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The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably, because of significant changes in design criteria and methods for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants should be reassessed to determine whether requalification is necessary.

The objective of technical studies performed under the Task Action Plan A-46 was to establish an explicit set of guidelines and acceptance criteria to judge the adequacy of equipment under seismic loading at all operating plants, in lieu of requiring qualification to the current criteria that are applied to new plants.

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Seismic Qualification of Equipment in Operating Nuclear Power Plants

Unresolved Safety Issue A-46

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

T. Y. Chang

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Seismic Qualification of Equipment in Operating Nuclear Power Plants

Unresolved Safety Issue A-46

Draft Report for Comment

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Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably, because of significant changes in design criteria and methods for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants should be reassessed to determine whether requalification is necessary.

The objective of technical studies performed under the Task Action Plan A-46 was to establish an explicit set of guidelines and acceptance criteria to judge the adequacy of equipment under seismic loading at all operating plants, in lieu of requiring qualification to the current criteria that are applied to new plants.

This report summarizes the work accomplished on USI A-46 by the Nuclear Regulatory Commission staff and its contractors, Idaho National Engineering Laboratory, Southwest Research Institute, Brookhaven National Laboratory, and Lawrence Livermore National Laboratory. In addition, the collection and review of seismic experience data by the Seismic Qualification Utility Group and the review and recommendations of a group of seismic consultants, the Senior Seismic Review Advisory Panel, are presented. Staff assessment of work accomplished under USI A-46 leads to the conclusion that the use of seismic experience data provides the most reasonable alternative to current qualification criteria. Consideration of seismic qualification by use of experience data was a specific task in USI A-46. Several other A-46 tasks serve to support the use of an experience data base.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
ANSI	American National Standards Institute
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CFR	Code of Federal Regulations
CQC	complete quadratic combination
DC	U.S. Department of Commerce
EERI	Earthquake Engineering Research Institute
EQ	environmental qualification
EQE	EQE Incorporated
ERS	experience response spectra
FRF	frequency response function
GDC	General Design Criterion
HVAC	heating, ventilating, and air conditioning
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory (EG&G Idaho, Inc.)
JAERI	Japan Atomic Energy Research Institute
LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
MCC	motor control center
MMI	modified Mercalli intensity
MPF	modal participation factor
NAS	National Academy of Sciences
NCEE	U.S. National Conference on Earthquake Engineering
NCEL	U.S. Naval Civil Engineering Laboratory
NEMA	National Electrical Manufacturers Association
NRC	Nuclear Regulatory Commission
OBE	operating basis earthquake
PRA	probabilistic risk assessment
PSD	power spectra density
PWR	pressurized water reactor
RES	Richardson Engineering Services, Inc.
RG	Regulatory Guide
RRS	required response spectra
SEP	Systematic Evaluation Program
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSE	safe shutdown earthquake
SSRAP	Senior Seismic Review Advisory Panel
SWRI	Southwest Research Institute
TAP	task action plan
UBC	Uniform Building Code
USI	unresolved safety issue

1 INTRODUCTION

1.1 Background

General Design Criterion (GDC) 2 in Appendix A to Title 10 of the Code of Federal Regulations (CFR) Part 50 (10 CFR 50) states that structures, systems, and components important to safety in nuclear power plants shall be designed to withstand the effects of natural phenomena, such as earthquakes, without a loss of capability to perform their safety functions. Section III of Appendix B to 10 CFR 50 states that design control measures shall provide for verifying or checking the adequacy of design by the performance of a suitable testing program. It also requires that this program include suitable qualification testing under the most adverse design conditions. These requirements point to the need for seismic qualification of safety-related electrical and mechanical equipment to ensure structural integrity and functional capability during and after a seismic event. Current criteria and methods of compliance are in the Nuclear Regulatory Commission (NRC) Revision 2 to Standard Review Plan (SRP) Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment" (NUREG-0800) (NRC, July 1981)* and NRC's Regulatory Guide (RG) 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants." With some exceptions, RG 1.100 basically endorses the Insitute of Electric and Electronics Engineers (IEEE) Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

On the basis of the requirements and recommendations of these criteria and methods, equipment is seismically qualified today by analysis and/or laboratory test. Analyses alone are acceptable only if the necessary functional capability of the equipment is ensured by its structural integrity. Otherwise, some testing is required using the required response spectra or required time histories for the seismic input motion to equipment. When equipment is tested, it is mounted on a shake table and subjected to certain types of excitation corresponding to a test response spectrum that envelopes the required response spectra. The equipment is tested in the operating condition. For equipment too large to fit on a shake table, a combined analysis and test procedure is used.

Since commercial nuclear power plants were first introduced, seismic qualification criteria have been changed to a significant degree. The analytical and experimental methods used to qualify equipment have also changed. Because of these changes the margins of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably and may not meet current seismic qualification criteria. Therefore, to ensure performance during and after a seismic event, seismic capability of equipment in operating plants must be reassessed.

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*References in this report are cited parenthetically by author. See Section 3, "References," for a complete citation.

It was also recognized that it may not be practical to qualify operating plant equipment using current seismic qualification criteria and methods because of (1) excessive plant down time, (2) difficulties in shipping irradiated equipment to a test laboratory, and (3) difficulties in acquiring identical vintage equipment for laboratory testing. In December 1980, the Nuclear Regulatory Commission designated "Seismic Qualification of Equipment in Operating Plants" as an unresolved safety issue (USI). The objective of USI A-46 is to develop alternative methods and acceptance criteria that can be used to assess the capability of mechanical and electrical equipment in operating nuclear power plants to perform the intended safety functions.

1.2 Description of A-46 Tasks

A task action plan (TAP) was developed for USI A-46 in the spring of 1981. Tasks for study were selected on the basis of their potential for providing reasonable alternatives to current requirements for seismic qualification. It was recognized that a utility always has the option to requalify equipment using procedures required for new plants. Only alternative procedures which provide some advantage over current requirements are likely to be used. In addition, any alternative procedure must be sufficiently rigorous to provide a level of safety comparable to that achieved by current requirements. A key element of the approach was to take advantage of experience gained by previous qualification tests and analysis, and experience with actual seismic events.

Tasks selected for study were:

(1) Identification of seismic-sensitive systems and equipment;
(2) Assessment of adequacy of existing seismic qualification;
(3) Development and assessment of in-situ testing methods to assist in qualification of equipment;
(4) Seismic qualification of equipment using seismic experience data;
(5) Development of methods to generate generic floor response spectra.

As work progressed it became increasingly apparent that Task 4, "Seismic Qualification of Equipment Using Seismic Experience Data" was the most likely alternative for assessing seismic capability, which is both economically attractive to the plant owners and acceptable from a public safety viewpoint. Lawrence Livermore National Laboratory (LLNL), under contract to NRC, conducted a feasibility study (NRC, August 1983) which concluded that use of seismic experience data is feasible and can be as effective as current qualification methods. This study is discussed in more detail later in Section 2.1.2. In addition, a utilities group, Seismic Qualification Utility Group (SQUG), in conjunction with its consultants EQE Incorporated, conducted a pilot program to independently demonstrate the feasibility of using seismic experience data. Their report was issued by EQE in September 1982. A more detailed discussion of this effort is presented in Section 2.1.3.

In March 1983, SQUG proposed to NRC management the formation of a Senior Seismic Review Advisory Panel (SSRAP) to provide consulting services and expert opinion on the use of experience data. This idea was endorsed by NRC management and SSRAP was subsequently formed in June 1983.

In February 1984, SSRAP released its report which describes the SSRAP findings and recommendations for using seismic expeience data for non-nuclear plants to evaluate seismic adequacy of equipment in operating nuclear plants. Conclusions on the use of seismic experience data including caveats and exclusions were presented in the SSRAP report. These technical findings are presented in Section 1.3 of this report. More detailed description of this study and its conclusions can be found in Sections 2.1.4 and 2.1.5 of this report.

Tasks 3 and 5, "Development and Assessment of In-situ Testing Methods To Assist in Qualification of Equipment" and "Development of Methods To Generate Generic Floor Response Spectra," play a supporting role. The emphasis on both tasks was focused to support use of an experience data base. Such emphasis is described in Sections 2.2 and 2.3 in this report.

Task 2, "Assessment of the Adequacy of Existing Seismic Qualification," was an effort to develop methods to evaluate the acceptability of qualification by procedures used before current requirements were instituted. For instance, a method was developed to assess results of a single axis test in terms of expected multiple axis response. Although Southwest Research Institute developed a procedure for such assessment, it is of limited immediate value in its present form because of the need to either know the fragility level or estimate the fragility of the equipment and know the required response spectra. It may be useful in special cases. Task 2 is described in more detail in Appendix A.

Task 1, "Identification of Seismic Risk Sensitive Systems and Equipment," was an attempt to develop, on a generic basis, a minimum equipment list. The study, performed by Brookhaven National Laboratory (BNL), was conducted on a hybrid model of a pressurized water reactor (PWR) plant and a hybrid model of a boiling water reactor (BWR) plant using a seismic probabilistic risk assessment (PRA) model. The contribution to risk of major systems and components was calculated and ordered by risk importance. Although this study did provide some insight into the risk importance of systems and components and demonstrated the effect of varying equipment fragility on overall risk, it is of limited usefulness in defining a generic equipment list. The major conclusion of the BNL study was that BNL had demonstrated a methodology that could be applied on a plant-specific basis to develop a risk-based minimum equipment list. For plants for which an existing seismic PRA model is available, it may be feasible to evaluate the necessity to qualify specific systems or components on the basis of risk contribution. This task is described in more detail in Appendix A.

1.3 A-46 Technical Findings

The principal technical finding of A-46 is that seismic experience data applied in accordance with the guidelines developed can be used to verify the seismic adequacy of mechanical and electrical equipment in operating nuclear plants. Explicit seismic qualification should be required only if seismic experience data or existing test data on similar components can not be shown to apply.

This finding is based primarily on the staff's review of the work accomplished by SQUG and SSRAP to develop a seismic experience data base and to develop guidance for its application. In addition to endorsement of the SSRAP conclusions, the staff has developed general guidance for extending the applicability of seismic experience data to other classes of components.

1.3.1 General Conclusions

The study completed by SQUG and SSRAP (SSRAP, February 1984) was limited to eight classes of equipment: motor control centers, low-voltage (480-V) switchgear, metal-clad (2.4 to 4-kV) switchgear, unit substation transformers, motor-operated valves, air-operated valves, horizontal pumps, and vertical pumps.

General conclusions of the study on these eight classes of equipment were summarized by SSRAP as follows:

(1) Equipment installed in nuclear power plants is generally similar to and at least as rugged as that installed in conventional power plants.

(2) This equipment, when properly anchored, and with some reservations, has an inherent seismic ruggedness and a demonstrated capability to withstand significant seismic motion without structural damage.

(3) For this equipment, functionality after the strong shaking has ended has also been demonstrated, but the absence of relay chatter during strong shaking has not been demonstrated.

(4) With several important caveats and exclusions, it is SSRAP's judgment that for excitations below certain seismic motion bounds, it is unnecessary to perform explicit seismic qualification of existing equipment in these eight classes for operating nuclear power plants to demonstrate functionality after the strong shaking has ended.

(5) The existing data base reasonably demonstrates the seismic ruggedness of this equipment up to these seismic motion bounds.

Furthermore, SSRAP believes that similar conclusions might apply to other classes of equipment, but such an extrapolation should only be made after a very detailed and careful review.

1.3.2 Scope of Seismic Adequacy Review

The staff concluded that it is unnecessary to verify the seismic adequacy of all plant equipment defined as seismic Class I in RG 1.29 (NRC, September 1978). This implies that only those systems, subsystems, and components required to bring the plant to a safe shutdown condition and to maintain it in that condition are important to assure safety during and after a seismic event. The scope of the seismic verification, therefore, can be limited to the minimum equipment necessary to perform the functions related to plant safe shutdown. This approach is consistent with seismic reviews conducted by the Systematic Evaluation Program (SEP) and with current staff thinking on simultaneous occurrence of a loss-of-coolant accident (LOCA) with a seismic event.

The initial intent of Task 1 of the A-46 TAP, "Identification of Seismic Risk Sensitive Systems and Equipment" was to develop a generic risk-ordered list of equipment. This effort however did not result in an equipment list that could be generally applied to operating plants. This effort is described in Appendix A.

The staff developed assumptions related to defining the equipment scope and guidance on required plant functions. The assumptions which dictate the systems and equipment that will be needed are:

(1) The seismic event does not cause a LOCA.

(2) The LOCA will not be postulated to occur simultaneously with or during a seismic event.

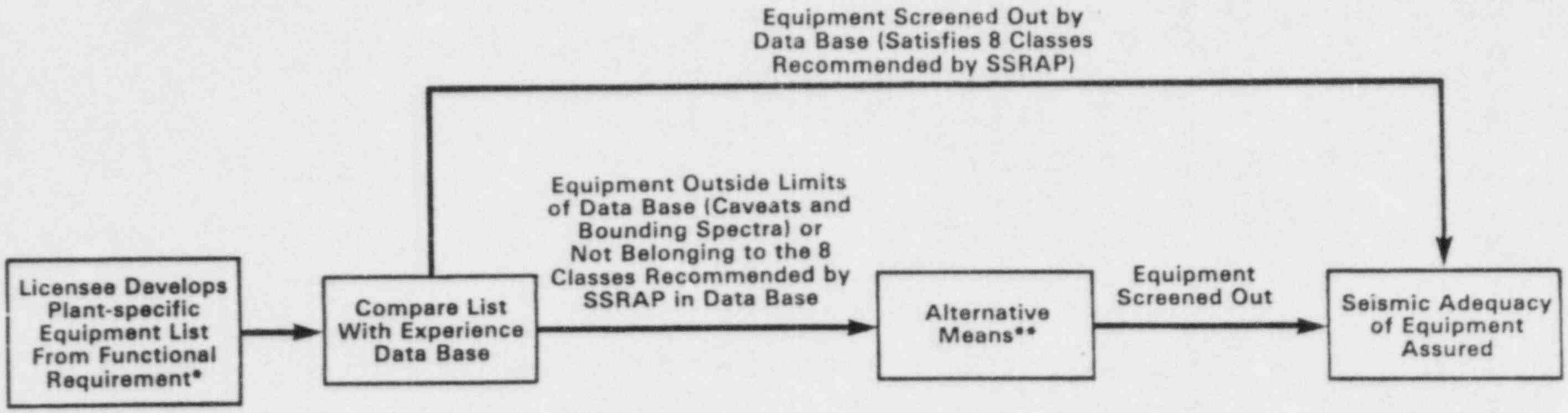
(3) Offsite power will be lost during and/or after a seismic event.

Given these basic assumptions, the scope of systems and equipment needed is less than currently required for new licenses.

1.3.3 Equipment Outside Applicability of Seismic Experience Data Base

Not all equipment required following a seismic event is within the defined scope of the experience data. The staff believes that an extension of the data base to cover additional classes of equipment or to extend the limits for the original eight equipment classes is feasible. In addition, other procedures can be developed. Suggested steps if equipment is not covered by the existing data base are listed below and shown graphically on Figure 1.3-1.

(1) Extend experience data to include additional classes.
(2) Find test data which are applicable to equipment.
(3) Develop other evidence of seismic ruggedness.
(4) Test prototype.
(5) Perform analysis and/or in-situ test to show seismic ruggedness or similarity with data base or test data (see Section 2.2).
(6) Perform simple modification to provide similarity with data base.
(7) Replace equipment with qualified equipment.
(8) Qualify to current requirements.



*From Section 1.3.2.

**1. Extend experience data which are comparable to SSRAP guidance and caveats.
2. Find test data which are applicable to equipment.
3. Develop other evidence of seismic ruggedness.
4. Test prototype.
5. Perform analysis and/or in-situ test to show seismic ruggedness or similarity with data base or test data.
6. Simple modification to provide similarity with data base.
7. Replacement by qualified equipment.
8. Qualify to current requirement.

Figure 1.3-1 USI A-46 screening procedure

2 SUMMARY OF TECHNICAL WORK WHICH SUPPORTS USI A-46 TECHNICAL RESOLUTION

As mentioned in Section 1.2, of the five tasks selected for study in USI A-46, the most practical proved to be Task 4 "Seismic Qualification of Equipment Using Seismic Experience Data." Tasks 3 and 5, "Development and Assessment of In-Situ Testing Methods To Assist in Qualification of Equipment" and "Development of Methods To Generate Generic Floor Response Spectra," play a supporting role. The emphasis on these two tasks was focused to support use of an experience data base. In the following paragraphs these three tasks are described. The other two tasks included in the task action plan did not directly contribute to resolution of USI A-46. They are discussed in Appendix A.

2.1 Seismic Qualification of Equipment Using Seismic Experience Data Base

2.1.1 Background

It is well known that many non-nuclear power plants and industrial facilities containing equipment similar to that in nuclear power plants experienced major earthquakes. It is also recognized that during the course of qualifying safety-related equipment for licensing nuclear plants in the last decade or so, numerous equipment items were tested on shake tables in laboratories for seismic capability. Therefore, there is a wealth of information regarding seismic experience that potentially can be utilized as an alternative to formal qualification of equipment in operating plants. To use this information the data must be collected and organized, and guidelines and criteria must be developed. Two independent efforts to develop a seismic experience data base were initiated. The SQUG (Table 2.1-1) conducted a pilot program, "Program for Development of an Alternative Approach to Seismic Equipment Qualification." The pilot program was completed by the SQUG contractor, EQE Incorporated. Results of this pilot program were recorded in a two-volume report issued in September 1982. A second effort was initiated by the NRC staff, with LLNL as the contractor. NRC published "Correlation of Seismic Experience Data in Non-Nuclear Facilities With Seismic Equipment Qualification in Nuclear Plants" in August 1983.

The results of both studies confirmed the feasibility of utilizing non-nuclear seismic experience data to verify seismic adequacy of equipment in operating nuclear power plants.

A group of seismic consultants, the Senior Seismic Review Advisory Panel (SSRAP) was formed by the SQUG in June 1983 to provide consulting services and expert opinion on the use of experience data. The staff worked closely with SQUG and SSRAP to develop an acceptable approach to using seismic experience data.

In February 1984, SSRAP released its report which describes the SSRAP findings and recommendations (SSRAP, February 1984). Conclusions on the use of seismic experience data including caveats and exclusions were presented in the report. The study included motor control centers, low-voltage (480-V) switchgear, metal-clad (2.4 to 4-kV) switchgear, unit substation transformers, motor-operated

Table 2.1-1 Seismic Qualification Utility Group members

Baltimore Gas and Electric Company
Commonwealth Edison Company
Consolidated Edison Company
Consumers Power Company
Detroit Edison Company
Duke Power Company
ENEL (Italy)
Florida Power Corporation
GPU Nuclear Corporation
Maine Yankee Atomic Power MPR Associates, Inc.
Nebraska Public Power District
Power Authority of State of New York
Northeast Utilities
Northern States Power Company
Philadelphia Electric Company
Rochester Gas and Electric Company
Sacramento Municipal Utility District
Southern California Edison Company
Wisconsin Electric Power Company
Yankee Atomic Electric Company

valves, air-operated valves, horizontal pumps and vertical pumps. General conclusions of SSRAP on these eight classes of equipment can be summarized as follows:

(1) Equipment installed in nuclear power plants is generally similar and at least as rugged as that installed in conventional power plants.

(2) This equipment, when properly anchored and with some reservations, has an inherent seismic ruggedness and has a demonstrated capability to withstand substantial seismic motion without structural damage.

(3) Functionality after the strong shaking has ended has also been demonstrated, but the absence of relay chatter during strong shaking has not been demonstrated.

(4) With several important caveats and exclusions, it is the SSRAP judgment that below certain seismic motion bounds it is unnecessary to perform explicit seismic qualification of existing equipment in these eight classes for operating nuclear power plants to demonstrate functionality after the strong shaking has ended.

(5) The existing data base reasonably demonstrates the seismic ruggedness of this equipment up to these seismic motion bounds.

Furthermore, SSRAP believes its conclusions can be extended to other classes of equipment, but only with further study on experience data and test data on a class-by-class basis. Possible sample candidate classes for extension of conclusions are: heat exchangers, diesel generators, electrical motors, air compressors, fans, HVAC (heating, ventilating, and air conditioning) ducts, piping, and cable trays.

2.1.2 Summary of LLNL Report, "Correlation of Seismic Experience Data in Non-Nuclear Facilities With Seismic Equipment Qualification in Nuclear Plants (A-46)"

The study was completed by LLNL and NRC issued a report (NUREG/CR-3017) (NRC, August 1983). This study was intended to answer the question: Is it feasible to use experience data on the performance of equipment in non-nuclear facilities during earthquakes in addressing issues concerning the seismic qualification of equipment in operating nuclear power plants located in the eastern United States?

The study shows that the answer to this question is affirmative. LLNL's general approach to the feasibility determination is based on the assumption that if experience data can be shown to be equivalent to current seismic equipment qualification requirements, then it is feasible to use experience data. The basic approach was to develop an overall summary statement evaluating seismic experience data and current requirements, as embodied in 12 different NRC Standard Review Plan sections, regulatory guides, and national standards. A comparison of the two summary statements provides the basis for the feasibility determination.

In LLNL's approach, 30 categories (issues) of possible seismic equipment qualification requirements are identified. That is, seismic equipment qualification standards might be (but presently are not) formulated in terms of requirements and criteria that address each of the 30 issues. Each of the 30 issues was ranked and a minimum set was identified. Table 2.1-2 lists the 30 issues and briefly describes each issue.

The 12 "current requirements" documents which are considered most important in terms of seismic equipment qualification for new plants are listed in Table 2.1-3.

LLNL's evaluation was performed by first reviewing the 12 current requirements in each of the 30 categories in Table 2.1-2, and then providing a comprehensive evaluation of these requirements. The evaluation was performed by ranking the current requirements in the 30 categories using the following numerical weights:

- <u>Adequate - 3</u>: This is the highest ranking. It is used to show that the current requirements are judged to adequately address the particular issue. "Adequately" means that "the issue is addressed as well as is needed." It should not be interpreted as "ideally" or "perfectly" addressed or that it "addresses the issue as perfectly as can be conceived."

- <u>Moderately Adequate - 2</u>: This is the next highest ranking.

- <u>Marginally Adequate - 1</u>: This is a poor ranking. The issue is addressed, but not very satisfactorily.

Table 2.1-2 Categories of possible seismic equipment qualification (EQ) requirements

Category of Possible Seismic EQ Requirement	Brief Description of Category
Physical attributes	
1. Sampling	For equipment items qualified by testing, only a limited number of the items installed in a plant are tested.
2. Similarity	The EQ for one item of equipment is sometimes extended to similar but different items.
3. Mounting simulation	The mounting and orientation used in the qualification of equipment may be different from those of installed equipment.
4. Peripheral attachments	Peripheral items, such as electrical cables, small control piping, large pipe, and so forth, are often attached to the major item of equipment.
5. Dummy components	Equipment is sometimes qualified by testing with a dummy item substituted for the actual item. For example, an electrical cabinet might be qualified with a dummy component substituted for a relay.
Seismic loads	
6. Generic loads	Generic loads (loads that envelop all the required design loads for a particular category of equipment) are sometimes defined.
7. Enveloping load assumption	It is often assumed that if an item of equipment is qualified for load L_1, then it is also qualified for load L_2, where L_1 is greater than L_2.
8. Required design load	Do the required design load and parameters adequately reflect EQ issues and concerns?
9. Margin	Is there sufficient margin in the capacity of the equipment?
10. Tolerances	Are tolerances specified for the required qualification load?

Table 2.1-2 Categories of possible seismic equipment qualification (EQ) requirements (continued)

Category of Possible Seismic EQ Requirement	Brief Description of Category
Seismic loads (continued)	
11. Single vs. multiaxis testing	How many independent test excitation axes are required?
12. Wave form	A number of issues are related to the waveform of the test motion imparted to equipment.
13. Fatigue	The fatigue requirements are considered here. An example is 5 OBE plus 1 SSE.
Strength/capacity	
14. Fragility	Do the EQ requirements address the strength of equipment, and if they do, how do they address it?
15. Failures	Addresses failures that occur during qualification testing.
16. Functional requirements	Addresses the functional performance of the equipment before, during, and after qualification testing.
17. Critical parameters	Addresses the parameters that are most important to the survivability or functionality of equipment.
18. Degradation under test	Has the qualification testing has been so severe that the capacity of the equipment to perform as required in the future can be questioned?
19. Response	Addresses the observed response of the equipment during qualification testing.
20. Unexpected results	Includes failures at unexpectedly low levels, unusual response patterns, and behavior that is inconsistent with predictions.
Seismic and other loads	
21. Load combination	Relates to appropriate combinations of loads such as seismic, thermal, and pressure.
22. Load sequencing	A variant of load combination.

Table 2.1-2 Categories of possible seismic equipment qualification (EQ) requirements (continued)

Category of Possible Seismic EQ Requirement	Brief Description of Category
Miscellaneous	
23. Errors	Includes design, qualification, construction, mounting, and maintenance errors.
24. Maintenance	Includes consideration of how normal (rather than erroneous) maintenance might affect the qualification status of equipment.
25. Mounting adequacy	Addresses the adequacy of the equipment mounting.
26. Post earthquake	Addresses the issue of assessing EQ subsequent to an earthquake.
27. Value/impact	Addresses the benefit of seismic EQ in risk reduction (value) versus the cost of such requirements (impact).
28. EQ by analysis	Addresses the issue of performing EQ by analysis rather than testing.
29. EQ by testing and analysis	Addresses the issue of performing EQ by a combination of testing and analysis.
30. In-situ testing	Addresses the issue of the possible role of in-situ testing in EQ.

° Inadequate - 0: This is the worst ranking. The issue is either not addressed at all or, if it is addressed, it is addressed poorly.

° Ranking not required: This ranking usually occurs when an issue that does not have to be addressed is included for completeness.

Next, the use of experience data was also evaluated for each of the 30 categories. The same ranking as above was used. These rankings were then weighted according to importance, and the two sums (current requirements and experience data) were compared to arrive at a feasibility judgment. The result of the evaluation is summarized in Table 2.1-4. Table 2.1-4 shows that when the current requirements in existing NRC and national standards were evaluated against the common set of 30 issues, they were estimated to score 91 out of 156 overall, or about 60%. Experience data were estimated to score 97 out of 156 overall, also about 60%. The fact that the current requirements and experience data score about the same

Table 2.1-3 Documents most important for seismic equipment qualification

- U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," NUREG-0800, Rev. 2, July 1981.

- U.S. Nuclear Regulatory Commission, Regulatory Guides:

    - 1.40 - "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," March 16 1973.

    - 1.73 - "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plans," January 1974.

    - 1.100 - "Seismic Qualification of Electric Equipment for Nuclear Power Plants," Rev. 1, August 1977.

    - 1.148 - "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants," March 1981.

- IEEE Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations, ANSI N41.9-1976, IEEE Std. 334-1974.

- IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, ANSI/IEEE Std. 344-1975.

- IEEE Standard for Qualification of Safety-Related Valve Actuators, IEEE Std. 382-1980.

- IEEE Standard Seismic Testing of Relays, IEEE Std. 501-1978.

- IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations, IEEE Std. 649-1980.

- Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard, ANSI N278.1-1975.

- Functional Qualification Requirements for Power-Operated Active Valve Assemblies for Nuclear Power Plants, ANSI B16.41, Draft 3, Rev. II, June 1981.

(60%), led to LLNL's conclusion that it was feasible to use experience data on seismic equipment qualification issues.

Besides the feasibility study, LLNL's report also addressed recommended guidelines for the use of experience data.

For all the categories considered to be the most important (those given an importance ranking of 3), guidelines were developed. Categories considered are:

Table 2.1-4a Summary of feasibility evaluation: Sources 1-7 (as listed in Table 2.1-3)*

Category	SRP 3.10	RG 1.40	RG 1.73	RG 1.100	RG 1.148	IEEE Std. 334-1974	IEEE Std. 344-1975
1. Sampling	Sampling is acceptable. Sample size is not defined.	A "prototype unit" to be tested under most adverse design conditions.	A "prototype unit" to be tested under most adverse design conditions.	A "prototype unit" to be tested under most adverse design conditions.			Implicit acceptance of sampling, at least for cases where fragility testing is performed.
2. Similarity							Extension of EQ by test to similar equipment is allowed using a combination of test and analysis.
3. Mounting simulation	The fixture design should simulate the actual service mounting.						The equipment shall be mounted in a manner that simulates the intended service mounting.
4. Peripheral attachments	Major peripheral attachments are addressed.						The effects of peripheral attachments must be considered.
5. Dummy components	Dummy specimens are allowed to simulate the mass effects and dynamic coupling to the supports.						Use of dummy specimens is allowed.
6. Generic loads							
7. Enveloping load assumption	Not clear whether the assumption is made.			The assumption is made.			The assumption is made.
8. Required design load							
9. Margin	Margins are required but not specified.			Margins are required but not specified.			10% margins are specified for the response spectrum at the mounting point of the equipment.

*A blank indicates no requirement was found.

Table 2.1-4a Summary of feasibility evaluation: Sources 1-7 (as listed in Table 2.1-3) (continued)

Category	SRP 3.10	RG 1.40	RG 1.73	RG 1.100	RG 1.148	IEEE Std. 334-1974	IEEE Std. 344-1975
10. Tolerances							
11. Single vs. multiaxis testing	Two simultaneous axes of input are generally required. General procedures are specified.						Multiaxis testing is suggested. Single-axis testing is allowed if conservative, or if the responses in the axes are independent.
12. Wave form	The characteristics of the required input should be specified by response spectrum or time history methods.						Requirements for simulating earthquake are given. Specific requirements for proof testing are specified.
13. Fatigue	Structural integrity and operability must be assumed under an SSE preceded by several OBEs.			Performance must must be assured during and after an SSE preceded by several OBEs.			The requirement is five OBEs plus an SSE.
14. Fragility							Fragility testing recommended, but not required, for equipment to be used in a number of applications.
15. Failures							
16. Functional requirements	Operationality should be verified during and/or after testing.			General, indirect references to functionality are given.	Reference is made to ANSI N278.1-1975.	Seismic input is assumed to occur with motor standstill, starting, running, or coasting down.	For devices (relays, motors, sensors), it is assumed that the seismic input can be imposed while simulating normal operative and sensing performance.
17. Critical parameters							Some parameters are suggested as possibly critical and are recommended for identification.

Table 2.1-4a Summary of feasibility evaluation: Sources 1-7 (as listed in Table 2.1-3) (continued)

Category	SRP 3.10	RG 1.40	RG 1.73	RG 1.100	RG 1.148	IEEE Std. 334-1974	IEEE Std. 344-1975
18. Degradation under test					Upon completion of the test, the motor shall be dismantled and inspected.		
19. Response							Monitoring is required, but specific requirements are not given.
20. Unexpected results							Analysis might be used to explain unexpected behavior during a test.
21. Load combina tion	It is not clear what combinations are acceptable.						Normal operating loads which adversely affect function must be combined with seismic loads.
22. Load sequencing	Load sequencing is to follow IEEE Std. 323-1974.			Load sequencing is indirectly addressed.			
23. Errors							
24. Maintenance							
25. Mounting adequacy	Requirements on mounting adequacy are given with respect to testing and/or analysis assessments.						The mounting method shall be the same as that recommended for active service.
26. Post earthquake							
27. Value/ impact							

Table 2.1-4a Summary of feasibility evaluation: Sources 1-7 (as listed in Table 2.1-3) (continued)

Category	SRP 3.10	RG 1.40	RG 1.73	RG 1.100	RG 1.148	IEEE Std. 334-1974	IEEE Std. 344-1975
28. EQ by analysis	EQ by testing is preferred.			EQ by other testing is implicitly accepted by IEEE Std. 344-1975.			EQ by analysis is not generally recommended without test except where structural integrity alone can ensure equipment function.
29. EQ by testing and analysis							EQ by combined testing and analysis is acceptable, but only vaguely defined.
30. In-situ testing	In-situ testing is not required, but it is allowed.						In-situ testing can be a part of EQ by combined testing and analysis.

Table 2.1-4b Summary of feasibility evaluation: Sources 8-12 (as listed in Table 2.1-3) and other data

Category	IEEE Std. 382-1980	IEEE Std. 501-1978	IEEE Std. 649-1980	ANSI N278.1-1975	ANSI B16.41-1981	Score on current requirement	Experience data	Score on experience data
1. Sampling	A procedure suggested for selecting the test units is given in App. A.	A minimum of 3 specimens is required.	At least one device must be tested, but not one motor control center.		Testing of at most one sample is acceptable.	3	Several units are commonly excited at once by an earthquake. Therefore experience data are potentially rich in sampling.	6
2. Similarity	Similarity is addressed in terms of generic groups of valve actuators from which test units are drawn.	Extension of qualified relays to relays not tested is allowed.	General guidelines are given to extend the qualification of motor control centers to other units.		Guidelines are given to extend qualification of value assemblies to similar units.	3	Equipment among non-nuclear facilities is usually quite similar. A casual comparison also indicates that the equipment is also quite similar to that in nuclear facilities.	6
3. Mounting simulation	The valve actuator is required to be mounted to the shaker table as it would be attached to the valve.	The relay must be mounted as it normally would be in service.	The motor control center must be mounted as it would be in a plant.			4	Experience data reflect the true mounting conditions. Therefore, mounting is not an issue for such data.	6
4. Peripheral attachments	Electrical, hydraulic, or pneumatic connections must be attached.		Anticipated additional weight and external connections shall be simulated.		Electrical, hydraulic, or pneumatic connections shall be required.	4	The credibility of effects from peripheral attachments is not an issue for experience data.	6
5. Dummy components						4	Dummy specimens do not represent an issue for experience data.	6
6. Generic loads	Generic loads for valve actuators are established for most plants.	Fragility testing is required for relays; therefore, generic loads are essentially required.	Generic load EQ techniques are allowed for groups of equipment.			Not required.		Not required.

Table 2.1-4b Summary of feasibility evaluation: Sources 8-12 (as listed in Table 2.1-3) and other data (continued)

Category	IEEE Std. 382-1980	IEEE Std. 501-1978	IEEE Std. 649-1980	ANSI N278.1-1975	ANSI B16.41-1981	Score on current requirement	Experience data	Score on experience data
7. Enveloping load assumption	Enveloping is probably established through generic loads.					2	Experience data could provide an indication of equipment performance at loads that envelope required loads for EQ.	2
8. Required design load	The required design load may be deficient.					6	Although loads experienced are realistic, the adequate reflection of such loads to areas of concern in EQ of nuclear plant equipment may be lacking.	3
9. Margin	Margins are included in the generic loads.	Fragility testing includes the concept of margins.	Margins are specified in Table 1.			3	Some evaluations indicate that some non-nuclear facilities have experienced seismic loadings in excess of design loadings in nuclear facilities.	6
10. Tolerances		Tolerances are specified for instrumentation.				Not required.		Not required.
11. Single vs. multiaxis	Biaxial testing required.	Triaxial testing is desired, but biaxial testing is acceptable.				6	Experience data generally consists of three-dimensional excitation.	6
12. Wave form	Requirements are consistent with IEEE Std.344-1974.	Two multi-frequency, standard response spectra are specified for qualification of relays.				9	Inputs in experience data can be either narrow banded if the equipment is mounted on a structure or piping system, or broad banded if mounted on the foundation.	6

Table 2.1-4b Summary of feasibility evaluation: Sources 8-12 (as listed in Table 2.1-3) and other data (continued)

Category	IEEE Std. 382-1980	IEEE Std. 501-1978	IEEE Std. 649-1980	ANSI N278.1-1975	ANSI B16.41-1981	Score on current requirement	Experience data	Score on experience data
13. Fatigue	OBE and SSE testing are required. Each test must be 15-s minimum.	Five OBE plus an SSE testing are required. Minimum duration is 15-s per test				3	Low-cycle fatigue may be revealed by experience data.	3
14. Fragility		Fragility testing is required for relays.				3	Present indications from a limited review of experience data suggest that few or no failures of equipment will be observed.	3
15. Failures		Determination of what constitutes failure for relays is given.				3	Failure information may be limited.	3
16. Functional requirements	Valve actuators must be functional before, during, and after testing.	Relays must be tested in the transition from nonoperating to operating condition.	Motor control center operational capacity must be demonstrated.	Valve assemblies must be operable during and after the test.	Functional requirements are given for valve assemblies.	9	Experience data on the functionality of equipment may be relatively scarce.	6
17. Critical parameters						1	Since few or no failures have been observed, it is unlikely that experience data will reveal critical parameters. The most important failures observed have been failures of mountings or attachments.	0
18. Degradation under test					Inspection of valve assemblies shall be performed before and after testing.	0	Degradation is generally not an issue for experience data.	

Table 2.1-4b Summary of feasibility evaluation: Sources 8-12 (as listed in Table 2.1-3) and other data (continued)

Category	IEEE Std. 382-1980	IEEE Std. 501-1978	IEEE Std. 649-1980	ANSI N278.1-1975	ANSI B16.41-1981	Score on current requirement	Experience data	Score on experience data
19. Response						Not required.		Not required.
20. Unexpected results						Not required.		Not required.
21. Load combination		Seismic testing of relays can be performed under prevailing ambient conditions of the test laboratory.				6	Normal operating loads are expected to be present already when an earthquake occurs.	4
22 Load sequencing	A standard load sequence is required.		Sequencing of preaging and seismic testing is specified.		A sequence of testing is specified for valve assemblies.	6	Equipment in operating plants can be expected to have normal environments, transients, and in-situ vibration.	4
23. Errors						0	Equipment in plants presumably has been installed with a more or less typical set of errors.	2
24. Maintenance	Maintenance to be performed during the test must be specified.	Maintenance can be performed after a given fragility test.	Modifications during testing shall be evaluated to determine their effect on the EQ.		If maintenance or adjustments are required during testing, acceptance of the test must be evaluated.	0	Experience data should be valuable in assessing if, and how maintenace affects seismic performance.	2
25. Mounting adequacy	The valve actuator must be mounted to the shake table as it would be mounted to a valve.	Recommended mounting hardware must be used.	Mounting must be by welding or bolting for seismic testing.		The valve assembly must be supported as required to permit testing in accordance with the standard.	9	Failure of mountings appears to be the single most important failure; therefore, experience data can be expected to provide useful information regarding mounting adequacy.	6

Table 2.1-4b Summary of feasibility evaluation: Sources 8-12 (as listed in Table 2.1-3) and other data (continued)

Category	IEEE Std. 382-1980	IEEE Std. 501-1978	IEEE Std. 649-1980	ANSI N278.1-1975	ANSI B16.41-1981	Score on current requirement	Experience data	Score on experience data
26. Post earthquake						2	Equipment exposed to an earthquake is subsequently subjected to normal operation, transients, etc. Therefore experience data should be useful for assessing post-earthquake behavior, but only partially.	1
27. Value/impact						Not required.		Not required.
28. EQ by analysis	EQ by analysis is allowed to extend qualification of a generic group to a specific application.	EQ by analysis is assumed to be possible.	EQ by analysis is allowed.		EQ by analysis is allowed.	1	Experience data are at least as amenable to analysis as EQ is through ordinary means.	1
29. EQ by testing and analysis	EQ by combination of test and analysis is allowed to extend EQ of a generic group to specific applications.	General requirements are given.	EQ by combination of test and analysis is allowed.		EQ by combination of test and analysis is allowed.	1	The use of combined test and analysis in experience data would have to be defined in more detail to make a good evaluation of its value.	1
30. In-situ testing						1	It should be possible to develop acceptable in-situ techniques for nonnuclear facilities and nuclear facilities alike.	3
Total						91		97

(1) Sampling
(2) Similarity
(3) Required design load
(4) Margin
(5) Single vs. multi-axis testing
(6) Wave form
(7) Fragility
(8) Failures
(9) Functional requirements
(10) Mounting adequacy

The guidelines, as taken directly from the LLNL report, are combined under the five headings as follows:

Sampling

(1) Experience data should be gathered on all non-nuclear facilities that have experienced (a) a significant earthquake, or (b) failures of any kind or either temporary or permanent loss of functional capability. LLNL anticipates that 10 to 50 facilities will fall into this class. If fewer than ten facilities, three significant earthquakes, or all facilities that have experienced some kind of mechanical, structural, or functional failure are included in the data base, LLNL does not recommend that the NRC accept experience data as fully as it has otherwise recommended.

(2) The numbers of each type and size of affected equipment should be obtained for each facility in (1). If fewer than three items of each type and size of interest are found, then a justification must be provided to extend the experience data.

Similarity

(3) The issue of the similarity of equipment in non-nuclear facilities to equipment in nuclear facilities must be addressed. However, exact similarity need not be established.

Rather, what is required is reasonable assurance that the equipment in non-nuclear facilities (a) is of the same type and basic design, and (b) was manufactured by the same manufacturers in the same period as the equipment of interest in nuclear facilities.

Required Design Load, Wave Form, and Dimensionality

(4) The approximate location of each item of equipment in non-nuclear facilities must be established in order to obtain a "rough" idea of the type of earthquake motion it experienced. "Rough" means that dynamic modeling or analysis is not required. Two categories are suggested:

(a) Dimensionality. Was the earthquake motion affecting the equipment predominantly one-, two-, or three-dimensional in nature?

(b) Wave form. Was the earthquake motion affecting the equipment:

- random like an earthquake (as for equipment in the foundation or free-field)
- random because of superposition of a number of narrow-band pass motions, each with a different center frequency (as for horizontal motions on equipment in the lower elevations of a structure)
- sinusoidally random, that is, essentially a single-band pass motion (as for horizontal motions on equipment in the higher elevations of a structure).

Criteria are difficult to establish in this area, because in some respects they are dependent on the motions expected for the equipment of interest in nuclear facilities. However, if the experience data indicate significant two- or three-dimensionality of motion and sinusoidally random motion with a mix of center frequencies, then the experience data are acceptable.

Margin

(5) The facilities in (1) should be selected in order of decreasing severity (for example, peak acceleration) of earthquake, that is, the most severe earthquake first. A reasonable assurance of margin for plants in the eastern U.S. is provided if the experience data are obtained from earthquakes with a peak acceleration greater than the SSE peak acceleration for the nuclear plants of interest and the duration is greater than 10 seconds.

However, inevitably questions will arise about the most detailed aspects of the motion affecting the equipment in non-nuclear facilities (for example, in-structure response spectra) and the relation to similar motions in nuclear facilities.

The staff believes that the above requirement for acceleration and duration provides reasonable assurance on the issue of margin, and nothing further is recommended. If, however, the NRC decides that more needs to be done on the margin issue, three steps are recommended:

(a) As a first step, realistic analyses can be performed on the non-nuclear facilities. For example, a comparison of realistic non-nuclear and nuclear design in-structure spectra, as in the EQ report of September 1982 may establish the required confidence in margin.

(b) If (a) is not chosen or if it does not indicate margin is present, then the following may be an acceptable alternative. Realistic, best-estimate analyses, with uncertainties explicitly characterized, as in the LLNL report of July 1981, should be performed on both the non-nuclear (for the earthquake that occurred) and nuclear (for design earthquakes) facilities. The median of the two results should be used as a measure of whether or not adequate margin exists. For example, median in-structure spectra from the two analyses can be compared.

(c) As part of either (a) or (b) above, margin is assured if, for example, margin exists at the frequencies of interest but not at some other frequencies in the spectra.

Fragility, Failures, Functional Requirements, and Mounting Adequacy

(6) A vigorous effort to seek out failures or incipient failures in experience data is required. In addition to mechanical or structural distress or failure, incipient or actual functional failures should also be sought. This effort includes examination of plant system logs and interviews with plant operators or other personnel present during the earthquake.

The six guidelines above are concerned with experience data obtained from non-nuclear facilities.

The next three guidelines are concerned with actions recommended by LLNL for nuclear facilities.

Functional or Other Failures

(7) Nuclear plant equipment should be examined very closely for any and all failures revealed in (6). For example, experience data suggest that mounting failure is the single most important cause of failure of equipment. All nuclear equipment of interest should be examined for adequacy of mounting or attachment.

(8) The NRC should develop a detailed and definitive check list to aid in a "walk-down" of equipment of interest in nuclear plants. Such a walk-down should then be performed in each operating nuclear power plant where there is concern about the seismic adequacy of equipment. The items and procedures in the checklist should be drawn from three sources:

(a) Information gathered from the collection of experience data;

(b) Information gathered from laboratories experienced in seismic equipment qualification testing;

(c) Recognized experts who have performed walk-downs in the past.

(9) A limited amount of shake table testing should be performed on equipment obtained from operating nuclear power plants to confirm the perceived strength of equipment. This testing should satisfy the following:

(a) The test objective is to obtain the "capacity" of each equipment item tested. Capacity includes:

° incipient or actual "structural" failure

° degradation of or loss of function

° identification of failure modes and key parameters related to failure or capacity

° anomalous behavior

An example of such testing can be found in the JAERI report of August 1979.

(b) The equipment should be tested while functioning or in such a manner that capability of function is assured.

(c) The equipment need not be artificially aged or subjected to loads or environments other than seismic.

(d) The equipment should be tested as is. That is, it should not be modified, adjusted, disassembled and tested separately, etc., after it is selected for removal or removed from the plant.

(e) The testing should be limited in the number of categories of equipment tested, but comprehensive in addressing each operating plant and category of equipment. For example, one item of each category of equipment should be obtained from each category of equipment, and the same test program executed for each.

(f) The number of categories of equipment should be limited. The selection of the category of equipment to be tested should be based on importance, estimated vulnerability, (that is, choose a category that is believed to be relatively weak rather than strong) and diversity of equipment type. For example, these objectives may be satisfied if the testing is limited to:

° 125-V vital bus (electrical equipment)
° motor-operated valves (mechanical equipment)

(g) The above requirements may lead to testing on the order of 100 items of equipment, depending on the number of plants involved. As an alternative to 100 tests on only 2 categories of equipment, as outlined above, a minimum of 5 tests on 20 or so categories would be acceptable.

2.1.3 Summary of EQE Report, "Pilot Program Report - Program for the Development of an Alternative Approach to Seismic Equipment Qualification"

Many non-nuclear power plants and industrial facilities containing equipment similar to that found in nuclear power plants have experienced major earthquakes. A sample of this experience is shown in Table 2.1-5. The SQUG with help from EQE, initiated a pilot program to evaluate the potential for using experience data as the basis for qualification. The results of this pilot program were documented in this EQE report (EQE, September 1982). Stated goals of the pilot program were:

(1) To develop a historical data base on the performance of equipment in power plants during and after strong earthquakes.

(2) To show that much of the equipment in those plants is similar to equipment found in nuclear power plants.

(3) To determine whether data from actual earthquakes are sufficient to conclude that seismic qualification by conventional methods is not necessary for certain classes of equipment.

(4) To develop a methodology for using earthquake data to evaluate the necessity for seismic qualification of specific items of equipment by conventional methods.

Table 2.1-5 Selected major earthquakes that have affected power and industrial facilities

Earthquake Location	Year	Approximate Richter Magnitude	Recorded Peak Ground Acceleration (g)	Estimated Number of Ground Motion Records	Power Plant Units Affected
1. Eureka, Ca	1980	7.0	0.15+	8	3
2. Imperial Valley, CA	1979	6.6	0.81+	50	4
3. Miyagi-Ken-Oki, Japan	1978	7.4	0.40	100+	10+
4. Friuli, Italy	1976	6.5	0.30+	30+	?
5. Eureka, CA	1975	5.5	0.35	Several*	3
6. Point Mugu, CA	1973	5.9	0.09	10+	4
7. Managua, Nicaragua	1972/3	6.2	0.60	4+	3
8. San Fernando, CA	1971	6.5	1.25	60+	20+
9. Caracas, Venezuela	1967	6.5		--	Several*
10. Seattle, WA	1965	6.5	0.08	3	Several*
11. Alaska	1964	8.4		--	7
12. Niigata, Japan	1964	7.5	0.18+	Several*	Several*
13. Chile	1960	8.5		None	Several*
14. Kern County, CA	1952	7.7	0.13	5+	1
15. Long Beach, CA	1933	6.3	0.15+	Several*	5

Source: EERI, 1981.

