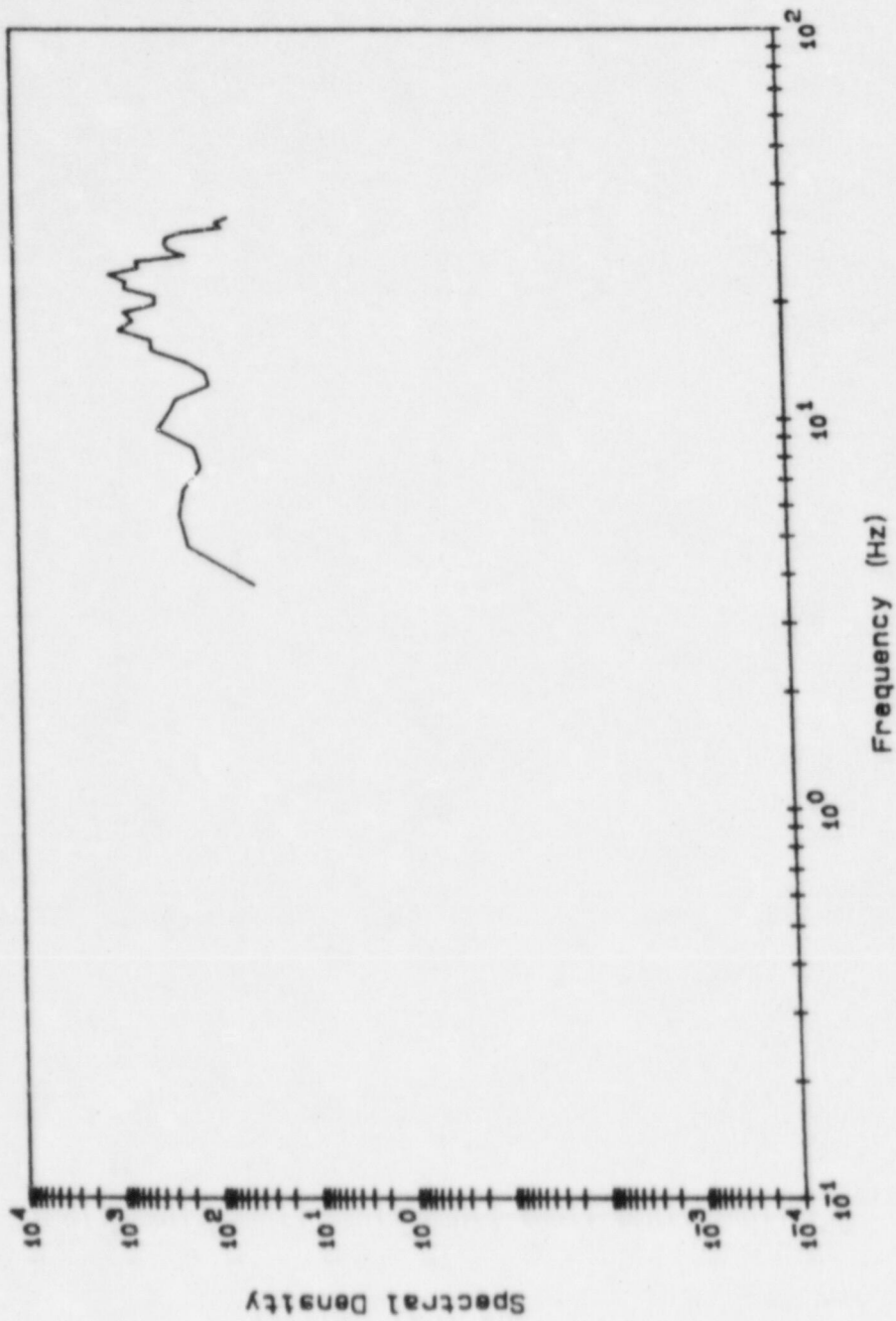
All spectral density values in units of
(in/min)/(gms²/s)

Fig. 4.11: Power Spectral Density Function, Perry (1986) Normalized to 1. g versus NRC Proposed Target (c) Vertical



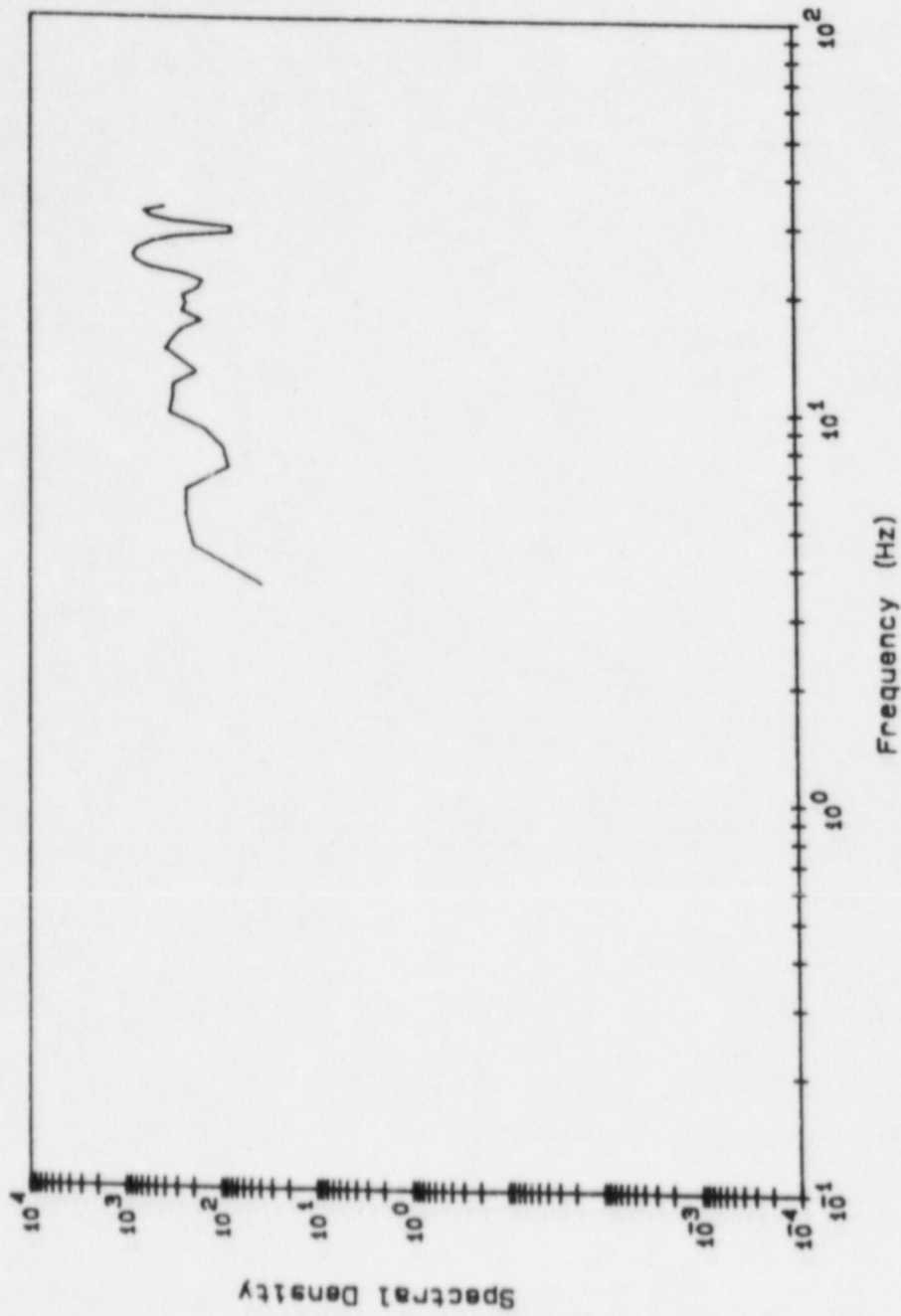
Notes:
 All spectral density values in units of
 (in/in) / (s²/s)