+Indicates equal to or greater than the number shown.
*Actual number not determined.

2.1.3.1 Methods Used in the Pilot Program

Two types of facilities were addressed in the pilot program: nuclear power plants and non-nuclear power facilities that have experienced strong earthquakes (also referred to as data base plants by SQUG).

The steps involved in collecting data from the data base plants and the nuclear power plants and in comparing the data are shown in Figure 2.1-1. Before walkdowns of the data base plants were conducted, available records of the seismic event at each site were collected. These data included ground motion traces recorded near the plant sites. Facilities that had experienced significant ground motion and that also appeared to contain equipment appropriate to the investigation were selected for visits and walkdowns.

Preliminary and final walkdowns were conducted at both the nuclear power plants and the non-nuclear facilities. Preliminary walkdowns at the nuclear power plants were used to identify types of commonly encountered safety-related equipment. Preliminary walkdowns at the non-nuclear facilities were used to record the locations of types of equipment that are similar to nuclear power plant equipment. Following the walkdowns, particular classes of equipment were selected to be the focus for the remainder of the pilot program. Final walkdowns were used for collection of detailed data, including conducting in-situ dynamic testing.

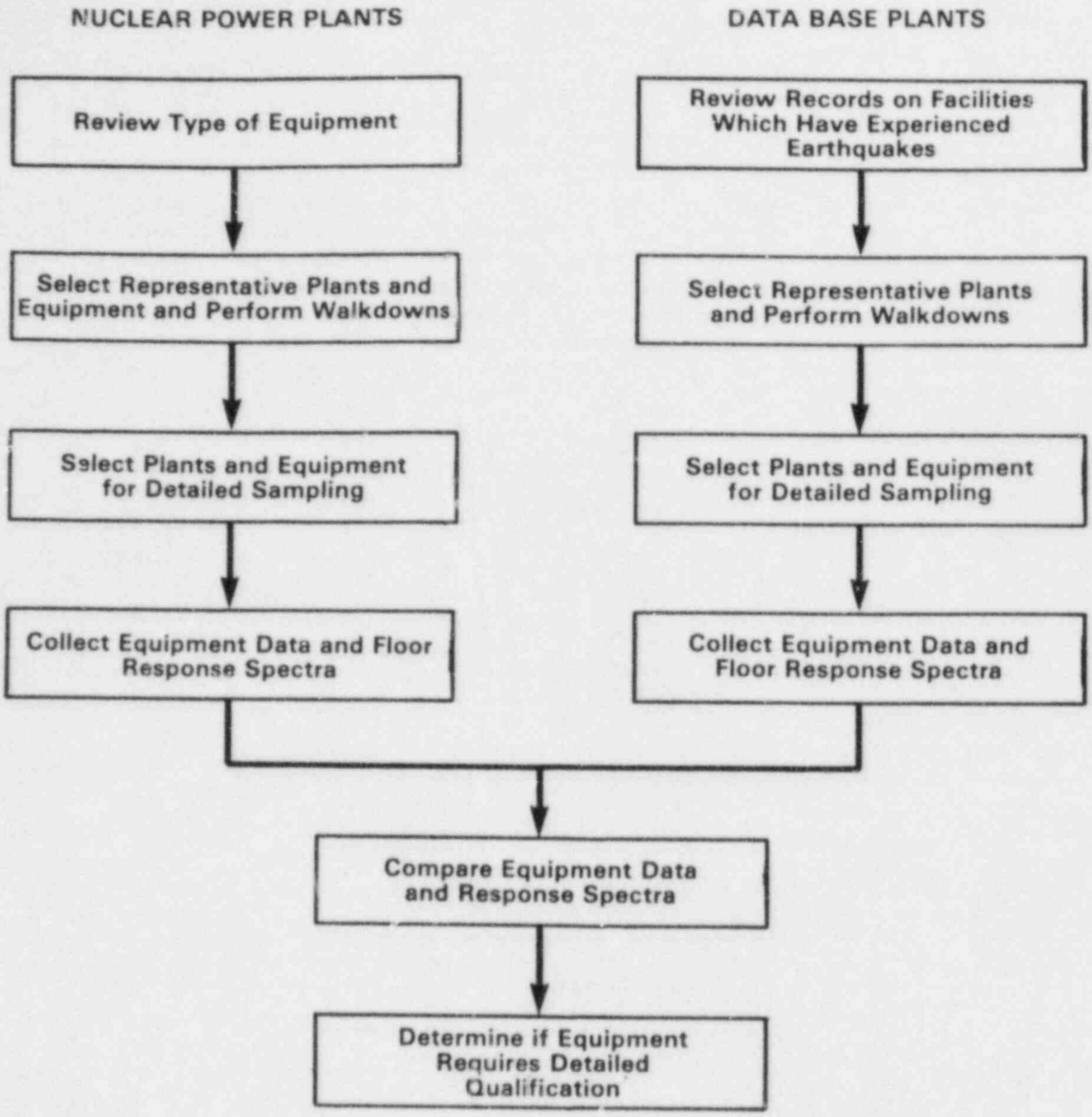


Figure 2.1-1 Methods used in pilot study

Low-excitation-level in-situ testing was conducted on approximately 200 pieces of equipment in the data base and nuclear power plants to determine approximate primary response frequencies and mode shapes. This permitted estimates to be made of equipment response to floor motion.

Seven classes of equipment* were selected for detailed study (see Table 2.1-6). Each class was reviewed to determine similarities between equipment in the two types of power plants. The following characteristics were examined to establish similarity: primary structural and functional characteristics; dimensions and

*An eighth equipment class was later added.

Table 2.1-6 Equipment selection for SQUG pilot program

Equipment selected:

- Motor control centers
- 480-V switchgear
- 2.4 to 4kV switchgear
- Motor-operated valves
- Air-operated valves
- Horizontal pumps
- Vertical pumps

Of seven nuclear power plants visited, three were selected for equipment data collection:

Plant	Design-basis SSE
Dresden 3	0.21 g
Calvert Cliffs 1	0.15 g
Pilgrim	0.15 g

name-plate data; and ranges of dynamic-response frequency. The response frequencies found during the in-situ testing were compared to determine whether the equipment in the data base plants and the nuclear plants could be expected to have similar dynamic response properties.

It was noted by SQUG that most of the equipment of interest in the data base plants is located at grade, in basements, or in the first two floors of the structure (up to the turbine decks). In addition, most of the data base structures are relatively stiff, many are either light concrete structures with shear walls or braced steel-frame structures. Therefore, SQUG concluded that no large amplification of ground motion by the structure was expected for the locations of most of the equipment of interest. Free-field ground spectra were used as conservative estimates of the floor response spectra for the data base structures that were not analyzed. Thus, amplification of the data base floor response spectra was excluded.

The floor response spectra required for the nuclear power plants were obtained from the operating utility. Wherever spectra were unavailable for a specific item, amplified floor spectra were assumed on the basis of nearby spectra.

The data base floor response spectra and the nuclear equipment required response spectra obtained as above are then compared to assure that floor response spectra of the data base envelope those of the nuclear equipment.

The performance of data base equipment during past earthquakes was evaluated and conclusions regarding the seismic resistance capability of similar nuclear equipment were reached. A typical comparison is shown in Table 2.1-7.

For the purpose of the pilot program, non-nuclear power plants and other facilities in southern California where significant earthquakes have occurred were chosen for the study. Table 2.1-8 shows the four earthquakes in southern

Table 2.1-7 Comparison of equipment data

Variable	Data Base Equipment	Nuclear Equipment
ITEM:	480-V motor control center cabinets 1VA-6VA, P3A & P4A (Eight Units)	480-V motor control center 39-3
PLANT:	Sylmar Converter Station	Dresden Nuclear Plant, Unit 3
MANUFACTURER:	General Electric 7700 Line Series, 1970	General Electric 7700 Line Series, 1971
LOCATON:	Sylmar Converter Station basement, facing northeast and soutnwest	Reactor bulding elevation 570 ft, facing east (grade is at elevation 517.5 ft)
FUNCTION/SYSTEM:	Control of pumps and valves for rectifier cooling systems	Control of various Class I mechanical systems
CABINET:	Each cabinet is four cubicles wide; the specific arrangement of starter units varies from cabinet to cabinet; they are otherwise very similar.	Cabinet is six cubicles wide. The cabinet contains starter units in cubicles of various sizes.
COMPONENTS:	A typical starter unit consists of a General Electric CR-106 magnetic contractor, a circuit breaker switch, a control transformer, on-off pushbuttons and a terminal block.	A typical starter unit consists of a General Electric CR-106 or CR-105 magnetic contractor, a circuit breaker switch, a control transformer, on-off pushbuttons, and a terminal block.
ANCHORAGE:	The bottom channel of the cabinet is tack welded to a baseplate embedded in the concrete floor. At least one cabinet was inadequately anchored at the time of the earthquake and slid a few inches.	The bottom channel is tack welded to an embedded baseplate, two welds at the base of each stack of cubicles, front and back.
APPLICABLE RESPONSE SPECTRA:	The records taken at Pacoima Dam are shown scaled to 40% of the measured amplitudes as a conservative estimate of the ground motion at Sylmar.	The calculated floor spectra for the reactor building, elevation 589 ft are shown. Spectra at elevation 570 ft were not generated.
EQUIPMENT STATUS DURING AND FOLLOWING THE EARTHQUAKE:	The MCCs were in operation at the time of the earthquake. No damage to either cabinet or components was reported. One cabinet slid a few inches due to lack of floor anchorage.	

Table 2.1-8 Summary of data base plants and earthquakes

Earthquake & Date	Facility		Estimated PGA
San Fernando 1971	1.	Sylmar Converter Station	0.50 - 0.75*
	2.	Valley Steam Plant	0.40*
	3.	Burbank Power Plant	0.35*
	4.	Glendale Power Plant	0.30*
	5.	Pasadena Power Plant	0.20*
	6.	Rinaldi Receiving	0.50*
	7.	Vincent Substation	0.20*
	8.	Saugus Substation	0.39**
Point Magu 1973	9.	Ormond Beach Plant	0.20*
	10.	Santa Clara Substation	0.10*
Santa Barbara 1978	11.	Goleta Substation	0.28**
	12.	Ellwood Peaker Plant	0.30 - 0.40*
Imperial Valley 1979	13.	El Centro Steam Plant	0.51**
	14.	Magmamax Geothermal Plant	0.20 - 0.30*

*Located near strong motion records.
**Recorded peak ground acceleration - at plant site.

California that were reviewed in detail in this program. The facilities that contained the largest number of equipment items of interest and were reviewed in detail are the Sylmar converter station, Valley steam plant, Burbank power plant, Glendale power plant, Pasadena power plant, and El Centro steam plant.

Seven nuclear power plants were visited, and three were selected for equipment data collection, they are Dresden Unit 3, Calvert Cliffs Unit 1, and Pilgrim. These plants were selected so that the equipment reviewed for the project would form a representative sample of a variety of nuclear plant characteristics, including reactor type and vintage. Only equipment required for safe shutdown was considered.

2.1.3.2 Conclusion and NRC Staff Comments

The goals of this pilot program were evaluated by SQUG against the results obtained from the study. Table 2.1-9 lists the goals, findings, and conclusion as seen by SQUG. Finally, SQUG reached the following two conclusions:

- The structural integrity of anchored power plant equipment and component is not compromised in strong earthquakes of up to 0.50 g peak ground acceleration.

- Typically, operability of power plant equipment is not comprised in strong earthquakes with peak ground acceleration of about 0.20 g to 0.30 g.

Although the staff is in general agreement with SQUG on the first overall point, it has some reservation on the second point.

Table 2.1-9 Major conclusions of SQUG

GOAL 1: Develop a historical data base on the performance of equipment in conventional power plants during and after strong earthquakes.

FINDINGS:

- Several power plants and other industrial facilities have experienced strong earthquakes exceeding the free-field safe-shutdown earthquakes required for the design of most U.S. nuclear power plants.
- The plants responded well to the earthquakes and usually continued to operate or were back on line shortly after the earthquakes.
- Many of the facilities were in operation at the time of the earthquakes; thus their equipment was subjected to normal operating loads in addition to the seismic loads from the earthquakes.
- With a few minor exceptions, the equipment contained in the power facilities was undamaged and was functional after the earthquakes. The equipment was not known to be modified because of the earthquakes.
- Sufficient data exist to estimate the spectra experienced by the plants and their equipment.
- There is a large, available data base, only a portion of which was sampled in this study, of power plant equipment that has been subjected to strong earthquakes.

CONCLUSION: There is a large body of available data on the performance of power plant equipment in strong earthquakes, including both mechanical and electrical equipment. Many conventional power plants and industrial facilities have experienced earthquakes that subjected their equipment to seismic environments equal to or exceeding seismic loads associated with safe shutdown earthquakes required for the design of most nuclear power plants.

GOAL 2: Show that much of the equipment investigated, which has experienced strong earthquakes, is similar to equipment found in nuclear power plants.

FINDINGS:

- A few major equipment manufacturers supply much of the equipment for both conventional and nuclear power plants.
- There is little observable difference between the measured dynamic response frequencies of equipment in nuclear power plants and those in conventional plants.

Table 2.1-9 Major conclusions of SQUG (continued)

GOAL 2: FINDINGS: (CONTINUED)	° There are no generic differences other than age between equipment found in conventional and nuclear power plants.
CONCLUSIONS:	Certain types of mechanical and electrical equipment found in nuclear power plants are very similar in configuration, function, manufacturer, and model to the types found in conventional plants. Much of the equipment in nuclear power plants and conventional power plants is the same.
GOAL 3:	Determine whether actual earthquake data are sufficient to conclude that seismic qualification of certain classes of equipment by conventional methods is not necessary.
FINDINGS:	° Excluding some unanchored equipment and one air-operated valve, no failures were reported in any of the seven types of equipment addressed in this study. ° With the possible exception of electrical relays, there is no evidence of malfunction of the reviewed equipment during the earthquakes. ° The estimated ground-response spectra from several California earthquakes and the conventional power plants affected by them envelop the floor-response spectra for the safe shutdown earthquakes required for nuclear power plants in the ranges of most equipment response frequencies. ° Conventional plants that were subjected to earthquakes with peak ground acceleration of about 0.30 g or lower generally continued to operate throughout the earthquakes.
CONCLUSION:	Seismic qualification of nuclear equipoment by conventional methods does not appear to be necessary for the classes of equipment evaluated for most levels of safe-shutdown earthquakes.
GOAL 4:	Develop a methodology for the use of actual earthquake data to determine whether seismic qualification of specific items of equipment by conventional methods is necessary.
FINDINGS:	° The seismic performance of the reviewed equipment appears to be independent from any of the following factors: ° Age of equipment ° Years of service

Table 2.1-9 Major conclusions of SQUG (continued)

GOAL 4: FINDINGS: (CONTINUED)	°	Manufacturer and model
	°	Mounting configuration
	°	Dynamic properties
	°	The methodology used in the pilot program to evaluate classes of equipment would be equally applicable to specific items of equipment.
CONCLUSION:	The pilot has demonstrated the methodology. There is an abundance of data that can be used to identify specific items of equipment that do not require additional seismic qualification.	

The NRC staff completed the review of the pilot program report, and concluded that it is feasible to accept experience data as a basis for seismic qualification. Staff comments on the SQUG pilot program were generally an assessment of what further work should be done to provide an acceptable experience data base. The comments were sent to SQUG in December 1982.

2.1.4 Summary of EQE Reports, "Seismic Experience Data Base--Data Base Tables for Seven Types of Equipment," "Seismic Experience Data Base--Average Horizontal Data Base Site Response Spectra," and "Investigation of Equipment Performance in Foreign Earthquakes and the 1964 Alaska Earthquake"

After reviewing the SQUG pilot program report, the staff concluded that it is feasible to accept experience data as a basis for seismic qualification, so long as some additional work is done to provide an acceptable data base. In a meeting with NRC management in March 1983, SQUG suggested the formation of a third-party Senior Seismic Review Advisory Panel (SSRAP) to provide consulting services and expert opinion for the further development of experience data. The members of SSRAP were to be five recognized experts in the field of seismic engineering, and in the design, operation and qualification of electrical and mechanical equipment in both nuclear and fossil power plants. The functions of SSRAP were to be:

(1) To review and comment on the validity of the conclusions reached by SQUG.

(2) To provide guidance in the use of earthquake experience data as a screening method to exclude certain classes of equipment from formal seismic qualification and focus qualification efforts on the more fragile equipment.

(3) To evaluate the data collection and review process and methods used by SQUG in the screening of equipment.

NRC management endorsed formation of SSRAP and the panel was subsequently formed in June 1983 and is organized as follows:

Chairman - Robert E. Kennedy (Structural Mechanics Associates)
Vice Chairman - Water A. Von Riesemann (Sandia National Laboratory)
Secretary - Paul Ibanez (ANCO Engineers, Inc.)
Member - Anshel J. Schiff (Purdue University)
Member - Loring A. Wyllie, Jr. (H. J. Degenkolb Associates, Engineers)

On July 8, 1983, SQUG presented its pilot program to the ACRS during the 279th ACRS meeting. The response from ACRS was generally favorable to the pilot program; however, the Committee observed that "more work is required to establish the operability of equipment during and after an earthquake and more data will be required to support conclusions drawn concerning the seismic resistance of the equipment investigated."

After a review of SQUG's pilot program report and the staff's comments on the report, SSRAP compiled a list of issues and requested additional information to help the panel in its review. Briefly, the requests and observations follow.

(1) Data Deaggregation. The SSRAP recommended that the data base be deaggregated to provide the following information.

   (a) average spectra for the two horizontal components for each plant, rather than the larger (or smaller) of the two;

   (b) a list of equipment by plant;

   (c) a list of equipment located more than 40 feet above grade in a structure whose first mode resonant frequency is below 3 Hz. Also, percentage of the data base, on an equipment category-by-category basis, above 40 feet. These data are needed to assess the significance of possible base isolation and spectra reduction effects of low-tuned buildings;

   (d) a breakdown of equipment by manufacturer/model, size, and type (e.g., gate versus butterfly valves).

(2) Data Base Extent. The SSRAP recommended that the current data base be extended to include the 1964 Alaska earthquake and 1983 Coalinga earthquake. These earthquakes should be reviewed largely with emphasis on investigating whether failures occurred or not. The Alaska event is particularly useful because of its long duration. These data will help satisfy the issue of repeated or longer duration shaking. Also, SSRAP recommended that knowledgeable U.S. power industry people be surveyed about their experiences in selected foreign earthquakes (including, at least, Fruili, Managua, and Miyagi-Ken-Oki). The emphasis should be to document, in writing, their experience as to whether a significant number of generic equipment failures occurred.

(3) SSRAP endorsed the SQUG pilot program in general, and agreed that the SQUG activity should be limited to the seven classes of equipment (see Table 2.1-6).

(4) The goal of the SSRAP review will be to establish, if possible, a set of screening criteria for the seven classes of equipment. The intent was to

avoid piece-by-piece comparison of equipment in the data base with equipment in the operating nuclear plants. No further seismic qualification of equipment should be required if it is satisfactorily established by the screening criteria that the equipment belongs to one of the seven classes of equipment. In order to make this approach feasible, SSRAP believed that a significant amount of data will be needed for each of the seven classes of equipment.

(5) Similarity and operability of equipment are the two most important issues to be resolved in developing the screening criteria. Operability of equipment must be more fully addressed. The conventional plant data do not yet indicate how phenomena such as relay chatter and breaker trip would affect operations in a nuclear plant. More data and study are needed, including studies of the differences in requirements between conventional and nuclear plants. Alternatively, specific relay qualification or replacement may be required.

(6) Generic qualification of the kind proposed may not be possible with structures containing certain brittle materials, such as cast iron and porcelain.

(7) Walkdown of nuclear plant equipment will probably be an essential part of a generic qualification procecure.

(8) More explanation is needed for the data on vertical pumps (e.g., nature of shaft supports and overall size).

(9) The data base needs to be expanded on motor-operated valves and vertical pumps.

(10) Adequate equipment anchorage should be established before equipment is screened.

SSRAP met with the NRC staff and SQUG seven times from June 1983 to January 1984, and reviewed, exchanged ideas, and commented on the SQUG study. In addition, walk-throughs of several of the non-nuclear facilities in the Los Angeles area used in the data base were conducted, and Zion and Dresden nuclear power plants were visited. During the November 1983 meeting, EQE provided SSRAP with the information it asked for in the form of three draft reports. Following are summaries of these reports.

2.1.4.1 EQE Report, "Seismic Experience Data Base--Data Base Tables for Seven Types of Equipment"

SSRAP asked SQUG to deaggregate the data base to provide the needed information. EQE, consultant to SQUG, prepared the report described here (EQE, November 1983c). This report not only deaggregated the data base but included the 1983 Coalinga earthquake.* Foreign earthquakes and the 1964 Alaska earthquake are surveyed in a separate EQE report, described in Section 2.1.4.3. Average horizontal spectra for each plant are covered in another EQE report, and are described in Section 2.1.4.2.

*The performance of equipment in the Coalinga earthquake is documented in an EQE report, dated August 1984 (see EQE, August 1984).

The tables in this EQE report include a count of equipment found within the power plants and industrial facilities studied. The count is limited to items of the seven types of equipment under study. For horizontal pumps and for air-operated valves the count is approximate and conservatively low because of the large number of these items found in the facilities surveyed. Small pumps, both vertical and horizontal, under 50 horsepower, were not included in the count. Data are included in the table entries in varying levels of detail. In general, more detail was collected on equipment which was most representative of that found in nuclear plants. All equipment listed survived the earthquake without damage, unless otherwise noted.

For each of the seven types of equipment, data are summarized in a series of columns. The data columns vary slightly among the different equipment types. The headings of columns are defined below.

(1) Location/Elevation - This entry locates the floor elevation of equipment with respect to grade elevation within the plant. If the equipment is located in the yard adjacent to the plant structures the location is designated as "ground level."

(2) Number of Assemblies (No. Asm.) - For electrical equipment, an assembly consists of multiple cubicles or cabinets mounted in vertical sections which are bolted together to form a single structure.

(3) Number of Units (No. Units/No. Un.) - For electrical equipment, a unit is defined as one circuit breaker cabinet or one motor controller cubicle mounted within an assembly.

(4) Estimated Peak Ground Acceleration (Est. PGA) - This is the peak horizontal ground acceleration estimated for the particular site as an average of two horizontal components (see Section 2.1.4.2).

(5) Size - For electrical equipment, size includes the width of the assembly in vertical sections. The dimensions of the assembly are also included, although for many entries these numbers are simply estimates based on standard cubicle dimensions. Motor control centers are designated as being double- or single-faced assemblies, with cubicles either mounted in both sides or in only one side of the assembly. For metal-clad switchgear, the operating voltage is noted as either 2.4 or 4.16 kV. Motor control centers and low voltage switchgear always operate at 480 V unless otherwise noted on the table. For pumps, size is designated by the motor horsepower (hp) and by the pump flow rate (gpm) and discharge pressure (in feet of head). The total height of vertical pumps is also included, measuring from the base plate to the top of the motor. The size of valves is designated by the pipe diameter and by the operator height measured from the pipe centerline to the top of the operator. Where accurate data are available, entries for valves include an estimate of the flexibility of the supporting line. Very flexible lines are those with measured or estimated frequencies less than 4 Hz. Moderately flexible lines are those with frequencies between 4 Hz and 10 Hz. Supporting lines would be considered stiff if they had no response frequencies below 10 Hz, and rigid if they had no frequencies below 33 Hz.

(6) Frequency - For a few sample items, measurements were made of the lowest response frequency as an indication of the typical flexibility of the

type of equipment. For electrical equipment, the rocking or overturning frequency of the assembly is noted where measured. For valve operators, the rocking or "cantilever" frequency is noted where measured. Valve operator cantilever frequencies correspond to the response of the operator relative to the supporting piping.

(7) Form - For electrical equipment, details of internal devices are provided where available. Data on specific components are given for typical cubicles or cabinets within an assembly. For example, the major components for a typical motor controller within a motor control center (MCC) may be listed, including the manufacturer and the model number if available. For switchgear, the model number of a typical circuit breaker in the assembly may be noted along with the types of door-mounted relays on the front face of the assembly. For vertical pumps, the type of pump is designated as either a turbine or a centrifugal pump. Vertical turbine pumps include the length of the shaft below the base plate, if known. The means of support for the suction line containing the shaft is also noted if known. Most vertical turbine pump suction casings are supported only at the pump base plate. The suction casing thus forms an inverted cantilever into the source of water below the pump motor. For horizontal pumps, the drive mechanism for the pump is noted as either electric motor, steam turbine, or diesel engine. The drive train is noted as either through a gearbox or transmission, or by a direct connection between motor and pump. The type of pump is noted as either a centrifugal single impeller, a multistage turbine pump, or a screw. For valves, the type of valve is designated (if not covered by insulation). The orientation of the attached operator is noted with respect to the valve.

(8) Attached Piping - For pumps, the diameters of the suction and discharge lines are listed if this information is available.

(9) Manufacturer, Model, Vintage - The manufacturer of the equipment is noted where nameplate data were collected. If a designation of model, size, or type was include on the nameplate, this is noted. The equipment vintage is usually estimated according to the year of construction of the particular unit of the plant.

(10) Internal Details - For electrical equipment a short description is provided of the units which make up the assembly, including variations in the size of cubicles, and the ratio of occupied to blank cubicles in the assembly. An assembly is listed as full if all or nearly all of its available cubicles contain motor controllers (in MCCs) or circuit breakers (in switchgear). Additional details are included, such as the pressure of door-mounted components such as relays, or the inclusion within the assembly of equipment such as transformers.

(11) Installation - The anchorage of the equipment is described where this information was collected. For some entries, the size of anchor bolts are estimates. Any additional supporting structure other than anchorage to the floor (or pipe) is noted.

(12) Photographs Available (Photo Avail.) - Photographs are available for nearly all equipment listed. Exceptions exist for a portion of the horizontal pumps and air-operated valves which are usually found to be repetitions in a particular facility. Where only a portion of the individual

items counted in a table entry have available photos, the photo inventory is listed as "partial."

(13) Catalogue Available (Cat. Avail.) - For a portion of the equipment, manufacturer's catalogues, equipment specifications, or drawings for the particular item have been collected.

A summary table is included in this EQE report for each of the seven types of equipment. The summary table provides a total count of equipment, broken down according to earthquake, data base plant, and elevation with respect to ground. A summary is also provided of the manufacturers and vintage of the equipment, and the performance of the data plant during the earthquake.

The tables are followed by a series of plots in which certain parameters for each equipment type are presented in graphic form.

Typical samples of tables and plots are presented in Tables 2.1-10 through 2.1-16 and Figures 2.1-2 through 2.1-9 for the seven types of equipment for a random selection from various data base plants.

2.1.4.2 EQE Report, "Seismic Experience Data Base--Average Horizontal Data Base Site Response Spectra"

For some of the facilities included in the seismic experience data base, ground motion records were not available at their specific locations. The nearest ground motion record was then used by EQE to extrapolate an estimate of the peak ground acceleration and the shape of the ground motion response spectra at the data base site. This EQE report (EQE, November 1983b) includes plots of the horizontal ground motion response spectra for the various data base sites used in the SQUG studies. The two horizontal ground motion response spectra are plotted as dashed lines for each record. The average response spectrum of the two horizontal components is plotted as a solid line. This average horizontal spectrum is then used for the various data base sites, multiplied by a scaling factor to account for the location of the data base site with respect to the causative fault or the epicenter. As an example, the development of the estimated data base site horizontal response spectra during the February 9, 1971 San Fernando earthquake is described below.

Scaling factors to estimate data base site response spectra for the San Fernando sites were developed by EQE in the following manner. Peak ground accelerations were obtained from the sites of actual ground motion records. These peak ground accelerations are the higher acceleration of the two horizontal components recorded. By comparing the location of the various data base sites with the locations of the records with respect to the causative fault, estimates were made of the peak ground acceleration at the data base sites. These estimates were based on past studies of ground motion attenuation as a function of distance from the fault. The average ground motion response spectrum for the nearest ground motion record was then scaled by the ratio of estimated peak ground acceleration at the data base site to the measured peak ground acceleration at the record site. For the data base sites in the San Fernando Valley, this procedure is summarized in Table 2.1-17.

Figure 2.1-10 shows a map of the San Fernando Valley included to locate data base and ground motion record sites. Figures 2.1-11 and 2.1-12 show the response spectra at Pacoima Dam and Sylmar Converter Station, respectively.

Table 2.1-10 Summary: Motor control centers

Earthquake	Location	Elevation	No. Asm.	No. Units	Est. PGA (g)	Manufacturer, Model, Vintage	Performance During Earthquake
San Fernando 1971	Sylmar	Basement	11	180	0.50	General Electric, Cutler Hammer, ~1970	Facility lost power for several months; no motor controllers required replacement; one assembly slid slightly.
		12 ft.	7	109			
		43 ft.	5	35			
	Valley	Ground floor	6	83	0.30	General Electric, Federal Pacific 1950s	Three units were on-line; two tripped off-line and lost power, one remained on-line. No damage to motor controllers.
		15 ft.	11	218			
	Burbank Olive Plant	Ground floor	5	126	0.32	Westinghouse, Cutler Hammer, ~1960 Electric Machinery late 1960s	Four units were on-line; two tripped off-line, two remained on. All shut down shortly after the earthquake as offsite power was lost. No damage to motor controllers.
	Glendale	Basement	16	162	0.27	Westinghouse, ~1963 General Electric, ~1959 Square D, ~1953	Three units were on-line; all remained on-line.
	Pasadena	Ground floor	1	24	0.18	General Electric, ~1965 Federal Pacific, ~1957	Two units were on-line; both remained on-line
		1[illegible] ft.	2	20			
		33 ft.	1	30			
Imperial Valley 1979	El Centro	Ground floor	3	30	0.42	Westinghouse, ~1957 Square D, ~1968	Two units were on-line; one lost power; one tripped off-line but continued to operate. No damage to motor controllers.
		20 ft.	2	26			
Coalinga 1983	Within 10 km of epicenter	Ground level	7	212	0.60	Nelson Electric, ~1970 Furnace Electric, ~1980 Westinghouse, ~1980 ITE, ~1972 and 1980	All facilities lost power. Two unanchored assemblies slid; two anchored assemblies failed anchorage and slipped. No damage to motor controllers.*
	Within 20 km of epicenter	Ground level	4	25	0.35	Westinghouse, ~1970 General Electric, ~1970	All pumping stations lost power. Motor controllers were not damaged.
Total	--	--	81	1280	--	--	--

*One motor controller unit at the Union Oil Butane Plant was inoperable following the earthquake because of a thermal overload relay that would not reset. Operators at the plant thought that the controller's condition had been noticed before the earthquake, but positive confirmation could not be made.

Table 2.1-11 Motor control centers at the Sylmar Converter Station

Location	No. Asm.	No. Un.	Est. PGA (g)	Size	Frequency (Hz)	Form	Manufacturer, Model, Vintage	Internal Details	Installation	Photo. Avail.	Cat. Avail.
Basement	8	110	0.50	4 sections wide; 90" x 20" x 80"; cubicles on 1 side.	Not measured	Typical unit contains GE CR-106 contactor, circuit breaker, control transformer, pilot lights, & push buttons.	General Electric, 7700 Line Series MCC, ~1970	Cubicles of 3 sizes; assemblies are 2/3 full; assembly includes a switchboard.	Tack welds to embedded base plate in concrete floor; about 6 per assembly.	Yes	Yes
Basement	2	10	0.50	2 sections wide; 90" x 20" x 40"; cubicles on 1 side.	Not measured	Typical unit contains CR-106 contactor, circuit breaker, control transformer, pilot lights, & push buttons.	General Electric, 7700 Line Series MCC, ~1970	Cubicles of 1 size; 1 section are spares.	Tack welds to embedded base plate in concrete floor; about 4 per assembly.	Yes	Yes
Basement	1	60	0.50	8 sections wide; 90" x 20" x 160"; cubicles on both sides.	Not measured	Internals not inspected.	Culter-Hammer, Unitrol, ~1970	Cubicles of 3 sizes; assemblies are 3/4 full; assembly includes a large transformer at one end.	Anchor bolts, 1/4" diameter, at corners of each section.	Yes	No
Second floor 12' above ground	1	32	0.50	5 sections wide 90" x 20" x 100"; cubicles on both sides.	Not measured	Internals not inspected.	Cutler-Hammer, Unitrol, ~1970	Cubicles of 2 sizes; assemblies are 3/4 full; 1 section supports instrumentation rather than motor controllers.	Anchor bolts, 1/4" diameter, at corners of each section.	Yes	No
Second floor 12' above ground	6	77	0.50	3 sections wide; 90" x 20" x 60"; cubicles on 1 side.	Not measured	Internals not inspected, but probably similar to other GE units.	General Electric, 7700 Line Series MCC, ~1970	Cubicles of 1 size; assemblies are full; 1 cubicle supports a door-mounted relay.	Anchor bolts, 1/4" diameter, across center of assembly base.	Yes	Yes
Fourth floor 43' above grade	5	35	0.50	2 sections wide; 90" x 20" x 40"; cubicles on 1 side.	Not measured	Internals not inspected.	Cutler-Hammer, Unitrol, ~1970	Cubicles of 2 sizes; assemblies are 1/2 full.	Anchor bolts, 1/4" diameter, at corners of each section.	Yes	Yes

Table 2.1-12 Summary: Motor-operated valves

Earthquake	Location	Elevation/Piping Flexibility	No. of Valves	Est. PGA (g)	Operator, Manufacturer, Model, Vintage	Performance During Earthquake
San Fernando 1971	Valley	El. 10 ft. Spring-supported feedwater lines	14	0.30	Limitorque, ~1953	Three units were on-line. Two tripped off-line and lost power; one remained on-line. No damage to valves.
		El. 20 ft. Spring-supported feedwater lines	17	0.30	McBain Torkmaster, ~1957	
	Burbank	Ground level Rigid 24" lines	2	0.32	Limitorque, ~1958	Four units were on-line. Two tripped off-line; two remained on. All units shut down shortly after the earthquake as offsite power was lost. No damage to valves.
		El. 20 ft. Very flexible lines	2	0.32	Limitorque, ~1958	
	Glendale	Basement Mezzanine Moderately flexible lines	4	0.27	Limitorque, ~1959	Three units were on-line; all remained on-line. No damage to valves.
		El. 6 ft. Very flexible line	1	0.27	Limitorque, ~1959	
		El. 20 ft. Adjacent to boiler	1	0.27	Limitorque, ~1953	
		El. 60 ft Adjacent to boiler	1	0.27	Limitorque, ~1965	
Imperial Valley 1979	El Centro	Ground level Rigid 24" lines	2	0.42	Limitorque	Two units were on-line. One unit lost power; one tripped off-line but continued operating. No damage to motor-operated valves.
		El. 80 ft. Adjacent to boiler	3	0.42	Limitorque, 1953 - 1968	
Coalinga 1983	Main oil pumping plant	Ground level Short piping runs probably rigid	55	0.60	Limitorque, 1967 - 1980	Plant lost power and all equipment shut down. Some damage to plastic conduit attached to valve motors.
	San Luis Canal pumping stations	Ground level Short piping runs probably rigid	29	0.35	Limitorque, 1963 - 1979	Stations lost power. No damage to valves.
Total		--	131	--	--	--

Table 2.1-13 Motor-operated valves at near-field sites near Coalinga

Location	No. of Valves	Est. PGA (g)	Size	Frequency (Hz)	Form	Operator, Manufacturer, Model, Vintage	Installation	Photo. Avail.	Cat. Avail.
Ground level, Main oil pumping plant	1	0.60	Pipe diameter = 12" Operator ht. = 75" Short span of pipe, probably rigid	Not measured	Gate valve, operator mounted above and to one side of the valve.	Limitorque, Type SMC-03, ~1980 100# wt.	Motor/gearbox bolted to yoke with four 1/2" bolts; shaft bolted to valve with 1/2" bolts.	Yes	Yes
Ground level, Main oil pumping plant	2	0.60	Pipe diameter = 12" Operator ht. = 60" Short span of pipe, probably rigid	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMB, Size 1, ~1967 400# wt.	Motor/gearbox bolted to yoke with eight 1/2" bolts; yoke bolted to valve with eight 1/2" bolts.	Yes	Yes
Ground level, Main oil pumping plant	4	0.60	Pipe diameter = 12" Operator ht. = 40" Short span of pipe, probably rigi	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMB Size 00 ~1967 200# wt.	Motor/gearbox bolted to yoke with eight 1/2" bolts; yoke bolted to valve with eight 1/2" bolts.	Yes	Yes
Ground level, Main oil pumping plant	7	0.60	Pipe diameter = 24" Operator ht. = 90" Short spans of pipe well supported, probably rigid	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMC-03 ~1980 100# wt.	Motor/gearbox bolted to yoke with eight 1/2" bolts; yoke bolted to valve with eight 1/2" bolts.	Yes	Yes
Ground level, Main oil pumping plant	4	0.60	Pipe diameter = 8" Operator ht. = 40" Short spans of pipe, probably rigid	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMC-03, ~1980 100# wt.	Motor/gearbox bolted to yoke with four 3/8" bolts; yoke bolted to valve with four 3/8" bolts.	Yes	Yes
Ground level, Main oil pumping plant	4	0.60	Pipe diameter = 10" Operator ht. = 20" Short spans of pipe, probably rigid	Not measured	Globe valves, operator mounted above and to one side of valves.	Limitorque, Ident. No. B76P0576M-WF ~1967	Motor/gearbox bolted to yoke with four 3/8" bolts; yoke bolted to valve with four 3/8" bolts.	Yes	No
Ground level, Main oil pumping plant	4	0.60	Buried pipe Operator ht. = 20" Short spans of pipe, probably rigid	Not measured	Gate valves, operator projects out of ground directly above valves.	Limitorque, Type SMC-03 ~1980 100# wt.	Motor/gearbox bolted to yoke with four 3/8" bolts; yoke bolted to valve with four 3/8" bolts.	Yes	Yes

Table 2.1-13 Motor-operated valves at near-field sites near Coalinga (continued)

Location	No. Valves	Est. PGA (g)	Size	Frequency (Hz)	Form	Operator, Manufacturer, Model, Vintage	Installation	Photo. Avail.	Cat. Avail.
Ground level, Main oil pumping plant	2	0.60	Buried pipe Operator ht. = 30" Above ground	Not measured	Gate valves, operator projects out of ground and is then offset to one side.	Limitorque, Ident. No. B76P0576M-WF ~1967	Motor/gearbox bolted to yoke with four 3/8" bolts; yoke bolted to valve with four 3/8" bolts.	Yes	No
Ground level, Main oil pumping plant	2	0.60	Pipe diameter = 24" Operator ht. = 70" Short spans of pipe, probably rigid	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMB Size 1 ~1967	Motor/gearbox bolted to yoke with eight 1/2" bolts; yoke bolted to valve with eight 1/2" bolts.	Yes	Yes
Ground level, Main oil pumping plant	1	0.60	Buried pipe Operator ht. 50"	Not measured	Gate valves, operator mounted directly above.	Limitorque Type SMC-03 ~1980 100# wt	Motor/gearbox bolted to yoke; yoke bolted to valve with four 3/8" bolts.	Yes	Yes
Ground level, Main oil pumping plant	5	0.60	Buried pipe Operator ht. = 30"	Not measured	Gate valves, operator mounted directly above.	Limitorque, Type SMC-03 ~1980 100# wt.	Motor/gearbox bolted to yoke; yoke bolted to valve with four 3/8" bolts.	Yes	Yes
Ground level, Main oil pumping plant	5	0.60	Pipe diameter = 24" Operator ht. = 96" Short spans of pipe, probably rigid	Not measured	Gate valves, operator mounted directly above.	Limitorque (no name-plate)	Motor/gearbox bolted to yoke; yoke bolted to valve with four 1/2" bolts.	Yes	No
Ground level, Main oil pumping plant	8	0.60	Buried pipe Operator ht. = 40" Above ground	Not measured	Gate valves, operator mounted directly above.	Limitorque Type SMC-03 ~1980 100# wt.	Motor/gearbox bolted to yoke; yoke bolted to valve with eight 5/8" bolts.	Yes	Yes
Ground level, Main oil pumping plant	6	0.60	Buried pipe Operator ht. = 36" Above ground	Not measured	Gate valves, operator mounted directly above.	Limitorque Type SMB Size 0 ~1967 300# wt.	Anchorage not visible.	Yes	Yes

Table 2.1-14 Motor-operated valves at far-field sites near Coalinga

Location	No. of Valves	Est. PGA (g)	Size	Frequency (Hz)	Form	Operator, Manufacturer, Model, Vintage	Installation	Photo. Avail.	Cat. Avail.
San Luis Canal Pumping Station 20-R	4	0.35	Pipe diameter = 8"-16" Operator ht. = 16" Short spans of pipe, probably rigid.	Not measured	Butterfly valves; operator mounted to one side.	Limitorque, Type SMC-04, 1979 100# wt.	Motor/gearbox mounted atop worm gear actuator; actuator bolted to valve flange with four 3/4" bolts.	Yes	Yes
San Luis Canal Pumping Stations 21-R & 22-R	8	0.35	Pipe diameter = 8"-16" Operator ht. = 18" Short spans of pipe, probably rigid.	Not measured	Butterfly valves; operator mounted to one side.	Limitorque, Type H1BC-SMB-00, 1976 200# wt.	Motor/gearbox mounted atop worm gear actuator; actuator bolted to valve flange with four 3/4" bolts.	Yes	Yes
Pumping Stations 16-RC & 14-RC	9	0.35	Pipe diameter = 8"-14" Operator ht. = 18" Short spans of pipe, probably rigid.	Not measured	Butterfly valves; operator mounted to one side.	Limitorque, Type SMC-04, 1978 100# wt.	Motor/gearbox mounted atop worm gear actuator; actuator bolted to valve flange with two 1/2" bolts.	Yes	Yes
Pumping Station 7-1	4	0.16	Pipe diameter = 24" Operator ht. = 24" Short spans of pipe, probably rigid.	Not measured	Butterfly valves; operator mounted to one side.	Limitorque, Type H, 1963. 200# wt.	Motor/gearbox mounted atop worm gear actuator; actuator bolted to valve flange with two 3/4" bolts.	Yes	Yes
Pumping Station 16-RA	4	0.35	Pipe diameter = 10"-20" Operator ht. = 18"	Not measured	Butterfly valves; operator mounted to one side.	Limitorque, Type SMB-00, 1979 200# wt.	Motor/gearbox mounted atop worn gear actuator; actuator bolted to valve flange with two 3/4" bolts.	Yes	Yes

Table 2.1-15 Vertical pumps

Earthquake	Location	Elevation	No. of Pumps 50-200 hp	No. of Pumps >200 hp	Est PGA (g)	Manufacturer, Model, Vintage	Performance During Earthquake
San Fernando 1971	Valley	Ground floor	20	4	0.30	Motors - General Electric, Elliot, Westinghouse, US Electric.	Three units were on-line. Two tripped off-line and lost power; one remained on-line. No damage to pumps.
		El. 20 ft	8	0		Pumps - Johnston, Byron-Jackson, Peerless, United. 1954-1956.	
	Burbank	Ground floor	4	2	0.32	Motors - Allis Chalmers, General Electric, US Electric. Pumps - Byron-Jackon, 1960.	Four units were on-line. Two tripped off-line; two remained on. All units shut down shortly after earthquake as off-site power was lost. No damage to pumps.
	Glendale	Basement	6	0	0.27	Motors - General Electric, Allis Chalmers.	Three units were on-line; all remained on-line.
		Ground level	1	2		Pumps - Byron-Jackson, Peerless, US Pump, 1941-1964.	
	Pasadena	Ground level	0	4	0.18	Motors - General Electric, Pumps - Foster-Wheeler, 1957.	Two units were on-line; both remained on-line.
Coalinga 1983	Facilities within 10 km of epicenter	Ground level	0	8	0.60	Motors - Westinghouse, Siemans-Allis, US Electric. Pumps - Byron-Jackson, Union, Verilline. 1967-1980.	All facilities lost power and shut down. No damage to pumps.
	Pleasant Valley	Ground level	0	9	0.49	Motors - Toshiba Shiburu. Pump - Ebaru, 1969.	Plant lost power and all equipment shut down. No damage to pumps.
	San Luis Canal	Ground level	29	27	0.35	Motors - General Electric, Westinghouse, US Electric. Pumps - Peabody Floway, 1970-1979.	Stations lost power and all equipment shut down. All pumps were operable following the earthquake. A few pumps displayed excessive vibration because of worn bearings.
	Pump Station 7-1	Ground level	0	4	0.16	Motors - General Electric. Pumps - Fairbanks, Morse, 1963.	Station was down at time of earthquake. No damage to equipment.
	Total		68	60			

Table 2.1-16 Vertical pumps at near-field sites near Coalinga

Location	No. Pumps	Est. PGA (g)	Size	Form	Attached Piping	Manufacturer, Model, Vintage	Installation	Photo. Avail.	Cat. Avail.
Ground level Main oil pumping plant	2	0.60	Motor - 300 hp. Pump - no nameplate. Total ht. = 8 ft.	Turbine pump; shaft length unknown.	12" suction 24" discharge	Motor - Westinghouse, Lifeline Induction Motor. Pump - no nameplate ~1967.	Base of pump anchored to concrete pad with twelve 1" bolts.	Yes	No
Ground level Main oil pumping plant	2	0.60	Motor - 500 hp. Pump - 3500 gpm, 271 ft. head. Total ht. = 9 ft.	Turbine pump; shaft length unknown.	16" suction 16" discharge	Motor - Seimans-Allis Induction Motor. Pump - Byron-Jackson ~1980.	Base of pump anchored to concrete pad with four 1" bolts.	Yes	No
Ground level Water filtration plant	4	0.60	Motor - 700 hp. Pump - no nameplate. Total ht. - 10 ft.	Turbine pump; shaft length = 20 ft.	12" discharge	Motor - U.S. Electric. Pump - Veriline Turbine Pump, ~1970.	Base of pump bolted to concrete with four 1/2" bolts.	Partial	No
Pleasant Valley Pumping Plant	9	0.49	Motor - 7000 hp. Pump - 225 ft.^{3}/sec., 197 ft. head.	Centrifugal pump; motor and pump on different floors, connected by a 30-ft. drive shaft.	Suction from canal, 36" discharge line	Motor - Toshiba Shiburu Type TAK. Pump - Ebaru centrifugal pump Type 54-39VLM, 1969.	The motors are built into a concrete pedestal on the ground floor; the pump is mounted on the basement floor below the canal water line.	Partial	No

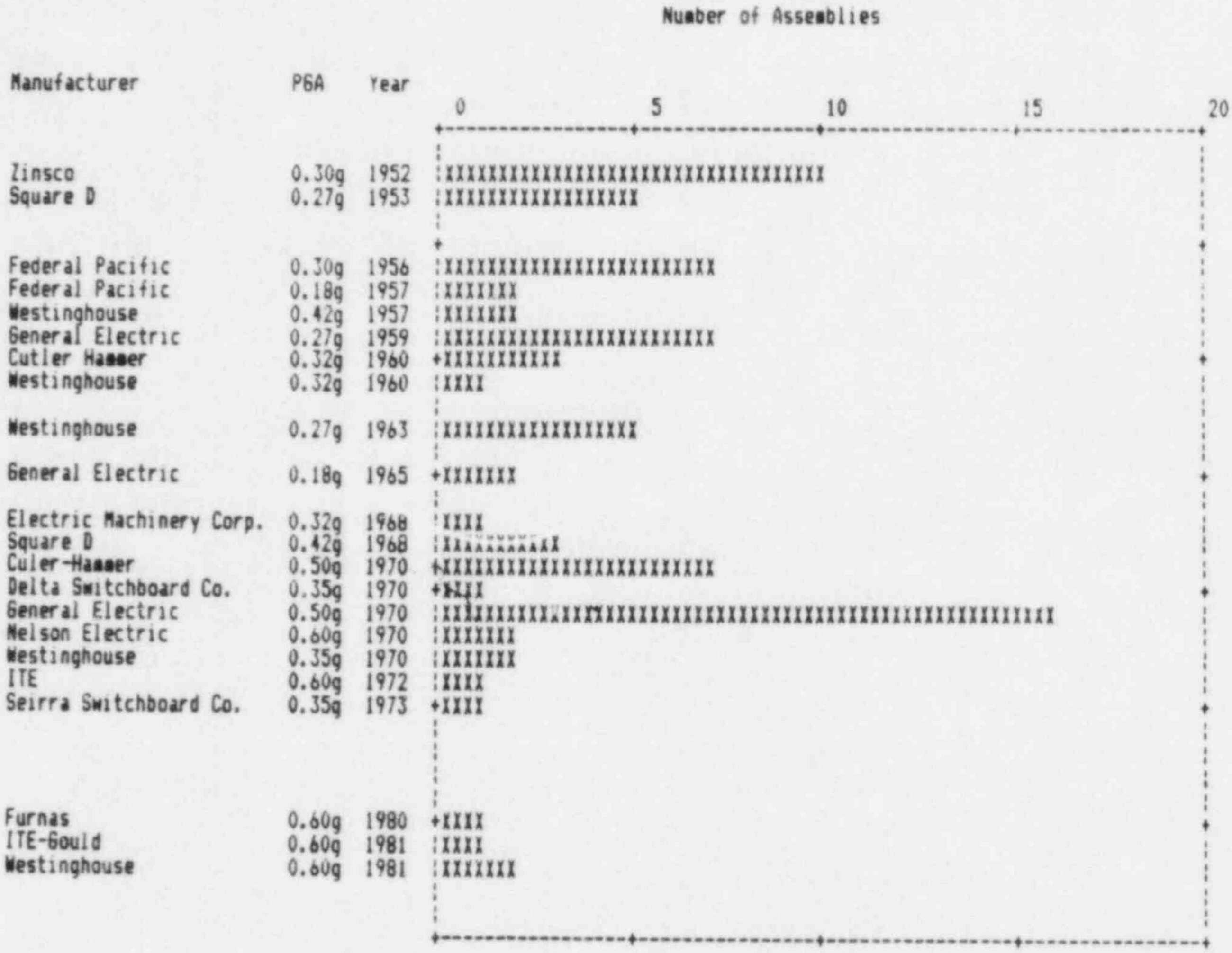


Figure 2.1-2 Distribution of motor control centers as a function of vintage, manufacturer, acceleration, and number of assemblies

2.1.4.3 EQE Report, "Investigation of Equipment Performance in Foreign Earthquakes and the 1964 Alaska Earthquake"

The equipment earthquake experience data base compiled by EQE for the SQUG project during the pilot program indicates a lack of failure for the seven types of equipment considered. The data base equipment was subjected to seismic motions comparable to the design earthquakes for the operating nuclear power plants in the eastern U.S. However, it does not include any data from earthquakes outside the U.S. The possibility of discovering numerous equipment failures during well-known earthquakes not investigated by the project is a serious concern on the part of SSRAP.

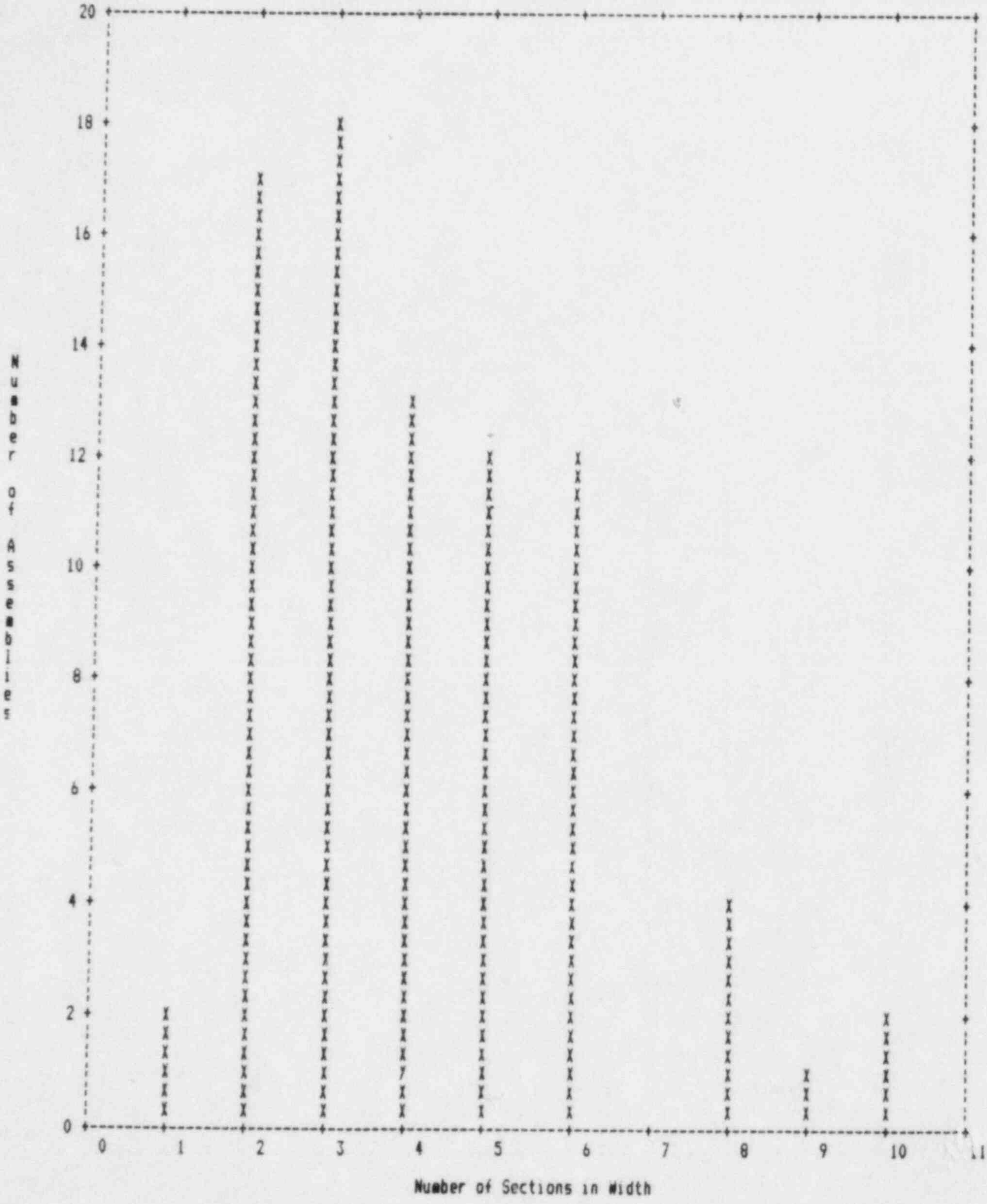


Figure 2.1-3 Motor control centers surviving PGA $\geq$ 0.18 g, data base of motor control centers plotted as a function of width in sections

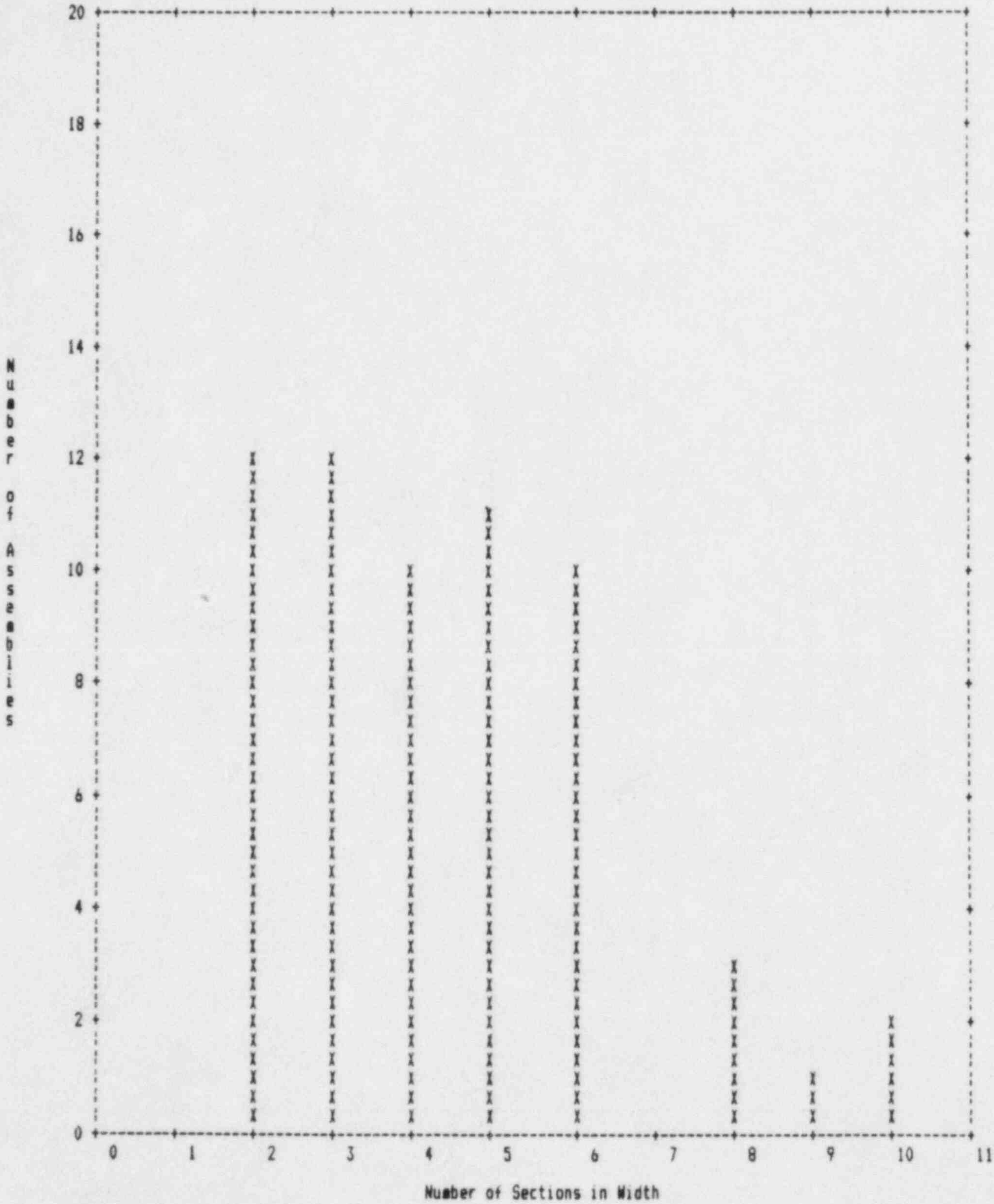


Figure 2.1-4 Motor control centers surviving PGA $\geq$ 0.28 g

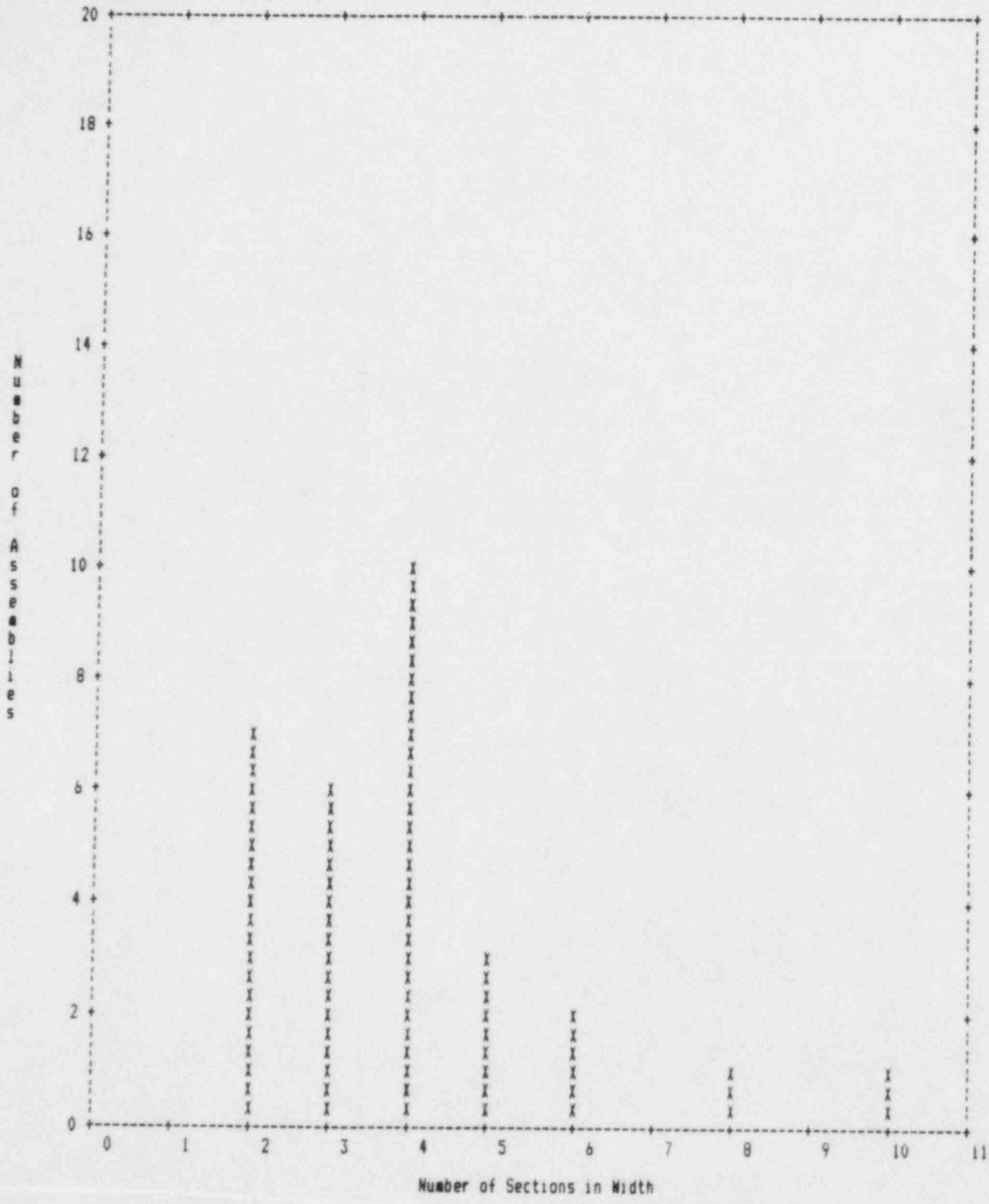
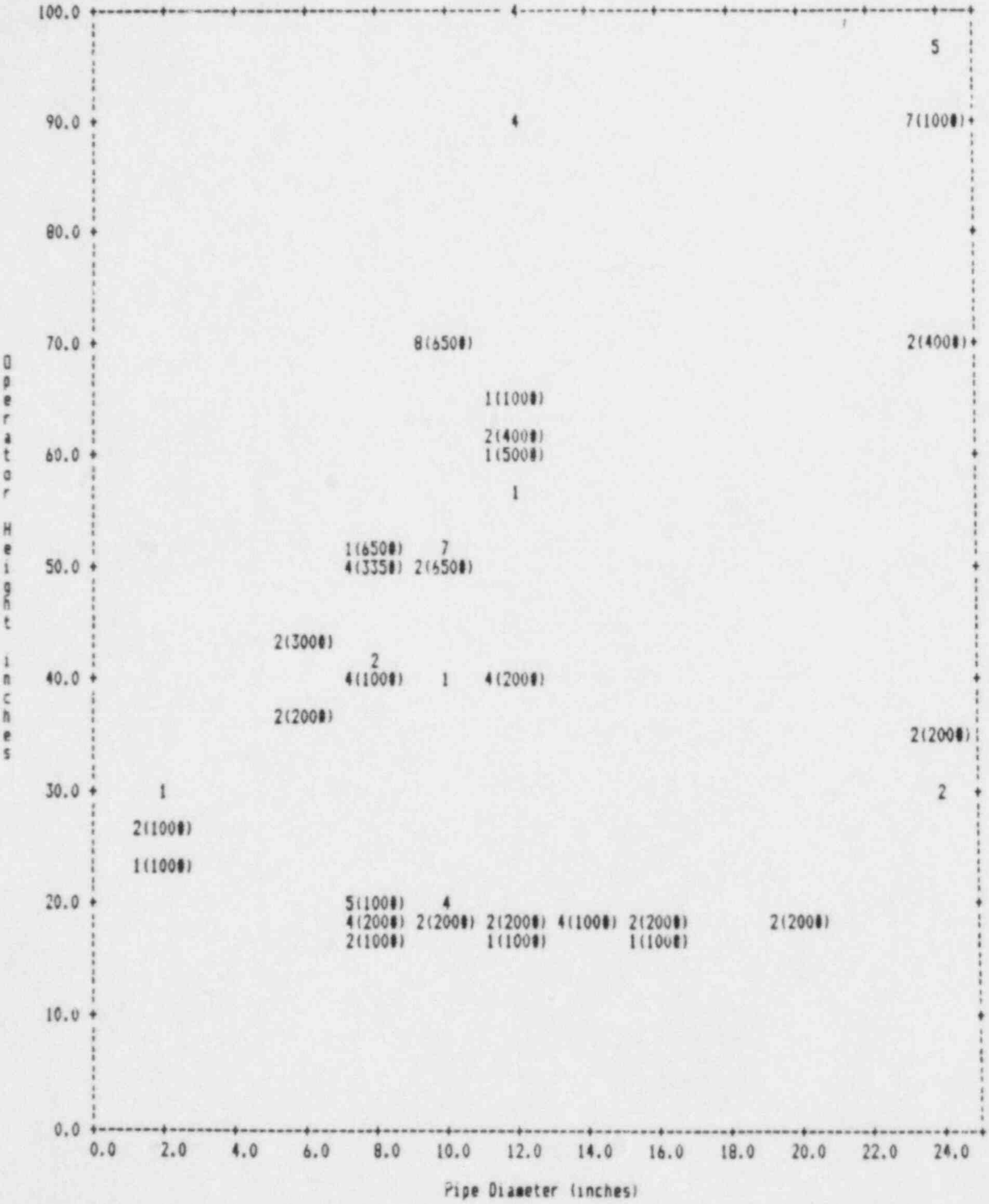
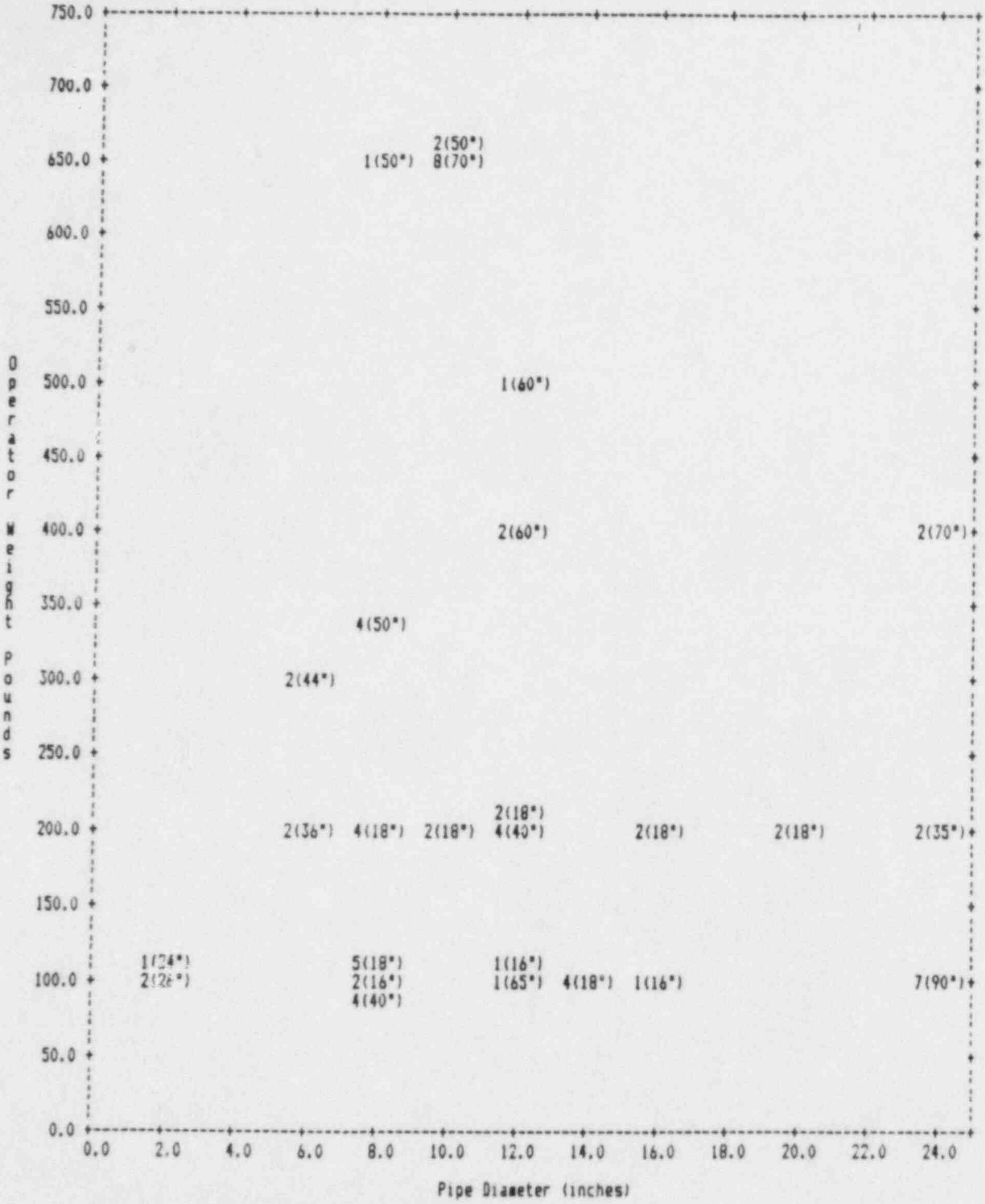


Figure 2.1-5 Motor control centers surviving PGA $\geq$ 0.45 g



Key: Number of valves (operator weight #)
Operator weight is omitted if data are not available.
Buried pipes are omitted

Figure 2.1-6 Motor-operated valves surviving PGA $\geq$ 0.18 g, data base of motor-operated valves plotted as a function of supporting pipe diameter and operator height



Key: Number of valves (operator height ")
Only valves for which operator weights and pipe diameters are known are plotted.

Figure 2.1-7 Motor-operated valves surviving PGA $\geq$ 0.18 g, data base of motor-operated valves plotted as a function of supporting pipe diameter and operator weight

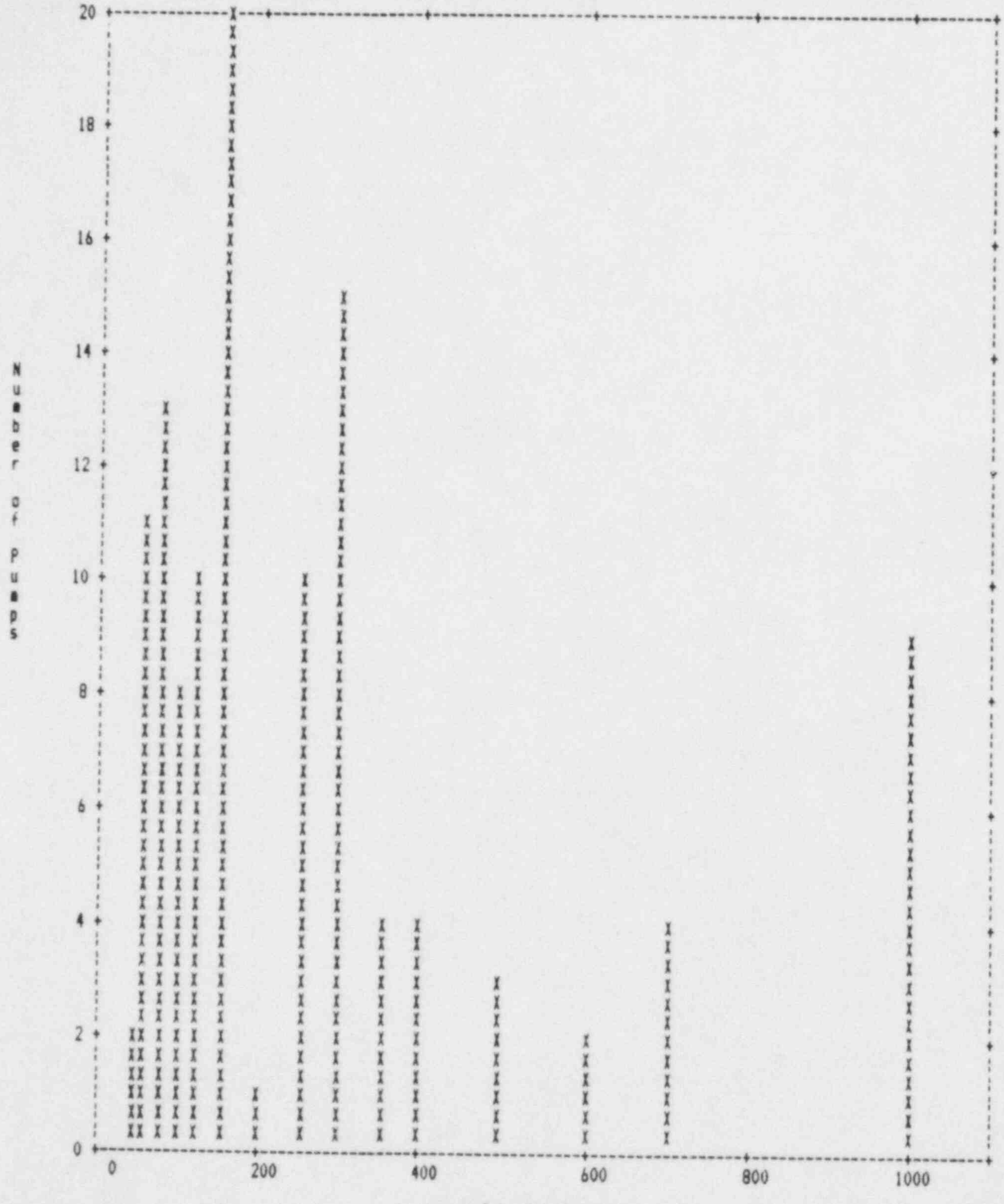
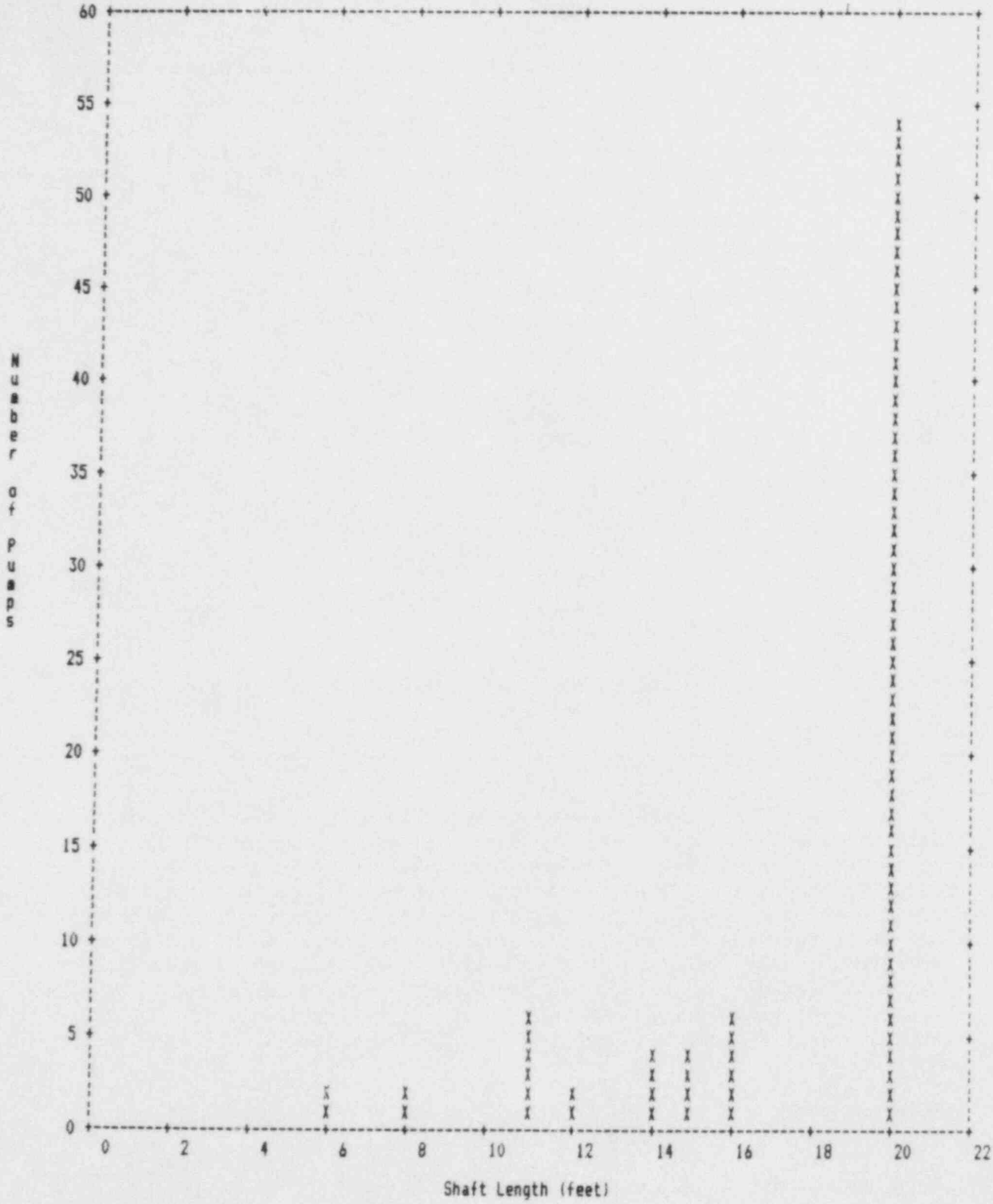


Figure 2.1-8 Vertical pumps surviving PGA $\geq$ 0.18 g, data base of vertical pumps plotted as a function of pump horsepower



NOTE: Only pumps for which shaft length data are available are plotted.

Figure 2.1-9 Vertical turbine pumps surviving PGA $\geq$ 0.18 g, data base of vertical turbine pumps plotted as a function of shaft length

Table 2.1-17 Procedure to estimate data base site response spectra

Ground Motion Record Site	Measured Peak Horizontal Ground Accelerations (two components)	Data Base Site	Estimated Peak Horizontal Ground Acceleration at Data Base Site	Scaling Factor for Response Spectrum	Estimated Average Horizontal Acceleration at Data Base Site
Pacoima Dam	1.25g, 1.24g	Sylmar Converter Station	0.50g	$\frac{0.50g}{1.25g}$	0.50g
Orion Blvd.	0.27g, 0.14g	Valley Generating Plant	0.40g	$\frac{0.40g}{0.27g}$	0.30g
Broadway Ave., Glendale	0.28g, 0.23g	Burbank Power Plant	0.35g	$\frac{0.35g}{0.28g}$	0.32g
		Glendale Power Plant	0.30g	$\frac{0.30g}{0.28g}$	0.27g
Milikan Library, Cal. Tech.	0.22g, 0.18g	Pasadena Power Plant	0.20g	$\frac{0.20g}{0.22g}$	0.18g

Because of this concern EQE studied selected foreign earthquakes and the 1964 Alaska earthquake (magnitude 8.4). The findings and conclusions of this study are covered in an EQE report titled "Investigation of Equipment Performance in Foreign Earthquakes and the 1964 Alaska Earthquake" (EQE, November 1983a). The Alaska earthquake is of interest mainly because of its long strong motion duration, which may be a characteristic of the larger eastern U.S. earthquakes. This study was not performed just to collect more detailed data similar to data already collected. This study was performed, however, to assure it is most unlikely in the future that numerous equipment failures will occur (during earthquakes) which have not been studied by the project. A summary of the report follows.

This study addressed the same seven equipment types considered in the SQUG pilot program. The study was undertaken in three parts:

(1) A survey of U.S. experts

(2) A literature survey of equipment performance in the 1964 Alaska earthquake

(3) A literature survey of equipment performance in foreign earthquakes.

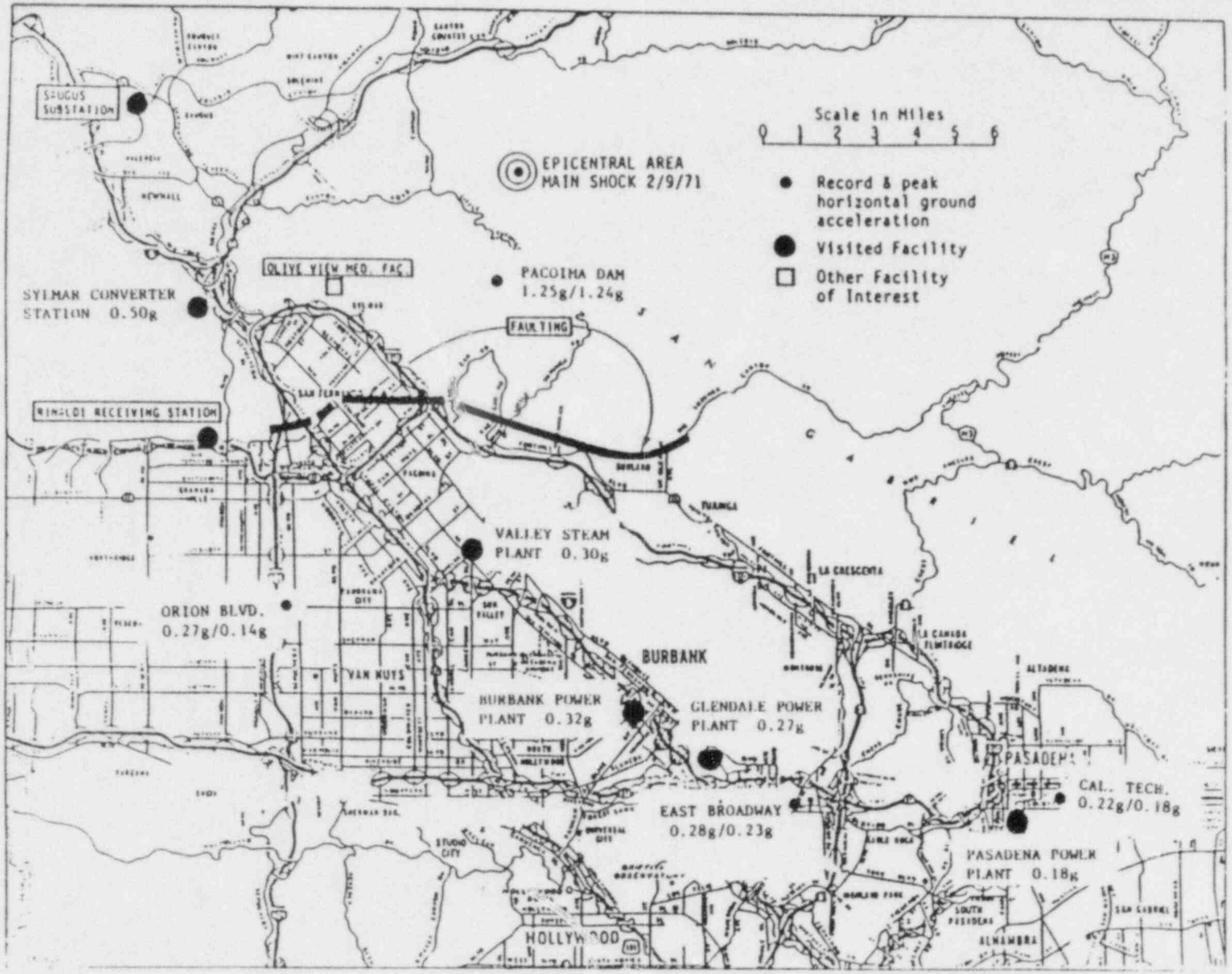


Figure 2.1-10 Location of the San Fernando Valley data base sites and the ground motion records which are the basis for the estimated average peak horizontal ground accelerations

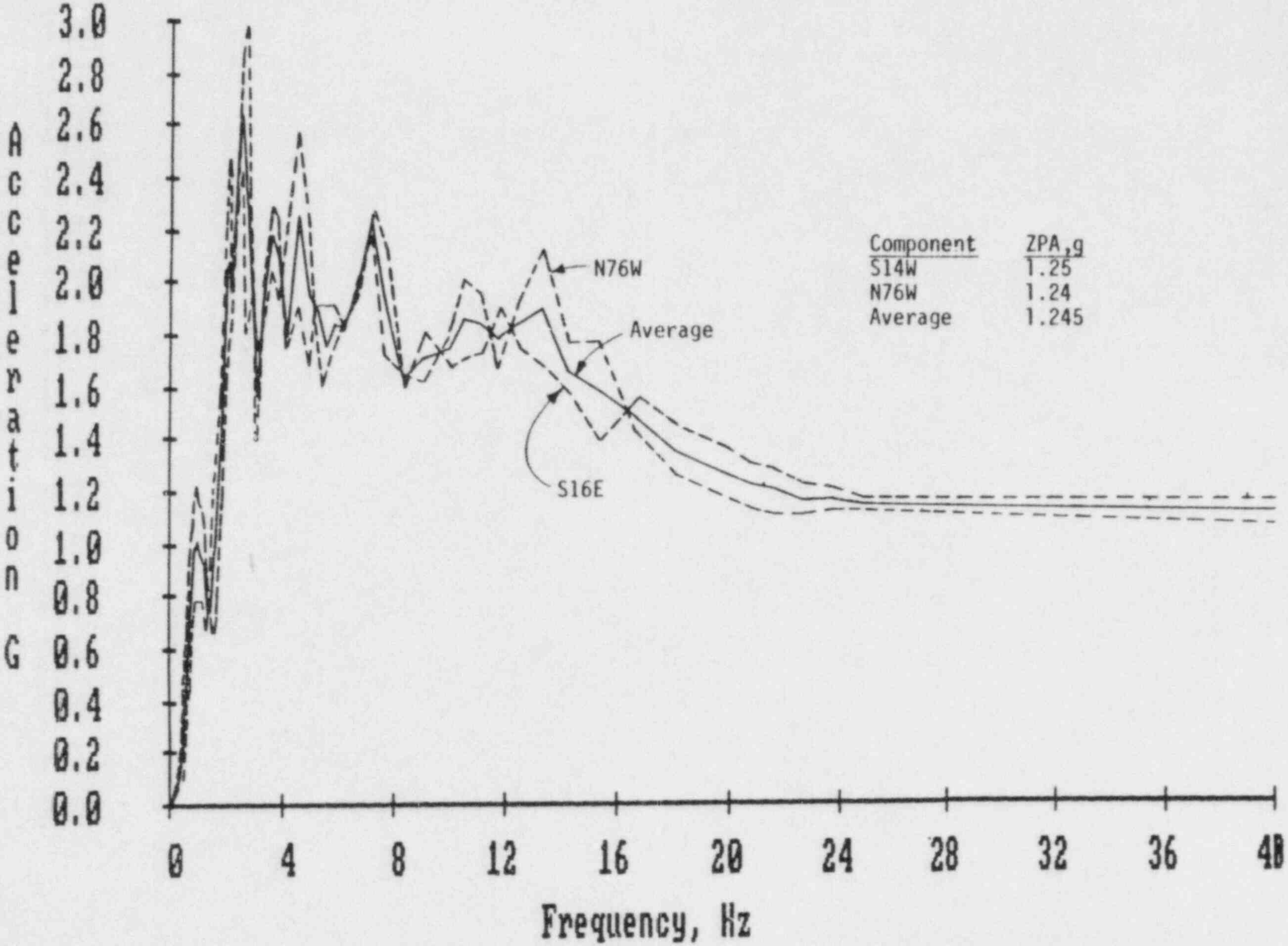


Figure 2.1-11 Response spectra for the ground motion record at Pacoima Dam, 5% damping, two horizontal components and their average

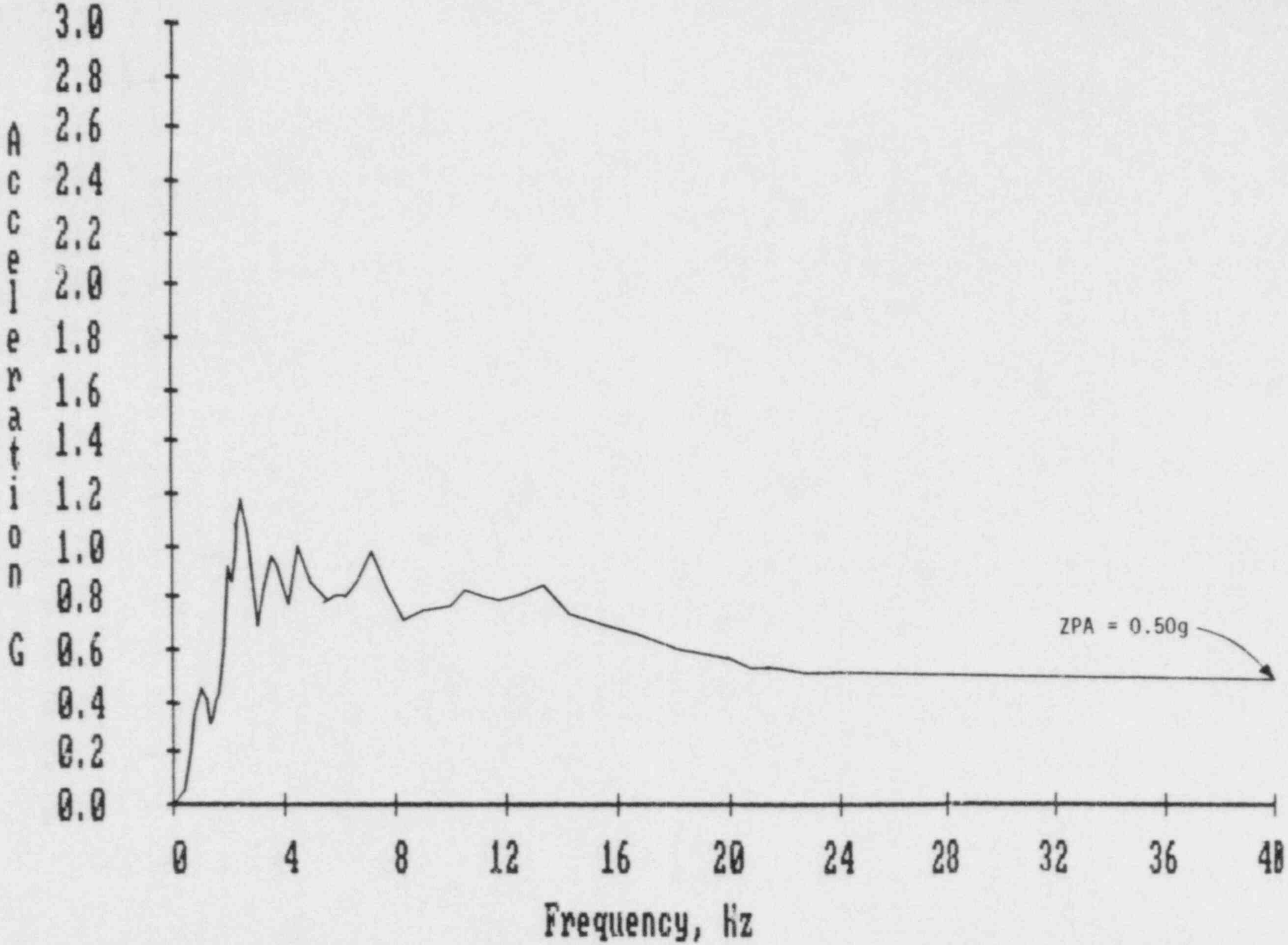


Figure 2.1-12 Average horizontal response spectrum, Sylmar Converter Station, 5% damping, based on the ground motion record at Pacoima Dam, scaled by a factor of 0.50/1.25

2.1.4.3.1 Survey of U.S. Experts

In this survey, a list of 36 U.S. power people with expertise and experience insofar as the equipment under consideration was developed. A questionnaire was prepared and mailed to all the experts, inquiring whether they have seen or are aware of failures of any of the equipment of concern in past earthquakes. A sample questionnaire is included as Figure 2.1-13. The questionnaires were followed up with telephone calls. When an expert returned the questionnaire indicating having seen or having knowledge of equipment failures, the indicated failures were researched further by EQE; most of the reported failures were for equipment other than those seven types included in the scope of this study such as substation equipment, ceramic failures, etc.) or were failures of equipment without adequate anchorage or no anchorage at all. Few failures were found relevant to this study.

In general, the experts expressed a very strong interest in the study. They also typically expressed a strong lack of confidence in the completeness of their observations. Most did not look at anything more than the performance of anchorages and attachments, such as pipes, nozzles, and valve operators. All of the experts were specifically asked if they were aware of severely damaged internals of equipment in the seven classes. Typically they were unaware of such damage and felt that they would have probably been informed at the time by the owners/operators of the equipment or would have certainly noticed gross failures. Several of the experts from the California utilities had specifically sought such information. Their comments have also been included in the findings of the survey.

EQE concluded that, on the basis of the returned questionnaires and the conversations with the experts, no major failures of the seven types of equipment in past earthquakes have been uncovered.

2.1.4.3.2 Literature Survey of Equipment Performance in the 1964 Alaska Earthquake

In this literature survey, EQE reviewed available investigative reports on the 1964 Alaska earthquake. Specifically, the effects of the earthquake on the power and industrial facilities were studied; any reports of equipment failures were noted. In addition, EQE made telephone inquiries to personnel from power plants affected by the 1964 earthquake.

The Alaska earthquake occurred at 5:36 p.m. AST on March 27, 1964. It had a Richter magnitude of about 8.4. The energy release was probably about twice that of the 1906 San Francisco earthquake. The quake produced shaking that lasted, at some locations, for 3 to 4 minutes with two strong segments caused by at least 6 separate earthquakes. The damaging portion of the earthquake is estimated to have lasted about a minute. No strong motion records were obtained. What is known about the nature of the earthquake was obtained or deduced from witnesses, damage investigations, and seismographic information. The peak ground acceleration at Anchorage was estimated to be about 0.2 g. The quake was caused by faulting along a plane extending from Kodiak Island to the vicinity of Valdez. The epicenter was in the Prince William Sound area and the main energy release was somewhat south of Montague Island.

1. Have you visited any of the areas affected by the following earthquakes?

Earthquake	Yes	No
1964 Alaska (Prince William Sound)	___	___
1972 Managua, Nicaragua	___	___
1976 Friuli, Italy	___	___
1978 Miyagi-Ken-Oki, Japan	___	___
1980 Campania-Basilicata, Italy	___	___
1983 Akita Prefecture, Japan	___	___

2. Have you visited any other earthquake areas? Please list:

____
____
____
____
____

3. In the areas you visited, did you survey any power plants or facilities housing the equipment of interest (see Question 4 for a list)?

Yes ___ No ___ If "Yes", how many facilities surveyed? ______

4. Have you seen or are you aware of failures due to earthquake of any of the following equipment types common to power plants?

Equipment	Any Failures? Yes	No	Where?
Motor control centers	___	___	________________
Low voltage (480 V) switchgear	___	___	________________
Metal-clad (2.4-4 kV) switchgear	___	___	________________
Motor-operated valves	___	___	________________
Air-operated valves	___	___	________________
Horizontal pumps and motors	___	___	________________
Vertical pumps and motors	___	___	________________

5. Please provide, to the extent possible, a description of the failures if you marked "Yes" for any of the equipment listed in Question 4 above. References to published material would be very helpful. If uncertain, please tell us all you can; we will search for the information.