Fig. 4.12: Power Spectral Density Function, Mitchell Lake Rd
 (a) 18 from North



Notes:
 All spectral density values in units of
 (inwin) / (s²ms)

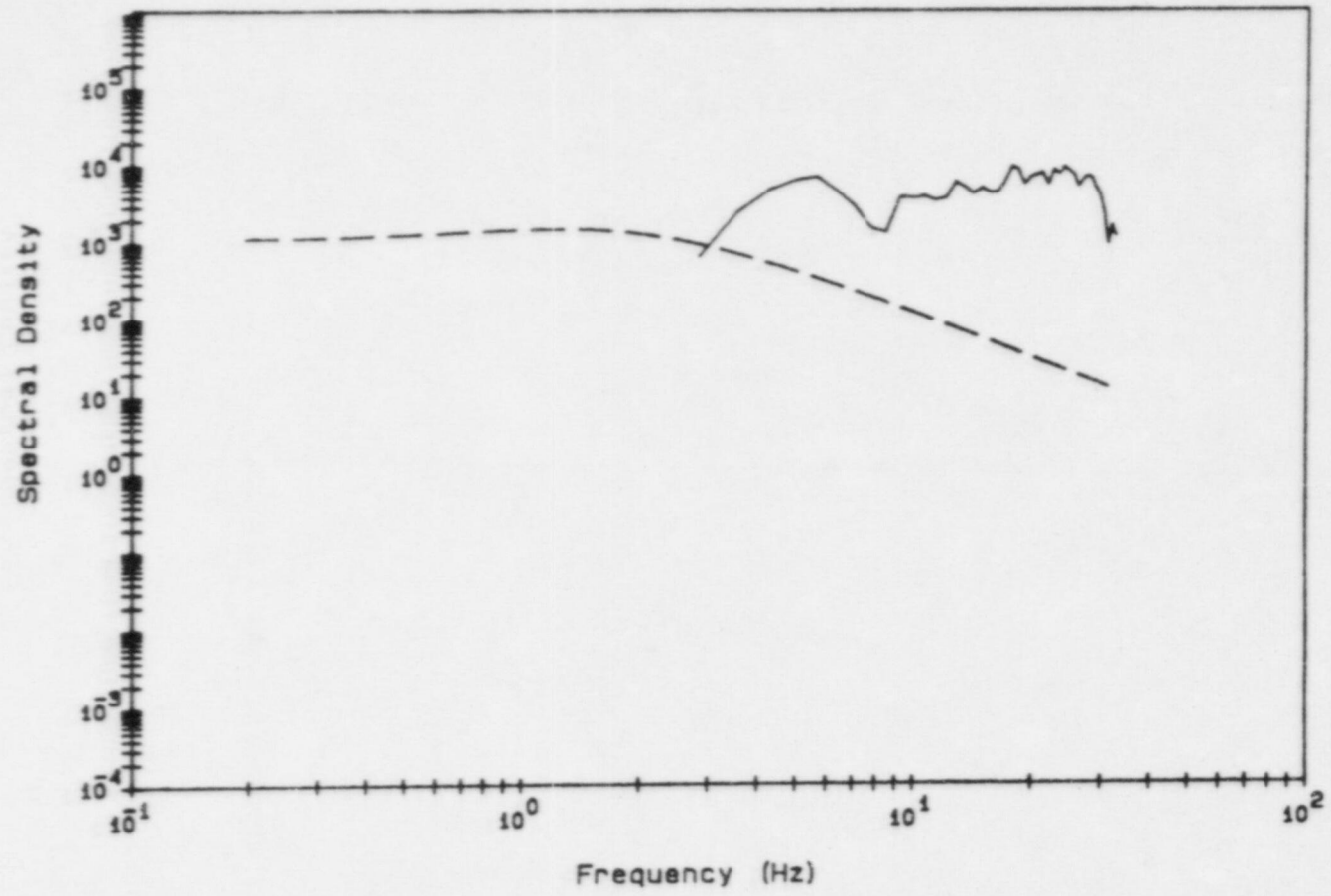
Fig. 4.12: Power Spectral Density Function, Mitchell Lake Rd
 (b) 288 from North



Notes:

All spectral density values in units of
(in²/in) / (s²/s²)

Fig. 4.12: Power Spectral Density Function, Mitchell Lake Rd
(c) Vertical



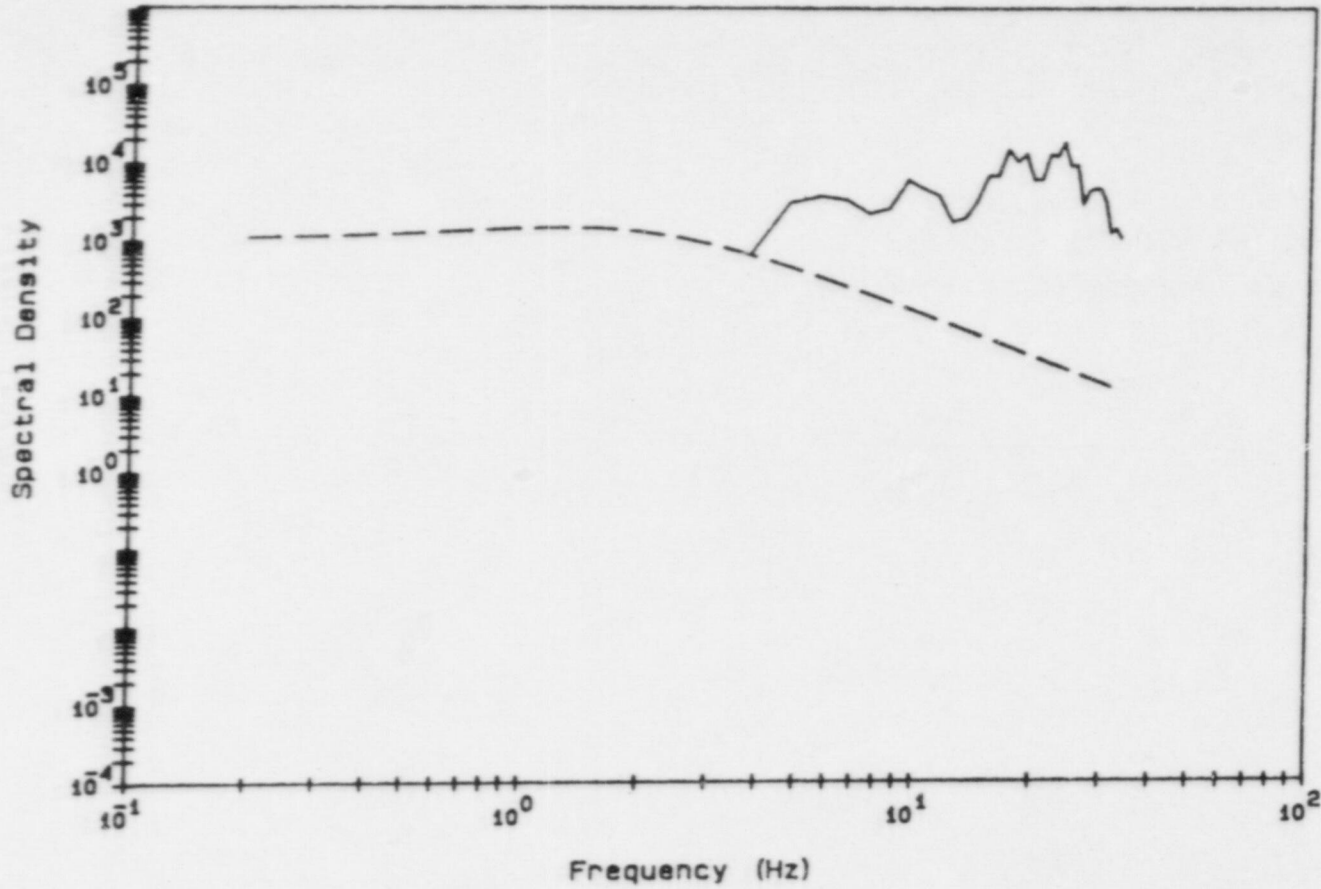
Legend:

Mitchell Lake Rd
 18 from North _____
 NRC Proposed Target - - - - -

Notes:

All spectral density values in units of
 $(in/in) / (s/s/s)$

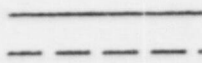
Fig. 4.13: Power Spectral Density Function, Mitchell Lake Rd
 Normalized to 1.g vs. NRC Proposed Target (a) North



Legend:

Mitchell Lake Rd
288 from North

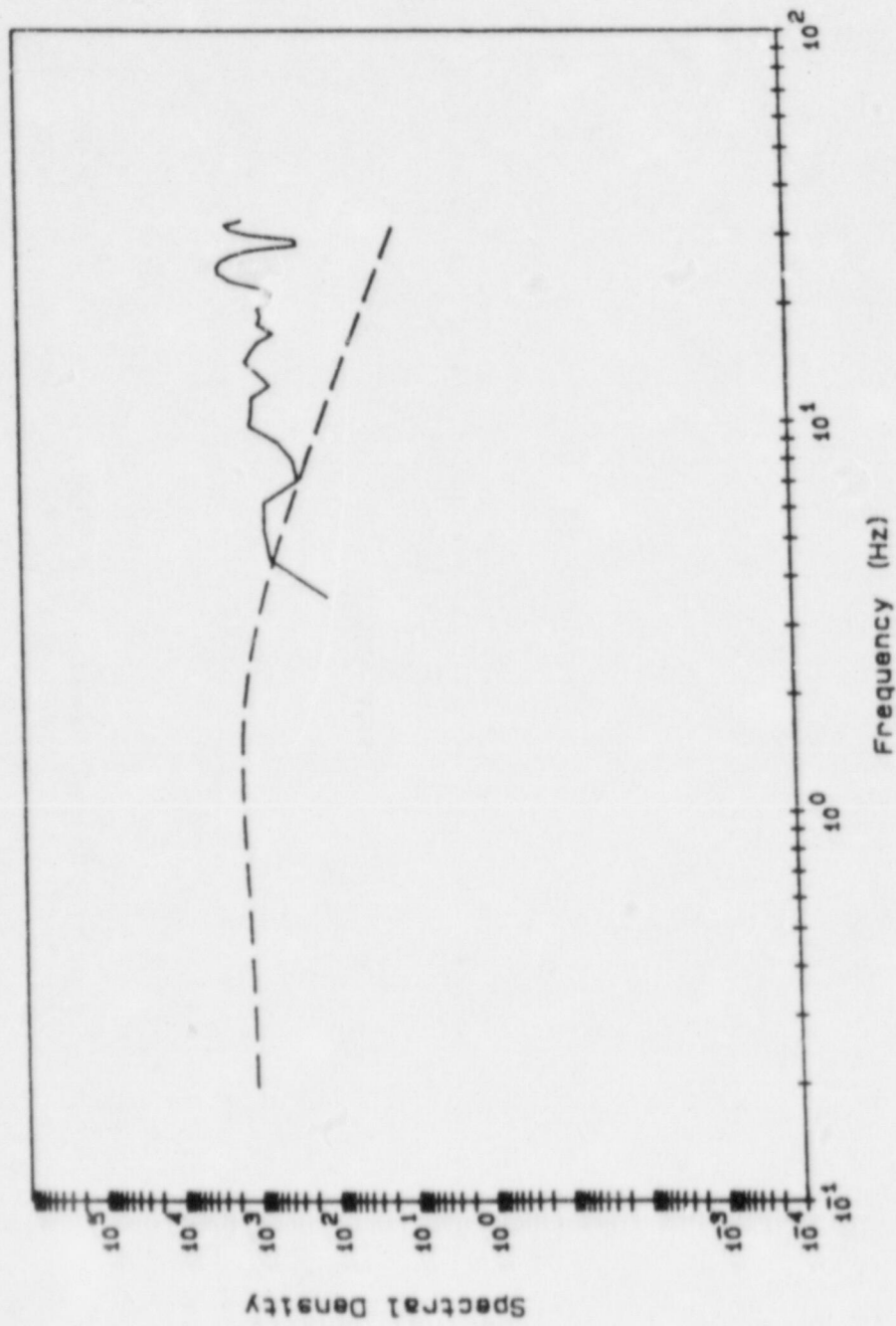
NRC Proposed Target



Notes:

All spectral density values in units of
(in/in) / (s*s*s)