Figure 2.1.13 Questionnaire

In spite of the great magnitude and destructive power of the earthquake, the number of communities and facilities affected was relatively small because the event occurred in a region with a population of only 140,000. It did cause considerable destruction and 131 people died, 115 in Alaska and 16 in British Columbia, Oregon, and California. A major portion of the damage was due to landslides and soil subsidence. Further damage was caused by tsunamis, particularly at Valdez, Seward, and Kodiak, in Alaska, and in Crescent City, California. Some facilities were directly damaged by vibration, which, although significant, was small compared with that from other causes. Many large and small buildings were severely damaged or collapsed as a result of the vibratory effects of the earthquake. Some of these buildings were presumably well designed and constructed.

Eight significant power facilities (all relatively small) and many minor facilities in the area were affected by the earthquake. The eight facilities were:

(1) City of Anchorage gas turbine plant in Anchorage
(2) Chugach Power Plant at Knik Arm
(3) Fort Richardson Heat and Power Plant (on northeast outskirts of Anchorage)
(4) Elmendorf Field Heat and Power Plant (on northwest outskirts of Anchorage)
(5) Bureau of Reclamation Eklutna hydroelectric plant at the end of Knik Arm
(6) Chugach Bernice Lake gas turbines near Kenai
(7) Chugach Cooper Lake hydroelectric plant 25 miles from Seward
(8) Port of Whittier Heating and Power Plant in Whittier.

The performance of these facilities was summarized by F.F. Mautz (EQE, November 1983a) reporting on his April 22-28, 1964, visit to Alaska:

> In summary, electric generating and distributing facilities in Alaska withstood the earthquake quite well. Except for very local damage to certain plants none of it was severe enough to cause the plant to be out of action for any length of time. All plants continued to operate for some length of time during and after the earthquake until forced off the line, generally by some circumstance or problem outside of the plant proper. In one case damage was due to earth sliding rather than earthquake shaking proper. All plants were quickly restored to operating condition, and at the time of inspection all were in full operating capability, even though emergency repairs were still being carried out in some cases.

In Appendix B the purpose of each of the eight electric power facilities listed above is given.

The reported instances of failures in Alaskan power plants related to the seven basic types of equipment being reviewed in the SQUG project are the following:

(1) Control panel at Elmendorf Air Force Base power plant upset. It had been anchored with one 1/2-in. bolt in each corner (NAS, 1973, p. 946).

(2) Air-operated valves at Elmendorf Air Force base power plant opened automatically on four tanks when an instrument air control line broke. This resulted in the loss of 60,000 gallons of treated boiler feedwater (NCEL, June 26, 1964, p. 4).

(3) Motor starter circuit breaker damaged at Fort Richardson (NAS, 1973, p. 402).

(4) A fan motor starter and two small motors burned out at Fort Richardson Power Plant. "Motor burnouts were also reported in other buildings where motors were damaged by falling debris but most burnouts were probably caused by motor starting under low-voltage conditions when power was restored after the earthquake." (NAS, 1973, p. 402).

(5) A control cabinet at the City of Anchorage gas turbine plant tore loose .t floor anchors and fell over (NAS, 1973, p. 1053, & DC, 1967, p. 29).

There were other failures in the plants and substations which caused problems, e.g., broken water lines, an ash hopper which fell, damaged buildings, and toppled regulators and transformers. All of these items had to be repaired before operations were restored to normal.

Some equipment failures were reported in buildings and facilities other than power plants:

(1) A control panel fell over at Elmendorf AFB hospital. It was not bolted down (NAS, 1973, p. 353).

(2) Relays were damaged at an L Street apartment building when a selector panel door on which they were mounted swung open (NAS, 1973, p. 353).

(3) A circuit breaker panel at Fort Richardson Barracks failed when the copper-bronze bolts failed on a copper bus (NAS, 1973, p. 399).

(4) Control panels toppled over at various places (unnamed). They were not anchored (NAS, 1973, p. 400).

(5) Valve (hand operated) opened slightly as a result of shaking at Whittier Union Oil Company West Camp area. This resulted in 3,000 barrels of leakage through a broken pressure relief line (NAS, 1973, p. 1101).

2.1.4.3.3 Literature Survey of Equipment Performance in Foreign Earthquakes

The purpose of the study was to investigate, from a review of the literature, the performance of power plant equipment in some significant earthquakes in foreign countries. The study is concerned specifically with major failures of the seven types of equipment currently considered by the SQUG program.

Four significant earthquakes were studied: 1972 Managua, Nicaragua (magnitude 6.2); 1976 Friuli, Italy (magnitude 6.5); 1978 Miyagi-Ken-Oki, Japan (magnitude 7.4); and the 1980 Campania-Basilicata, Italy (magnitude 6.8). For each of these, the earthquake characteristics, ground motion records, if any, and the general effects of the earthquake are reviewed. The effects of the earthquakes on power and industrial facilities are studied in more detail. Summary of this study is presented in Appendix C.

The study did not uncover any mass failures of the equipment of interest. Reports of failures were found; however, most of the reported damage for equipment was attributable to anchorage failures. Those few reports of failures that would concern the SQUG project are highlighted in the report.

2.1.4.3.4 Conclusions and Staff Comments on Alaskan and Foreign Earthquakes

The purpose of this effort was to ascertain that the earthquakes not studied in the SQUG pilot program have not caused numerous failures of any of the seven types of equipment. The study failed to discover numerous failures.

Through the survey of experts and the literature review studies, some reported failures were discovered. However, such failures were very few and did not indicate any trend. Whenever detailed information about the failures was available, such information was recorded.

Descriptions given in most reports were found incomplete, and in some cases, contradictory to other reports. This is mainly because, especially in the investigations of the earlier earthquakes, the investigators were rarely interested specifically in equipment performance other than anchorage failures. Also, very few reports are available specifically on equipment performance. Details of reported failures can be obtained only through onsite investigations.

2.1.5 Summary of SSRAP Report, "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants"

The SSRAP completed its study and made an oral presentation to the NRC staff and SQUG/EQE on December 15, 1983, to outline its conclusions and recommendations on the use of non-nuclear seismic experience data. The SSRAP conclusions and recommendations were documented in a report titled "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants," which was published in February 1984, and revised in August 1984 (SSRAP, February 1984).

The SQUG pilot program studied only seven classes of equipment. SSRAP, after its study, concluded that there are adequate data on "unit substation transformers" and included them in their recommendations.

The SSRAP assessment was primarily based upon past earthquake performance data provided to SSRAP by the Seismic Qualification Utility Group (SQUG) through its consultant, EQE Incorporated. Detailed reviews were conducted by EQE on the performance of the eight classes of equipment at:

(1) Several conventional power plants (Valley Steam Plant, Burbank Power Plant, Glendale Power Plant, and Pasadena Power Plant) and the Sylmar Converter Station subjected to the 1971 San Fernando earthquake (magnitude 6.5).

(2) The El Centro Steam Plant subjected to the 1979 Imperial Valley earthquake (magnitude 6.6).

(3) Pumping stations and refineries near the 1983 Coalinga earthquake (magnitude 6.5).

In addition, much more limited reviews were conducted at several electrical substations for the 1971 San Fernando earthquake, the Ormond Beach Plant and one substation subjected to the 1973 Point Mugu earthquake (magnitude 5.9), and at the Ellwood Peaker Plant and the Goleta Substation subjected to the 1978 Santa Barbara earthquake (magnitude 5.1). Limited literature reviews searching for reported failures of equipment in these eight classes were conducted for the 1964 Alaska (magnitude 8.4), 1952 Kern County (magnitude 7.4), 1978 Miyagi-Ken-Oki, Japan (magnitude 7.4), 1976 Friuli, Italy (magnitude 6.5), and 1972 Managua, Nicaragua (magnitude 6.2), earthquakes.

Some of this work was initiated at the request of SSRAP and all of this work was carefully reviewed by SSRAP.

All members of SSRAP performed walk-throughs of the Sylmar Converter Station, Valley Steam Plant, and Glendale Power Plant, and the SSRAP members spoke with operators present at the Sylmar Converter Station and the Glendale Power Plant during or shortly after the 1971 San Fernando earthquake. In addition, at least one SSRAP member is familiar with equipment in these eight classes at the El Centro Steam Plant and at some of the pumping stations and refineries used in the Coalinga data base. All members of SSRAP have conducted walk-throughs of at least three different types of nuclear power units for the purpose of reviewing these eight classes of equipment. Several members have conducted similar walk-throughs of many additional nuclear power plant units. The purpose of these walk-throughs was to judge similarity between the equipment in nuclear power plants and that in the conventional plants from which past earthquake experience data were collected. SSRAP and vendors of some of these classes of equipment discussed the similarity between equipment installed in nuclear plants and equipment in conventional plants. A partial list of the material reviewed by SSRAP is given in the bibliography to the report (SSRAP, February 1984). Lastly, SSRAP relied on the extensive collective experience of its five members with these eight classes of equipment.

After a detailed and careful review of the full range of the available experience data base, the SSRAP conclusions for these eight classes of equipment are:

(1) Equipment installed in nuclear power plants is generally similar to and at least as rugged as that installed in conventional power plants.

(2) This equipment, when properly anchored, and with some reservations (to be discussed later), has an inherent seismic ruggedness and a demonstrated capability to withstand significant seismic motion without structural damage.

(3) For this equipment, functionality after the strong shaking has ended has also been demonstrated, but the absence of relay chatter during strong shaking has not been demonstrated.

Therefore, with several important caveats and exclusions as discussed below, it is SSRAP's judgment that for excitations below the defined seismic motion bounds, it is unnecessary to perform explicit seismic qualification of existing equipment in these eight classes for operating nuclear power plants to demonstrate functionality after the strong shaking has ended. The existing data base reasonably demonstrates the seismic ruggedness of this equipment up to these seismic motion bounds. Secondly, it only applies to functionality after the strong shaking has ended. Third, there are exceptions as denoted in subsequent sections. Fourth, the conclusion is only applicable to those eight classes. However, SSRAP believes that similar conclusions might be applicable for some other classes of equipment, but such an extrapolation should only be made after a very detailed and careful review.

The data base is inadequate to preclude the possibility of an inadvertent change of function (breaker trip, etc.) due to causes such as relay chatter. This does not mean that SSRAP expects these problems to occur. It simply means that their preclusion has not been demonstrated by the available data base. The data base does demonstrate the breakers can be properly reset and the equipment functions properly after the earthquake.

SSRAP is particularly concerned with equipment anchorage and feels that any attempt to justify equipment for acceptable seismic performance must ensure adequate engineered anchorage. There are many examples of equipment sliding or overturning in earthquake exposure because of no anchorage or inadequate anchorage. Inadequate anchorage can include short, loose, or poorly installed bolts or expansion anchors, and improper welding or bending of sheet metal frames at anchors. SSRAP believes that equipment anchorage must not only be strong enough to resist the anticipated forces but also stiff enough to prevent excessive movement of the equipment and potential resonant response with the structure. It is SSRAP's opinion that any review program should include consideration of both strength and stiffness of the anchorage and its component parts.

Excluded from assessment in the SSRAP study are the utilities that might be connected to the classes of equipment under consideration. Examples include air, power, fuel, and cooling systems.

2.1.5.1 Seismic Motion Bounds

SSRAP uses three different seismic motion bounds (Type A, B, and C) in its report. These bounds are defined in terms of the 5% damped horizontal ground response spectra shown in Figure 2.1-14. The seismic motion bounds may be used for the equipment class as defined in the table that follows.

These spectra bounds are intended for comparison with the 5% damped design horizontal <u>ground</u> response spectrum at a given nuclear power plant. Alternately, one may compare 1.5 times these spectra with a given 5% damped horizontal floor spectra in the nuclear plant.

The comparison of these seismic bounds with design horizontal ground response spectra is judged by SSRAP to be acceptable for equipment mounted less than 40 feet above grade (the top of the ground surrounding the building) and for moderately stiff structures (fundamental frequency greater than 2 Hz). For equipment mounted more than 40 feet above grade, comparisons of 1.5 times these spectra with horizontal floor spectra is necessary. In all cases a comparison with floor spectra is acceptable.

The criteria are met so long as the 5% damped design horizontal spectrum lies below the appropriate bounding spectrum at frequencies greater than or equal to the fundamental frequency range of the equipment.

2.1.5.2 Motor Control Centers

On the basis of a review of the data base and anticipated variations in conditions, SSRAP is of the opinion that motor control centers are sufficiently rugged to survive a seismic event and remain operational thereafter, provided the following conditions exist in the nuclear facility:

Equipment Class	Seismic Motion Bound	Seismic Motion Bound Derived From*
Motor control centers Low-voltage (480-V) switchgear Metal-clad (2.4 to 4-kV) switchgear unit substation transformers	Type B	Sylmar Converter Station (San Fernando earthquake)
Motor-operated valves with large eccentric operator lengths to pipe diameter ratios (see Figure 2.1-14)	Type C	Valley Steam Plant and Burbank Power Plant (both for San Fernando earthquake)
Motor-operated valves (exclusive of those with large eccentric operator lengths to pipe diameter ratios) Air-operated valves Horizontal pumps Vertical pumps	Type A	El Centro Steam Plant (Imperial Valley earthquake) Pleasant Valley Pumping Plant (Coalinga earthquakes)

*Based on smoothed averaged horizontal ground 5% damped response spectra from actual ground motion records divided by 1.5.

(1) Motor control centers of the 600-V class (actual voltage is 480 V) are considered. The style of cabinets must be similar to those specified in NEMA Standards. This requirement is imposed to preclude unusual designs not covered in the data base. SSRAP feels that cabinets which are styled after NEMA Standards will perform well if they are properly anchored. Cabinet dimensions and material gauges need not match NEMA Standards.

(2) The cabinets have engineered anchorage. Both the strength and stiffness of the anchorage and its component parts must be considered. Stiffness can be evaluated by engineering judgment on the basis of the cabinet construction and the location and type of anchorage, giving special attention to the potential flexibility between the tiedown anchorage and the rigid walls of the cabinet. Adequate stiffness can also be shown by determining that the fundamental frequencies of the anchored cabinet under significant shaking in both horizontal directions is above approximately 8 Hz. It is the opinion of SSRAP that properly anchored cabinets will have a fundamental frequency greater than about 8 Hz.

(3) The intent of this requirement is to ensure that under earthquake excitations the natural frequency of the installed cabinet will not be in resonance with both the frequency content of the earthquake and the fundamental frequency of the structure.

(4) Cutouts in the cabinet sheathing are less than 6 inches wide and 12 inches high.

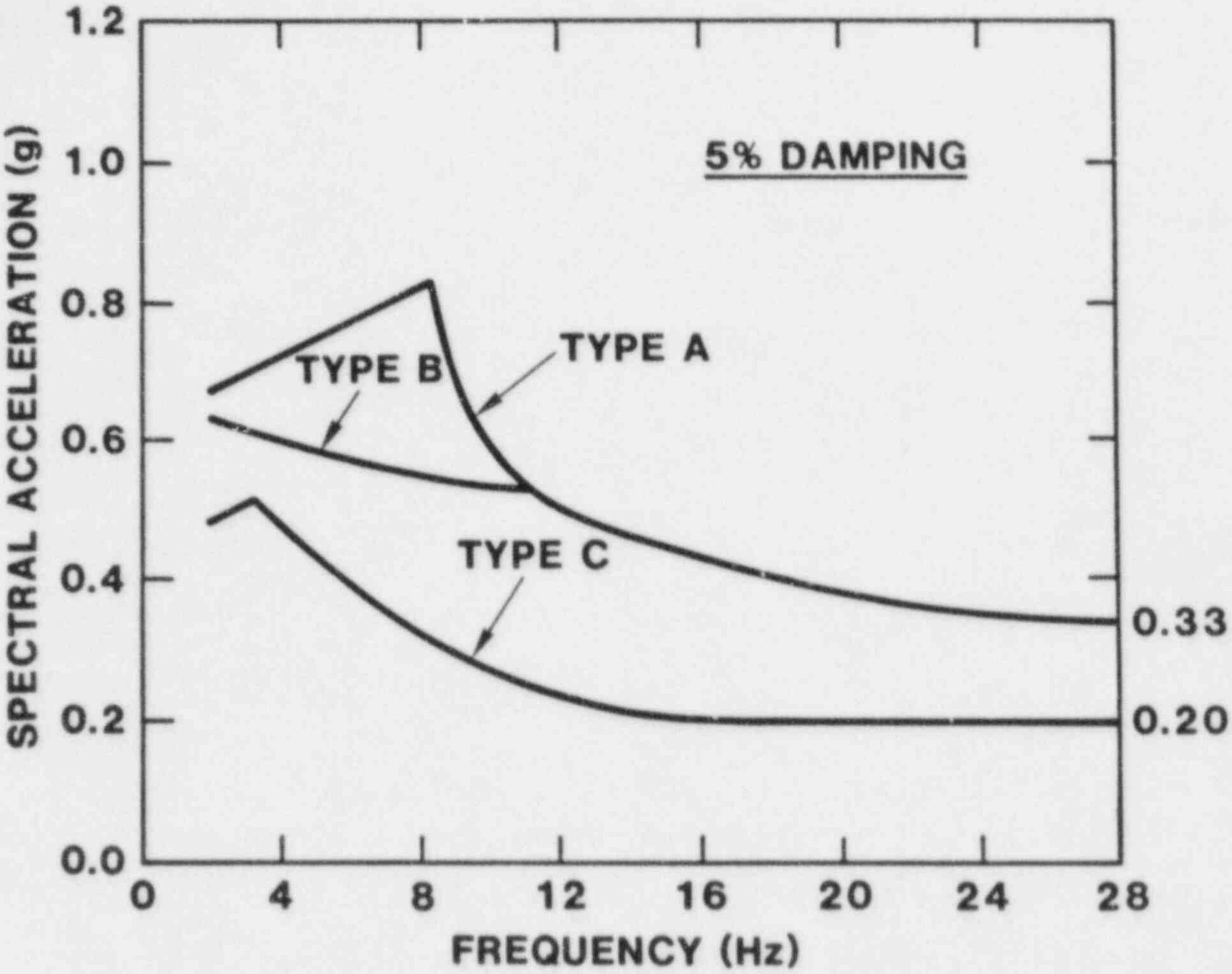


Figure 2.1-14 Seismic motion bounding spectra

(5) All internal subassemblies are securely attached to the motor control cabinets which contain them.

(6) Adjacent sections of multi-bay cabinet assemblies are bolted together.

(7) Equipment and their enclosures mounted externally to motor control center cabinets and supported by them have a total weight of less than 100 pounds.

SSRAP does not consider the functionality, that is, inadvertent changing or failure to change state on command of relays during an earthquake. The functionality must be established by other means. The structural integrity of relays contained in the motor control centers and their ability to function properly after earthquakes, as defined in Section 2, has been demonstrated.

2.1.5.3 Low-Voltage Switchgear

Low-voltage switchgear of the 600-V class (actual voltage is 480 V) is considered. The style of cabinets must be similar to those specified in ANSI C37. This requirement is imposed to preclude unusual designs not covered in the data base. SSRAP feels that cabinets which are styled after ANSI Standards will perform well if they are properly anchored. Cabinet dimensions and material gauge need not match the ANSI Standard.

All the conclusions, limitations, and bounding spectra for motor control centers are applicable to low-voltage switchgear.

2.1.5.4 Metal-Clad Switchgear

Metal-clad switchgear of 2.4 kV and 4.16 kV is considered. The style of cabinets must be similar to those specified in ANSI C37 Standards. This requirement is imposed to preclude unusual designs not covered in the data base. SSRAP feels that cabinets which are styled after ANSI Standards will perform well if they are properly anchored. Cabinet dimensions and material gauges need not match NEMA Standards.

All the conclusions, limitations, and bounding spectra for motor control centers are applicable to metal-clad switchgear, except that the cutouts in the cabinet sheathing shall be less than 12 inches by 12 inches.

2.1.5.5 Unit Substation Transformers

Unit substation transformers convert the distribution voltage to low voltage.

Unit substation transformers which convert 2.4-kV or 4.16-kV distribution voltages to 480 V are considered.

On the basis of a review of the data base and anticipated variation in conditions, SSRAP is of the opinion that unit substation transformers are sufficiently rugged to survive a seismic event and remain operational thereafter, provided that in the nuclear facility both unit substation transformer enclosures and the transformer itself have engineered anchorage.

The functionality of properly anchored unit substation transformers during and after earthquakes, as defined above, has been demonstrated.

2.1.5.6 Motor-Operated Valves

On the basis of a review of the data base and anticipated variations in conditions, SSRAP is of the opinion that motor-operated valves are sufficiently rugged to survive a seismic event and remain operational thereafter, provided the following conditions exist in the nuclear facility:

(1) The valve housing and yolk construction is not of cast iron.

(2) The valve is mounted on at least a 2-inch pipe.

(3) The actuator is supported by the pipe and not independently braced to or supported by the structure unless the pipe is also braced, immediately adjacent to the valve, to a common structure.

The limitations on operator weight and eccentric length relative to pipe diameter are derived from the data base for motor-operated valves that was provided by SQUG. The data base contains relatively few heavy operators and small pipe diameters subjected to severe ground shaking. These limitations could be less restrictive if more motor-operated valves had been located and documented in the areas of higher shaking. It is felt that additional data, either from other earthquake experience or seismic qualification tests, can expand the scope of these recommendations. These limitations are shown in Figures 2.1-15 and 2.1-16.

For motor-operated valves not complying with the above limitations, the seismic ruggedness for ground motion not exceeding the Type A bounding spectrum may be demonstrated by static tests. In these tests, a static force equal to three times the approximate operator weight shall be applied non-concurrently in each of the three orthogonal principal axes of the yoke. The limitations other than those related to the operator weight and distance from the top of the operator to the centerline of the pipe, given above, shall remain in effect.

2.1.5.7 Air-Operated Valves

On the basis of a review of the data base and anticipated variations in conditions, SSRAP is of the opinion that air-operated valves are sufficiently rugged to survive a seismic event and remain operational thereafter, provided the following conditions exist in the nuclear facility:

(1) The valve housing is not of cast iron.

(2) The valve is mounted on a pipe of 1-inch diameter or greater.

(3) Limitations on pipe diameter versus distance from centerline of pipe to top of operation are shown in Figure 2.1-17.

(4) The actuator is supported by the pipe and not independently braced to the structure or supported by the structure unless the pipe is also braced immediately adjacent to the valve to a common structure.

The air line and its connection are not included in this assessment.

For air-operated valves not complying with the above limitations, the seismic ruggedness for ground motion not exceeding the Type A bounding spectrum may be demonstrated by static tests. In these tests, a static force equal to three times the approximate operator weight shall be applied non-concurrently in each of the three orthogonal principal axes of the yoke. The limitations other than those related to the distance of the top of the operator to the centerline of the pipe, given above, shall remain in effect.

2.1.5.8 Horizontal and Vertical Pumps

SSRAP feels that horizontal pumps in their entirety, and vertical pumps above their flange are relatively stiff and very rugged devices because of their inherent design and operating requirements. However, the applicability of the data base is subject to the limitations set forth below.

For horizontal pumps, one must ensure that the drive (electric motor, turbine, etc.) and pump are rigidly connected through their base so as to prevent damaging relative motion. Of concern are intermediate flexible bases; these must be evaluated separately. Proper horizontal thrust load capacity must also be

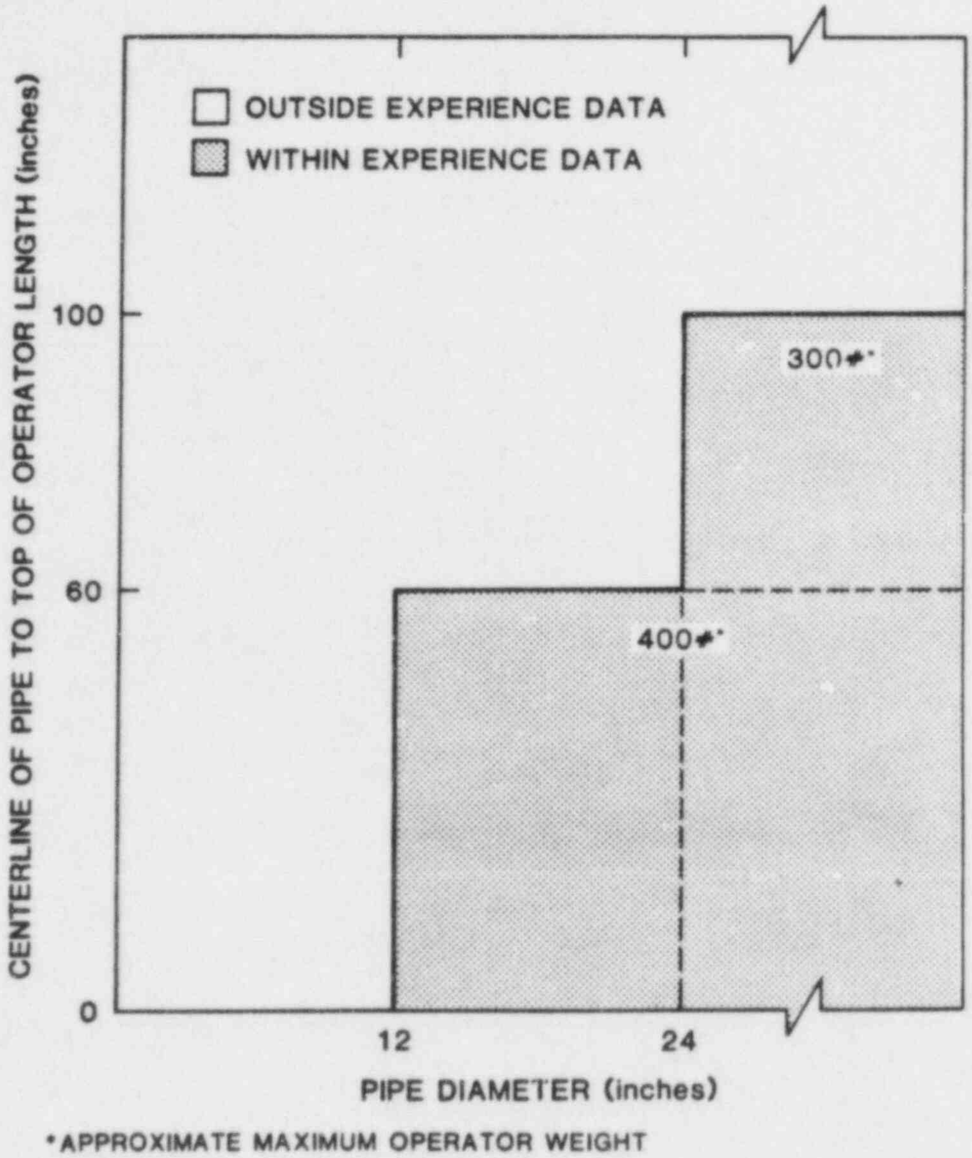


Figure 2.1-15 Motor-operated valves for which Type A spectrum is to be used

ensured in both axial directions. The data base covers pumps up to 2500 hp. However, SSRAP feels that the conclusions are equally valid for horizontal pumps of greater horsepower.

For vertical pumps, the data base has many entries up to 700 hp and several up to 6000 hp; however, SSRAP feels that safety-related vertical pumps, above the flange, of any size are sufficiently rugged to meet the Type A bounding spectrum.

SSRAP feels that the variety of vertical pump configurations and shaft lengths, below the flange, and the relatively small number of data base points in several categories, preclude the use of the data base to screen all vertical pumps. Vertical turbine pumps with cantilevered casings up to 20 feet in length and

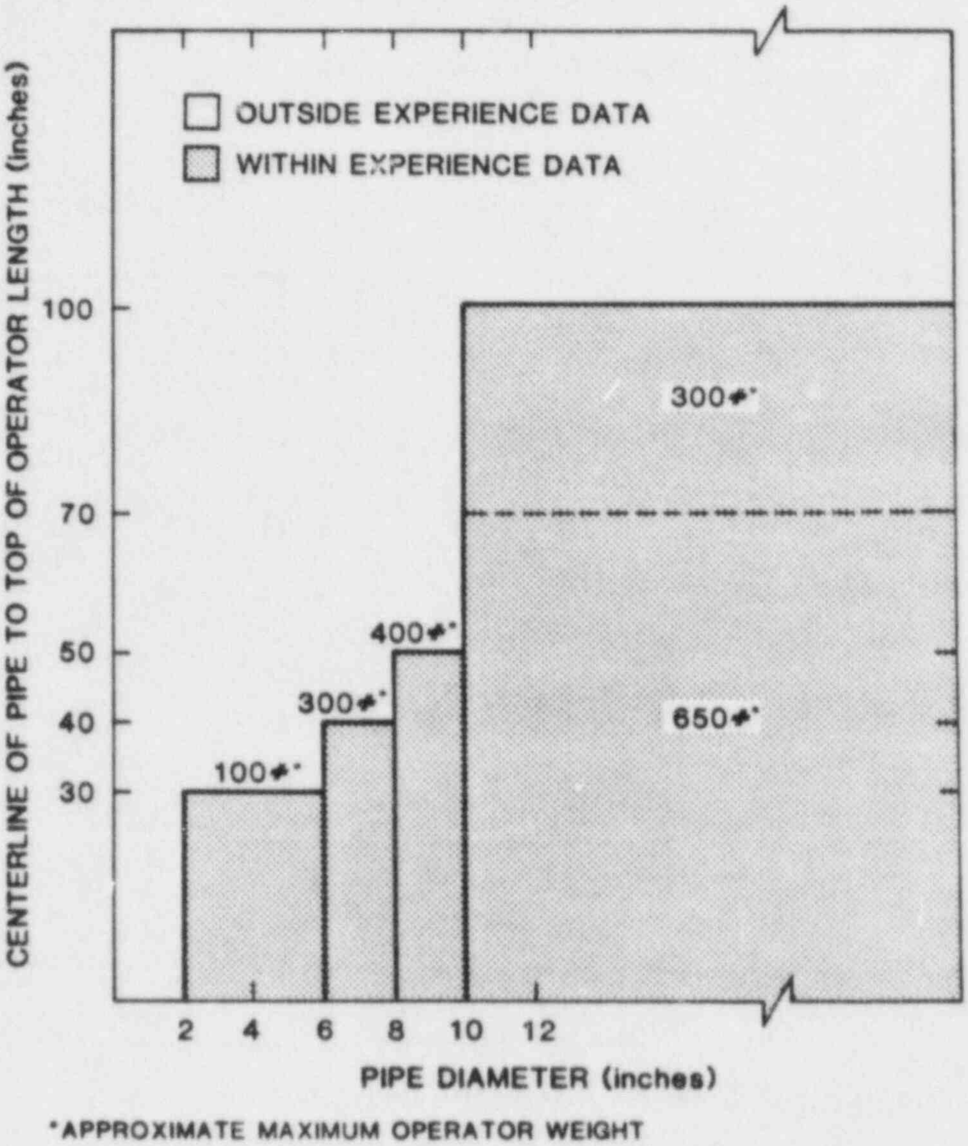


Figure 2.1-16 Motor-operated valves for which Type C spectrum is to be used

with bottom bearing support of the turbine to the casing appear well enough represented to meet the bounding criteria below the flange as well. SSRAP recommends either individual analysis or use of another method as a means of evaluating of other vertical pumps below the flange. The chief concerns would be damage to bearings from excessive loads, damage to the impeller from excessive displacement, and damage from inter-floor displacement on multi-floor supported pumps.

2.1.5.9 Conclusion and NRC Staff Comments

General conclusions arrived at by SSRAP after its study of the data base for the eight classes of equipment are summarized in Section 2.1.5.

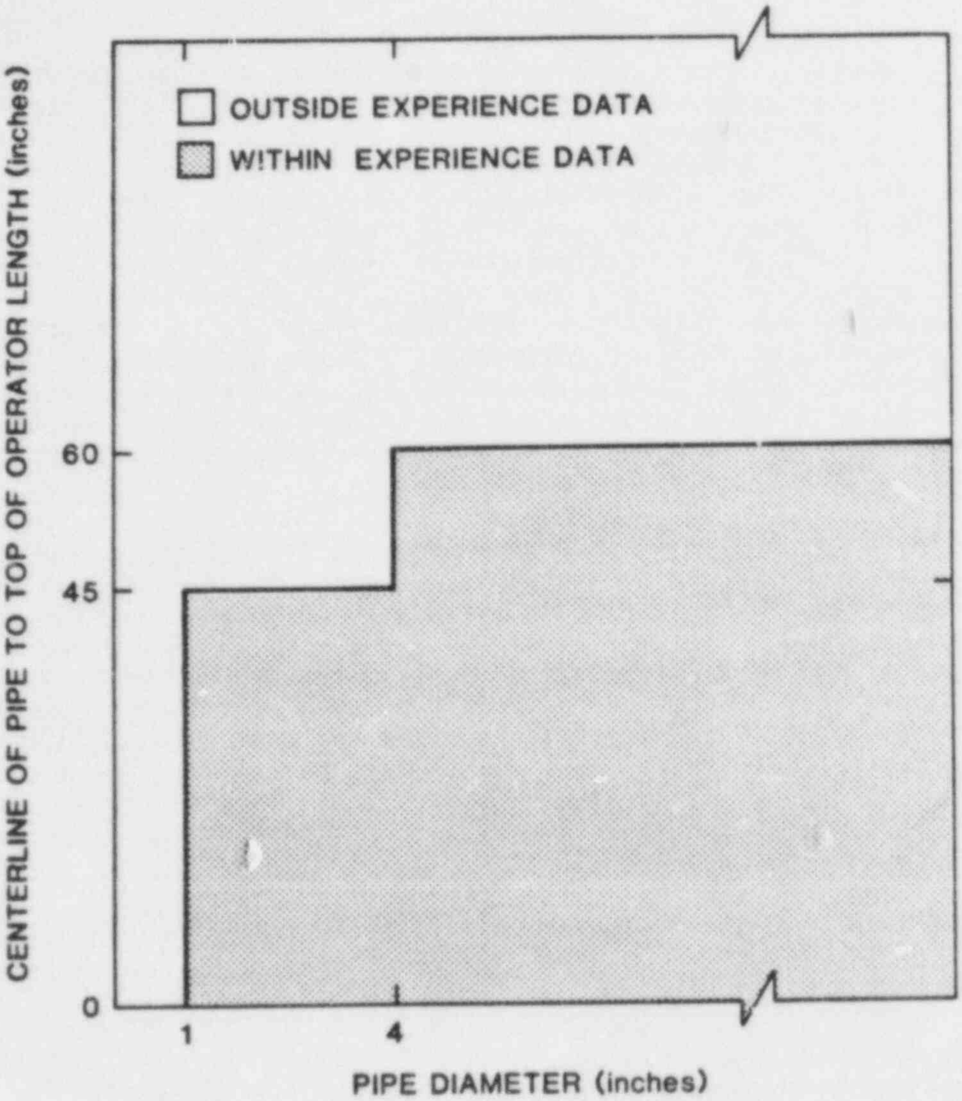


Figure 2.1-17 Air-operated valves for which Type A spectrum is to be used

SSRAP envisions that a seismic review of these items of equipment in an existing nuclear power plant will require a walk-through of the plant (1) to determine which equipment is within the limitations of these recommendations and, (2) to evaluate judgmentally other factors that may affect the seismic performance of the equipment, such as the evaluation of adjacent equipment and conditions to verify that impacts during a seismic event which might damage the safety-related equipment are precluded. It is expected that this evaluation will flag for special review any unusual or non-typical conditions such as major modifications to standard equipment or equipment that is unique.

The SSRAP recommendations are based on experience data which confirm that the equipment included within the limitations is rugged enough to maintain functionality after the strong shaking has ended. However, it has been brought to the attention of SSRAP that there apparently have been cases where maintenance personnel have noted increased wear in bearings of vertical pump shafts several weeks after the earthquake exposure. Because wear of bearings is a normal condition and because these pumps did operate for weeks after the earthquake before maintenance was required, SSRAP feels that this potential situation is within routine maintenance and not a matter of concern. It is mentioned only as an additional consideration for post-earthquake maintenance checks.

Much of the data base equipment was over 20 years old at the time of the earthquake exposure and some of this equipment is located in reasonably high thermal and corrosive environments, so the data base undoubtedly does address these aspects of equipment aging. However, none of the data base equipment was exposed to radiation, so the aging effects from radiation exposure upon the equipment are beyond the scope of this program.

As part of the data development for this program, literature reviews of several significant earthquakes were conducted to determine if failures had occurred which might contradict the lack of failures within the data base. References in Appendices B and C reported several isolated failures of equipment in the 1964 Alaskan earthquake and the 1972 Managua, Nicaragua earthquake. The original reports contain incomplete data, poorly documented, and most failures can conceivably be explained by conditions such as improper anchorage. Nevertheless, because of the overwhelming evidence in the data base, SSRAP has not altered its conclusions on the basis of these reports, but it is suggested that attempts be made to determine if more detailed information does exist to properly evaluate these reports.

The conclusions of the SSRAP study have been based largely on the data base that was provided. As previously noted, some items or portions of equipment have been excluded from the scope of the recommendations because of lack of information within the data base. For example, some motor-operated valves as well as the functionality of relays during the earthquake have been excluded. SSRAP believes that the limitations imposed on some of the eight classes of equipment can be relaxed with the use of seismic equipment qualification tests which undoubtedly have been performed or could be performed on an industrywide basis.

SSRAP believes that the approach to equipment evaluation for seismic performance utilized in this study can be extended to other classes of equipment. It is recommended that future studies utilize both earthquake experience data as well as seismic equipment qualification test data as appropriate. Each class of equipment must be carefully addressed on an individual basis to consider potential vulnerabilities and appropriate limitations. Although no detailed studies have been performed, SSRAP suggests that the following are examples of classes of equipment that may be amenable to this approach: air compressors, conduit and cable tray raceways, diesel generators, electric motors, fans, heat exchangers, HVAC systems, and piping.

The staff is in general agreement with the SSRAP conclusions, recommended caveats, and exclusions (as outlined in Sections 2.1.5, and 2.1.5.1 to 2.1.5.8).

2.2 Development and Assessment of In-Situ Testing Methods To Assist in Qualification of Equipment

2.2.1 Background

This task was selected for A-46, because the potential exists that in situ testing can be a promising tool in assisting the seismic qualification of equipment in operating plants. The task is conducted by Idaho National Engineering Laboratory (INEL), and was started in early 1982. The intent of this task is to investigate present in-situ testing methods and to evaluate the feasibility of using these methods to assist in requalifying equipment, and to develop methods, guidelines, and acceptance criteria for their use.

More specifically, the work scope for this task consisted of the following topics:

(1) Basic review of existing approaches to in-situ testing and identification of preliminary in-situ test methods for the qualification of equipment in plants which are currently licensed and operating.

(2) Review of approaches to laboratory testing and simulation of seismic events in the laboratory for qualification of equipment. Limitations on the use of current guidance was also studied.

(3) Review of the analysis procedures fundamental to in-situ testing methods. Review of use of subcomponent proof test and/or subcomponent fragility tests in the qualification process. Review of the qualification requirements for anchors.

(4) Investigation of techniques for assessing/monitoring the effects of chemical or metallurgic aging, mechanical fatigue, and wear during plant operation.

(5) Address adequacy, limitations and inherent shortcomings, and nonconservatisms of the various approaches above.

(6) Development of guidelines and acceptance criteria for use of in-situ testing to support alternative methods of seismic qualification of safety-related equipment.

(7) Definition of requirements for a test data base in support of seismic qualification of existing equipment in currently licensed operating plants.

(8) Development of cost estimate for alternate seismic qualification methods.

(9) Verification and further development of combined in-situ and analysis methods suitable for equipment qualification. Examination of limitations and pitfalls of applying in-situ testing methods in determining dynamic characteristics and evaluating component mountings of structures which support, contain, or position safety-related equipment in operating plants. Development of guidelines for minimum testing requirements and reporting requirements in qualification documentation.

2.2.2 Summary of INEL Report, "The Use of In-Situ Procedures for Seismic Qualification of Equipment in Currently Operating Plants"

Results of work on topics 1, 2, 3, 4, 5, 7, and 8 of Section 2.2.1 above are covered in the contractor report titled "The Use of In-Situ Procedures for Seismic Qualification of Equipment in Currently Operating Plants" (NUREG/CR-3575) (NRC, June 1984). This report is divided into four parts, each of which addresses a specific area. Following is a summary of these four parts.

2.2.2.1 Summary of Part A and Part B, "Preliminary Study of the Use of In-Situ Procedures for Seismic Equipment Qualification in Currently Operating Plants" and "Improved In-Situ Procedures and Analysis Methods"

The goal of this study was to examine the most important uses of in-situ testing employed to assist in requalification of safety-related equipment.

Theoretically, in-situ test procedures could be applied in the following three methods:

(1) Testing at full load level with equipment in place.

(2) Low load level testing with equipment in place.

(3) Periodic intermediate or low load level testing to support a continuing surveillance data base.

It is the conclusion of this study that among the three potential methods of in-situ test, only method 2 is normally practical and feasible. Method 1, which applies the dynamic load up to the safe shutdown earthquake (SSE) level, has to satisfy certain conditions. The required conditions are that:

(1) The motion applied to the equipment-supporting structure should not excessively load the appurtenances, the components mounted thereon or in the vicinity, and the equipment-supporting structure itself.

(2) Sufficient access must exist in order to load the equipment mounting.

(3) No damage occurs to the local area where load is applied.

(4) No significant mechanical aging degradation has occurred during testing, so that component can be employed in service for its nominally useful lifetime.

These conditions severely limit the usefulness of full load level in-situ tests. Valve operators are one equipment type that have been dynamically qualified in-situ by using a static load to perform an interference evaluation. However, the potential for performing full load level in-situ testing is so limited that it is not considered further.

Method 3 above could, in principle, be useful for identifying aging degradation. However, the contractor concluded that for the types of equipment of interest in this program, no potential applications are apparent. This is because changes significant to operability of safety-related equipment (particularly in a seismic environment) can not generally be detected by in-situ procedures.

The low load level in-situ tests are normally performed by applying hammer impact on equipment or supporting structures. Portable electromagnetic or hydraulic shakers can also be applied to equipment or equipment-supporting structures in place, in order to dynamically test them. The input force and output, normally acceleration, are recorded as loads are applied at various positions. The recorded quantities are converted from time histories to a frequency representation by use of the Fourier transform. Using the frequency representation, transfer functions are calculated between points of input and output. These calculations are typically performed with minicomputers which are part of the modal analyzer system. Software internal to these computers then identifies natural frequencies and mode shapes. The mode shapes encompass points on the structure where data were recorded.

The contractor concludes from his study that in-situ testing will be useful in the following areas related to equipment qualification:

(1) establishment of similarity between equipment with consideration of failure modes

(2) prediction of component-specific required response spectra (RRS)

(3) component mounting evaluation

(4) comparison of fundamental building frequency with equipment-supporting structure frequency

It was also concluded that in-situ testing will not be feasible and suitable for the following applications:

(1) to establish component/equipment seismic capacity
(2) to support a continuing surveillance data base

The applications of in-situ testing methods is further discussed below. Other related topics covered by this contractor's report are described in Appendix B.

(1) Establishment of Similarity Between Equipment With Consideration of Failure Modes

The most obvious application of in-situ testing to seismic qualification of equipment in operating plants is to establish dynamic similarity between pieces of equipment.

As mentioned in Section 1.2, after reviewing the results of all the tasks of A-46, the NRC staff concluded that seismic qualification using seismic experience data probably is the most likely approach to develop a qualification method which is both economically attractive to the plant owners and would be acceptable from a public safety viewpoint. Two conditions will have to be established before the experience data base can be utilized to help assess seismic adequacy of equipment in operating plants. They are:

(a) To establish that RRS of equipment in operating plant to be requalified is enveloped by the pertinent experience data base response spectra.

(b) To establish similarity between operating plant equipment to be requalified and equipment in the experience data base.

Condition (a) is addressed by No. 2 (immediately following) and also by Section 2.5. The staff's position on the definition of similarity was described as "for equipment to be similar for the purpose of qualifying an equipment item on the basis of experience data on another item, the safety function as well as the dynamic characteristics, should be similar. This means that the experience data must include data on performance both during and after a seismic event. Similarity parameters must include mass distribution, material, size, stiffness, configuration, restraints, and anchorage details...."

Similarity of dynamic characteristics can most effectively be addressed by conducting an in-situ test. Dynamic characteristics of equipment consist of mode shapes, natural frequencies, mass distribution, and damping. In-situ procedures identify the natural frequencies and mode shapes. In certain cases the mass distribution can also be estimated (alternate methods for determining the mass distribution are proposed by the contractor in his reports). It is also possible to characterize viscous damping by using in-situ tests that represent the damping that actually occurred during the test. Since damping may depend on response level, the contractor proposed that values obtained from low level in-situ tests may not necessarily be valid and Regulatory Guide 1.61 (NRC) is recommended for damping values.

The safety function aspect (operability and failure modes) of similarity is further discussed in paragraph 1 of Appendix B.

(2) Prediction of Component-Specific RRS

In order to seismically qualify a piece of equipment, it is first necessary to establish the specific RRS. For equipment mounted on a floor, the response can be predicted by the floor response spectra. However, because safety-related components are mounted on or attached to the equipment-supporting structures (such as electrical cabinets, racks, etc.), the RRS for these components will be different from the floor response spectra. In situations like these, three methods are studied and proposed by the contractor to establish component-specific RRS. Each method will utilize in-situ testing to a different extent.

(a) The first approach is to develop a finite element computer model of the equipment-supporting structure and the mounted equipment. The analysis procedures involved here are those of the typical time history method. In this process, (i) a synthetic time history is developed from a specific floor response, (ii) the modes, frequencies, and modal participation factors are calculated from the model,

(iii) a time history analysis is performed on each significant mode, (iv) the modes are algebraicly combined to determine total time histories, and (v) the time histories are converted to RRS for the components of interest. The contractor feels that this basic procedure is potentially unreliable because the system is complex and boundary condition modeling is unreliable. Consequently, it can only be used if the equipment is already installed and in-situ procedures are used to verify the calculated modal parameters. A major disadvantage of the approach is that it is relatively expensive because of the cost associated with developing a finite element model. An advantage is that if minor equipment modifications are made at a later date, the model can be updated and a new set of RRS can be calculated.

(b) The second method to generate component-specific RRS is an analysis method by utilizing modal parameters directly. The process involves using the frequencies and mode shapes determined from in-situ procedures directly in constructing a numerical solution. In this approach, the modal participation factors can either be estimated by using the definition for the modal participation factor and approximating it with discrete mode shape and modal mass, or using an approach which is based on reconstructing the force vector using the significant modes of the structure. The second method is judged by the contractor to provide the best possible estimate of the modal participation factor and is recommended by the contractor. When using this method to generate the RRS, there is no need to develop a finite element model. As with the finite element approach, the response of individual modes is calculated and then superimposed for the total response. The contractor offered several comments about using this method. First, as the natural frequency increases it becomes more difficult for in-situ procedures to resolve the associated mode shapes. For seismic analysis it is felt that higher modes, or modes with several antinodes will result in low or negligible modal participation factors. Consequently, it will probably only be necessary to accurately calculate the lower mode shapes. The situation must be checked for every individual case. The second comment concerns closely spaced modes. The decomposition of the total frequency response into a modal frequency response function is one step in the development of the mode shapes. Closely spaced mode shapes reduce the accuracy with which the modal frequency response functions are calculated from the experimental transfer functions. The existence of closely spaced significant modes could render the direct use of modal parameters infeasible. It is anticipated that this situation will occur infrequently in which case the alternative of method "a" above can be used to determine RRS. The advantage of the direct use of modal parameters is that the modal parameters are relatively inexpensive to generate experimentally. Generation of modal parameters by the finite element method will require substantially more expense.

(c) The third method involves response spectra transfer based on the application of random vibration theory. When applied to seismic environments, this normally implies that the mean square response is used as the basis for predicting peak response values. The application of random vibration theory to a particular process is simplified if the process is Gaussian, zero mean, and stationary, because power spectra density (PSD) function completely defines the process under these restrictions. The contractor suggested that earthquakes are Gaussian in character because of broad frequency content and the random phasing of the frequency components, and they are obviously zero mean. Furthermore, the contractor suggested that earthquakes may be considered as a finite duration segment in a stationary process and corrections can be applied to structural response for the non-stationary effect of duration. Under these conditions, the statistical properties of the output can, in theory, be inferred from the input using the properties (natural frequencies, mode shapes, modal participation factors, and modal dampings) of the intervening structures.