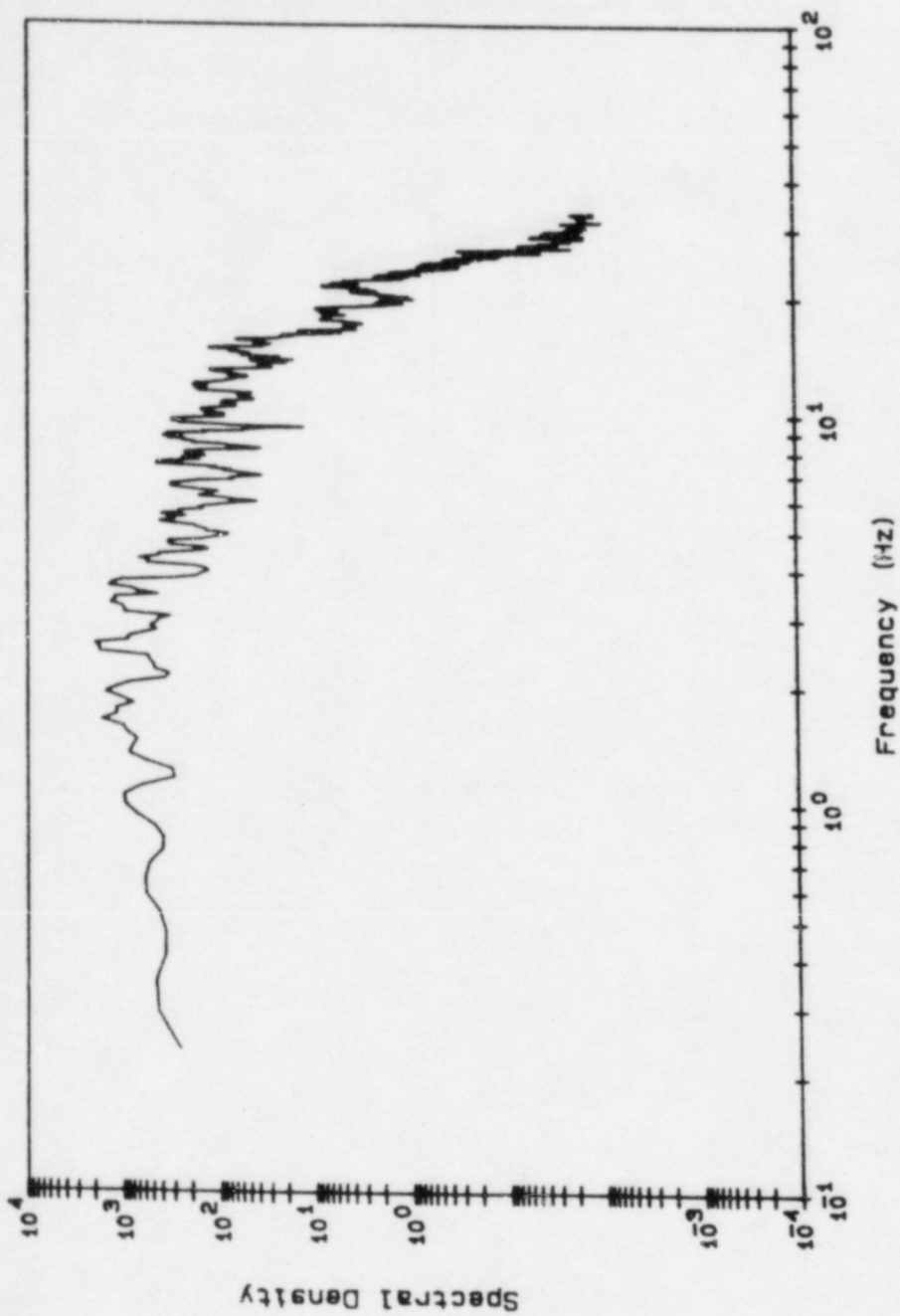
Fig. 4.13: Power Spectral Density Function, Mitchell Lake Rd
Normalized to 1.g vs. NRC Proposed Target (b) West



Legend:
Mitchell Lake Rd Vertical _____
NRC Proposed Target - - - - -

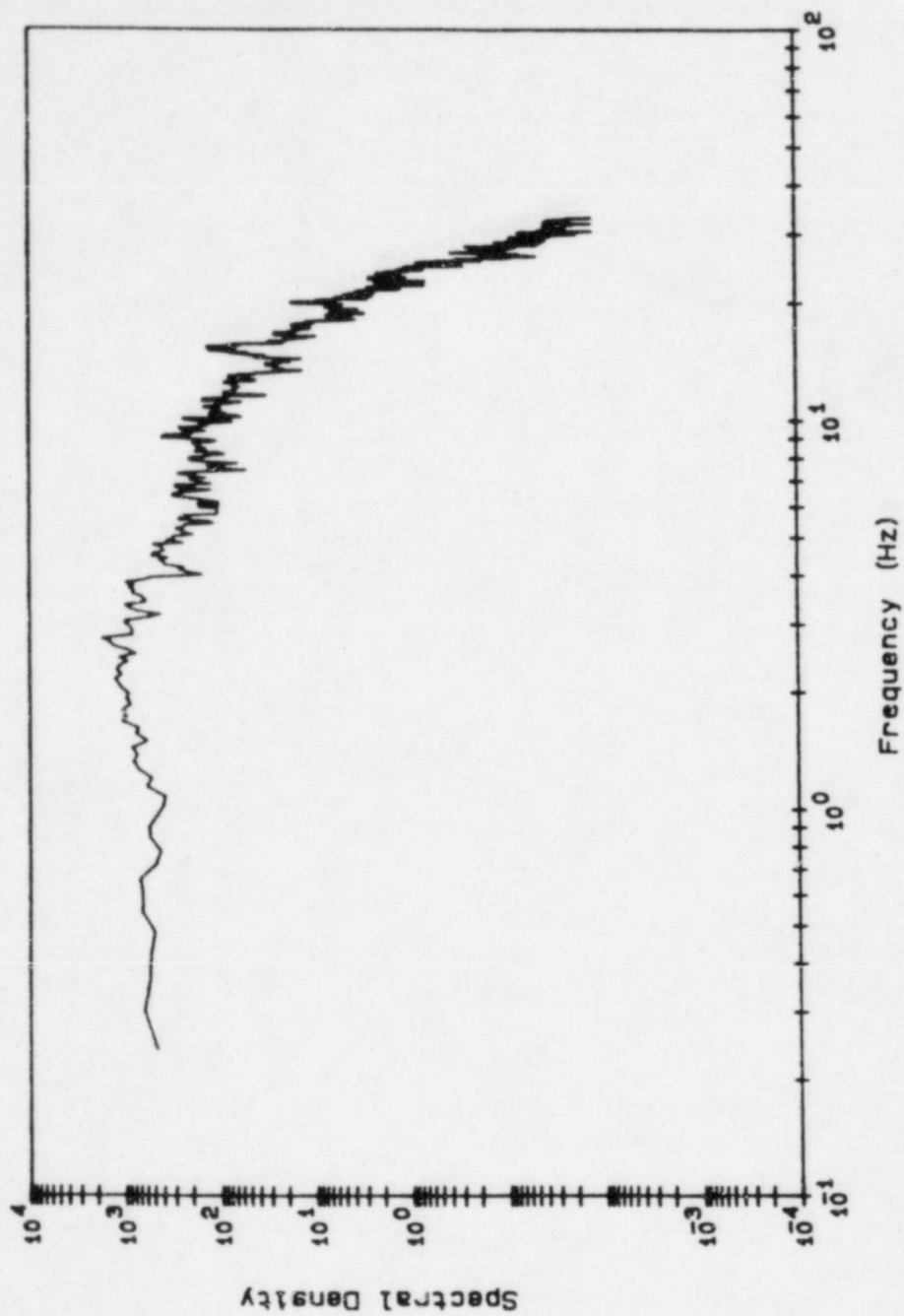
Notes:
All spectral density values in units of (in*in)/(s*ms)