One difficulty in seismic analysis arises from the structural motion starting from zero initial conditions. Correction factors must be used to correct for the differences between steady state response and response from realistic initial conditions.

On the basis of the above discussion, the contractor proposed a procedure for response spectra transfer using random vibration theory. The recommended procedure is to develop a response-spectrum-consistent PSD using an appropriate correction for duration, calculating the output PSD including the effects of all cross-modal terms and multiple directions of excitation through the use of transfer functions, integrating this PSD to determine the mean square response, and finally determining the response spectrum value from the root mean square response and an appropriate peak value factor. Details of the procedure are described in the INEL report of October 1983, "Improved In-Situ Procedures and Analysis Methods for Seismic Equipment Qualification in Currently Operating Nuclear Power Plants."

3) Component Mounting Evaluations

Mounting inadequacy has been a major cause of retrofit and retest in qualification programs. The current qualification process essentially qualifies mountings during shake table testing. For operating plants several options are available. Analysis procedures using data from in-situ testing can predict the maximum acceleration of equipment. Thus, the loads that mountings must transmit can be predicted. It should be a straightforward process to assess existing designs. The main distraction is the large number of mountings that exist. Enveloping the maximum acceleration could be an approach to reducing this workload.

Examining mountings on a theoretical basis may not address some (perhaps the major) problems. The contractor points out that quality of installation or use of problem-prone designs may be a stronger influence on mounting adequacy than strength considerations. To address these concerns, the contractor suggests a physical mounting review by practitioners experienced in both seismic qualification testing and current mounting design practice would be an effective mounting evaluation measure. This process would be enhanced if the reviewers were supplied with an equipment table identifying an enveloping acceleration, equipment weight, and a simple description of the mounting. The plant walkthrough would then screen mountings for those requiring in-depth review or retrofit. The effectiveness of this process is that it screens out items which are clearly adequate and concentrates more costly review on questionable items.

(4) Comparison of Fundamental Building Frequencies With Equipment-Supporting Structure Frequencies

The level of equipment-supporting structure response during a seismic event can be related to the corresponding floor response spectra. The design floor response will generally contain a region with significantly amplified magnitude. The center of this amplified region will generally lie between 2 and 10 Hz and coincides with the fundamental frequency of the building. The motion of the equipment-supporting structure is reckoned as a combination of its free vibration modes whose maximum values are determined from the floor-response spectra. Generally the first mode has the largest modal participation factor and is the most important. Knowing the first mode frequency and its modal participation factor, the maximum response is estimated readily from the floor-response spectra.

Tuning of the equipment-supporting structure and the building containing it occurs when a natural modal frequency of this equipment-supporting structure coincides with the fundamental building modal frequency. As an example, cabinet frequencies between 5-15 Hz are typical so that tuning is possible. In case tuning occurs, the floor-response spectra may result in a response level 2-5 times the predicted non-tuned response. A complicating factor is that the lowest natural frequency of an equipment-supporting structure depends on how it is attached to the floor as well as its physical properties. For instance a welded mounting will result in a higher frequency than a mounting with a minimum number of bolts. Thus, for operating plants, uncertainties relating to equipment-supporting structures include both physical properties and the mounting boundary condition.

Hence, the design environment of equipment will depend heavily on the relationship between the equipment-supporting structure and a building's fundamental frequencies. It is clear that most of the safety-related systems were not intentionally designed to function in highly amplified dynamic environments (i.e., tuned conditions). The contractor suggests that systems which may be subject to these loads should be identified by in-situ procedures. Here an abbreviated process can be followed in which all the equipment-supporting structure's natural frequencies below 15 Hz are experimentally determined. Mode shape need not be determined. A modal analysis crew should be able to check a number of cabinets in a single day, so cost is not an overwhelming burden. Where amplified equipment-supporting structure response is identified, two options are recommended. Regardless of the criteria applied to other equipment in operating plants, the contractor recommends that this equipment should be qualified vigorously. The first option is to determine the design-basis environment (or component-specific RRS) and qualify equipment to that environment. The second option is to modify the equipment-supporting structure, depending upon which is appropriate. That a lower response is assured should be verified by in-situ procedures.

2.2.2.2 Summary of Part C, "Guidance and Acceptance Criteria for Application of Combined In-Situ and Analysis Procedures"

This part covers Topics 6 and 9 defined in Section 2.2.1 of this report. Fourteen technical areas are identified.

Following is a summary of the guidance and acceptance criteria in the fourteen technical areas. Details can be found in the INEL report (NRC, June 1984).

(1) Dynamic Parameters From Tests. Guidance is required on the number and position of nodal points for describing the mode shape. Node points are to be located at all significant masses (>5% of total system mass), and there should be no less than four node points between modal antinodes for the significant mode with the largest natural frequency.

Assurance must be provided that all modes in the frequency range of interest have been determined. Additional guidance concerning natural frequencies is included in Items 8 and 14, that follow.

(2) Analytically Determined Dynamic Parameters. Guidance relating to analytically determined equipment-supporting structure models is that these models are to be verified by comparing computed and experimentally determined natural frequencies. The analytic and experimental frequencies must correlate to a reasonable tolerance - say 10%, for frequencies in the range of interest.

(3) Analysis Methods for Generating Device Location Required Response Spectra (RRS)

The time history analysis method is currently accepted (NRC, RG 1.92) and the same guidance should be applied to operating plants. Response-spectra transfer using random vibration methods is acceptable; the complete mean square response must be employed, the peak value factors must be justified, the modal participation factors employed must meet the criteria in Item 7, and all significant modes must be included in the structural model. Additional details are available in the INEL report of October 1983, "Improved In-Situ Procedures and Analysis Methods for Seismic Equipment Qualification in Currently Operating Nuclear Power Plants."

(4) Modal Participation Factor (MPF)

Proposed guidance is to determine the mass matrix ([M]) from physical characteristics of the system and calculate MPF according to the following equation

$$\{\Phi\}_i^T\ [M]\ \{1\} = MPF_i$$

An alternative method is to use the equation (where $\{\Gamma\}$ is the vector of MPFs where $[\Phi]^*$ represent the incomplete modes)

$$\{\Gamma\}^* = ([\Phi]^{*T}\ [\Phi]^*)^{-1}\ [\Phi]^{*T}\ \{I\}$$

and verify that the body force load is well simulated, i.e.,

$\{R\}/\{I\} \leq 0.05.$

where $\{R\}$ is an error vector,

Other methods for approximating the MPF must be justified and will be evaluated on a case-by-case basis.

(5) Determination of Fundamental Frequency of Equipment-Supporting Structure

The frequencies of equipment-supporting structures are acceptable if the transfer function in the frequency range of interest is determined from data maintaining a coherence of 0.8 or greater at the natural frequencies.

Another acceptable approach is to document that the magnitude and phase angle of the driving point frequency response functions (FRF) follow rules consistent with the absence of a natural frequency.

Other methods of establishing the low frequency range containing no natural frequencies will be evaluated on a case-by-case basis until experience warrants the development of general guidelines.

(6) Frequency Margin

As stated in Item 8 the exact value for the fundamental frequency can play a large role when modal parameters are combined with analysis procedures near a floor response spectrum peak, small errors in the in-situ frequency estimates can result in significant errors in the calculated RRS. There are potential sources of uncertainty in the frequency estimate, and the introduction of margin may be required to ensure conservative results.

The approach incorporates an uncertainty of ±10% in dynamic parameters determined using in-situ procedures. In this guidance it is assumed a time history or PSD consistent with an unbroadened floor response spectrum is employed.

In Figure 2.2-1 several frequency regions are defined on a line graph. If ω_s is the best estimate of a building's fundamental frequency and ω_c is the best estimate of a support structure's frequency, then Region 1 is $0.85\ \omega_s \leq \omega \leq 1.15\ \omega_s$, Region 2 is $0.9\ \omega_c \leq \omega \leq 1.1\ \omega_c$, and ΔD is the distance, measured in frequency (Hz) between the two regions as shown in Figure 2.2-1. If $\Delta D > 0.1\ \omega_c$ then the two regions are considered to be well spaced, otherwise they are considered to be coupled. One set of guidance applies if the regions are well spaced and a separate set applies to coupled regions. As noted earlier all guidance presented herein is based on unbroadened floor response.

For well spaced frequencies, either time history or mean square response (i.e., during PSD function) analysis procedures may be used. The input to the support structure is consistent with the unbroadened response spectra with peak at ω_s The structure for which an in-structure response

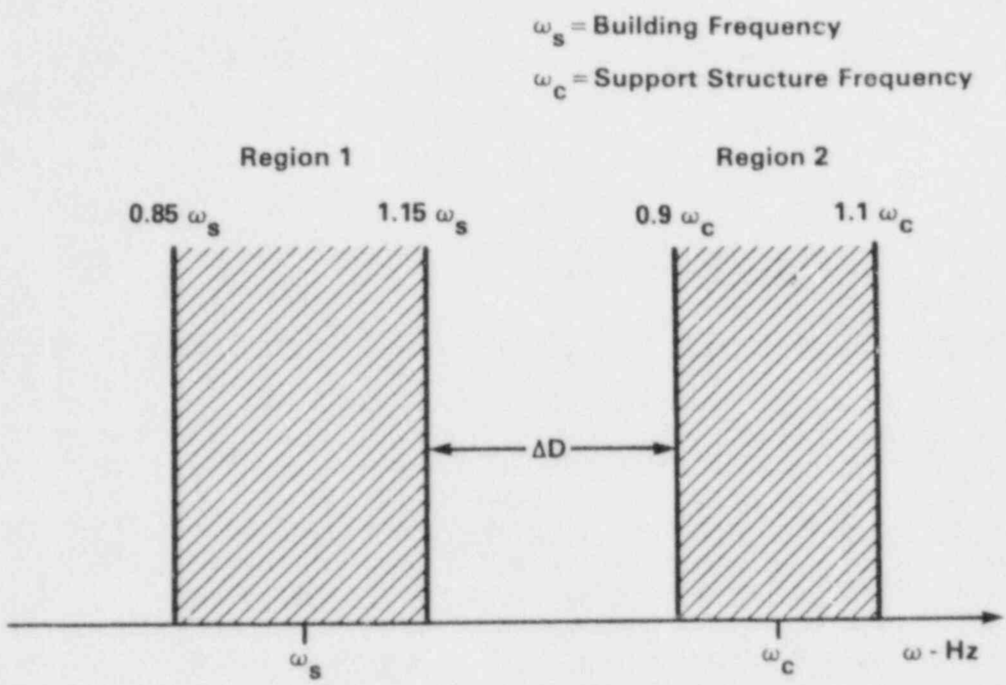


Figure 2.2-1 Line graph definition of Region 1, Region 2, and frequency separation ΔD

spectrum is sought is modeled with its best estimate modal properties. These estimates must be consistent with guidelines presented elsewhere in this document or in existing regulatory guidelines. The required in-structure responses are predicted using time history or root mean square procedures. Figure 2.2-2 shows the expected features of the in-structure response spectrum. The response spectrum peaks are horizontally extended across Region 1 and Region 2 to apply margin, and the remainder of the spectrum is formed in conformance with NRC Regulatory Guide 1.122.

For the situation in which Region 1 and Region 2 couple, the procedure is somewhat different. Time history methods are not practical because three separate spectra-consistent floor time histories are required to use the procedures to be described. Coupling or tuning of building and support structure is not expected to occur frequently. SQUG experience data investigations show support structures natural frequencies above 6 Hz to be the typical situation. This is signficant because incorporating margins for building modal parameters and support structure modal parameters is relatively more complicated for the condition where Region 1 and Region 2 couple.

The methodology for estimating secondary response spectra with the incorporation of margin on support structure frequency is now described. Two procedures are required. One for the case in which coupling occurs without overlapping. Three floor response spectra are defined. These response spectra have peaks at ω_s, 0.85 ω_s, and 1.15 ω_s, respectively. A spectrum-consistent PSD is calculated (NRC, June 1984) for each response spectrum. Several versions of the support structure's modal model are generated. The

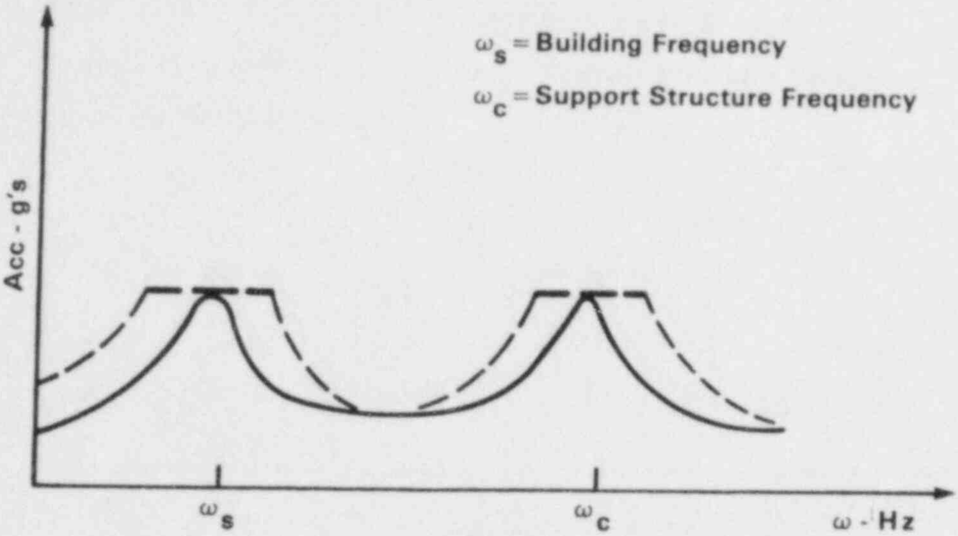


Figure 2.2-2 Best estimate in structure response spectra and broadened response spectra

mode shapes are not modified. One modal model has a set of natural frequencies in which the first mode frequency is 0.90 ω_c. A second model employs a first mode natural frequency of 1.1 ω_c. If Region 1 and Region 2 do not overlap, no other support structure models need be considered. The floor input PSD for 0.85 ω_s is combined with the support device structural model using 0.90 ω_c as its fundamental frequency and a response spectrum is generated using the root mean square approach. A second in-structure response spectrum using a PSD for 1.15 ω_s and fundamental support structure frequency of 1.1 ω_c is constructed. A third in-structure response spectrum using a PSD for 1.15 ω_s and fundamental support structure frequency of 0.9 ω_s is constructed. Finally, a combined response spectrum enveloping these two response spectra is formed and this response spectrum incorporates margin on both building properties and support structure properties. These three spectra are employed to generate the enveloping RS.

If Region 1 and Region 2 overlap, then a calculation in addition to the two described above is required. It is assumed the only practical situation is shown in Figure 2.2-3. An input PSD is generated for floor response spectra whose peak is at 1.15 ω_s. This input is applied to a structural model with fundamental frequency also at 1.15 ω_s. A second in-structure RS is calculated as follows. An input PSD is generated for floor response spectra whose peak is at 0/9 ω_c. This input is applied to a structural model with fundamental frequency also at 0.9 ω_c. As before, the RS are superimposed and an envelope is formed.

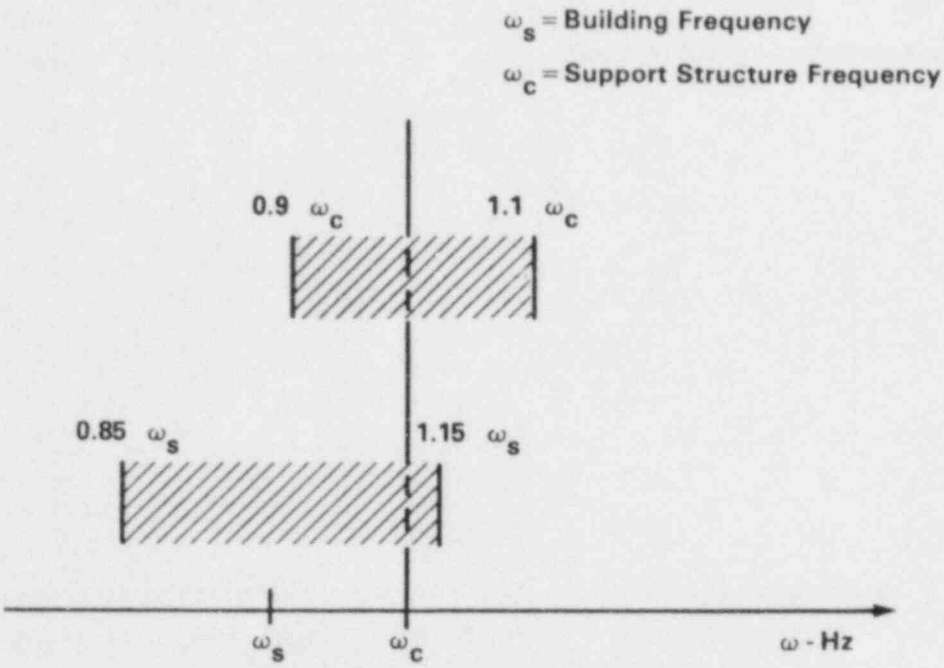


Figure 2.2-3 Coupled building and support structure natural frequencies

(7) Equipment-Supporting Structure Linearity

Support structure attached to the floor using bolt attachments must justify that installation preloads are not reduced by more than 70% during the SSE environment.

(8) Enveloping Criteria

As with current criteria, the experience response spectra (ERS) for rigid equipment must envelope the RRS at the ZPA. Envelopment at lower frequencies is not essential. For the structural integrity of equipment-supporting structure, envelopment is required only at frequencies greater than the fundamental frequency of support structures (with 15% margin on frequency). See Figure 2.2-4.

If justification can be provided that equipment is not specifically sensitive to low frequency inputs (i.e., so that the input does not have to be rich in low frequency content to perform a qualification test), envelopment can be restricted to the remaining frequency range.

(9) Component Mounting Structural Integrity

Loads on component mounting can be calculated using dynamic parameters developed from in-situ procedures. An acceptable maximum acceleration is calculated using the peak-broadened FRS, the modal parameters, and the

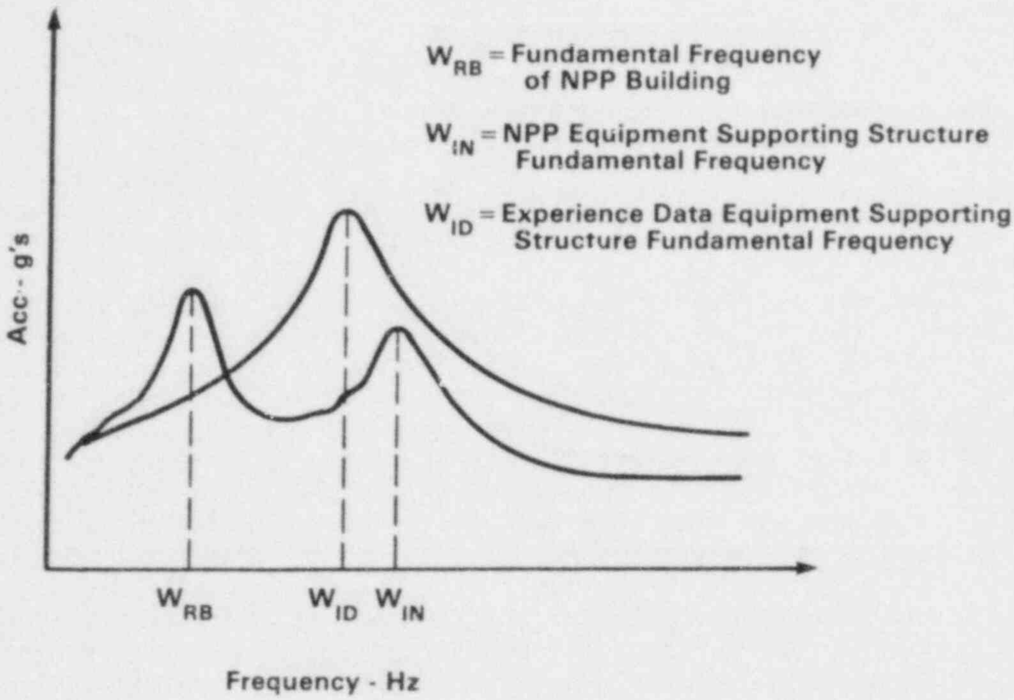


Figure 2.2-4 Comparison of envelopment

analysis methods discussed in NRC Regulatory Guide 1.92. The mass is taken as the sum of the components and mounting fixture masses.

(10) Calibrational Certification of Equipment, Instrumentation, and Computer Software

Guidance with respect to calibration of equipment and instruments is that the calibration procedures used must be recorded and included with the test documentation. These procedures should be referenced to an applicable testing standard if possible. The methods of calibration (system or component), the instrument calibrations and the calibrated range, and manufacturer's specifications for calibration should be included in test documentation. Manufacturers' specifications for instruments (including weight and rated operating range) and equipment should be included with test documentation. A driving point frequency response function measured during the initial stages of testing should be repeated at the completion of testing. These two measurements of the same frequency response function at the driving point must compare within acceptable limits to verify stability of measurements. The modal extraction software employed should have been certified by the solution of a standard problem. Software certification is discussed further in Item 14. A sketch of the system tested showing overall dimensions, location of seismic Category I equipment, instrumented positions, and detailing of anchorage must be included with documentation.

(11) Pretest Evaluations

The major item to be resolved during pretest evaluations is identifying the appropriate method, locations, and directions for exciting the structure. To ensure that all natural frequencies have been determined, excitation must be applied at a minimum of three positions for each principal horizontal direction. At these positions, frequency response functions at the driving point should provide the complete set of natural frequencies.

The excitor location to be used in generating the complete set of FRFs should maintain an acceptable value of coherence over the frequency range of interest (0.8 or greater). A coherence check at the natural frequencies between the input point and a remote accelerometer position is also required. In this case it is expected that the coherence will be lower in frequency ranges where the FRF indicates an antinode (a small modal coefficient for a given mode). Over the remainder of the frequency range of interest, the coherence must meet the same standard as the standard imposed at the driving point.

The reciprocity (output at 1 for an input at 2 versus output at 2 for an input at 1) between excitation location and a remote point should be verified. The comparison between FRFs should be sufficiently close to indicate that the same load paths are operating for both cases. Finally, the most representative frequency response function at the driving point should be evaluated at several levels of loading. The purpose is to demonstrate, in combination with the reciprocity check, that the natural frequencies and mode shapes will remain relatively invariant with excitation level.

(12) Data Collection

The qualification documentation should record the following information: (a) total number of data points in sample, (b) number of samples used to develop FRFs, (c) anti-aliasing filter employed, (d) windowing (if used) to prevent leakage in data, and (e) the sampling frequency.

(13) Calculation of Frequency Response Functions (FRFs) From Recorded Data

It is considered that no special guidance or acceptance criterion is necessary. A requirement to develop FRFs for a standard set of data could be imposed if the NRC staff felt that this level of certification was necessary. If the NRC staff felt certification of software was necessary, then a one-time requirement for development of accurate FRFs from a standard set of data could be imposed.

(14) Modal Extraction

The contractor should identify the developer of the software and the basis for choosing the modal extraction process used.

The major item in auditability of the modal extraction process is validation of the software used in modal extraction. The theory of steady state linear vibrations, Fourier transforms, linear algebra, etc., provides the common basis for modal extraction. However, numerous details are involved in developing computer software for application to modal extraction. Hence a direct check on software accuracy is desirable. In-situ test contractors should certify their software to one or more standard problems. This certification should be maintained by the utility for each such contractor retained for performance of in-situ investigations. Furthermore, it is recommended that the standard problem use data recorded during testing of an equipment-supporting structure typical to those found in nuclear power plants.

2.2.2.3 Summary of Part D, "Seismic Qualification Cost Estimating Task"

The objective of this task was to estimate costs associated with the steps of implementation of alternative seismic qualification methods as depicted in Figure 2.2-5. A table of estimated costs is given in this report (NRC, June 1984) and is shown here as Table 2.2-1. It should be cautioned, however, that initial comments on this cost table by an industry group indicate that equipment replacement costs are low by a factor of 3 to 5 and in some cases as high as 9.

Assumptions used to develop the cost estimates are described below.

Equipment List

The equipment list was obtained by modifying the list offered in the report "Survey of Methods for Seismic Qualification on Nuclear Plant Equipment and Components." The modifications resulted from a comparison of the list with two complete lists of safety-related equipment for two new plants--one PWR, one BWR.

Analysis

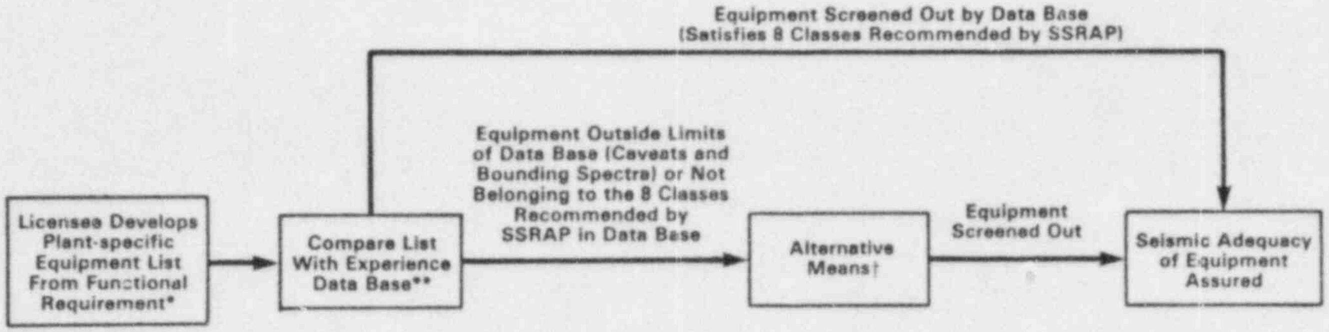
The "analysis" cost estimates were based on experience in estimating analysis jobs and on reviews of such analyses performed during staff audits of new plants for licensing reviews. Equipment which has no estimate for analysis is not suitable for qualification by analysis.

Test and Analysis

The numbers under "test and analysis" represent the cost to determine equipment/support dynamic characteristics via in-situ testing. These numbers were based on an attachment to the contractor's report (NRC, June 1984). Cost of labor, travel of personnel, and transportation of test equipment are included in the estimates.

Replacement

"Replacement" is the cost incurred to replace equipment with qualified equipment. This includes purchase of the equipment with qualification documentation and installation. It does not include freight charges. Estimates are primarily based on "Process Plant Construction Estimating Standards," by Richardson Engineering Services, Inc. (RES). Two editions of the standard were used, one dated 1975 and the other 1981. Estimates taken from the 1975 edition were increased by 30% to account for inflation. Two components on the list (MSIV & CRDM) were not covered by the standard. Estimates for these two were obtained from equipment vendors.



*From Section 1.3.2 (beyond the scope of this work).

**An estimate was made for the cost of comparing dynamic and functional characteristics of equipment in plant and that in the data base.

†a. Extend experience data which are comparable to SSRAP guidance and caveats.
 b. Find test data which are applicable to equipment.
 c. Develop other evidence of seismic ruggedness.
 d. Test prototype.
 e. Perform analysis and/or in-situ test to show seismic ruggedness or similarity with data base or test data (see NOTE 1, below).
 f. Simple modification to provide similarity with data base 2 (see NOTE 2, below).
 g. Replacement by qualified equipment (an estimate of replacement cost was made).
 h. Qualify to current requirement.

NOTE 1: An estimate was made of the cost of determining equipment/support dynamic characteristics via in-situ testing. Supports are typically either included in the qualification of equipment (e.g., diesel generator skid) or qualified as separate equipment (e.g., panels, racks, cabinets).

NOTE 2: A cost estimate of simple support modifications to obtain similarity with the data base was made. These numbers represent the cost of providing simple support modifications to obtain similarity with the data base equipment. They were calculated using the following formula:

$$\text{Cost} = (1.5\ L_i \times W) + 0.1\ C_i + 200$$

where

L_i = the number of manhours required for installation of a new piece of equipment (the "average" L_i is twice the "low" L_u and one-half the "high" L_i)

W = hourly wage of installation labor ($20/hr was used)

C_i = base cost of a new piece of equipment.

The first term of the equation ($1.5\ L_i \times W$) represents the labor cost to make the modification. The second term ($0.1\ C_i$) is the material cost. The third term (200) represents four hours of an engineer's time at $50/hr.

Figure 2.2-5 USI A-46 screening procedure

Table 2.2-1 Cost estimates[a]

Equipment Type	Analysis High	Analysis Low	Analysis Average	Test and Analysis High	Test and Analysis Low	Test and Analysis Average	Replacement High	Replacement Low	Replacement Average	Comparison High	Comparison Low	Comparison Average	Support Modification High	Support Modification Low	Support Modification Average
Air Circ Fan/Motor	10,000	6,000	8,000	44,500	9,900	15,300	75,000	3,500	13,500	600	100	200	7,000	1,300	2,600
Air Cond Unit	200,000	75,000	100,000	118,000	26,200	40,600	260,000	28,000	115,000	1,600	400	800	15,000	2,400	7,000
Cabinet[b]	13,000	7,000	9,000	44,500	9,900	15,300	4,500	1,000	2,500	600	100	200	850	350	500
Circuit Board	--	--	--	--	--	--	600	90	400	600	100	200	350	230	275
CRDM[c]	--	--	--	44,500	9,900	15,300	32,580K	2,450K	27,000K	600	100	200	33,700	5,800	13,800
Diesel Generator	200,000	75,000	100,000	118,000	26,200	40,600	750,000	250,000	500,000	2,000	400	1,200	88,600	24,800	49,400
Inverter	--	--	--	--	--	--	1,300	200	900	600	100	200	370	240	300
MSIV	18,000	12,000	15,000	53,600	11,900	18,400	350,000	140,000	200,000	600	100	200	37,400	13,100	21,600
Panels	13,000	7,000	9,000	44,500	9,900	15,300	30,000	1,000	7,000	600	100	200	1,870	360	710
Small Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	95,000	6,000	54,000	1,200	200	400	8,100	1,460	4,400
Medium Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	160,000	17,000	78,000	1,200	200	400	16,800	3,400	8,400
Large Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	245,000	31,000	125,000	1,200	200	400	25,200	5,200	12,800
Small Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	42,000	7,000	24,000	900	100	300	12,100	3,040	6,300
Medium Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	87,000	30,000	59,000	900	100	300	18,900	5,200	10,200
Large Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	160,000	50,000	100,000	900	100	300	31,800	8,500	16,800
Racks (Instr.)	13,000	7,000	9,000	44,500	9,900	15,300	3,300	750	1,900	600	100	200	800	350	510
Racks (Bat.)	13,000	7,000	9,000	44,500	9,900	15,300	5,000	1,100	2,800	600	100	200	870	360	540
Strip Chart Rec.	--	--	--	--	--	--	7,500	800	3,400	600	100	200	970	350	570
Relays	--	--	--	--	--	--	800	130	560	600	100	200	350	230	280
Metal-Clad Switchgear	--	--	--	53,600	11,900	18,400	73,000	12,000	42,500	600	100	200	9,000	2,140	4,800
Voltage Switchgear	--	--	--	--	--	--	7,100	300[d]	3,200	600	100	200	680	230	430
Motor Control Center	--	--	--	--	--	--	10,700	350[e]	3,650	600	100	200	1,270	270	410
Transducer	--	--	--	--	--	--	1,300	500	1,000	600	100	200	370	250	300
Transformer	--	--	--	27,400	6,100	9,400	8,500	1,500	5,500	600	100	200	1,530	500	920
Check Valve	6,000	2,000	4,000	27,400	6,100	9,400	9,000	150	4,800	600	100	200	1,150	350	700
Small Instr. Valve	6,400	3,200	4,800	26,800	6,000	9,200	300	90	125	600	100	200	330	230	260
Small Relief Valve	13,000	8,500	11,000	44,500	9,900	15,300	15,000	1,300	8,000	600	100	200	1,150	340	700
Large Relief Valve	13,000	8,500	11,000	53,600	11,900	18,400	45,000	5,200	25,500	600	100	200	3,400	760	1,920
Small Safety Valve	11,000	6,500	9,000	44,500	9,900	15,300	6,000	2,800	4,500	600	100	200	1,030	460	670
Large Safety Valve	11,000	6,500	9,000	53,600	11,900	18,400	35,000	6,000	14,000	600	100	200	2,500	660	1,200

a. Equipment with no estimate for a particular method is not suitable for qualification by that method.

b. Cabinet only. Contents of cabinet not included.

c. K = x 1,000

d. 15 amp-240 V ac 3-pole circuit breaker.

e. 600 V 3-phase ac @ 2 hp motor starter.

Qualification documentation was assumed to cost 150% of the cost of the unqualified components for all but three of the components--small instrument valves, transducers, and relays. These components are produced in large quantities and required in large quantities in typical plants. Their qualification documentation is assumed to be less costly--50% of the cost of the unqualified component.

Comparison

The "comparison" estimate is the cost of comparing dynamic and functional characteristics between equipment in plant and that in the data base. The estimate is based on the assumption that necessary data are readily available. Therefore, no costs resulting from analysis or in-situ testing have been included.

Table 2.2-1 is a summary of cost estimates taken from this contractor's report.

2.2.3 Staff Conclusions

As mentioned in Section 1.3.3, if there are items of equipment that can not be screened out by the data base, either because they are outside the limits of the data base (caveats and bounding spectra) or they do not belong to the eight classes of equipment recommended by SSRAP in the data base, then one of the alternatives is to perform analysis and/or in-situ tests to show seismic ruggedness or similarity with data base or test data. Section 2.2.2.1 addressed this alternative. Figure 2.2-5 shows schematically the steps suggested by the staff if equipment is not covered by the existing data base.

2.3 Development of Methods To Generate Generic Floor Response Spectra

2.3.1 Background

In the current practice of seismic qualification of safety-related equipment (either by analysis or by testing), when the dynamic characteristics of a piece of equipment are known, the required input seismic loading to the equipment, or more exactly, the information necessary to evaluate the response of the equipment to a seismic loading, usually is contained in the form of a set of required response spectra (RRS). If this equipment or component is attached to a floor, these RRS are the same as the "floor response spectra." In the case that this equipment or component is attached to an equipment-supporting structure (such as a rack, a cabinet, etc.), floor response spectra usually are still the starting point of analysis whereby the RRS at the equipment or component attachment locations can be obtained. Floor response spectra, therefore, are essential elements for the qualification of equipment in nuclear power plants.

To determine specific floor motion or equipment-supporting structure motion which is applicable to the development of equipment or component RRS, an expensive and time-consuming time history finite element analysis generally is required. For many operating nuclear power plants, the information on floor response spectra may not have been developed according to the current requirements. In other cases, the information is simply no longer available. The objective of this task was to develop a set of "generic floor response spectra" which can be utilized for qualifying equipment.

The task of developing generic floor response spectra was undertaken by Brookhaven National Laboratory (BNL). The task now is complete. NRC issued a report in September 1983. Following is a summary of this contractor report (NUREG/CR-3266).

2.3.2 Summary of BNL Report, "Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment in Operating Nuclear Power Plants"

The development of generic floor response spectra starts with the concept that there is a degree of boundedness to the structural responses. This report (NUREG/CR-3266) (NRC, September 1983) follows this concept and shows that the response can be bounded within a useful range.

The general approach was to study the effects on the dynamic characteristics of each of the elements in the chain of events that goes between the applied loads and the responses. This includes the seismic loads, the soils, and the structures. Two actual structural models, one BWR and one PWR, were used in the study. For the BWR model (Model 3), a Mark I containment structure is modeled as a single stick, as shown in Figure 2.3-1. For the PWR model (Model 4), the system is modeled as three separate structures on a common foundation. Three stick models are used to represent the shield structure, the steel containment, and the internal structure. Figure 2.3-2 shows this PWR model.

Free-field earthquake response spectra from the El Centro earthquake were used to generate horizontal earthquake time histories. Vertical spectra were not developed in this program. The peak acceleration of this input time history was scaled to a 1-g level as a normalization procedure to study the response. In reporting the proposed generic response spectra, the peak values were normalized to a more realistic time history peak of 0.1 g. The excitation was applied through the soil and into the various structures to produce responses in equipment at each level. An entire range of soil conditions was used with each structure, from soft soil (with a shear wave velocity of 800 ft/sec) to solid rock (shear wave velocity of infinity) in seven steps. For both the BWR and PWR models, stiffness properties were varied, with the same mass, to extend the fundamental base structure natural frequency from 2 Hz to 36 Hz. This resulted in fundamental mode coupled natural frequencies as low as 0.86 Hz and as high as 30 Hz. From all of these models of soils and structures, floor response spectra were generated at each floor level.

The proposed spectra were reported for the top level of a generic structure, based on an earthquake time history with a peak acceleration of 0.1 g. Reduction factors are applied to the peak accelerations to account for the site-specific time history maximum acceleration. A second factor was obtained which recognizes a reduced level of acceleration for equipment located at lower elevations.

Figure 2.3-3 is the maximum generic floor response spectra which were deduced from this study. The curves apply to the top of the structure, which is the point of maximum acceleration. They were normalized from an earthquake time history with a peak acceleration of 0.1 g. These spectra are for five different classes of soils (shear wave velocity from 800 ft/sec to infinity). As shown in the figure, curves A through E are associated with interaction frequencies

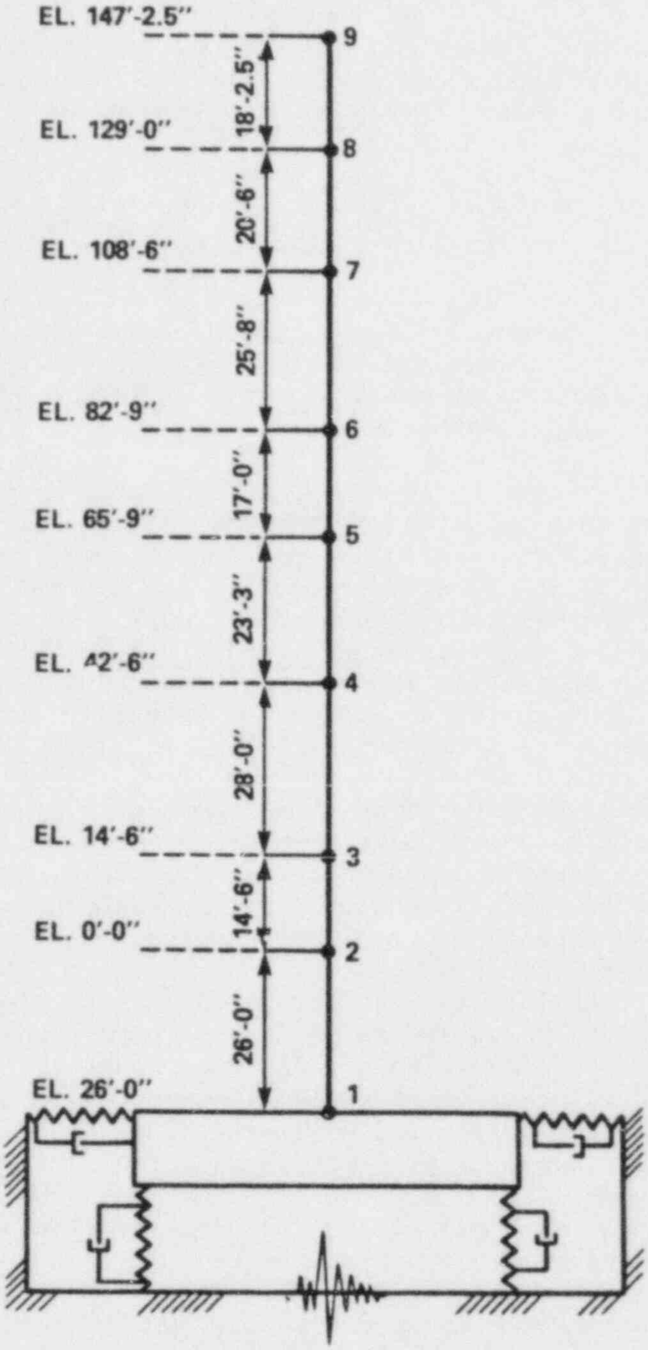


Figure 2.3-1 Model 3

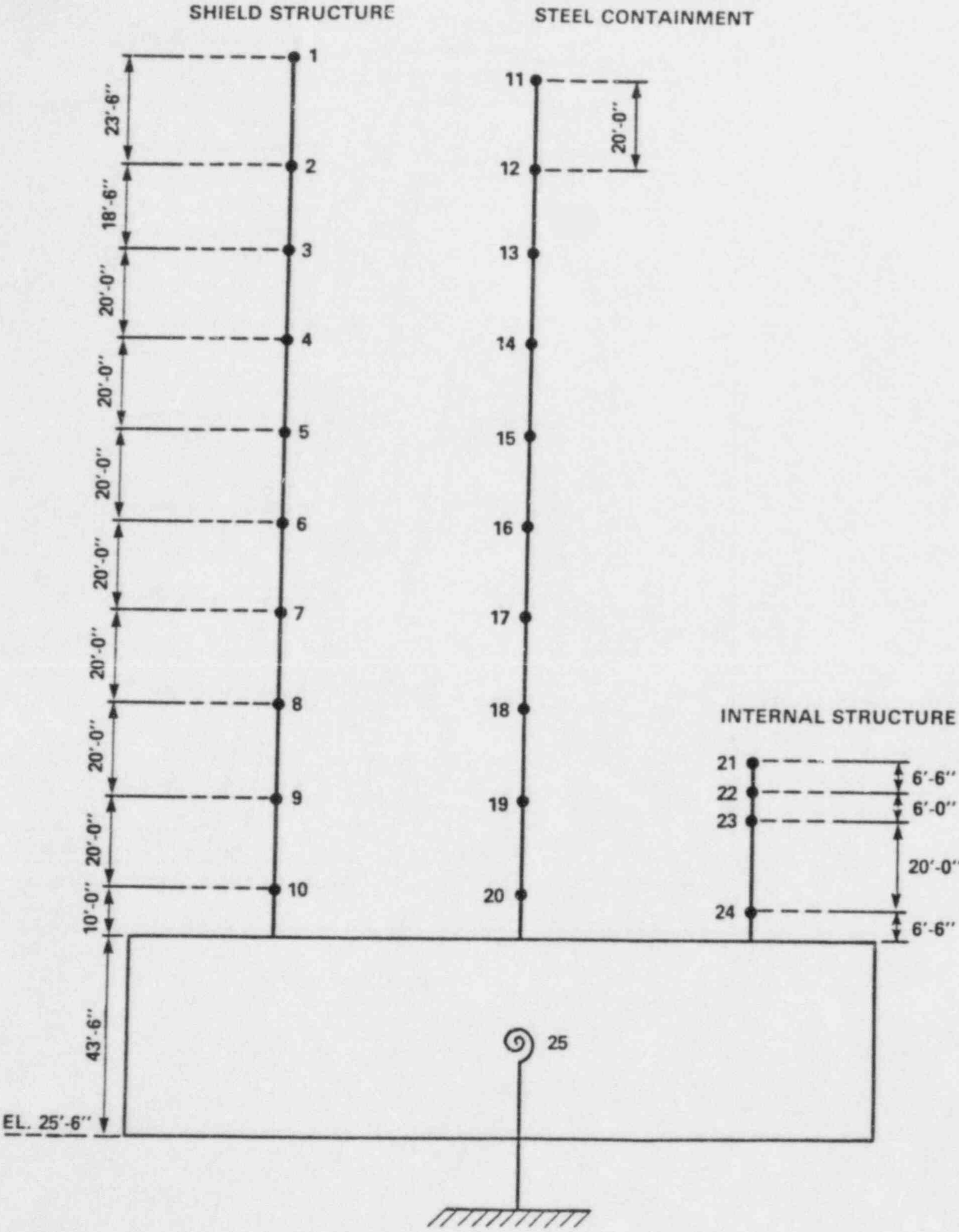


Figure 2.3-2 Model 4

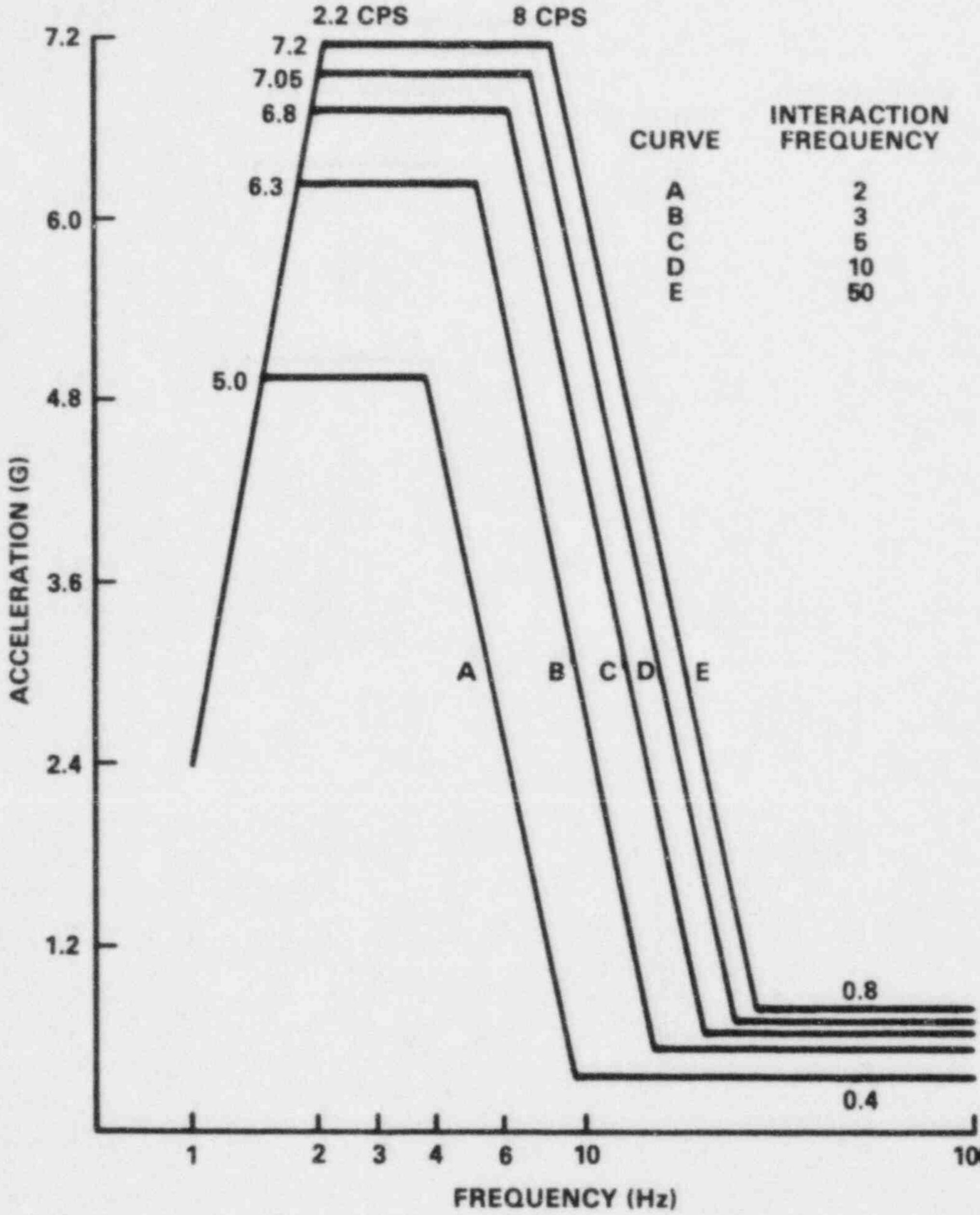


Figure 2.3-3 Generic floor response spectra

(a natural frequency calculation obtained by taking the square root of the ratio of soil stiffness to an equivalent mass of the soil and structure) of 2 Hz through greater than 50 Hz, or from soft soil through solid rock, respectively.

Figure 2.3-4 shows the reduced peak acceleration values that apply to the accelerations in the response spectra at different floor levels. This figure corresponds to soil condition of solid rock (Case E) which has a maximum peak acceleration of 7.2 g at the top level for a 0.1-g earthquake. The peak was calculated to be 6.0 g for a 0.1-g earthquake. This was increased by 20% to 7.2 g because only one earthquake time history was used for the horizontal spectra. As the shear wave velocity of the soil decreases (softer soil), the maximum floor response acceleration decreases. The peak acceleration at the top level of a structure on soft soil was taken to be 5.0 g. This is 30% less than the peak floor response acceleration of 7.2 g at the same elevation for a solid rock soil.

In summary, this report established a procedure for generating the horizontal generic floor response spectra to any operating plant. The procedure allows a utility to use as much or as little information as is available. The conservatisms of the spectra generated increase if little seismic data are available. Generic spectra in the vertical direction were not developed in this program. Because of the conservatism accumulated by this approach every step along the way, the NRC staff believes that conservative vertical generic floor spectra can be reasonably estimated by taking two-thirds of the values of generic floor spectra in the horizontal direction.

2.3.3 Staff Conclusions

Required response spectra (RRS) are needed whether analysis, test, or experience data are used for the qualification. If equipment is attached to the floor, the floor response spectra will be the RRS. If equipment is attached to a supporting structure, the RRS at the point where the equipment is attached can be generated by a variety of ways (see Section 2.2) from the floor response spectra.

By using the methodology described in this section, the floor response spectra can conceivably be generated with reasonable conservatism without having to go through the rigorous time history and finite element analyses normally required. However, the staff believes that this approach will have its limitations, and these limitations should be spelled out clearly.

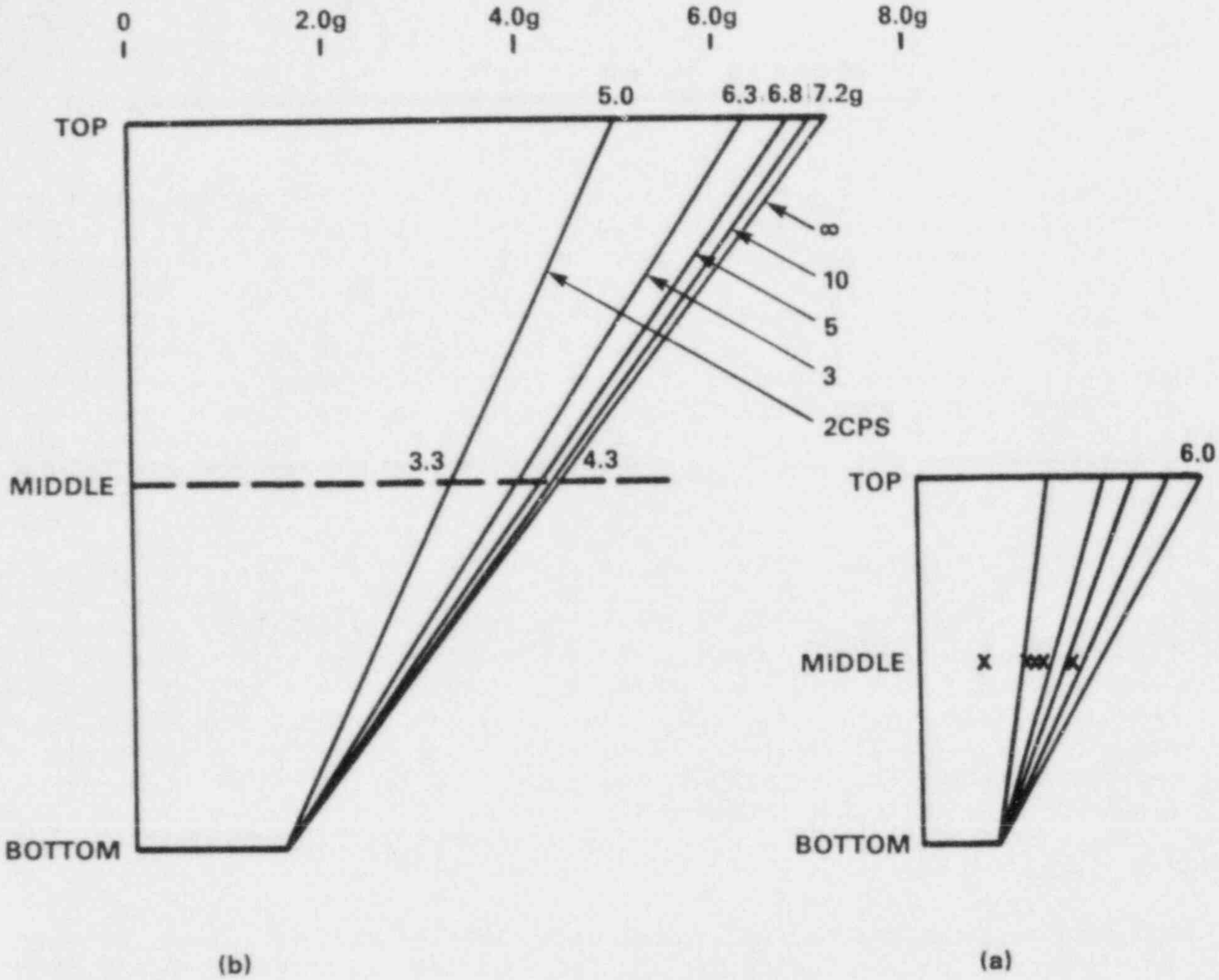


Figure 2.3-4 Generic peak responses at top, middle, and bottom levels

3 REFERENCES

ANSI (American National Standards Institute)

B16.41-1981, "Functional Qualification Requirements for Power-Operated Active Valve Assemblies for Nuclear Power Plants." Draft 3, Rev. II.

N41.9-1976, (or, IEEE Std. 334-1974), "IEEE Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations."

N278.1-1975, "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard."

DC (U.S. Department of Commerce)

(1967), Wood, F. J. (Ed.), "The Prince William Sound, Alaska, Earthquake of 1964 and Aftershocks," Vol. II, Part A, Washington, DC.

EERI (Earthquake Engineering Research Institute)

(May 1973), Meehan, J. F., L. S. Cluff, H. J. Degenkolb, G. A. Carver, D. F. Moran, R. B. Matthiesen, K. V. Steinbrugge, C. F. Knudson, "Managua, Nicaragua Earthquake of December 23, 1972," Reconnaissance Report, Berkeley, CA.

(December 1978), Yanev, P.I. (Ed.), "Miyagi-Ken-Oki, Japan Earthquake, June 12, 1978," Reconnaissance Report, Berkeley, CA.

(1981a), Amrhein, J. E., G. A. Hegemier, and G. Krishnamoorthy, "Performance of Native Construction, Masonry Structures and Special Structures in Managua, Nicaragua Earthquake of December 23, 1972," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. I, San Francisco, CA.

(1981b), Cajina, A., "The Managua Earthquake and Its Effects on the Water Supply System," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. II, San Francisco, CA.

(1981c), Ferver G. W., "Managua: Effects on Systems," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. II, San Francisco, CA.

(1981d), Hanson, R. D., and S. C. Goel, "Behavior of the ENALUF Office Building in the Managua Earthquake of December 23, 1972," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. II, San Francisco, CA.

(1981e), Klopfenstein, A., and B. V. Palk, "Effects of the Managua Earthquake on the Electrical Power System," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. II, San Francisco, CA.

(1981f), Knudsen, C.F., and H. A. Francisco, "Accelerograph and Seismoscope Records from Managua, Nicaragua Earthquakes," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. I, San Francisco, CA.

(1981g), Yanev, P.I., "Industrial Damage," EERI Conference Proceedings: Managua, Nicaragua Earthquake of December 23, 1972, November 29 and 30, 1973, Vol. II, San Francisco, CA.

(July 1981), Lagorio, H. J., and G. G. Mader, "Earthquake in Campaia-Basilicata, Italy, November 23, 1980, Architectural and Planning Aspects," Berkeley, CA.

EQE (EQE Incorporated)

(September 1982), Yanev, P. I., S. W. Swan, "Program for the Development of an Alternative Approach to Seismic Equipment Qualification," Vols. 1 and 2, San Francisco, CA.

(November 1983a), "Investigation of Equipment Performance in Foreign Earthquakes and the 1964 Alaska Earthquake," San Francisco, CA.

(November 1983b), "Seismic Experience Data Base--Average Horizontal Data Base Site Response Spectra," San Francisco, CA.

(November 1983c), "Seismic Experience Data Base--Data Base Tables for Seven Types of Equipment," San Francisco, CA.

(August 1984), "The Performance of Industrial Facilities and Their Equipment in the Coalinga, California, Earthquake of May 2, 1983," with Addendum "Summary of Ground Motion Intensities from the Coalinga, California, Earthquake of May 2, 1983: Based on Observed Ground, Structural, and Equipment Response," San Francisco, CA.

IEEE (Institute of Electrical and Electronics Engineers)

Std. 344-1974 (or ANSI N41.9-1976), "IEEE Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations."

Std. 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

Std. 382-1980, "IEEE Standard for Qualification of Safety-Related Valve Actuators."

Std. 501-1978, "IEEE Standard Seismic Testing of Relays."

Std. 649-1980, "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations."

(November/December 1980), Carfagno, S. P., G. H. Herberlein, Jr., "A Study of the Effect of Aging on the Operating of Switching Devices," IEEE Transactions on Power Apparatus and Systems, Vol. PAS-99, No. 6.

JAERI (Japan Atomic Energy Research Institute)

(August 1979), Uga, T., K. Shiraki, T. Homma, H. Inazuka, N. Nakagima, "Operating Function Tests of the PWR Type RHR Pump for Engineered Safety System Under Simulated Strong Ground Excitation," JAERI-M8354.

LLNL (Lawrence Livermore National Laboratory)

(July 1981), Johnson, J. J., G. L. Goudreau, S. E. Bumpus, O. R. Maslenikov, "Seismic Safety Margins Research Program, Phase I Final Report-SMACS."

NAS (National Academy of Sciences)

(1973), Committee on the Alaska Earthquake, Division of Earth Sciences, National Research Council, The Great Alaska Earthquake of 1964, Engineering Volume, Washington, DC.

NCEL (U.S. Naval Civil Engineering Laboratory)

(June 26, 1964), Stephenson, J. M., "Earthquake Damage to Anchorage Area Utilities - March 1964," Technical Note N-607, Port Hueneme, CA.

NRC (U.S. Nuclear Regulatory Commission)

(October 1975), "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), Washington, DC.

(July 1981), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, NUREG-0800, Washington, DC.

(June 1983), Brookhaven National Laboratory, "Identification of Seismically Risk Sensitive Systems and Components in Nuclear Power Plants Feasibility Study," NUREG/CR-3357, Washington, DC.

(August 1983), Lawrence Livermore National Laboratory, "Correlation of Seismic Experience Data in Non-Nuclear Facilities With Seismic Equipment Qualification in Nuclear Plants (A-46)," NUREG/CR-3017, Washington, DC.

(September 1983), Brookhaven National Laboratory, "Seismic and Dynamic Qualification of Safety Related Electrical and Mechanical Equipment in Operating Nuclear Power Plants," NUREG/CR-3266, Washington, DC.

(June 1984), Idaho National Engineering Laboratory, "The Use of In-Situ Procedures for Seismic Qualification of Equipment in Currently Operating Plants," NUREG/CR-3875, Washington, DC.

(August 1984), Southwest Research Institute, "A Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment," NUREG/CR-3892, Washington, DC.

(RG 1.29), "Seismic Design Classification," Rev. 3, Washington, DC, Sept. 1978.

(RG 1.40), "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," Washington, DC, March 1973.

(RG-1.61), "Damping Values for Seismic Design of Nuclear Power Plants," Washington, DC, October 1973.

(RG 1.73), "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants," Washington, DC, January 1974.

(RG 1.92), "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 1, Washington, DC, February 1976.

(RG 1.100), "Seismic Qualification of Electric Equipment for Nuclear Power Plants," Revision 1, Washington, DC, August 1977.

(RG 1.122), "Development of Floor Design Reponse Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1, Washington, DC, February 1978.

(RG 1.148), "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants," Washington, DC, March 1981.

RES (Richardson Engineering Services, Inc.)

(1975), "Process Plant Constrction Estimating Standards."

(1981), "Process Plant Construction Estimating Standards."

SSRAP (Senior Seismic Review Advisory Panel)

(February 1984, Revised August 1984), Kennedy, R. F., W. A. Von Riesmann, P. Ibanez, A. J. Schiff, L. A. Wyllie, Jr., "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants."

APPENDIX A

SUMMARY OF TECHNICAL WORK COMPLETED THAT IS NOT IMPLEMENTED IN USI A-46 RESOLUTION

In this appendix a summary of work done and major conclusions is presented. Detailed discussions of certain tasks are then included as separate appendices.

The following sections summarize contractors' results and conclusions of the various tasks. Unless otherwise stated, they represent the contractors' viewpoints and recommendations.

A.1 Identification of Seismic Risk Sensitive Systems and Equipment

A.1.1 Background

The objective of this task was to investigate possible methods of developing a generic minimum equipment list. If a methodology could be developed to evaluate the risk importance of safety systems and equipment, equipment could be ordered by the contribution to risk. Equipment whose failure resulted in a small change in risk could then be culled from the qualification list.

A.1.2 Summary of Task

Brookhaven National Laboratory (BNL) under contract to the NRC conducted a study (NRC, June 1983) to evaluate the seismic risk sensitivity of system and components in a PWR and a BWR. Both plant models used were hybrids in that they are not representative of any existing plant. The PWR model consisted of modified Surry Plant fault trees and event trees from the WASH-1400 study and used fragility data developed for the Zion plant. The BWR model consisted of modified WASH-1400 (NRC, October 1975) Peach Bottom risk models and Oyster Creek fragility data.

The intent of this study was initially to develop a generic risk-ordered list of plant equipment which could be applied to specific plants with some additional guidelines to develop plant-specific minimum equipment lists. However, BNL concluded, and the staff agrees, that results of the study should not be used generically. BNL's conclusion states that the study presents a methodology that can be applied on a plant-specific basis to develop a risk-ordered equipment list.

A.1.3 Staff Position on Task

For plants with existing seismic probabilistic risk assessment (PRA) studies, the staff believes it may be possible in some cases to eliminate components from the seismic qualification program on the basis of low risk sensitivity. If a utility should decide to conduct a PRA study using the methodology developed by BNL, the staff would consider it to be an acceptable method subject to the analysis assumptions and inherent uncertainties.

A.2 Assessment of Adequacy of Existing Seismic Qualification

A.2.1 Background

This task involves a study by Southwest Research Institute (SWRI) to evaluate past and present methods to qualify mechanical and electrical equipment to withstand seismic events. Conclusions have been documented in a contractor report titled "A Research Program for Seismic Qualification of Nuclear Plant Electrical and Mechanical Equipment" (NUREG/CR-3892) (NRC, August 1984). Some examples demonstrating the application of this approach are included in that report.

A.2.2 Summary of Work Accomplished

The concept of vibration equivalence is a key factor in development of the correlation of methodologies for seismic qualification of equipment. Vibrational equivalence forms the basis for a damage comparison between two different motions. In the qualification of nuclear power plant equipment, a great variety of physical failure mechanisms may occur. Therefore, the concept of vibration equivalence was generalized to include an arbitrary type of failure or malfunction, that can always be established by input vibrational conditions denoted as the fragility levels. It is understood that the failure or malfunction may or may not impart permanent damage to the equipment.

The conceptual approach for applying vibrational equivalence to correlation of equipment qualification by test is shown in Figure A.2-1. The upper and lower halves of the diagram (conditions 1 and 2, respectively) each represent the independent establishment of a fragility, or threshold of failure level, in equipment which is subject to a dynamic excitation at location x. The effect of the response at location y is to actuate a failure mechanism which depends on the equipment. This arbitrary failure mechanism is dependent on the response amplitude failure mechanism and is dependent on time. Thus, the failure is indirectly dependent on the excitation amplitude, frequency, and time. If the excitation is manipulated so that failure barely occurs, then the threshold of failure, or fragility function F_{xy} (f,t) is generated. This function represents a surface, any point on which corresponds to failure of the equipment. If more than one physical failure mechanism at more than one response point is present, then each possesses a failure surface, and the minimum value composite failure surface becomes of concern. The central assumption of the vibration equivalence concept is then postulated: the establishment of failure conditions (see Figure A.2-1 for excitation conditions 1 and 2) is possible by various types of vibration excitations, and the corresponding amplitude, frequencies, and time durations constitute equivalent excitations.

Generally, the information on failure, or malfunction, is not required as part of an equipment qualification process. On the other hand, functionality of an item of equipment at specified excitation levels is required for qualification. Functionality and fragility are very much related--fragility is the upper limit of functionality. Conversely, existing qualification data, which include excitation levels and functionality data, may be useful as a lower bound for fragility. Thus, since fragility data are necessary for a general application of the vibrational equivalence concept, use of such existing

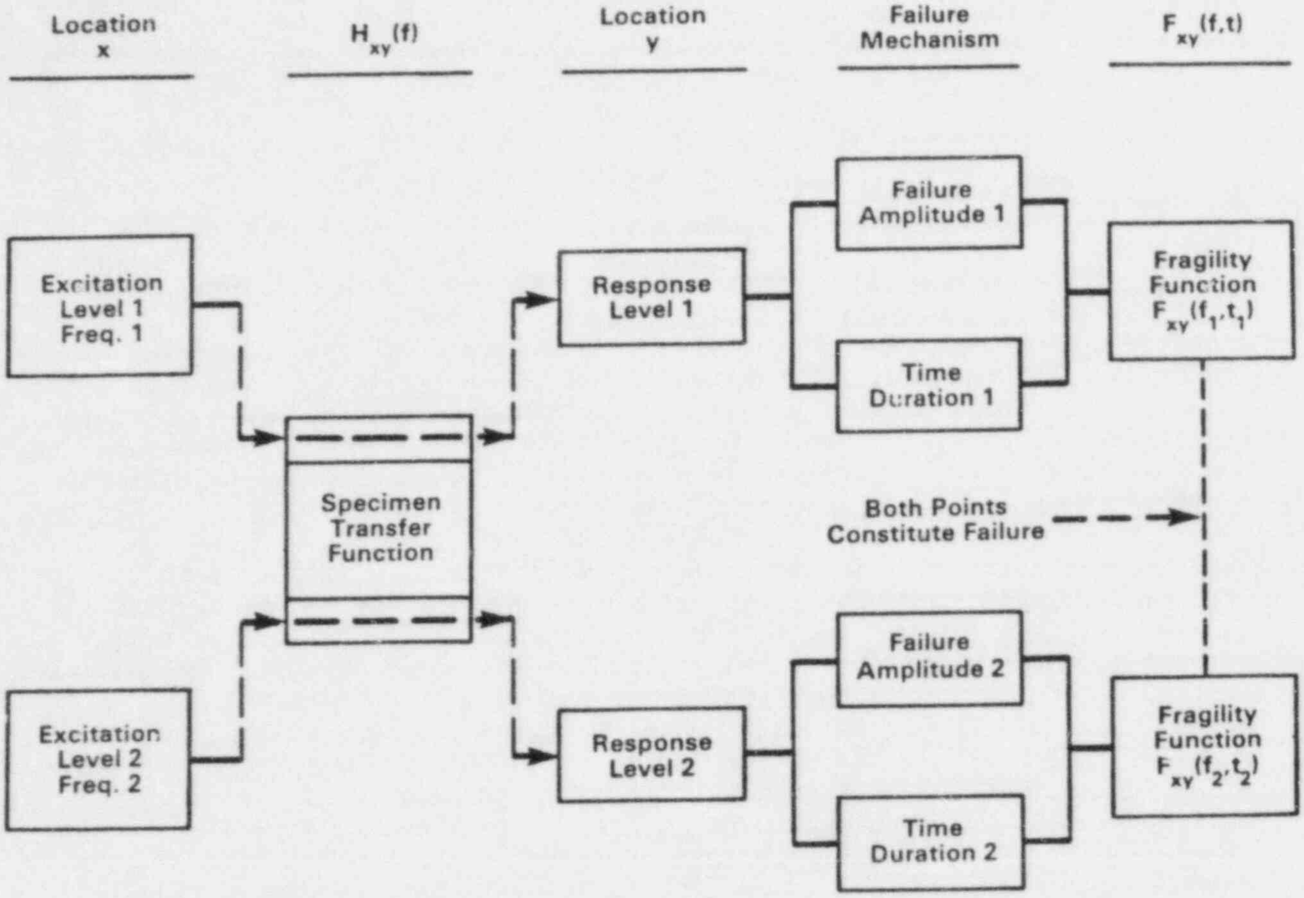


Figure A.2-1 Conceptual approach to vibration correlation

qualification data, where possible, is highly desirable to avoid the necessity of generating or collecting more precise fragility information for the great variety of equipment typically contained in a nuclear power plant.

The most general description of a fragility concept is shown in Figure A.2-2 as a fragility surface. This surface can be represented as a function F_{xy} (f, t) = M_f (f, t), where M_f (f, t), measured at the fragility surface, can be in terms of the amplitude of the excitation, the response spectrum power spectrum, or a variety of other parameters which may be used, or have been used, in typical equipment qualification procedures. The true surface may be quite complex, but a simpler lower bound surface can be defined conservatively from existing qualification information which is acceptable for practical engineering purposes.

A convenient method of measuring the onset of failure is proposed by the contractor as the damage fragility ratio

$$D_{fr} = \frac{M(f,t)}{M_f(f,t)} \leq 1$$

where M(f,t) is the value of the actual excitation function and M_f (f, t) is the value of the fragility function at the same conditions of frequency and time. This is shown in Figure A.2-3. A damage fragility equivalence similar to that described in Figure A.2-1 can then be stated as:

$$\frac{M(f,t)}{M_f(f,t)} = \frac{M(f_2,t_2)}{M_f(f_2,t_2)}$$

This is the general basis for comparing various test motions.

The report then proceeded to define simple systems and complex systems. A simple system is one whose fragility function is influenced by a single resonance, and therefore can be generated by a slowly swept sine or narrow band random excitation. A complex system is one where several failure modes can occur as the result of multiaxis and/or multimode response, and interaction between responses is included. Because of the difficulties involved when considering complex systems, it is advantageous to develop approximations as required to reduce the system to a simple one.

A number of procedures have been developed in structural analysis to look at the combined effects of multiaxis and multimode response. These procedures, such as absolute sum method, square root of the sum of the squares (SRSS) method, double sum method, closely spaced modes method, grouping method, ten percent method, Lin's method, and complete quadratic combination (CQC) method, are all generally based on modal or response spectrum analysis. Any one of these methods will give an estimation of the combined maximum peak response of a complex systems. In developing a fragility surface for existing qualification data, it was recommended by the contractor that a correction factor, generated from resonance search data, be used to modify the level of qualification excitation in order to develop an approximate lower bound fragility function.

The next step is to establish a correlation between the approximate fragility function (namely, existing qualification information) and the qualification corresponding to a different set of criteria. In a specific application, some judgment must be used, the detail of which may vary with each case. Several examples which demonstrate the application of these methodologies are included in the contractor's report. (See Figure A.2-4 for possible combinations of fragility function and qualification parameters.)

In summary, the results of a previous qualification are used first to establish some form of an approximate or acceptable fragility function. Then, the new criteria are compared to this acceptable fragility function to determine whether a more severe or less severe test is implied. If result shows a less severe test is implied by applying the new criteria, then it can be concluded that this equipment is still qualified to the new set of criteria. In some cases, a more accurate fragility function may need to be established in order to provide a final determination of the comparison. In these cases, the contractor suggested that it may be more practical to consider a completely new requalification.

The contractor also surmised that much of the previously qualified equipment will be able to be requalified to new criteria by the analytical method developed. His belief is based on the fact that many qualification tests prior to 1975 included sine wave and sine beat excitations of some form. The comparison of relative damage severity indicated that such motions produce

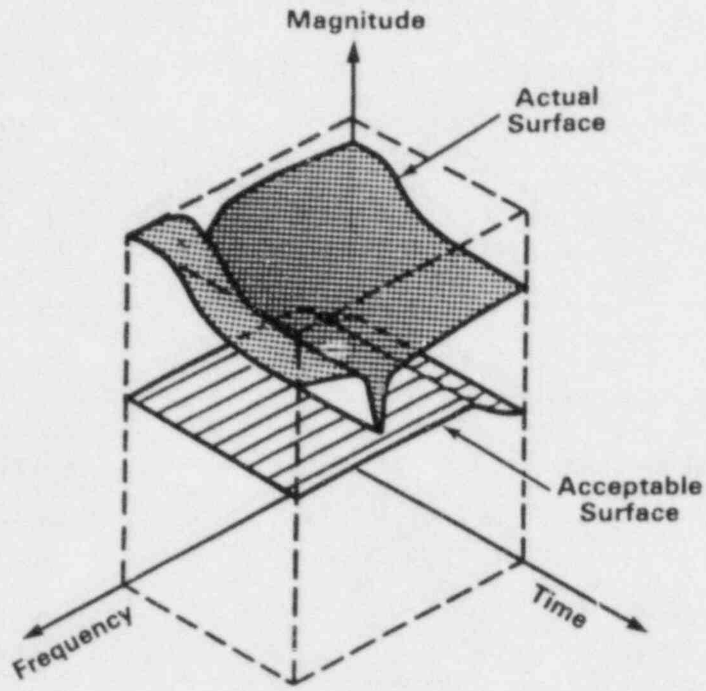


Figure A.2-2 Comparison of actual with acceptable fragility surface

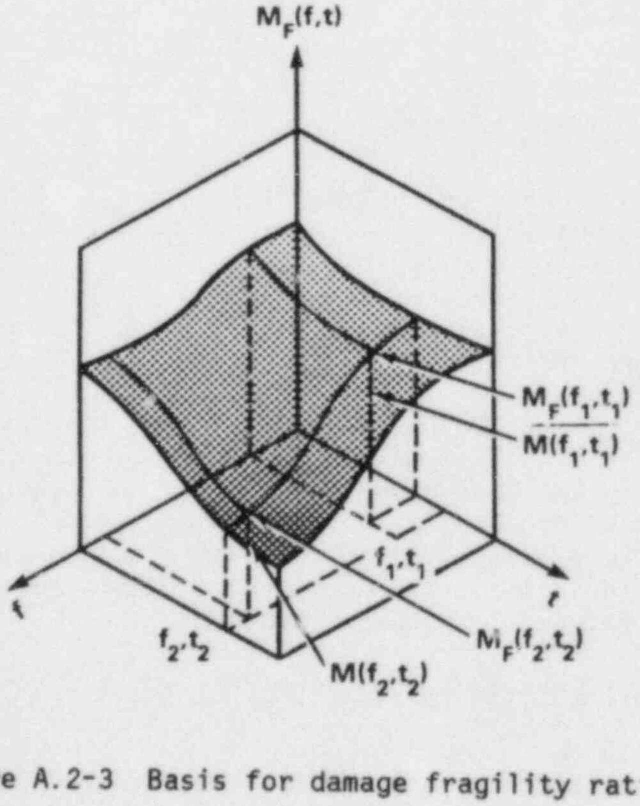


Figure A.2-3 Basis for damage fragility ratio

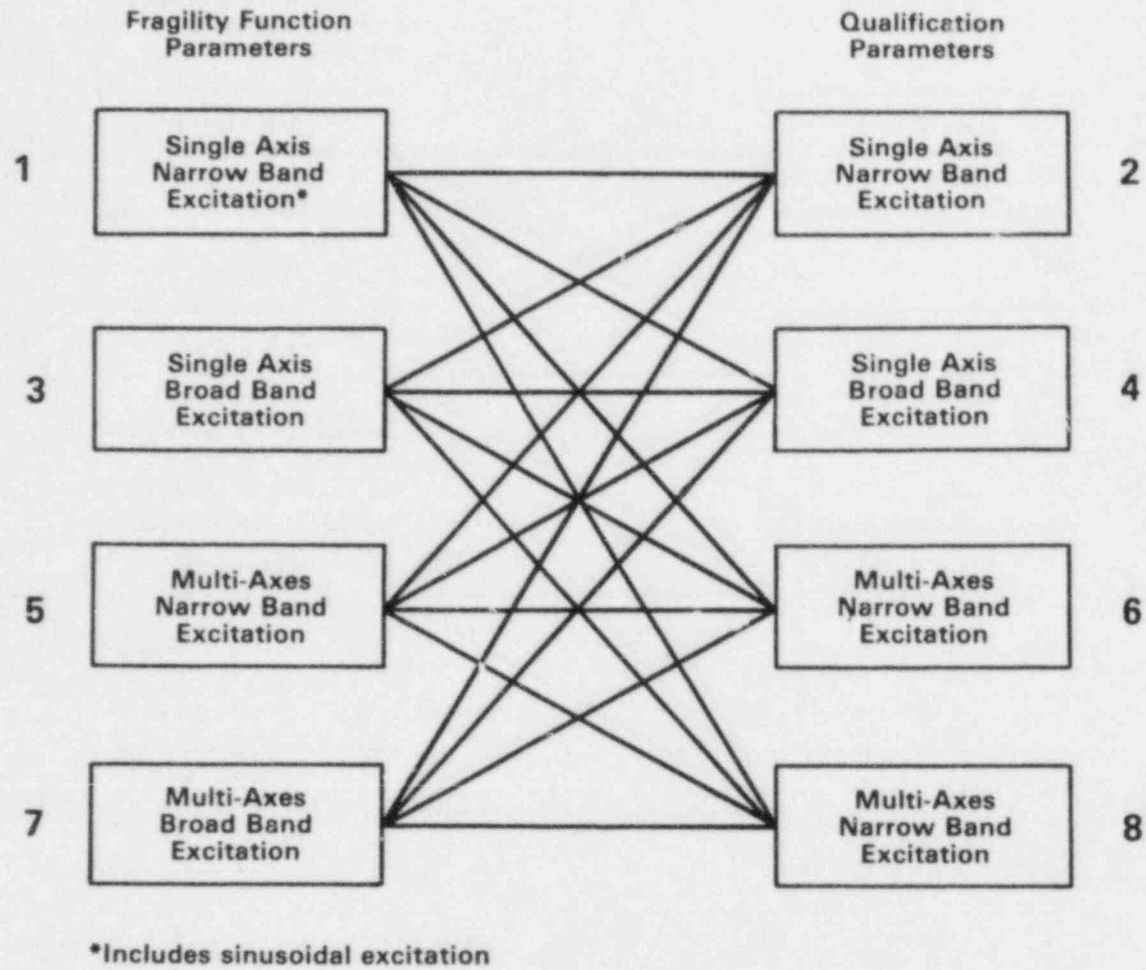


Figure A.2-4 Possible combinations of fragility function and qualification parameters

significantly more potential damage than do typical random motion simulations that have been more generally used after 1975.

A.2.3 Staff Conclusion

The technical basis and general methodology to correlate seismic qualification tests have been developed and demonstrated, but are of limited practical value in their present form because of the need to either know the fragility level or estimate the fragility of the equipment and know the required response spectra. It may be useful in special cases.

A.3 Related Topics Covered by the INEL Contractor's Report on In-Situ Testing

Even though the contractor report (NUREG/CR-3875) (NRC, June 1984) is concerned mainly with how to utilize in-situ testing to assist in performing seismic qualification of equipment, the contractor studied other related topics. Among them are the following.

A.3.1 Operability and Failure Modes:

In order to develop methods to utilize experience data to qualify equipment, the contractor suggested that a systematic treatment of operability is necessary. The failure modes which result in inoperability, from the contractor's viewpoint, are an essential ingredient to these methods. The contractor first defined inoperability and its causes and then identified all possible failure modes that may cause inoperability during an earthquake.

Inoperability is defined as any action or interaction of component parts or interfaces which prevents a component from performing an active operation or maintaining a state continuously. Inoperability can result from:

- inability to monitor the control condition
- inability to change states when so directed
- inability to maintain the current state when no change of state is directed

The contractor suggested that inoperability during an earthquake occurs through the following modes:

- structural integrity - stress limits are exceeded, permanent deformation occurs, flaw initiation or extension occurs.

- operability loss due to temporary or permanent reconfiguration - vibratory elastic motion results in a change of state or prevents a change of state from occurring.

- structural interference - excessive relative motion results in a tolerance mismatch.

- nonstructural changes in state-peizoelectric effects, effects of dynamics on contact resistance, and others; anywhere a fundamental nonstructural response is affected by vibration or stress.

The contractor then proposed that similarity between two equipment designs can be defined as similarity in potential failure modes. The basic premise involves two pieces of non-identical equipment having a common critical failure mode. The first piece has been qualification proof tested and its controlling design features are either identical to or inherently more fragile than the equipment in question. In that case, qualifying the first, amounts to qualifying the other to the same environment. The contractor suggests the procedures below to establish seismic capacity based on similarity.

- Specify operability requirements, take into account whether equipment is required to operate and/or maintain a continuous state during earthquakes. If there are no requirements during the earthquake, certain failure modes will be eliminated and qualification is simplied.

- Identify the design features/subcomponents which affect operability. The procedure will be impractical if there are too many.

- Identify similar pieces of equipment, i.e., equipment with nominally the same or less seismic capacity in the potential failure model(s). Some form of design evaluation/comparison will be required in making this assessment. Equipment used for comparison must be of known seismic capacity. The staff believes that in-situ testing will be a valuable tool to establish dynamic similarity between equipment through the comparison of the dynamic characteristics (mode shapes, natural frequencies, damping, size, shape, weight, etc.).

A.3.2 Environmental Aging Consideration:

The environmental history of a piece of equipment can produce changes in properties and dimensions which affect its seismic capacity. Addressing the total environmental qualification of equipment in operating plants is impractical. The contractor adopted an approach based on the interaction of aging and seismic capacity. Such an approach suggests that since some aging mechanisms will not affect seismic capacity, these cases need not be considered in seismic qualification.

The contractor considered the use of in-situ testing in evaluating the effects of aging on seismic qualification, however, no well developed technologies were identified. Consequently, aging has been examined in a broader context where:

- The consequences of aging degradation are examined. This allows the relationship between dynamic qualification and aging degradation to be organized in a fashion which more clearly demonstrates the interaction.

- Alternate criteria based on failure mode and similarity analysis. This provides both an organized aging assessment procedure and a method for using test data from "similar" equipment.

- Equipment without specific operability requirements during seismic events has been identified as less vulnerable to aging.

The effect of aging on seismic capacity is illustrated in Figure A.3-1. A systematic basis for evaluating aging degradation is provided by the failure mode analysis and the procedures embodied in Figure A.3-1. This method as proposed by the contractor is as follows. First, a determination of any aging effects produced by the design-basis environments should be conducted. This involves listing all vulnerable materials and examining environmental data for each. Presently, such data are only available for some materials. Those components demonstrating no environmental aging require no further examination. For components containing materials affected by the design environments, the aging mechanisms are defined and categorized by the contractor as follows.

- <u>Category I aging</u>: This includes all aging mechanisms which modify the dynamic response. The changes in dynamic response can affect all four failure modes defined earlier. Each failure mode must be examined in light of the anticipated degradation. If it cannot be established that no significant change in seismic capacity occurs, then the critical failure modes should be established. A similar system with a known aged seismic capacity may provide data on which to base the aged seismic capacity. Adversely affected items should be qualified to current criteria.

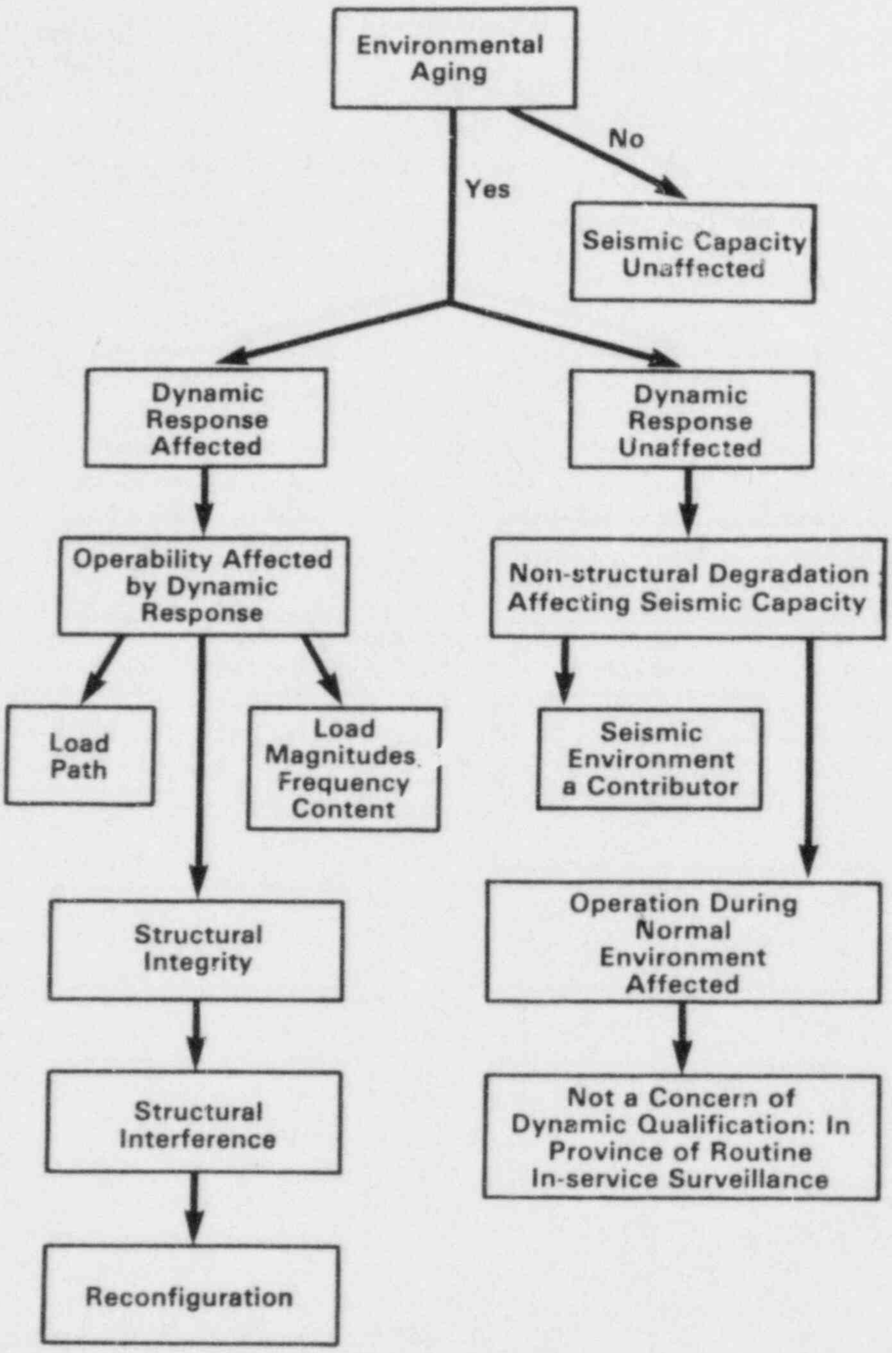


Figure A.3-1 Effect of aging on seismic capacity

- Category II aging: This is any aging mechanism which could affect the operability of safety equipment when combined with the predicted seismic loads. It is assumed that the dynamic response has not been affected. This is a type of aging mechanism which impacts only the nonstructural effects. It need only be examined if a known aging effect exists in a component. Again, seismic capacity can be inferred from tests on similar equipment. However, the requirements on similarity are somewhat more stringent in this case. Any loss of seismic capacity will be due to degradation combined with local structural dynamics. Thus, similarity requires that both be simulated.

- Category III aging: The mechanisms of this category are those identified which have no effect on seismic qualification (IEEE, November/December 1980). For a typical component many mechanisms would fall in this category.

The application of the above approach would probably be most economical if conducted in stages. The contractor proposed that initially all equipment would have a cursory examination for (a) no aging, (b) some aging, though with no effect on seismic capacity, (c) aging with a potential effect on seismic capacity, or (d) too complex to determine easily. For situations where further consideration is warranted, the steps are similar to those as described in the first paragraph of this appendix. The failure modes are used to establish similarity, and data from similar equipment are transferred to the equipment in question. The important factor is that much equipment will exhibit no significant seismic aging interaction of concern and, thus, screening can narrow the field effectively without overlooking substantial aging degradation.

APPENDIX B

PERFORMANCE OF POWER FACILITIES DURING THE 1964 ALASKA EARTHQUAKE

City of Anchorage Gas Turbine Plant

The plant contained two gas turbines rated at 15,000 kW and six older diesel generators. Three reports give different versions of what happened at the plant (NAS, 1973, p. 1053; NCEL, June 26, 1964; and F.F. Mautz, cited in EQE, November 1983a). Apparently, at least one unit operated through the earthquake even though a control cabinet or transformer toppled over. One unit was switched to diesel oil and started to supply power, but it was shut down when diesel fuel supply was lost because the fuel storage tank failed. Unit 2 became unbalanced several weeks later because of aftershocks. It was realigned and put back in operation.

Chugach Power Plant at Knik Arm

The plant had three coal-fired boilers. Only one unit was operating at the time of the earthquake. It continued to operate for about five minutes after the earthquake and then was shut down by some disturbance outside the plant. There was no other power available so it could not be restarted (Mautz, in EQE, November 1983a). The facility suffered structural damage in the coal bunker bay and an ash hopper fell (NAS, 1973, p. 255). One boiler fired up in 24 hours. One week later 3 boilers were in operation, and 5 days later the plant was back in normal production while structural repairs progressed. The turbine bay was undamaged. There was no electrical damage. One compressor was put out of service either by vibration or foundation settlement. Filter mixing tanks which were not bolted down fell over and ripped out piping. Pipe hangers also failed.

Fort Richardson Coal-Fired Steam Plant

This plant had a generating capacity of 18,000 kW and a heating capacity of 1,080,000 lb/hr of steam; it had five turbine generators and eight coal-fired boilers (NAS, 1973, p. 911). It produced steam without interruption and could have produced power but the receiving stations were not functioning. The plant had limited structural damage and extensive nonstructural damage, mainly to the superheater tubes and tile bricks.

Elmendorf AFB Coal-Fired Steam Plant

This plant had a capacity of 22,500 kW electric and a heating capacity of 950,000 lb/hr steam (NAS, 1973, p. 934). It had three 7,500-kW generators and six boilers. The plant operated through the earthquake, was shut down about an hour later (Mautz, in EQE, November 1983a), and was back in operation in 5 hours. The shutdown was caused by a loss of circulating water from the failure of buried transit pipe. A break in one air control line also contributed to the shutdown. Structural and nonstructural damage was similar to that at Fort Richardson.

Eklutna Hydroelectric Plant

This is a 30,000-kW hydroelectric plant. The plant apparently operated through the earthquake (NAS, 1973, p. 464 and NCEL, June 26, 1964, p. 3). The intake structure and conduit were damaged by soil consolidation and there was air circuit breaker and transformer damage to porcelain insulator columns.

Chugach Bernice Lake Gas Turbines

Continued to operate (NAS, 1973, p. 1073).

Chugach Cooper Lake Hydroelectric Plant

Undamaged, but transmission line down (NAS, 1973, p. 1062).

Port of Whittier Heating and Power Plant

This plant had three steam turbine generators--two 2,000 kW and one 2,500 kW. Damage to the plant was minor. One condensate line and two 10-in. water supply lines were broken (NAS, 1973, p. 1079).

Cordova Diesel Engine Plant

No report of damage (NAS, 1973, p. 1067).

Kodiak

No report of damage to power generators (NAS, 1973, p. 1070).

Homer

No report of damage to power generators (NAS, 1973, p. 1071).

Seldovia

No report of damage to power generators (NAS, 1973, p. 1071).

Kenai

This town had an old substation with skid-mounted transformers and regulators. Some transformers and regulators tipped over causing short circuiting (NAS, 1973, p. 1072).

APPENDIX C

PERFORMANCE OF POWER AND INDUSTRIAL FACILITIES DURING SOME FOREIGN EARTHQUAKES

C.1 Managua, Nicaragua, Earthquake of December 23, 1972

C.1-1 Earthquake Data

Magnitude:	Ms = 6.2, Mb = 5.6
Time:	December 23, 1972, at 6 hr 29 min GMT
Location:	Beneath the center of Managua
Depth:	8 km

C.1-2 Ground Motion Records

Four strong motion accelerograms and nine seismoscope records were obtained from a series of earthquakes that occurred in December 1972 and January 1973. One accelerograph and 13 seismoscopes recorded the main shock on December 23 (EERI, 1981f). The only accelerograph from this shock was recorded at the Esso Refinery; peak ground accelerations were 0.38 g in E-W direction, 0.34 g in N-S direction, and 0.33 g vertically.

An estimate of the ground motion to which Managua's major industrial facilities were exposed is provided in a report by P. I. Yanev (EERI, 1981g): "It is the author's estimate, based on the accelerogram taken at Esso refinery and on judgment, that the industrial facilities experienced an earthquake of moderate duration with the peak ground acceleration exceeding .25 g. Some facilities experienced accelerations exceeding .60 g."

C.1-3 General Effects of the Earthquake

Approximately 10,000 people died. Many structures collapsed completely and economic loss was heavy. The downtown business area, the industrial areas, and the surrounding residential areas were most seriously affected. The downtown area was almost totally destroyed, but most modern high-rise structures sustained the shock without collapse and often without significant structural damage. The architectural and other nonstructural components of these newer buildings were often damaged severely. Mechanical systems in buildings were generally inoperative after the earthquake (EERI, 1981g).

Power and industrial facilities suffered considerably lower losses. Damage to equipment and equipment systems was responsible for the greatest part of the industrial loss. Much of the damage and consequent delays in operation could have been prevented with improved equipment anchorages and other minor details. Few industrial facilities were left undisturbed by the earthquake (EERI, 1981g).

C.1-4 Electric Power Facilities

Two hydroelectric plants, each with two 25-mW units, were located 80 and 100 km from Managua. 100 km northeast of Managua there was a 15-mW gas turbine generator. None of these plants were damaged; however, they all disconnected electrically from Managua.

The Enaluf power plant in the City of Managua is a thermal electric power plant with one 40-mW and two 15-mW steam turbine generators. The plant is located on the shore of Lake Managua, immediately adjacent to (or possibly even on top of) the Tiscapa fault, which caused the event. Displacements of 10 in. along this section of the fault were reported within 200 m of the plant site.

It is reported that the facility was designed for a static-equivalent lateral load coefficient of 10%. Most of the equipment was anchored to the floor and experienced no damage. Some of the worst damage occurred to unanchored equipment which was free to displace or fall (EERI, 1981g).

The main shock caused generators to trip off-line by protective relays either through legitimate protective measures or through malfunctions due to vibration of mechanical contacts (EERI, 1981e). One of the units was back in service in two weeks and the second in three weeks. The third unit was not operative for several months because of greater damage and misalignment of the turbine shaft (EERI, 1981g).

Arturo Roja, General Manager of Enaluf (EERI, 1981c), prepared a list of the equipment damaged in the earthquake. This list is presented in Table C.1-1. Some of the reported damage relevant to the SQUG project is discussed below.

- All three deaerators moved on their bases. The Unit 3 deaerator also sustained a broken air pipe connected to the deareator and damage to refactory lining (EERI, 1981e).

- Draft fans, motors, and vents associated with the boiler and exhaust system did not suffer significant damage. Several of these shifted out of alignment (EERI, 1981e).

- All three steam turbine generators sustained sufficient damage to incapacitate them. Bearings in Unit 3 were badly worn when the emergency oil pump motors lost their DC power when the battery racks failed and the batteries broke. Misalignment and broken turbine blades were common to all three generators. There was also some relative movement between the turbine generator supports and the floor which resulted in further damage and misalignments (EERI, 1981e).

- The condensers associated with the 15-mW generators shifted 6 inches. This broke the valve between the pump and the pipe to the condenser (EERI, 1981e, Figure 19).