Fig. 4.13: Power Spectral Density Function, Mitchell Lake Rd Normalized to 1.g versus NRC Proposed Target (c) Vertical



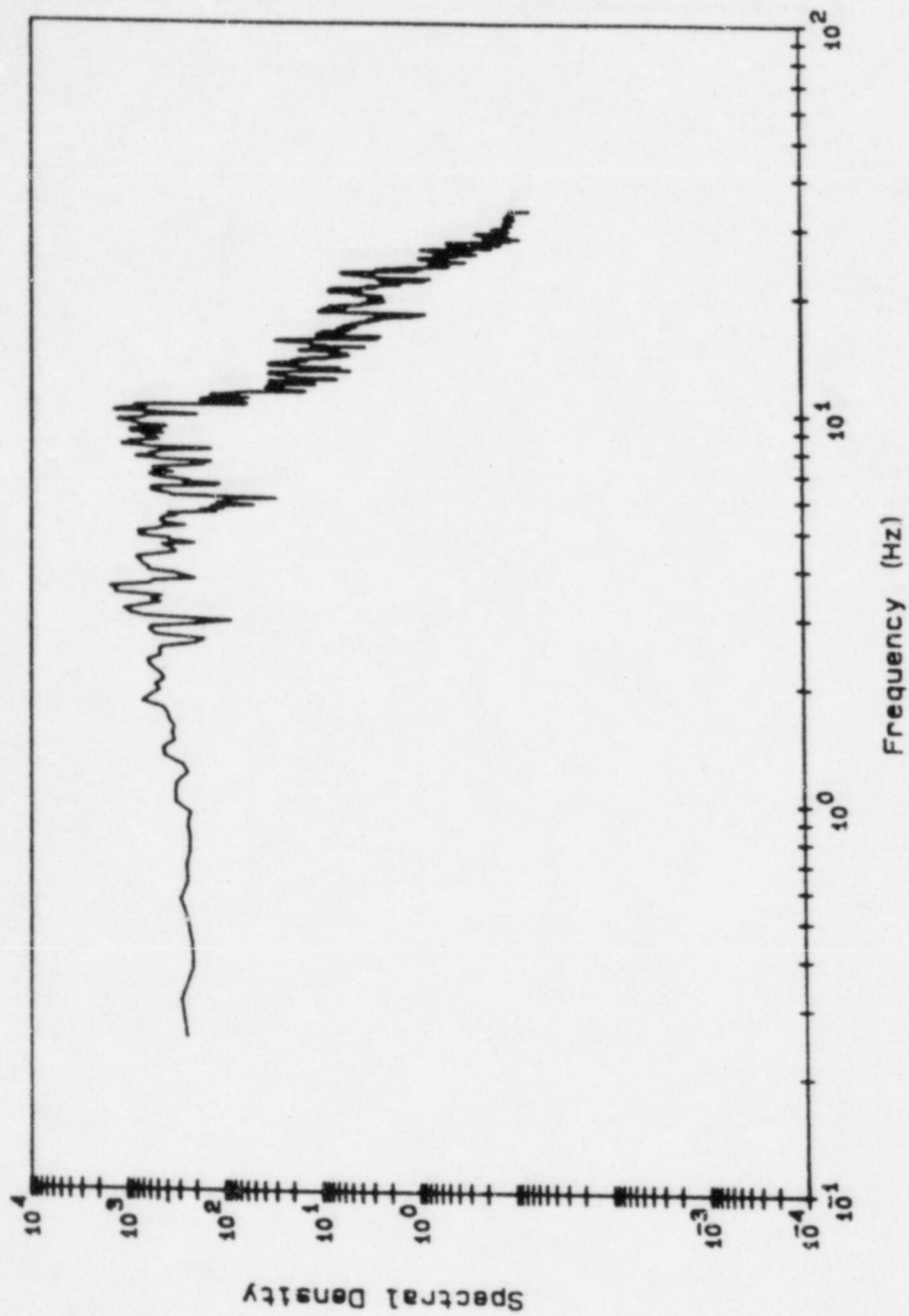
Notes:
 All spectral density values in units of
 (in*in)/(s*s*s)