- The obvious pipe damage discovered on Unit 3 steam system included broken piping in the boiler. A pipe connected to the saturated vapor valve of the deaerator was broken. The high pressure pipe of the primary element of the three recirculating valves for the water seating pumps was bent. In addition, three recirculating valves suffered cracks on their interior sections. The condenser had air pipe damage.

Table C.1-1 Damage to Enaluf Steam Plant

Siemens Unit No. 1 (15 mW)

1. Generator case: Shafts displaced.
2. Forced draft fan out of alignment.
3. Induced draft fan out of alignment.
4. Condensate pump: Burned-out bearing.*
5. 440-V ac Panel No. 2: Fallen.*
6. Condensate pump intake valve broken.*
7. Boiler No. 1: Tubing broken and refractory walls fallen.
8. Deaerator No. 1: Fallen from its base.
9. Chimney of Boiler No. 1: Anchor bolts broken and stack leaning.

Siemens Unit No. 2 (15 mW)

1. Generator case: Shafts displaced.
2. Draft fan forced out of alignment.
3. Induced draft fan out of alignment.
4. Boiler No. 2: Refractory walls fallen.
5. Deaerator No. 2: Fallen from its base.
6. Intake valve of condensate pump broken.*

Franco Tossi Unit No. 3 (40 mW)

1. 440-V ac control center: Fallen.*
2. Main transformer bushings broken.
3. Starting transformer bushings broken.
4. Exciter transformer bushings broken.
5. Unit transformer bushings broken.
6. Ljungstrom pre-heater seals damaged.
7. Four turbine bearings burned out. (Batteries broken, cutting off supply to DC-powered emergency lube oil pump.)
8. 69-kV switch bushings broken.

* - Denotes equipment failures of particular interest to SQUG.

- A motor control center (EERI, 1981e, Figure 23) fell over with many of the drawers coming loose from the main cabinets. After the earthquake, the cabinets were uprighted and the system was checked out and placed back into service. Before the earthquake, the cabinets had been secured in place with small bolts in concrete anchors which were not capable of resisting the overturning forces.

C.1-5 Industrial Facilities

Throughout the area the performance of industrial buildings ranged from complete collapse, such as the Pepsi-Cola Building, to structures with no damage. such as the Esso and Siemens industrial buildings. The degree of damage to the buildings related directly to the quality of design and construction, the distance from the fault, and the ground accelerations (EERI, 1981a).

Esso Refinery - The Esso oil refinery is located on the east side of Lake Asososca. Two seismoscopes and an AR240 strong motion seismograph located at this refinery provided the only records of the Managua, Nicaragua, earthquake. The record from the AR240 seismograph indicated a 30% to 40% ground acceleration both horizontally and vertically (EERI, 1981a).

The plant was built in two stages during the mid-1950s and early 1960s and was designed to meet UBC (Uniform Building Code) requirements. All detailing reflected the latest U.S. design procedures. At that time no specific provisions were added for existing seismic hazards. All equipment was tied to its foundations, piping systems were braced, etc. Some difficulties arose after a 1968 earthquake; consequently, the plant was apparently redesigned to withstand 20 g.

Damage at the refinery was minimal. At the time of the shock, half of the facility was shut down for maintenance. Damage to administration and equipment facilities was not significant and operations were resumed within 24 hours. Many grout pads at the supports of vertical steel vessels were spalled. Some piping in the low ground-level pipeway trenches jumped from saddle supports. Piping on the second floor of the concrete pipeway structure and floor drains for a heat exchanger shifted (EERI, 1981g).

Fabritex Textile Mill Complex - This facility is composed of several large industrial buildings of various sizes and construction types. None of the buildings suffered serious damage. Acoustical tiles fell, creating problems in putting the equipment back on line. Whole and broken tiles showered on equipment, falling inside intricate machinery. Inadequately braced machines were thrown out of alignment. The machines themselves were unharmed, but many bobbins and spools fell to the floor and were damaged. Some equipment displacements (sliding) caused pipe breaks throughout the system (EERI, 1981g).

Tanic Cigarette Factory - This factory is located 3 miles east of Managua and was less than 5 years old at the time of the earthquake. Construction is a heavy reinforced-concrete frame with fragile curtain walls of hollow clay tile blocks. Some shear cracking occurred in the walls, but the cracks usually did not penetrate the concrete frames and damage was minimal. Most equipment in the factory was light, low-profile, unanchored equipment. Movements of 3 to 4 inches at the bases of the equipment were common. No severe mechanical damage was incurred (EERI, 1981g).

Pepsi-Cola Bottling Plant - This plant suffered extensive structural and equipment damage and was inoperative for more than 1 month. The long shutdown was caused primarily by the failure of the reinforced concrete building which housed bottling and production equipment. Anchored equipment not damaged by the falling debris generally survived without significant damage (EERI, 1981g).

C.1-6 Water Supply System

Potable water is pumped up several hundred feet from Lake Asosoca, a caldera located about 3 miles southwest of downtown Managua, by five 500-hp vertical turbine pumps submerged several meters into the water (EERI, 1981g). The main pumping plant is located at the lake level and pumps water up the steep incline of the caldera in two steel pipes. Some earth slides occurred on the steep slopes and partially blocked an access road but did not damage the pipes or pumps. Anchor bolts holding down the surge tank were elongated as a result of the earthquake. The roof of the materials warehouse at the pumping plant collapsed (EERI, May 1973).

The 2500-kVA transformers of from 13,200 to 24,000 V, which feed the principal pumping station and the transformers of the booster stations, suffered damage in the secondary insulators. The GE control boards of the main pumping station suffered misalignments impeding the starting of the equipment and were repaired provisionally by means of flexible jumpers (EERI, 1981b).

C.1-7 Enaluf Office Building

In the penthouse, equipment that was sitting on the floor but not connected to the structure was displaced. The air conditioning unit slipped from its isolation pads causing the base to translate and rotate relative to its floor support and causing the metal cabinet to move relative to the base. An electric motor fell from its support, but the switch racks to which it was connected were not displaced. Some pipes failed. Roof acceleration is estimated at 1.16 g. Overall building performance was excellent. Nonstructural damage was minimal and structural damage was isolated to floor diaphragm cracks through the weakest part of the floor system. Damage to equipment in the penthouse could have been reduced by appropriate connection of the equipment to the structure (EERI, 1981d).

C.2 Friuli, Italy, Earthquake of May 6, 1976

C.2-1 Earthquake Data

Magnitude: Richter Scale 6.5 for the May 6 shock and 6.0 for the two aftershocks of September 15.

Time: Main shock on May 5, 1976 at 9:00 p.m., local time; aftershocks on September 15, 1976 at 3:15 a.m. and 9:21 a.m., local time.

Location: Northeastern Italy

C.2-2 Ground Motion Records

The ground motions from the main shock and the aftershocks were recorded by a number of accelerograph stations. A maximum peak ground acceleration of 0.37 g was recorded at Forgaria from the May 6 event.

C.2-3 General Effects of the Earthquake

The event was centered in an area of high density of towns and villages. 1,000 deaths and 5,000 injuries were reported. Most construction was old (approximately 100 years); however, there were some new industrial and residential complexes in the area. In total, 42,000 structures were destroyed. The preponderance of damage was in residential areas and to older homes.

C.2-4 Electric Power Facilities

There are a number of steam-generating stations and hydroelectric power plants in the region owned by ENEL. All generating stations in the region tripped.

The same happened to the interconnecting and distribution transformers. Following is a description of the effects of the earthquake on these facilities.

Somplago Plant - This is a 180-mVA hydroelectric power plant. The plant buildings suffered some damage in the form of cracks in the roof of the switchboard room, the workshop, the dining hall, and the storehouse; there was also damage from landslide action.

Electrical switchyard equipment was severely damaged: 18 out of the 21 single-pole oil circuit breakers came down; the same happened to 209 insulator elements out of a total of 580 because of porcelain cracking. There were also breaks in the contact sections of the disconnecting devices, in the pneumatic operating mechanisms, and associated pressure lines. The busses were overstressed at the joints.

Compagnola Hydroelectric Plant - The main damage was incurred by the brickwork and by the hydraulic structures with splits and displacements along the head race, in the wicket gates of the overtaking duct, and in the control building. A set of batteries fell off its stand. The upsetting and displacement of transformers was also noted.

Pireda Plant - There were breaks along the wall of the bypass canal and in the control building. There was minor equipment damage.

Campolessi Plant - There was some building damage. Damaged batteries were also reported.

C.2-5 Power Distribution Systems

In the S. Daniele and Buia primary cabins the high voltage transformers weighing 70 to 100 tons were displaced and derailed; in other substations the destruction of insulating elements and the overturning of the battery racks was almost complete. The distribution cabins suffered substantial damage.

C.3 Miyagi-Ken-Oki, Japan, Earthquake of June 12, 1978

C.3-1 Earthquake Data

Magnitude: Richter Scale 7.4
Time: June 12, 1979, at 17h 14m Japanese Standard Time (8h 14m GMT)
Location: 38 degrees 09 minutes N latitude
142 degrees 13 minutes E longitude
Focal Depth: 30 km

C.3-2 Ground Motion Records

Many strong motion instruments recorded the event. The maximum recorded peak ground acceleration was about 0.4 g at Sendai Kokuketsu Building (NRC, June 1983). Intensities 4-5 on the Japan Meteorological Agency scale or 7-8 in MMI occurred in worst hit areas (T. Okubo and O. Masamitsu, cited in EQE, November 1983a).

C.3-3 General Effects of the Earthquake

There were 28 deaths and 11,028 injuries (almost all occurred in Miyagi Prefecture) as a result of the seismic event. Sendai, a modern city of 615,000 people, suffered surprisingly small damage. Most of the damage seemed to correlate with poor local geologic and soil conditions (EERI, December 1978).

C.3-4 Electric Power Facilities

Sendai, a large industrial city, had more than 6500 business and manufacturing firms at the time of the earthquake; the facilities investigated represent only a small sample of the structures that were damaged by the earthquake. The degree of damage observed ranged from negligible (at the Fukushima Nuclear Power Plant) to severe (at the Sendai Gas Facility).

Electric power system damage to utilities was concentrated in Miyagi Prefecture. Before the earthquake, the Tohoku Electric Power Co. was delivering 4900 mW to the northern portion of Honshu Island. There was approximately a 1,500-mW decrease in demand after the earthquake, including the interruption of some 1,130 mW of supply. System frequency momentarily fluctuated from 50.00 Hz to 50.58 Hz, then returned to normal in 5 minutes. Power service of an estimated 681,600 customers was affected by seismic damage to power system facilities and by operation of relays triggered by the earthquake. These relays were reported to have normally operated and protected the equipment from electrical faults in the system before any equipment was structurally damaged.

Fukushima Nuclear Power Plant Complex - The site is on the Pacific coast, approximately 140 km from the epicenter. Faulting may have extended 60 km west of the epicenter, in which case the plant site may be located about 80 km from the nearest source of energy.

The complex has six nuclear units for a total of 4,700 mW and is the largest nuclear power complex in the world. Units 1 and 6 were instrumented with between 20 and 30 strong motion accelerometers and much valuable information was obtained from the earthquake. The recorded peak ground accleration, which could be considered to be a "free field" acceleration, was 0.125 g. The corresponding accelerations in the north/south direction and up/down directions were 0.100 g and 0.050 g. The strong motion exceeded 30 seconds in duration. The records were obtained from instruments located on the base slabs of the two units and at downhole instruments, about 30 to 40 m below two of the containments. The reported maximum response accelerations in the buildings were about 0.50 g.

At the time of the visit by a U.S. reconnaissance team (June 23, 1978--11 days after the earthquake), Units 1, 2, 3, and 6 were operating; Unit 5 was still under construction but was essentially completed, and it is believed that Unit 4 was scheduled to go into commercial production soon (EERI, December 1978). The plants are founded on a competent soft mudstone formation with a thickness in excess of 300 m. Unit 1 was designed for a peak ground acceleration of 0.18 g and a response spectrum based on Taft record from the southern California (Kern County) earthquake of 1952.

The reconnaissance team inspected the exterior of Unit 1 and the exterior and interior of Unit 6, including the containment structure, the reactor vessel pedestal, some of the equipment on the refueling floor, some of the equipment in the reactor building, the underside of the control rod drive in the containment, miscellaneous critical and non-critical piping, various critical and non-critical cable trays, the reactor building, the turbine building, the overhead crane, and various auxiliary buildings, the turbines and tanks. There was no damage or evidence of working of connections in any of the inspected areas. The only reported damage to the complex was to some non-critical electrical insulators (EERI, December 1978, Figure 57) some distance to the west of Units 1 and 2.

New Sendai Power Plant, Tohoku Electric Power Co. - This plant is located on the Pacific coast and has two Mitsubishi oil-fired boilers. Unit 1 was completed in 1971 and has a generating capacity of 350 mW; the 600-mW Unit 2 was completed in 1973 and was the largest of the company's units. The plant's seismic alarm located at the level of the turbine operating floor was triggered at approximately 0.15 g. Because the plant is closer to the epicenter of the earthquake and the assumed area of faulting, it may be assumed that the ground motion was somewhat stronger at the plant than at the city of Sendai, where the recorded peak ground accelerations varied between 0.2 and 0.4 g. The plant is located in an area of recent alluvium and on filled land; the depth of unconsolidated sand is approximately 15 m.

Both units were damaged and the plant was shut down for 6 days. Three types of damage occurred at the plant: (1) damage from local, minor settlement, (2) damage to the structural and architectural elements of buildings which was minor, and (3) damage to the equipment, which constituted the bulk of the loss.

In Units 1 and 2 tubing inside the boilers was damaged. A small furnace platen cooler tube inside the slag screen was sheared in the Unit 1 boiler. A similar failure occurred in the boiler of Unit 2 to one of the reheater spacer tubes. The suspended boilers and their structural supports also pounded against one another and sustained some damage. There was no other reported significant damage. The turbine pedestal and operating floor of the turbine buildings in Japan are usually separated by a 3- to 4-in. gap, and, in this case, there was no pounding between the two structures.

C.3-5 Electrical Substations

A total of 18 substations sustained equipment damage to varying degrees, including two 275-kV, seven 154-kV, and nine 66-kV or lower voltage substations. The primary cause of extensive power outages in the Sendai area was severe damage to electrical equipment at two of the key bulk power substations, including Sendai Substation. Most of the damage to equipment at these substations was associated with failures of porcelain components.

Sendai Substations, Izumi - This is a multilevel facility built on a site with extensive cut and fill work. Yard equipment in all parts of the facility was extensively damaged. Most damage occurred to various ceramic insulators, lightning arrestors, cicuit breakers, and transformers.

C.3-6 Industrial Facilities

Haranomachi Plant of Sendai City Gas Bureau, Sendai - This facility suffered major damage. The total collapse of a large propane gas holder was primarily responsible for the stoppage of gas service for the city. The collapsed tank caught fire shortly after failure, and all the stored gas was consumed. The fire was extinguished about 25 minutes later. The collapsing tank struck nearby pipeways and other piping systems and equipment, causing much additional damage to the facility. There was evidence of other kinds of damage throughout the facility; however, none of the other tanks at the facility are believed to have suffered major damage.

Sendai Refinery, Tohoku Oil Co., Ltd. - This facility suffered extensive damage from tank ruptures and massive oil spills on the site.

C.3-7 Water Supply System

Sendai City bureau of water supply provides potable water to some 200,000 customers from 3 treatment facilities, having a maximum daily capacity of 320,000 cubic meters. Facilities for collection, storage, transmission, and treatment work survived the earthquake without any substantial damage. Power required at treatment facilities was obtained from emergency power units and power outages did not affect service to customers.

C.3-8 Sewer System

The sewer system of Sendai serves approximately 60% of the city's population. The system has 11 main pumping stations where sewage is boosted to a single treatment plant. Although various types of damage were inflicted upon the sewerage system, the single most important seismic effect was the disablement of several pumping stations caused by power outages.

C.4 Campania-Basilicata, Italy, Earthquake of November 23, 1980

C.4-1 Earthquake Data

Magnitude: Richter Scale 6.8
Time: November 23, 1980, at 19h 34m local time (18h 34m GMT)

Location: 40 degrees 46 minutes latitude
15 degrees 18 minutes longitude
100 km east of Naples
Depth: 10 km

C.4-2 Ground Motion Records

The earthquake triggered a number of strong motion accelerographs. There were five shocks in less than 2.5 minutes. The strongest recorded motion was 0.35 g at Sturno. The range of recorded ground motions varied from 0.1 g to 0.35 g. The first shock of the five was the largest. The total duration of the five shocks (acceleration greater than 0.05 g) was 147 seconds. A peak ground acceleration of between 0.6 g and 0.7 g was estimated at the epicenter of the event (EERI, July 1981).

C.4-3 General Effects of the Earthquake

The earthquake killed approximately 3,000 people and injured about 9,000. The damaged area covered more than 10,000 square kilometers. Damage to housing was severe because of the multiple strong shocks and the lack of seismic resistance for the structures. Much of the damge to lifeline facilities was caused by building failures and the movement of building debris down slopes in the mountain villages.

C.4-4 Electric Power Facilities

Most power outages (caused when insulators and conductors broke in the epicentral region) were caused by buildings falling on distribution lines. Two hydroelectric power plants, Tanagro Hydrostation with 27-km epicentral distance and Agri Generating Plant at 100-km epicentral distance, suffered no damage or interruptions. At Calore Generation Station (43-km epicentral distance) lightning arrestors were damaged and conductors were broken.

The Garigliano Nuclear Power Plant located at Sessa Aurunca (125-km epicentral distance) felt the earthquake. The plant is a 150 net MWe General Electric BWR completed in 1962 and is similar to the Dresden 1 (U.S.) plant. Although the plant was in a shutdown condition, the control rod scram system, set at 0.05 g, was actuated by a 0.051 g signal from the vectorial sum seismic device. This plant was not damaged.

The earthquake was also felt at the Latina Nuclear Power Plant, a 150 net MWe, graphite-moderated, gas-cooled reactor unit completed in 1962, located 217 km away from the epicenter. This plant was also in a shutdown condition for maintenance. However, the safety system was actuated by spurious signals below the set value of 0.03 g, causing the insertion of the (safety) control rods. No evidence of damage or malfunction was found at the plant.

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The margin of safety provided in existing nuclear power plant equipment to resist seismically induced loads and perform their intended safety functions may vary considerably, because of significant changes in design criteria and methods for the seismic qualification of equipment over the years. Therefore, the seismic qualification of equipment in operating plants should be reassessed to determine whether requalification is necessary.

The objective of technical studies performed under the Task Action Plan A-46 was to establish an explicit set of guidelines and acceptance criteria to judge the adequacy of equipment under seismic loading at all operating plants, in lieu of requiring qualification to the current criteria that are applied to new plants.

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Regulatory Analysis for Proposed Resolution of Unresolved Safety Issue A-46 Seismic Qualification of Equipment in Operating Plants

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

August 1985

Note: Attachment to NUREG-1030

REGULATORY ANALYSIS FOR PROPOSED RESOLUTION OF UNRESOLVED SAFETY ISSUE A-46 SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS

I. STATEMENT OF THE PROBLEM

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the history of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform their intended safety functions may vary considerably. The seismic capability of equipment in operating plants therefore must be reassessed to assure the capability to bring the plant to a safe shutdown condition when subjected to a seismic event.

The need for such a reassessment was identified as a result of experience with the Systematic Evaluation Program (SEP) for eleven older operating plants and the staff's Seismic Qualification Review Team (SQRT) reviews on operating license applications. During the course of the SEP and SQRT reviews, the staff identified a concern with the anchoring and supports for electrical equipment in the SEP plants. An information notice concerning this issue was sent to all other operating plants. (IE Information Notice 80-21, "Anchorage and Support of Safety-Related Electrical Equipment," dated May 16, 1980.)

The USI A-46 program did investigate the adequacy of seismic qualification methods used for electrical and mechanical equipment installed in older nuclear plants, and determined that it is necessary to develop proposed requirements which could be implemented in a practical cost beneficial way to assure that equipment in older plants can adequately withstand a seismic event and ensure the capability to safely shut down the plant.

II. OBJECTIVES

The proposed regulatory requirement is needed to verify the seismic adequacy of mechanical and electrical equipment which is required to safely bring the reactor and plant to a safe shutdown condition and to maintain it in a safe condition. The specific objective of the A-46 task was to develop viable, cost effective alternatives to current seismic qualification licensing requirements to be applied to operating nuclear power plants.

III. SUMMARY OF A-46 TASKS

A-46 tasks included investigation of several alternative procedures for assuring seismic adequacy of equipment needed to cope with a seismic event. Some of the alternatives studied did not contribute significantly to the proposed resolution. Each of the tasks are described in the A-46 technical findings report, NUREG-1030 and in the references cited in that NUREG. Tasks included in the A-46 program were as follows:

1. Identification of Seismic Sensitive Systems and Equipment

The objective of this task was to develop possible methods of generating a generic minimum equipment list. If a methodology could be developed to evaluate the risk importance of safety systems and equipment then equipment could be ordered by contribution to risk.

2. Assessment of Adequacy of Existing Seismic Qualification Methods

This task involved a study to evaluate past and present methods to qualify mechanical and electrical equipment. The intent was to determine if older qualification procedures could be shown to provide adequate assurance of seismic adequacy.

3. Development and Assessment of In-Situ Test Procedures to Assist in Qualification of Equipment

This task was intended to develop guidelines for obtaining and using dynamic characteristics of equipment to assist in verifying seismic adequacy.

4. Seismic Qualification of Equipment Using Seismic Experience Data

This task was based on the experience data collected by the Seismic Qualification Utilities Group (SQUG) and recommendations made by the Senior Seismic Review Advisory Panel (SSRAP).

5. Development of Methods to Generate Generic Floor Response Spectra

This task led to development of, and guidelines for using, generic floor response spectra. These generic spectra can be used in lieu of calculating response spectra for use in determining seismic adequacy.

Task 4, the use of seismic experience data, proved to be the most reasonable alternative for verifying seismic adequacy. The other four tasks either play supporting roles or would be used to a limited extent if the seismic experience data base does not pertain to a particular item. The A-46 implementation plan presented in the following paragraphs therefore is based primarily on work completed in Task 4.

IV. PROPOSED IMPLEMENTATION PROCEDURE

Based on results of the A-46 tasks summarized above, an implementation plan was developed. Each licensee of an operating plant which has not been previously reviewed to current licensing criteria would be required to perform a seismic verification review and report the results. The verification review procedure is outlined below.

1. Plants Affected

The current requirements for qualification of equipment in licensing plants are defined in Regulatory Guide 1.100, IEEE Standard 344/1975 and Standard

Review Plan 3.10. The importance of equipment support to the qualification of equipment is recognized in current requirements, as evidenced by the following statement: "The equipment to be tested shall be mounted on the vibration generator in a manner that simulates the intended service mounting. The mounting method shall be the same as that recommended for actual service." The staff believes that plants reviewed to current requirements and with the implementation audited by the Seismic Qualification Review Team (SQRT) as is presently done have been confirmed to have an adequate level of protection for SSE level seismic events.

All plants not reviewed to these current equipment qualification requirements as documented by plant Safety Evaluation Reports (SER's), are included in the A-46 review. For plants reviewed under the Systematic Evaluation Program (SEP), structural integrity of equipment has already been covered, therefore these SEP plants will be reviewed for functional capability only. A list of plants affected is included as an enclosure.

For replacement of equipment and/or parts in plants subject to A-46 requirements, future replacements must be verified for seismic adequacy either by using A-46 criteria and methods or as an option, qualification by current licensing criteria.

2. Scope of Seismic Adequacy Review

Each licensee will be required to determine the systems, subsystems, components, and instrumentation and controls needed during and following a safe shutdown earthquake event using the following assumptions.

(1) The seismic event does not cause a LOCA and a LOCA does not occur simultaneously with or during a seismic event.

(2) Offsite power will be lost during or following a seismic event; and

(3) Plant must be capable of being brought to a safe shutdown condition following a design basis seismic event.

The equipment to be included in this implementation plan is limited to active mechanical and electrical components. Piping, tanks and heat exchangers are not included except that those tanks and heat exchangers that are required to achieve and maintain safe shutdown must be reviewed for adequate anchorage. Lessons learned from studies of nuclear and nonnuclear facilities under earthquake loading indicate that the effect of failure of certain items, such as suspended ceilings, and light fixtures could influence the operability of equipment within the scope of review. This concern is addressed in USI A-17, "Systems Interaction," and is therefore not further considered in implementing A-46. The failure of masonry walls that could affect the operability of nearby safety-related equipment is also of concern. However, this concern has been addressed by IE Bulletin 80-11, which requires that all such masonry walls be identified and re-evaluated to confirm their design adequacy under postulated loads and load combinations. This concern is therefore not considered as part of A-46 implementation.

For some pressurized water reactor plants, the seismic adequacy of Auxiliary Feed Water Systems (AFW) has been verified by licensee actions taken in response to generic letter 81-14 dated February 10, 1981. Review of the AFW may be deleted from consideration under A-46 if staff acceptance has been documented in an SER, or if the licensee has committed to meet the requirements of the generic letter.

For the purpose of this implementation plan, safe shutdown means bringing the plant to a hot shutdown condition and maintaining it there for a minimum of 72 hours. The 72 hour time period is sufficient for inspection of equipment and minor repairs if necessary following an SSE or to provide additional source(s) of water for decay heat removal if needed to extend the time at hot shutdown. Equipment required includes that necessary to maintain required supporting functions for safe shutdown. For all equipment within the defined scope, the verification should closely follow the procedure outlined in paragraph 4 below.

Studies are currently being done as part of USI A-45 "Decay Heat Removal Requirements" to review the risk associated with shutdown and decay heat removal systems. Part of the A-45 study involves a study to determine the risk associated with cold shutdown including seismic risk. This is a probabilistic risk assessment study and as such includes consideration of seismic hazard well above the SSE level (up to 5 times the SSE). Seven plant-specific PRA studies will be conducted under A-45. For each of these studies, plant-specific equipment fragilities are being generated from plant inspections of the equipment. These plant reviews are specifically looking for anchorage deficiencies and off normal equipment configurations. Concerns regarding seismic qualification of cold shutdown equipment are best addressed under USI A-45. If further A-45 studies show that there is an important reduction in core melt probability if equipment required to reach cold shutdown is seismically qualified to the SSE level, the implementation of these results will be made separately under USI A-45.

Accident mitigating systems were not included within the scope for two reasons:

(1) Experience data collected by SQUG and others, and high level seismic tests on piping conducted in foreign countries and in the USA show that piping is not susceptible to failure due to seismic inertia loads. The only observed instances of piping failure during the SQUG program to collect seismic experience data was due to relative movement of anchor points and inadequate or nonexistent anchorage of tanks or equipment for sites with zero period acceleration between 0.25g and 0.6g.

    In general, piping is found to have a high margin of safety for almost all the piping if only seismically induced inertia loads are considered. High stresses arise where piping runs through walls, or is attached to a large vessel resulting in relative displacements. In piping design, seismic stresses are usually held to a small percentage (say 15%) of the overall allowable stress. In addition, Seismic risk studies completed to date show that piping is not predicted to fail even at levels 2 to 5 times the SSE level.

Furthermore, IE Bulletin 79-02 requires review of as-built pipe support base plate designs using concrete expansion anchor bolts. IE Bulletin 79-07 requires review of the proper combination of the intramodal responses due to the spatial components of a multidimentional earthquake, and the verification of piping system computer codes. IE Bulletin 79-14 requires the confirmation of "as built" configuration of safety-related piping systems to their design/analysis configuration. The piping systems, including their restraints, were reviewed to the requirements of these IE Bulletins and all operating plants either met these requirements or were modified to meet these requirements.

(2) Seismic experience data collected by SQUG and reviewed by SSRAP, supplemented by reviews and literature surveys of strong motion earthquakes indicate that mechanical and electrical equipment of types commonly used in nuclear power plants are unlikely to fail at earthquake levels typical of SSEs at U. S. plants east of California. There is strong evidence that accident mitigating systems would function as designed in the unlikely event they are required following a SSE. In almost all cases where equipment damage has occurred it was due to failure of the anchorage or to displacement of unanchored equipment. It was also observed that some equipment with minimal anchorage did not move even though it was subjected to accelerations as high as 0.5g.

3. Requirements for Plant Shutdown

The time a plant can remain at hot shutdown after a seismic event without restoring offsite power is plant-specific. Each licensee must show practical means of staying at hot shutdown for a minimum of 72 hours. In the event that maintaining safe shutdown is dependent on a single (not redundant) component whose failure, either due to seismic loads or random failure, would preclude decay heat removal by the identified means, the licensee should show that at least one practical alternative for achieving and maintaining safe shutdown exists which is not dependent on that component.

The equipment to be considered depends on the functions required to be performed. Typical plant functions would include:

(1) bring the plant to hot shutdown and establish heat removal;

(2) maintain support systems necessary to establish and maintain hot shutdown;

(3) maintain control room functions and instrumentation and controls necessary to monitor hot shutdown;

(4) provide alternating current and direct current emergency power.

4. General Verification Procedure for Plant-Specific Review (refer to SSRAP Report*)

The general verification procedure for plant-specific review is described below. Figure 1 outlines this procedure. It should be noted that this figure depicts the implementation steps for Generic Resolution (see paragraph 6 below). The results of the SQUG (Generic Group) and EPRI/RES study will be accessible to all utilities, therefore with some differences in the areas of staff review/audit and utility reporting procedures (see paragraphs 6 and 7 below), Figure 1 generally applies to utilities who are not participating members of the Generic Group. The implementation should include: development of an equipment list; comparison of site spectra with appropriate bounding spectra; walk-through inspection including review of anchorages, review of equipment functional capability and review of equipment unique to nuclear plants.

<u>DEVELOPMENT OF EQUIPMENT LIST</u>

Each licensee will be required to develop an equipment list that includes all equipment items identified as necessary to perform functions related to plant hot shutdown (see Section 2, "Scope of Seismic Adequacy Review" above).

*SSRAP Report, "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants," January 1985. During implementation it might be necessary to modify the SSRAP recommendations on a plant-specific basis.

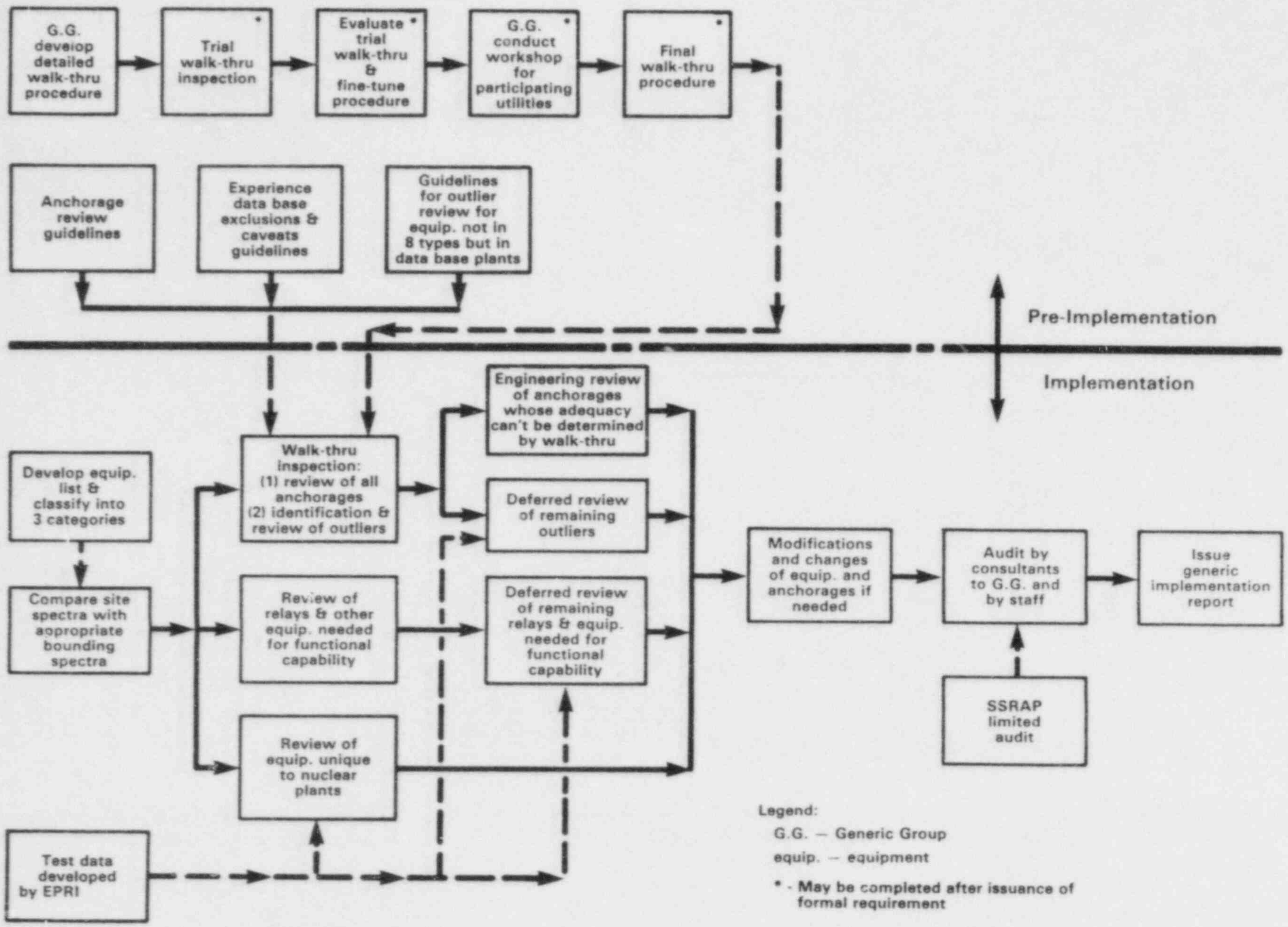

Figure 1 USI A-46 Implementation

Table 1
Typical Equipment List for USI A-46

1. Mechanical Equipment

    1. Vertical pumps and motors*
    2. Horizontal pumps and motors*
    3. Motor-operated valves*
    4. Air-operated valves* (including solenoid valves)
    5. Heating, ventilation and Air Conditioning (HVAC such as fans, blowers, chillers, filters, etc.)
    6. Pumps (turbine driven, diesel driven, and reciprocating positive displacement type)
    7. MSIVs (Main Steam Isolation Valves)
    8. Pilot-operated safety/relief valves
    9. Spring-operated safety/relief valves
    10. NSSS mechanical equipment (Control Rod Drive Mechanisms)
    11. PORVs (Power Operating Relief Valves)
    12. Air compressors and air accumulators
    13. Heat exchangers, tanks (anchorage review only)
    14. Atmospheric steam dump valves

2. Electrical Equipment

    1. Low voltage switchgear*
    2. Metal clad switchgear*
    3. MCCs* (Motor Control Centers)
    4. Transformers* (unit substation type)
    5. Motor-generator sets
    6. Distribution panels - AC and DC
    7. Batteries and battery racks
    8. Battery chargers
    9. Inverters
    10. Diesel generators and associated equipment
    11. Electrical penetration assemblies
    12. Transformers (other than unit substations)
    13. Automatic transfer switches
    14. Remote shutdown panels

3. Instrumentation

    1. Transmitters (pressure, temperature, level, flow)
    2. Switches (pressure, temperature, level, flow)
    3. Resistance temperature detectors and thermal couples (RTDs and T/Cs)
    4. Relays
    5. Control panels and associated components
    6. Instrument racks and associated components
    7. Instrument readouts (displays, indicators such as meters, recorders, etc.)
    8. Neutron detectors

NOTE: * The eight equipment types already included in the SQUG pilot program. (Seismic Experience Data Base)

A list of typical equipment required for plant hot shutdown is shown in Table 1. This list will be classified into three categories of equipment. These categories are based on the extent to which the experience data applies to them. They are defined as follows:

(1) equipment belonging to the eight types in the seismic experience data base (see Table 1);
(2) equipment of type not included in the eight types in the seismic experience data base but which are present in the data base plants.
(3) equipment unique to nuclear plants (i.e., no seismic experience available).

COMPARISON OF SITE SPECTRA WITH APPROPRIATE BOUNDING SPECTRA

The licensee will verify that the appropriate data base bounding spectra envelope the site free field spectra at ground surface defined for the plant. He will identify all equipment on his equipment list which is located at an elevation higher than forty feet above grade level. For equipment above forty feet, one and one half times the appropriate data base bounding spectra must envelope the floor response spectra for the equipment. For those cases where floor response spectra are needed, NUREG/CR-3266 entitled, "Seismic and Dynamic Qualification of Safety Related Equipment in Operating Nuclear Power Plants-Development of a Method to Generate Generic Floor Response Spectra" may be used as one alternative to develop the necessary floor response spectra on a case specific basis. The appropriate bounding spectra for equipment belonging to the eight types in the data base are defined in the SSRAP Report. For equipment not included in the eight types in the data base but which exist in the data base plants, and for equipment unique to nuclear plants, appropriate generic bounding spectra are under development. These generic bounding spectra will not exceed the type A bounding spectra defined for the data base plants unless test data collected by EPRI/RES verifies significantly higher seismic capacity. For equipment outside the data base for which the type A bounding spectrum is used, any caveats or exclusions developed prior to the implementation of USI A-46 by the SQUG/SSRAP and the NRC must be taken into account. The bounding spectra will be developed by SQUG, endorsed by SSRAP and reviewed and approved by the Staff prior to implementation.

WALK-THROUGH INSPECTION

Each licensee will be required to conduct a plant walk-through and visual inspection of all equipment items in the equipment scope. The inspection team should consist as a minimum of:

(1) plant operations supervisor or a licensed Senior Reactor Operator;
(2) an experienced structural engineer familiar with seismic anchorage requirements;
(3) an experienced mechanical engineer familiar with plant mechanical equipment; and
(4) an experienced electrical engineer familiar with plant electrical equipment.

As an alternative, licensees may use consultants instead of their staff for (2), (3), and (4) above.

The walk-through inspection should cover anchorage review, and identification of potential deficiencies and outliers.

(1) Anchorage Review
For all equipment within scope, verify equipment anchorage (including required tanks and heat exchangers) using guidance provided in paragraph 5 below.

(2) Identification and Review of "Deficiencies" and "Outliers"
Deficiency in this context means equipment, components, and their anchorages/supports which are identified to be inadequate by the A-46 criteria during plant-specific walk-through reviews. Outlier in this context means equipment items that are subject to the caveats and exclusions defined in the proposed generic letter, or are otherwise not

covered by the experience data. Potential deficiencies should be identified for all equipment within the scope. The treatment of deficiencies is further described in paragraph 6 below.

For equipment belonging to the eight types in the seismic data base, identify data base exclusions and caveats from guidance provided in paragraph 5 of the enclosure to the proposed generic letter.

For equipment which exists in data base plants but does not belong to the eight types, collection of additional seismic experience data is not required, however the basis for seismic adequacy must be documented for each equipment type. Guidelines provided in paragraph 6 of the enclosure to the proposed generic letter should be used for identification and review of "outliers" during the walk-through inspection of this equipment. These outlier review guidelines are based on experience accumulated by licensees and staff. (SEP and Seismic Qualification Review Team (SQRT) reviews).

All identified outliers can be deferred for implementation for a time period not to exceed 28 months from the date of issuance of the USI A-46 final resolution, at which time additional test experience data will be available from the EPRI/RES program.

The EPRI/RES test data collection program is described below. This program was initiated by EPRI (Electric Power Research Institute) in 1984 and RES (Office of Nuclear Regulatory Research of NRC) in 1985. RES and its contractors will collect and compile existing seismic fragility data on nuclear plant equipment. These test data will be used to improve the currently available information on equipment fragility and seismic margins. EPRI and its contractor will collect and compile existing seismic test data on nuclear plant equipment for A-46. The main objectives of the EPRI programs are: (1) to establish Generic Equipment Ruggedness Spectra (GERS), which can be used to demonstrate seismic adequacy of equipment, and (2) to demonstrate functional capability of equipment or components (e.g., relays) which are required to function during the strong motion part of an SSE.

Even though the objectives of RES and EPRI programs are different, the data collection process and the source organizations for the data are mostly the same. The cooperative program calls for exchanging collected data and coordinating collection activities by both organizations in order to minimize cost, prevent duplication, and maximize the use of available data sources. It has been agreed that EPRI will primarily collect data from utilities and west coast testing laboratories, and RES will primarily collect data from vendors and east coast testing laboratories.

REVIEW OF EQUIPMENT FUNCTIONAL CAPABILITY

There may be equipment or components which are required to function during the strong ground motion part of an SSE. In these cases, functional capability of the equipment or components must be established. Several options are acceptable: Comparison with test experience data base being developed by EPRI, qualify by test in a manner consistent with current licensing requirements (i.e., SRP 3.10, Reg. Guide 1.100/IEEE Standard 344-1975), or provide other evidence of functional capability.

Of particular concern are electromechanical devices such as relays, switches and contactors. Mercury switches are known to malfunction during testing and should be replaced by other types of qualified switches.

The review of electrical relays should follow the guidelines outlined below.

(1) All relays associated with the functioning of equipment necessary to bring the plant to a hot shutdown condition must be identified. Those relays which must function during the first 30 seconds of an SSE must be qualified by test, verified by comparison with the test data base being developed by EPRI/RES or replaced by relays qualified to current licensing requirements (i.e., SRP 3.10, Regulatory Guide 1.100/ IEEE Standard 344-1975).

(2) All relays, which could potentially change state during an SSE due to contact chatter and preclude the use of equipment needed for shutdown after the SSE, must be identified. Consideration must be given to all potential operational states of the relay (i.e., energized, de-energized, tripped or non-tripped) with respect to the availability of the equipment they control following an earthquake. These relays must also be qualified by test, by comparison with test data (comparison of spectra at mounting location) or replaced by relays qualified to current licensing requirements. As an alternative, the licensee may show that chattering or change of state of the relays does not preclude subsequent equipment or system functions.

(3) Seismic verification of relays may be deferred until the EPRI test data base is fully developed, provided that the seismic verification be completed no later than 28 months from the date of issuance of the USI A-46 final resolution.

REVIEW OF EQUIPMENT UNIQUE TO NUCLEAR PLANTS

For equipment unique to nuclear plants such as control rod drive mechanisms, power operated relief valves and main steam isolation valves, etc., test experience data base being developed by EPRI/RES or qualification records for similar items may be used to verify seismic adequacy.

REPLACEMENT PARTS

Component in this context means equipment and assemblies such as pumps and motor control centers, and sub-assemblies and devices such as motors and relays which are part of assemblies. In the event that components are modified or replaced by the utility as a result of A-46 review, each modification or replacement (assembly, subassembly, device) must be verified for seismic adequacy either by using A-46 criteria and methods or as an option, qualification by current licensing criteria.

5. Verification of Anchorage

To verify acceptable seismic performance, adequate engineered anchorage must be provided. There are numerous examples of equipment sliding or overturning under seismic loading due to lack of anchorage or inadequate anchorage. Inadequate anchorage can include short, loose or poorly installed bolts or expansion anchors, inadequate torque on bolts, and improper welding or bending of sheet metal frames at anchors. Torque on bolts can normally be ensured by a preventive maintenance and inspection program.

In general, checking of equipment anchorages requires one to estimate the equipment weight and its approximate center of gravity. Also, one will have to either estimate the equipment fundamental frequency so as to obtain the spectral acceleration at this frequency or else use the highest spectral acceleration for all frequencies. When horizontal floor spectra exist, these spectra may be used to obtain the equipment spectral acceleration.

Alternatively, for equipment mounted less than about 40 feet above grade, 1.5 times the free-field horizontal design ground spectrum may be used to conservatively estimate the equipment spectral acceleration. For equipment mounted more than about 40 feet above grade, floor spectra must be used.

Equipment anchorage must be not only strong enough to resist the anticipated forces but also be stiff enough to prevent excessive movement of the equipment and potential resonant response with the supporting structure. Review of anchorages should include consideration of both strength and stiffness of the anchorage and its component parts.

Additional discussions on seismic motion bounds and equipment supports and anchorage for each of the eight classes of equipment in the experience data base is included in Paragraph 6 of the enclosure to the proposed Generic Letter.

During the walk-through inspection, anchors and supports of all equipment within the scope of review will be carefully inspected using the detailed guidance provided.[-1] If adequacy of supports and anchors cannot be determined by inspection, an engineering review of the anchorage or support will be required. This engineering review will include review of design calculations or performance of new calculations and/or verification of fundamental frequency of equipment to ensure adequate restraint and stiffness. Physical modifications may be necessary if engineering review determined the anchorage or support to be inadequate.

6. Generic Resolution

The NRC will endorse and encourage a generic resolution of USI A-46 provided the guidelines presented below are followed.

(1) All member utilities of the SQUG would be eligible to participate.

(2) The Generic Group would be responsible to (a) develop procedures to identify relays to be evaluated, (b) to define the functionality requirements and to develop evaluation procedures for these relays. This procedure should be reviewed and endorsed by SSRAP and the NRC staff.

(3) The Generic Group would assume responsibility for the implementation and would make provisions for systematic and consistent plant specific reviews. Discussions between the Staff, the SQUG and SSRAP have resulted in a tentative procedure for conducting a generic implementation. The procedure is summarized as follows:

   (a) The Generic Group would submit to the NRC a generic schedule for the implementation of the A-46 requirements within 90 days of receipt of the A-46 generic letter. The schedule should apply to all participating utilities.

-1 The detailed guidance will be developed jointly by SQUG/SSRAP, EPRI and the NRC staff and will be available prior to implementation.

(b) The Generic Group would develop a detailed walk-through procedure based on the implementation requirement defined in the generic letter.

(c) A trial walk-through inspection would be performed by Generic Group consultants with NRC participation.

(d) An evaluation of the trial walk-through would be made and the procedure fine-tuned.

(e) The Generic Group would then conduct workshops for participating utilities.

(f) Individual utilities would then perform the plant specific implementation review. This review would generally follow the guidance given in paragraphs 4 and 5 above.

(g) Each individual utility should submit to the NRC an inspection report which should include: certification of completion of review, identification of deficiencies and outliers, justification for continued operation (JCO) for identified deficiencies, modifications and replacements of equipment/anchorages (and supports) made as a result of the reviews, and the proposed schedule for required modifications and replacements not completed at the time of the report submittal.

The objective of this requirement is to provide assurance that the plant can continue to be operated without endangering the health and safety of the public during the time period required to correct the identified deficiency.

The JCO may consider arguments such as imposition of administrative controls or limiting conditions for operation (LCO) or consideration of the importance of the safety function involved and/or identification of alternate means to perform that function.

(h) Consultants to the Generic Group would perform audits of plant-specific review. All plants would be audited. The NRC staff will participate in plant audits on a selective basis. The Generic Group must submit a generic implementation review report to the NRC certifying that the walk-through inspection has been completed by the individual utility and that the audit has been completed. This report covers all participating utilities, and must be endorsed by the SSRAP. The NRC staff involved in plant audits should have appropriate background and experience. As a minimum they will participate in the Generic Group workshop.

(i) The SSRAP and the NRC staff would perform a limited review of the generic group audit process to evaluate effectiveness.

(j) Final approval of the implementation will be made by the NRC following receipt of a final report from individual utilities certifying completion of implementation reviews and equipment/anchorage modifications and replacements.

(4) The Generic Group must provide for the continuation of the SSRAP as an independent review body. The SSRAP would be consulted during development of the generic program and walk-through procedure, and implementation audit.

(5) NRC staff members would be invited to participate in all meetings between the Generic Group, their consultants, and the SSRAP.

7. Provisions for Resolution for Individual Utilities

The generic resolution described in paragraph 6 above is the method preferred by the NRC for the resolution of A-46. This paragraph offers provisions for resolution of A-46 for individual utilities not participating in the Generic Group.

Each utility is required to perform plant-specific verification reviews according to guidance in paragraphs 4 and 5. He is also required to maintain an auditable record of implementation of USI A-46.

Within 45 days of receipt of the A-46 generic letter, the utility should submit to the NRC a schedule for implementation of the A-46 requirements.