Fig. 4.14: Power Spectral Density Function. Perry Design
 (a) North - South



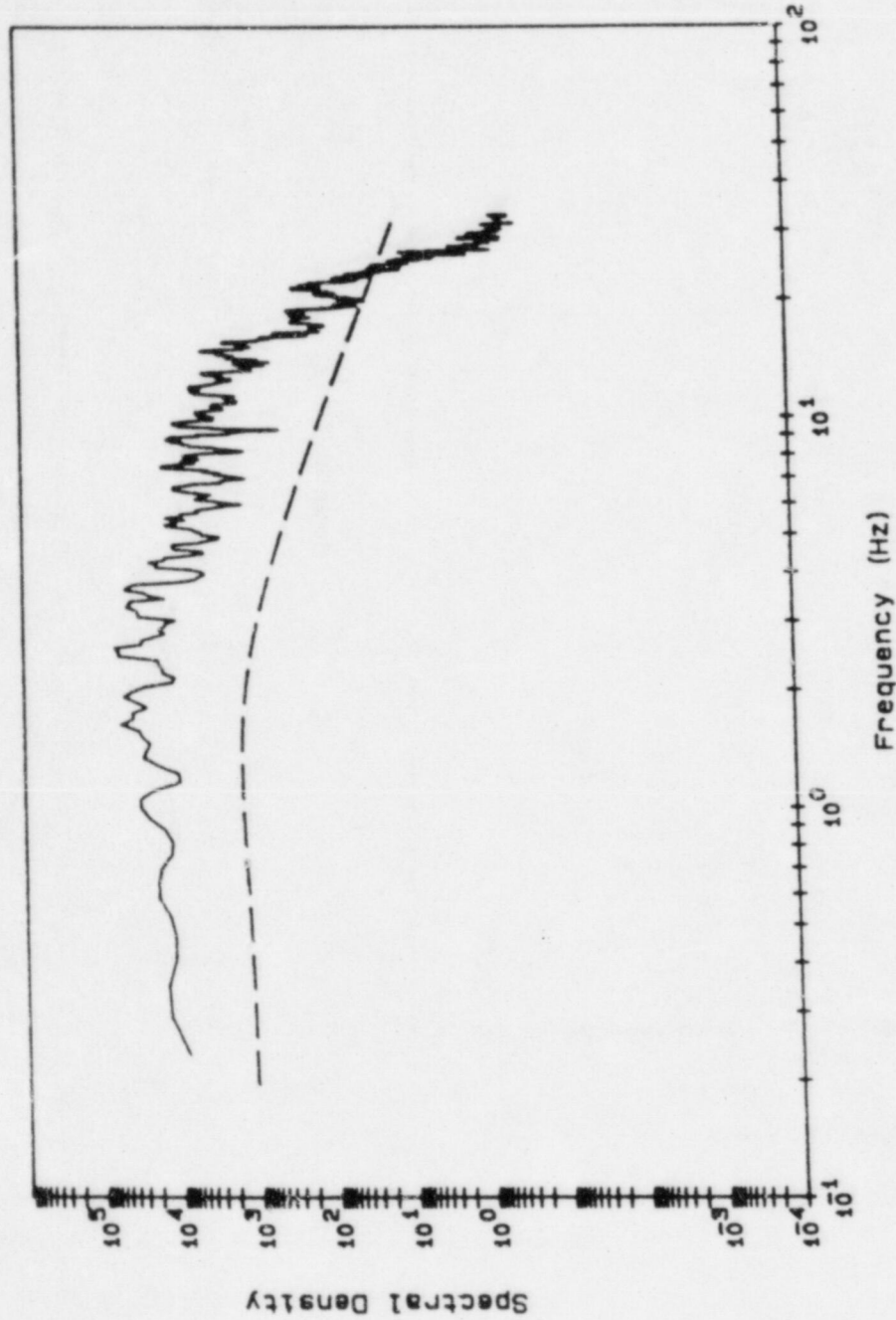
Notes:
 All spectral density values in units of
 (in/in) / (GMS/GS)

Fig. 4.14: Power Spectral Density Function. Perry Design
 (b) East - West



Notes:
 All spectral density values in units of
 (in/in) / (g*ms)

Fig. 4.14: Power Spectral Density Function, Perry Design
 (c) Vertical



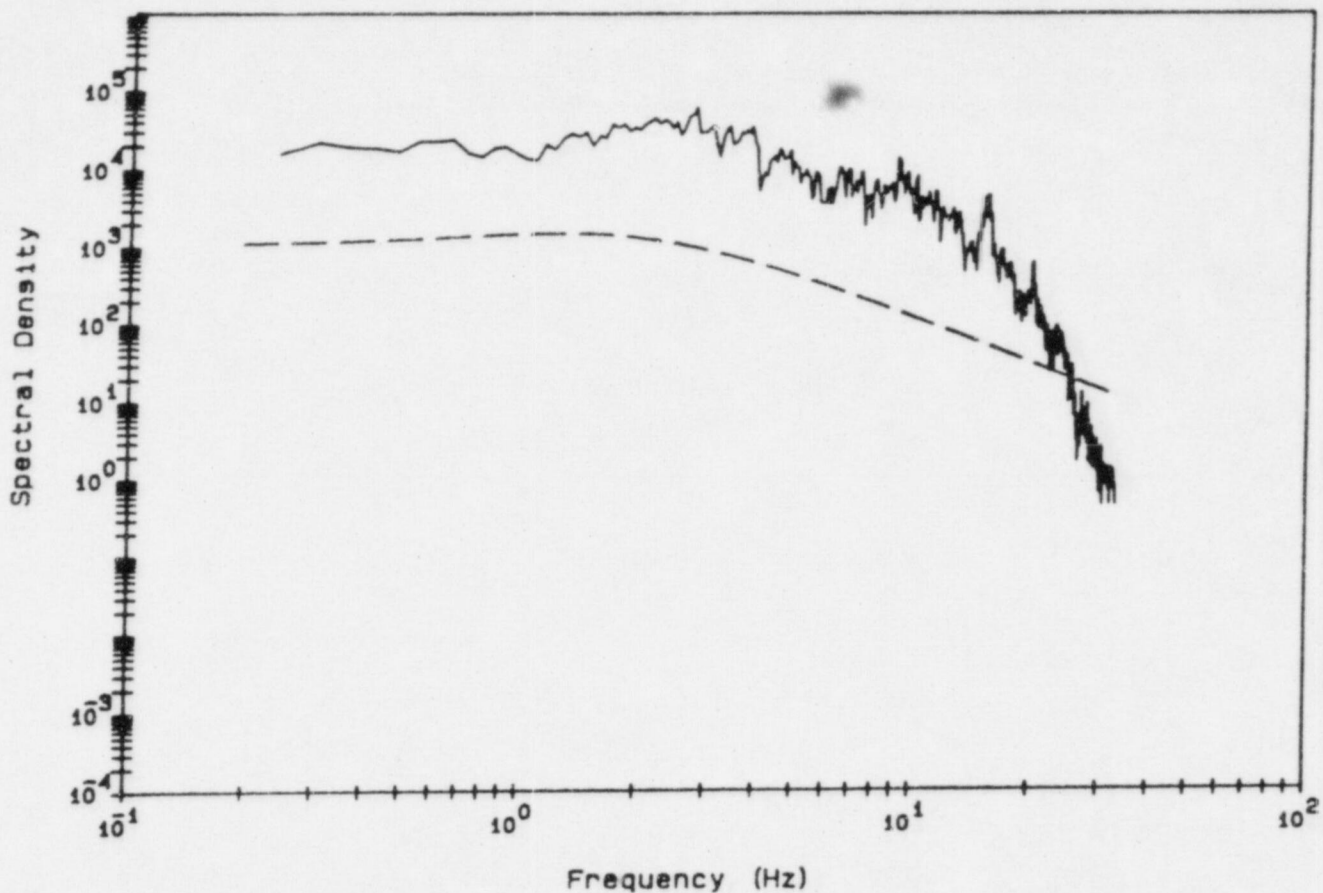
Legend:

- Perry Foundation Design N - S _____
- NRC Proposed Target - - - - -

Notes:

All spectral density values in units of (in/in) / (g*sqrt(s))

Fig. 4.15: Power Spectral Density Function, Perry Design Normalized to 1.0 g versus NRC Proposed Target (a) North - South



Legend:

Perry Foundation

Design E - W

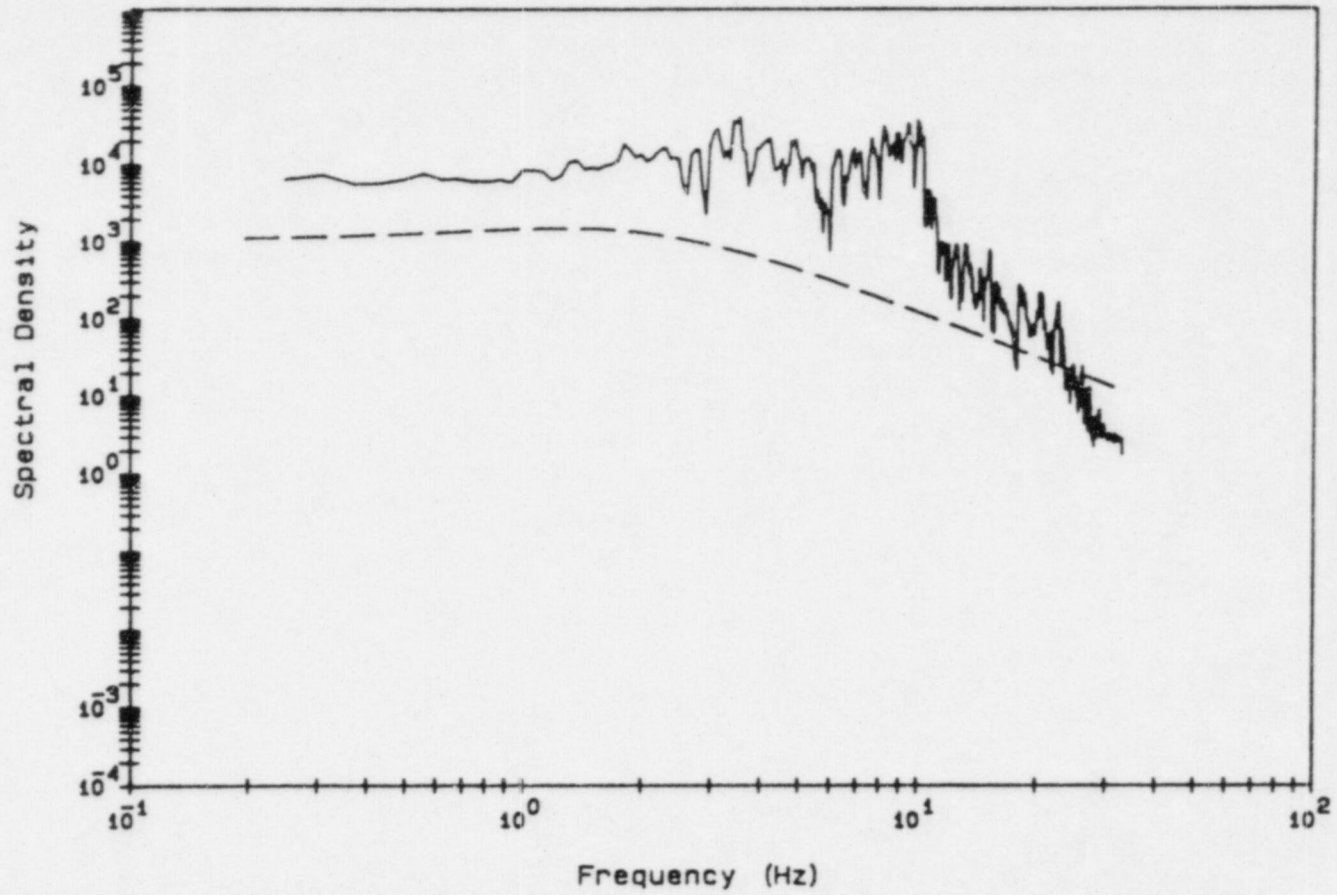
NRC Proposed Target



Notes:

All spectral density values in units of
 $(\text{in}^2/\text{s}^2)/(\text{cm}^2/\text{s}^2)$

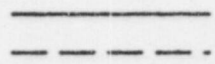
Fig. 4.15: Power Spectral Density Function, Perry Design Normalized to 1. g versus NRC Proposed Target (b) East - West



Legend:

Perry Foundation
Design Vertical

NRC Proposed Target



Notes:

All spectral density values in units of
(in*in) / (s*s*s*s)

Fig. 4.15: Power Spectral Density Function, Perry Design Normalized to 1. g versus NRC Proposed Target (c) Vertical

4.4 Nonlinear Response of Single Degree-of-Freedom Systems

Past history as documented by Kennedy et al. (Ref. 4) has shown that linear elastic analysis has not led to good predictions of structure behavior when failure is the principal interest. In particular, the damage potential of earthquakes is not well-described by low-damped elastic response spectra. Kennedy et al. indicated that well-designed structures can easily survive ground motions 2.5 times the motion that would cause yield in structural members. Inelastic response of shear walls begins as soon as extensive concrete cracking occurs. For heavily reinforced shear walls (steel percentages of about one percent) typical of those found in nuclear power plants, inelastic behavior occurs before the code specified minimum capacity is reached. Kennedy et al. assessed the damage potential of ground motions considering nonlinear behavior of simple single degree-of-freedom systems. Their investigation sought scale factors by which earthquake records must be scaled to induce specified levels of nonlinear deformation. Two levels of nonlinear deformation were considered, as defined by ductility ratios of 1.85 and 4.27. A ductility level of about 1.85 represents a reasonable lower estimate of the inelastic deformations which would occur in a shear wall designed for static lateral loads to the ACI - 349 code ultimate capacity. Current elastic design analysis methods when carried to code ultimate capacities are judged to lead to roughly this level of inelastic deformation.

Kennedy et al. analyzed single degree-of-freedom models representative of the fundamental frequency characteristics of stiff structures with elastically calculated frequencies varying from 2.14 Hz to 8.54 Hz. The inelastic behavior of these models was described by a shear wall model exhibiting stiffness degradation after yield and pinching of the hysteresis loop during loading direction reversal. The stiffness of the shear wall model beyond yield was taken to be 10% of the elastic stiffness. A modified version of the computer program DRAIN 2D as described in Ref. 13 was used to perform the analyses. The eleven

recorded motions and the artificial acceleration time history were considered in their study. A result of their analyses was scale factors which must be applied to each record and for each structure model to achieve deformations corresponding to ductility ratios of 1.85 and 4.27. The resulting scale factors may be used to scale the records up to achieve the desired result or their inverse may be used to scale down the record and estimate an effective peak acceleration. Note the scale factors are dependent on the specific records, the desired ductility level, and the structural model elastic frequencies.

A similar study was performed here to assess the potential effects of the Perry (1986) recorded motions on the Perry structures. The study followed the procedure used by Kennedy et al. (Ref. 13). Two single degree-of-freedom systems were considered. Their elastic frequencies were 5.4 Hz. and 8.9 Hz. approximating the fundamental frequencies of the Perry drywell and the Perry auxiliary building, respectively. These modes were considered representative of horizontal behavior. In the present study six records were considered--the two horizontal components of the Perry (1986) records, the two horizontal components of the Mitchell Lake Road, New Brunswick (1982) records, and the two horizontal components of the Perry foundation design motion. The nonlinear shear wall stiffness element described above was used. The yield force value was developed based on the following assumptions. The structure was designed to a force level corresponding to the spectral acceleration of the design ground response spectra (Regulatory Guide 1.60) at the fundamental frequency of the structure and for 7% damping. The design force level leads to a deformation corresponding to 1.85 times the yield displacement. The stiffness of the shear wall model beyond yield is 10% of the elastic stiffness. Nonlinear time history analyses were then performed as described in Ref. 13. The analyses were performed for the portions of the time histories defined by the strong motion duration T'_D . A series of nonlinear analyses was then performed to calculate the scale factors necessary to scale the records and achieve nonlinear deformation corresponding to 1.85 times the yield displacement, i.e., to achieve the deformations expected when the design level forces are

deformation corresponding to 1.85 times the yield displacement, i.e., to achieve the deformations expected when the design level forces are reached for the records considered. Table 4.4 contains the results. For the 5.4 Hz structure, the Perry (1986) N-S and E-W components would need to be scaled by 5.3 and 5.5 respectively to induce design level deformations in the structure. Ground motion acceleration time histories identical to the Perry (1986) recorded motions must be scaled to peak accelerations of 0.95g for the N-S component and 0.57g for the E-W component to induce design level deformations in the 5.4 Hz simple structural representation. Alternatively, an effective peak acceleration of these records and for the 5.4 Hz structural model would be 0.034g and 0.018g for the N-S and E-W components, respectively.

Table 4.4 presents results for the twelve cases considered. Note, the definition of the scale factor calculated here is slightly different than the scale factor reported in Ref. 13. Here, both elastic and inelastic response contribute to the scale factor. The elastic portion scales the 7% damped spectral acceleration of the record of interest at the frequency of interest to the design response spectra. The inelastic portion is the scale factor necessary to achieve the specified level of nonlinear deformation. The inelastic portion varied from 1.57 to 2.16 for the recorded motions (Perry, 1986 and Mitchell Lake Road, New Brunswick, 1982). The scale factors for the Perry foundation design motion are near one.

Table 4.4

Scale Factors to Achieve Ductility Ratios of 1.85 and
Corresponding Effective Peak Accelerations for Selected
Acceleration Time Histories

Earthquake Record	Model Structure Frequency			
	5.4 Hz		8.9 Hz	
	Scale Factor	Eff Accel. (g)	Scale Factor	Eff Accel. (g)
Perry, 1986 (N-S)	5.3	.034	6.7	.027
Perry, 1986 (E-W)	5.5	.018	4.3	.024
Mitchell Lake Rd New Brunswick, 1982, (18° from North)	4.9	.031	3.3	.046
Mitchell Lake Rd New Brunswick, 1982 (288° from North)	3.6	.067	2.8	.086
Perry Foundation Design Motion (N-S)	1.2	.15	1.1	.16
Perry Foundation Design Motion (E-W)	1.2	.14	1.1	.15

5. SUMMARY AND CONCLUSIONS

A summary of the results of the independent studies performed to evaluate the effects of the January 31, 1986 Ohio earthquake on the Perry Nuclear Power Plant structures is as follows:

- The total energy of the Perry (1986) recorded motions was calculated for each of the three components. They were compared with the energy in fourteen other recorded motions (eleven from a previous study and three for the Mitchell Lake Road, New Brunswick, 1982 earthquake), and four artificial time histories (one from a previous study and the three components of Perry foundation design motion). The energy in the Perry (1986) records was less than that in any of the records considered--much less for the longer duration motions. Along with the total energy, the strong motion durations, power, and RMS accelerations of the records were estimated. Relatively short duration, low power, and low RMS accelerations were calculated.
- Response spectra and PSD functions of the Perry (1986) three components of motion, the Mitchell Lake Road, New Brunswick (1982) three components of motion, and the Perry foundation design motions were calculated. The PSDs showed the concentration of energy in the high frequency range for the two sets of recorded motions. Comparing the normalized PSDs to the US NRC target PSD shows their values to be above the target in the high frequency range.
- Nonlinear analyses were performed on single degree-of-freedom models representing the fundamental horizontal

frequency of the drywell (5.4 Hz) and the auxiliary building (8.9 Hz). Scale factors were calculated which when applied to the input motions would achieve a nonlinear deformation in the simple model of 1.85 times the yield displacement. This level of nonlinearity is expected when typical concrete shear walls are loaded to their design level. The scale factors calculated here included an elastic and inelastic portion. The scale factors for the Perry (1986) records were: (5.3, 5.5) for the (N-S, E-W) components and the 5.4 Hz model; and (6.7, 4.3) for the (N-S, E-W) components and 8.9 Hz model. A measure of the effective peak acceleration of these records is the instrumental peak accelerations divided by the scale factors.

- The SSI and structure response aspect of the investigation confirmed that SSI was not an important phenomenon for the rock-founded Perry structures. Neither kinematic nor inertial interaction appears to have occurred to a significant extent. The effect of non-vertically incident waves on the response of the containment vessel was assessed. Based on in-structure response spectra in the containment vessel, vertically incident waves is the most likely wave propagation mechanism.
- Comparing predicted and measured response in the containment vessel at elevation 688', the frequency characteristics of the N-S, E-W, and vertical directions correspond well with one exception, i.e., the low frequency E-W component. Response predictions are within 30-35% with variations being primarily an under-prediction for E-W and vertical. It is well-recognized that peak spectral amplifications are uncertain, hence, the match is adequate.

Based on these studies, a review of CEI's evaluations and other technical information, the January 31, 1986 earthquake is judged to have had an insignificant effect on the Perry Nuclear Power Plant structures. Further, it is judged that the Perry seismic analysis models adequately predict the behavior of the reactor building when subjected to this event. Although, it is recognized that a portion of the high frequency motion recorded on the containment vessel may have been due to secondary effects, such as polar crane vibration or impact, and this remains an open question. Finally, the plant design of the structures is judged to be acceptable and unaffected by the event.

The effect of the January 31, 1986 on equipment was assessed by the US NRC in a separate effort.

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Supplement No. 10 to the Safety Evaluation Report (NUREG-0887) on the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or (CAPCO), as applicant's and owners for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441) has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This supplement reports the status of certain issues and action items that had not been resolved or completed at the time of publication of the Safety Evaluation Report and Supplement Nos. 1 through 9 to that report.

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