An inspection report should be submitted by the utility to the NRC following the plant-specific walk-through inspection. It should consist of the following:

(1) Certification of completion of the walk-through inspection and a description of the procedures used.
(2) List of equipment included in the review scope. Equipment required to function during the strong shaking period should be identified.
(3) Identified deficiencies.
(4) Identified outliers.
(5) Modifications and replacements of equipment/anchorages (and supports) made as a result of the inspection.
(6) Proposed schedule for future modifications and replacements.
(7) A justification for continued operation (JCO) for identified deficiencies.

Following the completion of all necessary modifications and replacements of equipment/anchorages, a final report should be submitted by the utility to the NRC. A description of the procedures used for the implementation reviews, and modifications and replacements should be included.

The NRC will review both the inspection report and the final report and will audit <u>all</u> plant-specific reviews prior to final NRC approval.

V. Value-Impact Analysis

Value impact analyses normally involve determination of the net safety benefit achieved from implementing a proposed resolution which is usually a physical change to the plant or a procedural change. The cost of implementing the proposed resolution is then estimated and the recommended implementation plan is based on the cost effectiveness considering how much risk reduction is achieved for the money spent.

1. Safety Benefit

The safety benefit of verifying the seismic adequacy of equipment in operating plants was not quantified in terms of risk reduction. PRA analyses were conducted and the relative importance of major safety systems and components determined. However, it proved impractical to quantify the results in a manner which would show the net safety benefit in terms of risk due to "qualifying" or verifying the seismic adequacy of equipment. These analyses are discussed in Section VII below.

Three factors influenced the staff judgment on safety benefit.

First, subject to certain exceptions and caveats, the staff has concluded that equipment installed in nuclear power plants is inherently rugged and not susceptible to seismic damage.

In the SQUG pilot program, the eight types of equipment (for which seismic experience data were collected to form the experience data base) are representative of mechanical and electrical equipment in both nuclear and non-nuclear plants. These eight types of equipment generally constitute, in a numerical sense, a large percentage of all safety-related equipment in a nuclear power plant. While conducting the pilot program study, the SQUG looked for equipment damage of all types of equipment due to seismic loading. This search was not restricted to the eight types considered for the data base. Of the approximately 3000 pieces of mechanical and electrical equipment surveyed in the data base plants, only one equipment item (an air operated valve) was damaged due to impact against a nearby structural girder during the earthquake. Although instances of overturned cabinets (such as switchgears and motor control centers) were found, they could all be attributed to inadequate anchorage and restraint. In most instances equipment functioned after the cabinets were made upright. SQUG therefore concluded that, subject to adequate anchorage and support, equipment found in the data base plants is inherently rugged and not susceptible to damage at the seismic levels experienced. Because of the similarity to equipment installed in nuclear plants, this conclusion was extended to nuclear power plants.

The review of seismic experience data (of eight classes of equipment) by a panel of seismic experts (SSRAP) also resulted in similar conclusions and they are:

(1) Equipment installed in nuclear power plants is generally similar to, and at least as rugged as, that installed in conventional power plants.

(2) This equipment, when properly anchored, and with some reservations has an inherent seismic ruggedness and a demonstrated capability to withstand significant seismic motion without structural damage.

(3) For this equipment, functionality after the strong shaking has ended has also been demonstrated, but the absence of relay chatter during strong shaking has not been demonstrated.

The NRC staff has closely followed the SSRAP work and is in broad agreement with their conclusions. Given that the SSRAP spectral conditions are met, the staff has concluded that it is generally unnecessary to perform explicit seismic qualification on the eight classes of equipment studied. Based on the equipment damage survey conducted in the data base plants and a broad damage survey of strong motion earthquakes around the world, the staff has further concluded that there is no need to collect additional seismic experience data on the remaining type of equipment, provided: (1) anchorage and support adequacy of equipment is assured, (2) certain caveats or exclusions for this equipment (derived from licensee, SEP and SQRT review experience) as outlined in paragraph 6 of the enclosure to proposed generic letter are addressed, (3) that the SQUG documents the basis for seismic capability of each equipment types not included in the original eight types for which detailed data were collected.

Second, although equipment is inherently rugged and not susceptible to seismic damage, failures due to seismic loads are likely to occur if equipment is not adequately supported or anchored. The need to review anchorage and supports was identified during the Systematic Evaluation Program (SEP) review. Structural adequacy of equipment, including supports and anchorage, was reviewed at each of the SEP plants. These reviews included a plant walk-through by a team of seismic experts. Results of these walk-through inspections included: identification of potential anchorage and support deficiencies such as lack of longitudinal restraints on battery racks at Millstone 1; anchor bolts overstressed on the containment spray heat exchanger and isolation condenser, need for positive anchors on switchgear panels, and need for evaluation of diesel generator anchorage at Oyster Creek; and strengthening of anchors on battery racks and need for a general engineering review of anchors at Dresden 2. As a result of the SEP experience, IE Information Notice 80-21 was issued which informed all licensees of the potential problem with anchorages. However, a recent survey of seven operating plants for the purpose of developing plant specific equipment fragilities indicated that anchorage deficiencies still exist in operating reactors.

The proposed requirement is based on the need to ensure that equipment is adequately anchored and supported, that certain equipment (primarily electrical relays) function as required during the shaking motion, and that other identified exceptions and caveats detailed in the SSRAP Report* and developed by the staff and SQUG are addressed.

The safety benefit of verifying the seismic adequacy of equipment by performing the proposed anchorage inspection procedure is principally to ensure that equipment needed to safely shutdown the plant does not fail due to failure of the anchorage or support or due to mounting configurations or geometry which make them susceptible to seismic damage. Unanchored equipment or improperly anchored equipment may overturn or move during seismic shaking. Numerous instances of overturned and displaced equipment and tanks due to improper or non-existent anchorage were found during the review of experience data base plants. This was particularly evident in the review of the Coalinga earthquake data.**

Although equipment anchorages have previously been identified as a problem area, there is evidence that anchorage deficiencies still exist in operating reactors. An inspection program to verify anchorage and supports of safety equipment would ensure that equipment failures due to seismic motion would be highly unlikely.

Third, although functional capability after the strong shaking has ended has been demonstrated by the seismic experience data, functional capability (such as absence of relay chatter) during the strong shaking motion (first 30 seconds of an earthquake) can not be demonstrated by seismic experience data. There is also some equipment that is unique to nuclear plants, for which the seismic experience data base does not apply.

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*SSRAP Report, "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants," February 1984, revised January 1985.

**See EQE report, "The Performance of Industrial Facilities and their Equipment in the Coalinga California Earthquake of May 1, 1983" dated August 1984.

Therefore, functional capability of all required equipment and the seismic adequacy of equipment which is unique to nuclear plants can be verified by test experience data. EPRI and RES (Office of Nuclear Regulatory Research, NRC) are currently conducting a program for the collection of test experience data. This program is designed to specifically support A-46 implementation.

2. Consideration of Alternatives

Several of the proposed procedures investigated as part of A-46 were determined to be not feasible, or useful only to support other methods. They are discussed in detail in NUREG-1030. As previously stated, the use of seismic experience data was determined to be the most practical way to demonstrate seismic adequacy. Only three alternative courses of action were considered:

(1) Not require any action of licensees.

This alternative was seriously considered because of the conclusion that equipment in nuclear plants is inherently rugged. The survey of seismic experience in non-nuclear facilities that had undergone significant earthquakes, indicated that if equipment were properly supported and anchored, it would be expected to survive without damage. Much of this equipment is identical or similar to nuclear plant equipment. This alternative was rejected because; (1) the SEP experience during seismic reviews showed that there were some equipment seismic deficiencies particularly with respect to anchorages; (2) there were several incidences of unanchored or improperly anchored equipment overturning or moving during seismic event in the data base plants; and (3) because of staff consideration of the recommendation of SSRAP. In addition, recent experience in nuclear plants discussed previously indicates that anchorage deficiencies still exist.

(2) Require operating plants to comply with current licensing criteria.

It was recognized from the start that it may not be cost effective or practical to qualify operating plant equipment using current seismic qualification criteria and methods due to excessive plant down time, difficulties in shipping irradiated equipment to a test laboratory and in acquiring identical old vintage equipment for laboratory testing. In addition, the cost of meeting current criteria would be much greater than for a new plant. Meeting current criteria however would meet the safety objective therefore cost estimates are provided below for this alternative.

(3) Require verification of seismic adequacy by performing an on-site inspection of anchorage and supports and verifying equipment functional capability during the strong shaking motion utilizing seismic experience data and/or test experience data.

This alternative takes advantage of experience gained from review of facilities that have experienced strong motion earthquakes and also provides for assuring that supports and anchorages are adequate. For equipment not in the seismic experience data plants or for equipment unique to nuclear plants, or equipment needed to function during the first 30 seconds of an SSE, the test experience data base being developed by EPRI/RES can be used to assess equipment seismic adequacy and/or functional capability during an SSE.

3. Costs of Alternative

(1) <u>Not require any action of licensees</u>

There is no utility cost associated with this option.

(2) <u>Require operating plants to comply with current criteria</u>

Experience gained from the application of current criteria on new licenses is extrapolated to estimate cost for operating plants. The use of current requirements presents several complicating factors.

(a) Equipment would have to be removed from the plant and sent to a test laboratory for testing.

(b) Qualification procedures could result in costly plant shutdowns.

(c) Some of the equipment would be irradiated which would require special procedures for removal, testing shipping and reinstallation.

It is estimated that the qualification procedure would involve about 40 pieces of electrical equipment and 70 pieces of mechanical equipment. This is based on the assumption that only equipment required to bring the reactor to a safe shutdown condition is included.

A rough estimate of the projected cost to upgrade an operating reactor to meet IEEE 344/75 is approximately 10 million dollars. This is based on the following assumptions and estimates.

(1) 75% of equipment would require tests and analysis (i.e., structural integrity will not ensure functionality).

(2) average cost of test and analysis per piece of equipment (from table 2) is $17,300. This number assumes equipment can be tested in place, i.e., in-situ testing. If a component is removed, shipped to a test laboratory and tested the cost would be much higher. In-situ testing would be practical only in a limited number of cases. Recent experience of one utility is that for an AC distribution panel or an instrument rack, the testing cost is $30,000. This does not include removal or

shipping which would at least double the cost. For purposes of this estimate, $50,000 per component is assumed for a total of ($50,000)(75%)(110 components) or approximately 4 million dollars.

(3) 10% of equipment would need to be replaced. The average cost of replacement based on Table 2 and correcting for a more realistic escalation due to inflation is: ($500,000 per piece of equipment) (10%) (110 components) or approximately 5.5 million dollars.

(4) Average analysis cost for the 25% of equipment where analysis alone would be acceptable is approximately $20,000 per item or ($20,000) (.25) (110) or $550,000.

This estimate seems reasonable in light of industry experience solicited by the staff on approximate costs to comply with IEEE-323/74 and IEEE 344/75, for both environmental and seismic qualification. This experience includes:

(1) The upgrade of reactor building pressure transmitters at a multi-unit operating PWR will cost about $200,000 for 9 transmitters. The upgrade includes both environmental and seismic and replacement of units.

(2) The upgrade of power systems equipment to IEEE 323/74 (mainly documentation) at a new PWR will cost the utility about $3.5 million.

(3) One utility estimated that their share of the cost of an NSSS program to upgrade IEEE equipment to IEEE 323/74 will cost $4 million.

(4) At a CE system 80 plant the estimate to update NSSS IEEE equipment to IEEE 323/74 will cost $15.0 million.

(5) At a multi-unit operating PWR, the utility estimated that it would cost $20-$30 million to upgrade equipment to IEEE 344/75 (seismic qualifica-tion only). This estimate did not include documentation or plant down time.

TABLE 2

COST ESTIMATES OF SEISMIC VERIFICATION ALTERNATIVES[a]

Equipment Type	Analysis High	Analysis Low	Analysis Average	Test and Analysis High	Test and Analysis Low	Test and Analysis Average	Replacement High	Replacement Low	Replacement Average	Comparison High	Comparison Low	Comparison Average	Support Modification High	Support Modification Low	Support Modification Average
Air Circ Fan/Motor	10,000	6,000	8,000	44,500	9,900	15,300	75,000	3,500	13,500	600	100	200	7,000	1,300	2,600
Air Cond Unit	200,000	75,000	100,000	118,000	26,200	40,600	260,000	28,000	115,000	1,600	400	800	15,000	2,400	7,000
Cabinet[b]	13,000	7,000	9,000	44,500	9,900	15,300	4,500	1,000	2,500	600	100	200	850	350	500
Circuit Board	--	--	--	--	--	--	600	90	400	600	100	200	350	230	275
CRDM[c]	--	--	--	44,500	9,900	15,300	32,580K	2,450K	27,000K	600	100	200	33,700	5,800	13,800
Diesel Generator	200,000	75,000	100,000	118,000	26,200	40,600	750,000	250,000	500,000	2,000	400	1,200	88,600	24,800	49,400
Inverter	--	--	--	--	--	--	1,300	200	900	600	100	200	370	240	300
MSIV	18,000	12,000	15,000	53,600	11,900	18,400	350,000	140,000	200,000	600	100	200	37,400	13,100	21,600
Panels	13,000	7,000	9,000	44,500	9,900	15,300	30,000	1,000	7,000	600	100	200	1,870	360	710
Small Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	95,000	6,000	54,000	1,200	200	400	8,100	1,460	4,400
Medium Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	160,000	17,000	78,000	1,200	200	400	16,800	3,400	8,400
Large Horiz Pump/ Motor	23,000	14,000	17,000	44,500	9,900	15,300	245,000	31,000	125,000	1,200	200	400	25,200	5,200	12,800
Small Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	42,000	7,000	24,000	900	100	300	12,100	3,040	6,300
Medium Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	87,000	30,000	59,000	900	100	300	18,900	5,200	10,200
Large Vert Pump/ Motor	26,000	17,500	22,000	44,500	9,900	15,300	160,000	50,000	100,000	900	100	300	31,800	8,500	16,800
Racks (Instr.)	13,000	7,000	9,000	44,500	9,900	15,300	3,300	750	1,900	600	100	200	800	350	510
Racks (Bat.)	13,000	7,000	9,000	44,500	9,900	15,300	5,000	1,100	2,800	600	100	200	870	360	540
Strip Chart Rec.	--	--	--	--	--	--	7,500	800	3,400	600	100	200	970	350	570
Relays	--	--	--	--	--	--	800	130	560	600	100	200	350	230	280
Metal-Clad Switchgear	--	--	--	53,600	11,900	18,400	73,000	12,000	42,500	600	100	200	9,000	2,140	4,800
Voltage Switchgear	--	--	--	--	--	--	7,100	300[d]	3,200	600	100	200	680	230	430
Motor Control Center	--	--	--	--	--	--	10,700	350[e]	3,650	600	100	200	1,270	270	410
Transducer	--	--	--	--	--	--	1,300	500	1,000	600	100	200	370	250	300
Transformer	--	--	--	27,400	6,100	9,400	8,500	1,500	5,500	600	100	200	1,530	500	920
Check Valve	6,000	2,000	4,000	27,400	6,100	9,400	9,000	150	4,800	600	100	200	1,150	350	700
Small Instr. Valve	6,400	3,200	4,800	26,800	6,000	9,200	300	90	125	600	100	200	330	230	260
Small Relief Valve	13,000	8,500	11,000	44,500	9,900	15,300	15,000	1,300	8,000	600	100	200	1,150	340	700
Large Relief Valve	13,000	8,500	11,000	53,600	11,900	18,400	45,000	5,200	25,500	600	100	200	3,400	760	1,920
Small Safety Valve	11,000	6,500	9,000	44,500	9,900	15,300	6,000	2,800	4,500	600	100	200	1,030	460	670
Large Safety Valve	11,000	6,500	9,000	53,600	11,900	18,400	35,000	6,000	14,000	600	100	200	2,500	660	1,200

a. Equipment with no estimate for a particular method is not suitable for qualification by that method.

b. Cabinet only. Contents of cabinet not included.

c. K = x 1,000

d. 15 amp-240 V ac 3-pole circuit breaker.

e. 600 V 3-phase ac @ 2 hp motor starter.

Table 2 presents representative costs to verify seismic adequacy. Initial comments on the cost table by an industry group indicate that equipment replacement costs are low by a factor of 3 to 5 and in some cases as high as 9. An explanation of each of the columns in Table 2 follows:

Analysis

The Analysis cost estimates were based on experience in estimating analysis jobs and on reviews of such analysis performed during Seismic Qualification Review Team (SQRT) audits of qualification reviews performed for operating license applications. Equipment which has no estimate for analysis is not suitable for qualification by analysis.

Test and Analysis

The figures under Test and Analysis include the cost estimated by a NRC contractor to determine equipment/support dynamic characteristics via in-situ testing. The analysis effort is greatly reduced by using dynamic parameters determined by test. This estimate was compared to actual cost data from the private sector and shown to be high. This was attributed to two factors. First, the estimate was based on a single test per trip, while the actual data involved multiple tests per trip. Second, the estimate was based on a full reduction of data, which yields full mass and stiffness matrices in addition to the natural frequency, mode shape, and damping data actually obtained. The numbers in the estimate were reduced by a constant multiplier to account for these factors. Numbers in the "Low" column were obtained by a multiplier that yielded an estimate within 5% of the actual cost for a test contract involving 17 tests in a single trip. Numbers in the "High" column were obtained with a multiplier to account for the more complete data reduction included in the estimate. The numbers in the "Average" column were obtained with a multiplier to account for the more complete data reduction and to adjust the estimate to a five test per trip basis.

Replacement

Replacement is the cost incurred to replace equipment with qualified equipment. This includes purchase of the equipment with qualification documentation and installation. It does not include freight charges. Estimates are primarily based on "Process Plant Construction Estimating Standards," by Richardson Engineering Services, Inc. Two editions of the standard were used, one dated 1975 and the other 1981. Estimates taken from the 1975 edition were increased by 30%* to account for inflation. Two components on the list (MSIV and CRDM) were not covered by the standard. Estimates for these two were obtained by contact with vendors.

Qualification documentation was assumed to cost 150% of the cost of the unqualified components for all but three of the components--small instrument valves, transducers, and relays. These components are produced in large quantities and required in large quantities in typical plants. Their qualification documentation is assumed to be less costly--50% of the cost of the unqualified component.

(3) Require verification of seismic adequacy and equipment functional capability by performing an on-site inspection of anchors and supports and comparing plant equipment with seismic experience data and/or test experience data

Two alternatives are considered. If a utility participates in a generic program, the cost will be substantially less than for a utility who elects to not participate in a generic program.

*Industry comments indicate that actual escalation rates between 1975 and 1984 may be as high as 90%. For nuclear estimations the 90% rate is usually multiplied by 3 to 5 since it does not cover health physics, decontamination, respirator work, etc.

The Comparison estimate in Table 2 is the cost of comparing dynamic and functional characteristics between equipment in-plant and that in the data base. The estimate is based on the assumption that necessary data is readily available. Therefore, no costs resulting from analysis or in-situ testing have been included. The estimated costs to licensee by using this alternative is discussed in Section 4 below.

4. Estimated Costs to Licensee

The least expensive procedure for verifying the seismic adequacy of components is by comparison with the experience data base. This procedure will work for many components, however, it is possible that additional steps will be required for some components. The estimates presented in Table 2 assume comparison of the required response spectra and dynamic characteristics of each component with the experience data base. A direct comparison on a component-by-component basis will probably be required for 10% or less of the components.

In the event that the utilities choose to adopt the generic implementation, costs to individual utilities would be much lower than the cost for each utility to provide a plant-specific verification of seismic adequacy. Shared costs of a generic resolution would depend on the number of utilities participating. The utility cost for a plant-specific verification will vary from plant to plant depending on the seismic design basis, the location of equipment and the type of plant. The following estimate therefore presents a range of costs for each item. Most equipment is located in plant areas where radiation does not present serious problems for inspection or modification.

The cost estimates therefore do not include special considerations for radiation protection. Labor cost of $100,000 per man-year is assumed.

Following estimates do not include plant down time, and are for a single power plant unit participating in a generic effort.

Item	Estimated Cost (Dollars)/plant
Define systems, subsystems and components required and develop equipment list (2 people, 1-2 month)	$17,000 - $35,000
Compare data base spectra with site spectra (1 person, 1/2 to 1 month)	$ 4,000 - $10,000
Conduct plant walk-through (4 people for 1 to 2 month)	$32,000 - $80,000
Repairs to anchorage and supports (average of $40,000 for support of a electrical or mechanical equipment, for 5 to 10 pieces)	$200,000 - $400,000
Identify relays needed to function during the 30 seconds of strong motion earthquake, and relays which could potentially chatter or change state during earthquake (2 people, 1-2 months and $1000/relay (assume 10-30 relays))	$27,000 - $55,000
Miscellaneous modifications to components to fit experience data ($2,000 for 5 to 10 items)	$10,000 - $20,000

Collection of test experience data (assumes cost to a single utility participating in a generic effort)	\$50,000 - \$100,000
Generation of floor response spectra (assumes simplified analysis is used in lieu of full fledged soil-structure interaction and finite element analysis)	\$50,000 - \$100,000
Auditing performed by an independent contractor (at 2 people, 1/2 month to 1 month which includes preparation before audit and documentation for audit	\$ 8,000 - \$20,000
Preparation and submission of report to NRC	\$10,000 - \$20,000
TOTAL	\$401,000 - \$840,000

The industry cost based on the above estimates for costs to a utility participating in a generic program would be 28 to 59 million dollars for the approximately 70 plants (units) involved.

In addition, the SQUG utilities have spent approximately $200,000 each developing the experience data base and anticipate spending an additional $35,000 each prior to plant specific implementation. The additional costs are for development of detailed walk-through procedures, pilot walk-throughs, holding implementation workshops and for documenting the basis for seismic capacity of equipment classes which are not treated in detail in the experience data base. Total SQUG expenditures are about $2,500,000.

In the event a utility decides to not participate in the proposed generic resolution, additional costs of preparing and submitting a plant specific report and the review and audit by the NRC staff would be incurred. This would add an estimated $50,000 to $100,000 to each utility's cost and $10,000 to $30,000 per utility to NRC staff costs.

In addition, a utility not participating in a generic resolution would be required to develop their own detailed walkdown procedure and spend considerable resources in planning and executing the implementation. This plant specific implementation procedure would need to be reviewed in detail and probably result in several iterations. The data bases and SSRAP and EQE data reports would of course be available to all utilities.

5. Costs to the NRC

The principal cost to NRC (for a utility not participating in a generic implementation) will be review of the reports submitted by the licensees and participation in the plant audit. It is estimated that about 70 plants would be required to submit reports. It is estimated that it would require 0.6 staff months to review each report and 0.5 staff months to prepare an SER, for a total expenditure of 77 staff months. At an estimate rate of $100,000 per staff year, the cost would be $640,000 total.

If a generic implementation is implemented by SQUG or a similar utility group, the cost to the NRC would be substantially reduced.

6. Safety Benefit Compared to Costs

The safety benefit of the proposed seismic verification program is reduced likelihood of core melt and radiation release due to seismic failure of equipment required to safely shutdown the plant following a seismic event. The principal concern is equipment failure or loss of equipment function due to failure of anchorage or supports or loss of shutdown system functions due to relay chatter. The experience data base plus the survey of strong motion

earthquakes conducted by SQUG and SSRAP indicates that anchorage failures are possible. Experience gained from SEP reviews and recent staff surveys also indicates that some anchorages in nuclear plants may be susceptible to seismic failure.
Although the incremental risk has not been quantified in this study, the potential for safety improvement exists. The staff concludes that the inspection and verification program outlined would result in significant safety improvement.

Approximate costs to achieve the safety benefit are:

(1) impose current licensing requirements — $10,000,000/per plant
(2) generic program using experience data — $401,000 to $840,000/per plant

Because of the lower cost and the more effective treatment of anchorages, the staff recommends that the proposed seismic verification program be implemented.

7. Impacts on Other Requirement

The proposed requirement would have no impact on current licensing requirements since it would not change the implementation of current requirements on existing license applicants or new license applicants.

8. Constraints

Implementation of the requirement could be affected by the limited amount and range of experience data presently included in the data base. Also, applicable test data has not been collected and organized. The implementation plan and schedule has therefore been developed with the assumption that additional test data will be collected and used to verify seismic adequacy.

VI. IMPLEMENTATION

1. The proposed method of implementation is by issuance of a generic letter under the provisions of 10 CFR 50.54(f).
   The generic letter was selected rather than a Standard Review Plan or Regulatory Guide because the proposed requirements apply only to operating plants not reviewed to current licensing criteria and will be a one-time review of operating facilities rather than a continuing requirement.

2. Schedule for Implementing the Proposed Requirement

The proposed resolution presents the procedure for verifying the seismic adequacy of equipment using seismic experience from non-nuclear facilities that have experienced strong motion earthquakes as well as equipment test data. The SQUG program to date has been limited to eight classes of equipment. Implementation on the initial eight classes of equipment and review of all anchorages can proceed as soon as the requirement is established. Additional work is required to collect test data on a number of equipment types. The SQUG has initiated a program to develop the additional information and to continue the SSRAP as an independent review group. Follow-on work needed to complete implementation is:

(1) Develop basis for seismic adequacy of equipment not included in the 8 types in the seismic experience data base, but which exist, in the seismic experience data plants and equipment unique to nuclear plants.
(2) Add qualification test data to the data base.

The implementation schedule will be negotiated with the Generic Group taking into consideration the NRC policy on integrated schedules for plant modifications as stated in generic letter 83-20 dated May 9, 1983. This policy was reiterated in NRC 1985 Policy and Planning Guidance dated February 26, 1985. Utilities electing to not participate in a generic resolution are invited to individually negotiate with the staff on their implementation schedule.

For equipment for which collection of test data is needed, the implementation may be deferred until the EPRI test data base is fully developed, provided that
the seismic verification be completed no later than 28 months from the date of issuance of the USI A-46 final resolution.

Actual schedule dates will be based on final approval of the proposed requirement and the results of negotiations with the Generic Group or individual utilities. The elapsed times shown are estimated from the date of issuance of the requirement.

Item	Elapsed Time from Date of Requirement (Months)
Generic Group complete final walkdown procedure and conduct workshops	4
Start implementation. Identify systems, subsystems and components and conduct walk-through inspection of all anchorages and equipment other than those required to collect test data.	9
Complete necessary modifications to all anchorages and equipment.	15

Assess seismic adequacy and/or functional capability of equipment and component (including relays) for which collection of test data is required.	28
Provide report to NRC.	32

3. Relationship to Other Existing or Proposed Requirements

The proposed requirement would be imposed on existing plants which were not reviewed to current requirements as an alternative to requiring those plants to meet current requirements.

VII. SUMMARY OF A-46 RISK ANALYSIS

An attempt was made to develop a quantitative basis for estimating the risk reduction due to qualifying equipment but the results were inconclusive. The results of these probabilistic risk assessment analyses which were conducted by Brookhaven National Laboratory (BNL) (NUREG/CR-3357, Reference 8 in Enclosure 3) did not provide sufficient risk information to estimate incremental releases due to use of qualified versus unqualified equipment. The BNL analyses did, however, predict the percentage risk contribution attributable to major safety systems and components. The risk analysis results included random failures as well as seismic failure of the equipment and were estimated in terms of percentage of total risk.

The risk contribution due to seismic induced equipment failure is dependent on the assignment of a fragility curve. To determine the risk reduction due to qualifying equipment, the fragility of "qualified" and of "unqualified" equipment must be known. Very little actual fragility data exists. Most qualification tests are "proof" tests where the test item is subjected to a required test severity to prove it will survive and/or function at that level. To obtain fragility information, a large number of test specimens must be tested to failure for each failure mode considered to obtain the mean failure level and the associated uncertainties. The procedure of qualifying equipment does not change the fragility. The fragility will change only if the item is replaced or is physically modified in some manner.

In addition, the BNL analysis results are applicable only to the models used. They are model-specific with regard to the plant systems, the assumed fragilities and site soil conditions and seismicity. The BNL study does however provide some pertinent insights.

(1) The risk importance of specific systems or components can be calculated on a plant-specific model by varying the fragility and noting the change in predicted release. BNL has estimated that the cost to conduct such an analysis (assuming an internal event PRA model exists) ranges from $330,000 to $490,000 and would take 33 to 49 months to accomplish (BNL Draft Report, "Guidelines for Identification of Seismically Risk-Sensitive Systems and Components in a Nuclear Power Plant," dated September 1983). These figures include developing cost estimates and performing a value/impact analysis.

(2) Results of the BNL analysis indicate that both core melt frequency and risk for Boiling Water Reactors are dominated by structural failures. Structural failures accounted for 85% of core melt frequency and 97.5% of risk in the Boiling Water Reactor model and 38% of core melt frequency and 33.6% of risk in the Pressurized Water Reactor model. BNL concluded, in part that, "...overall seismic induced structure failures and random failures (due to non seismic causes) contribute more significantly to core melt probability and risk than seismic-induced equipment failure." However, in Board Notification 83-01A "Seismic Risk to BWR Plants", the staff concluded that meaningful conclusions

regarding overall risk from BWR plants or comparisons between PWR and BWR plants cannot be drawn from this BNL report.

(3) BNL points out in their conclusions that their analysis results basically indicate that upgrading equipment from the baseline (assumed fragility levels) without upgrading the structure enclosing the equipment does not introduce a marked reduction in risk, whereas a more demonstrated increase in risk is calculated when the equipment is more "fragile" than the baseline case (see Figure 1). However, the sensitivity study from which this conclusion is based does not account for the large uncertainties associated with the mean fragility levels used and thus makes the conclusion somewhat questionable. The equipment fragility data was developed by R. P. Kennedy, et al. and published as NUREG/CR-2405, "Subsystem Fragility" dated February 1982. This fragility data is the source of fragilities used in the Seismic Safety Margins Research Program seismic risk model as well as the Indian Point Safety Study and Oyster Creek Station Risk Assessment. The data was developed by review of available test data and analysis supplemented by consensus of expert opinion. It includes fragility test data developed by the Department of the Army in their blast effects program.

The staff has concluded, based on the considerations discussed above that it is not feasible to provide a quantitative estimate of net safety benefit in terms of risk to the public.

FIGURE 2

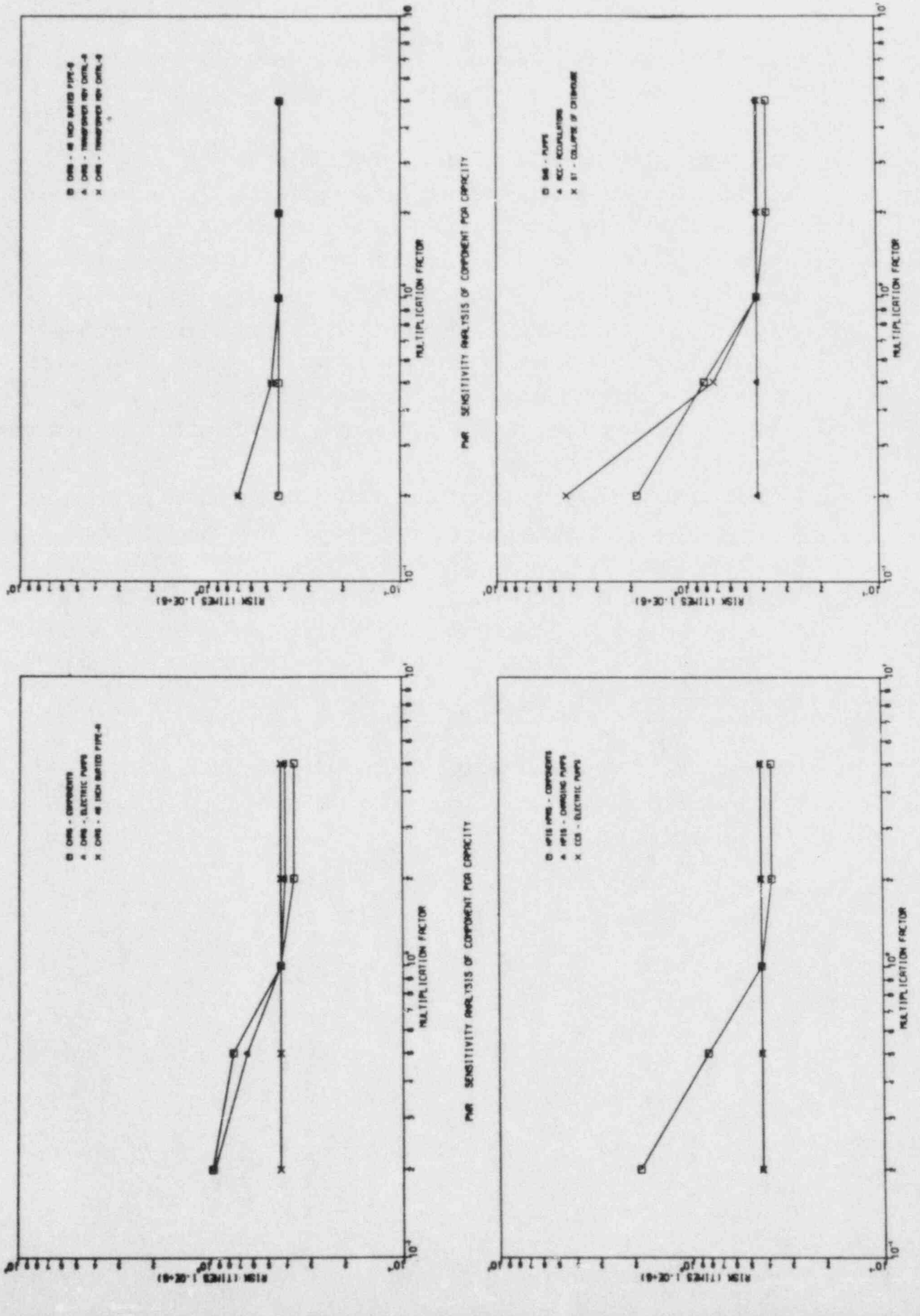
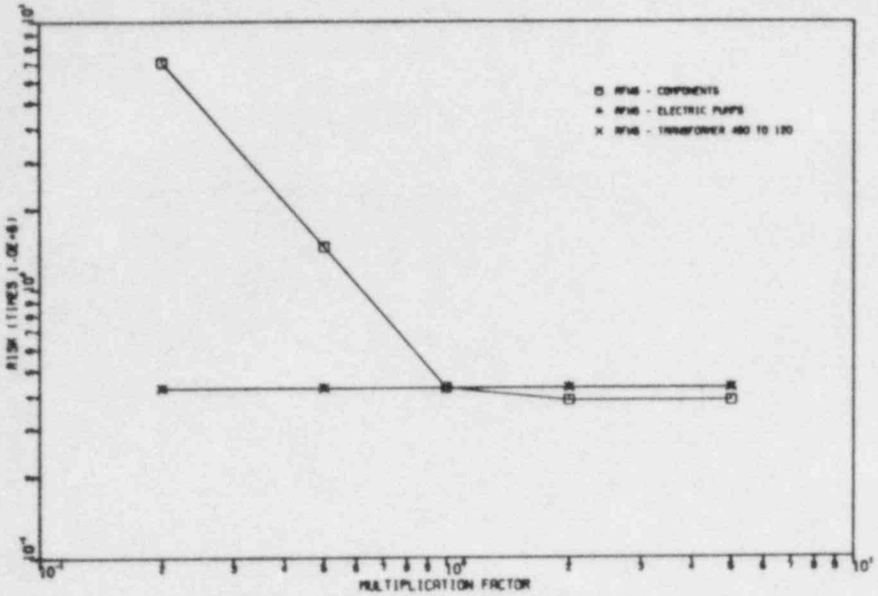
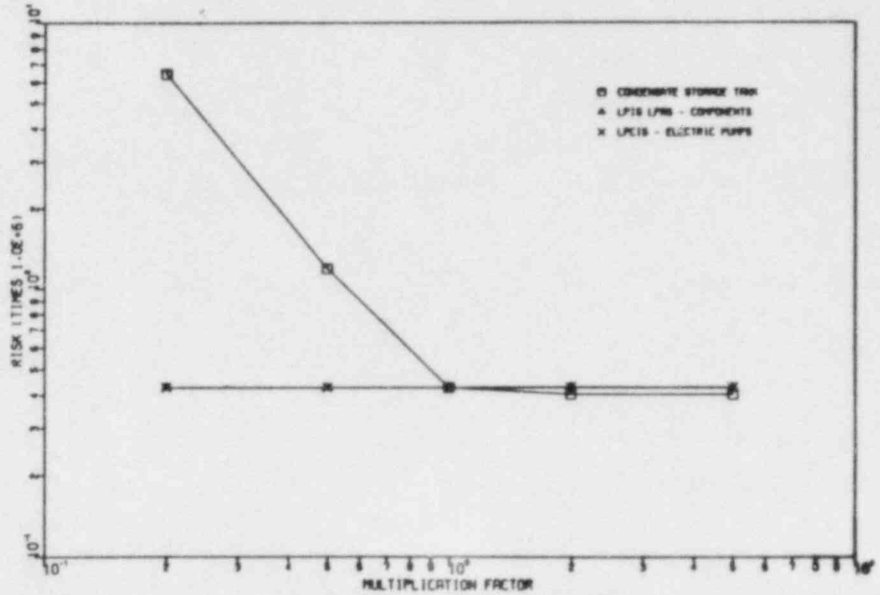

RISK (TIMES 1.0E-6)
MULTIPLICATION FACTOR
PWR SENSITIVITY ANALYSIS OF COMPONENT PGA CAPACITY

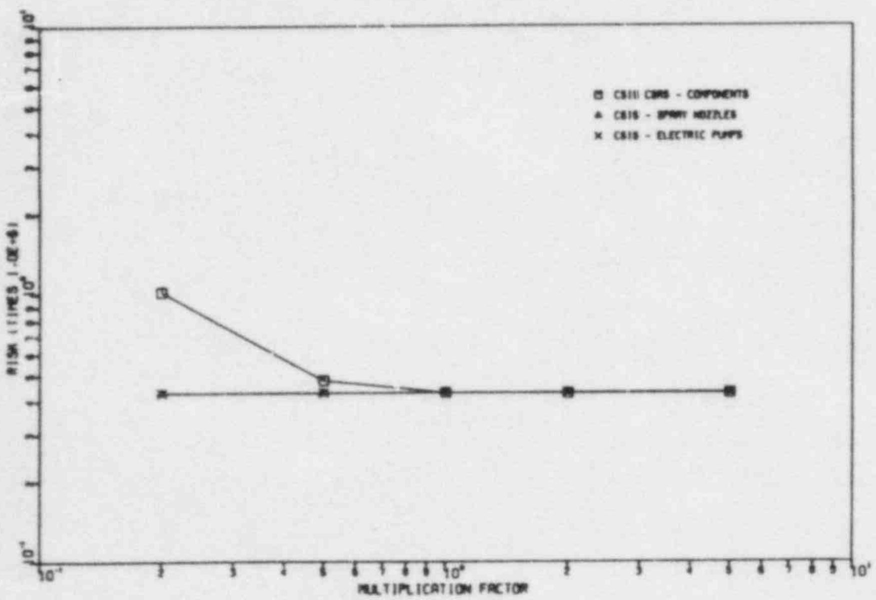
FIGURE 2 (CONTINUED)



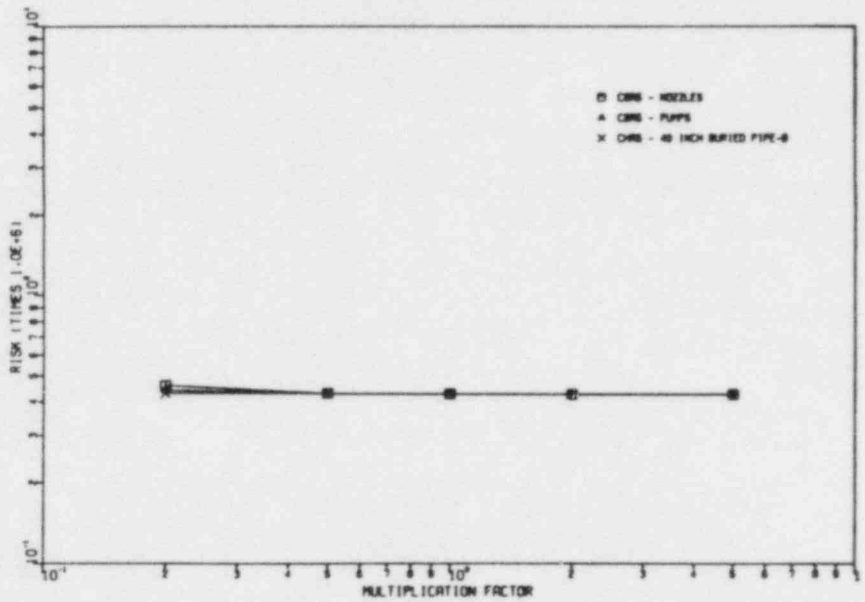
PWR SENSITIVITY ANALYSIS OF COMPONENT POR CAPACITY



PWR SENSITIVITY ANALYSIS OF COMPONENT POR CAPACITY

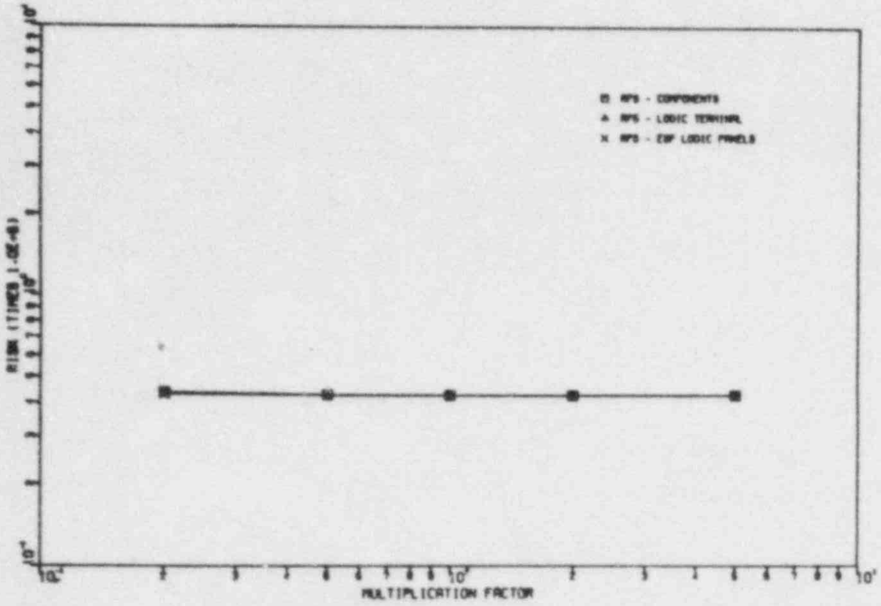


PWR SENSITIVITY ANALYSIS OF COMPONENT POR CAPACITY

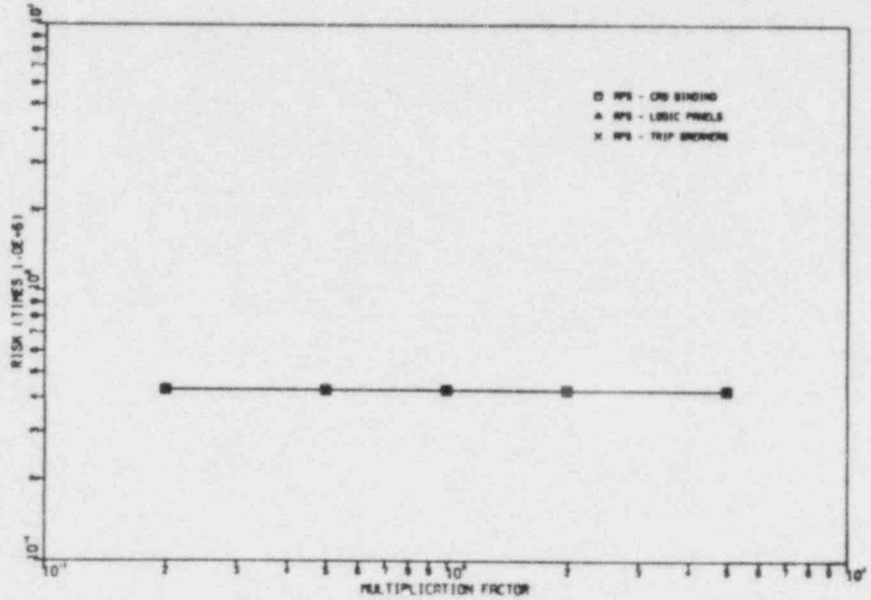


PWR SENSITIVITY ANALYSIS OF COMPONENT POR CAPACITY

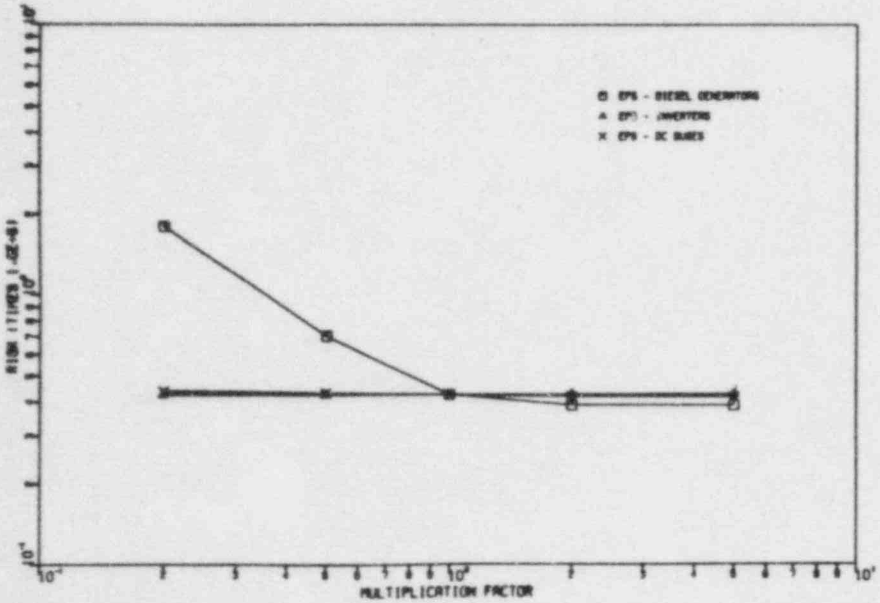
FIGURE 2 (CONTINUED)



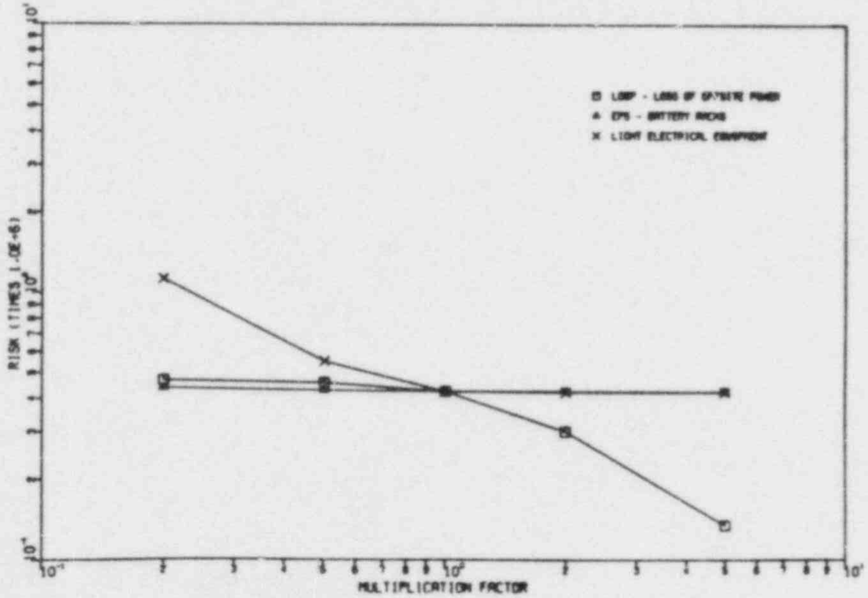
PWR SENSITIVITY ANALYSIS OF COMPONENT PGA CAPACITY



PWR SENSITIVITY ANALYSIS OF COMPONENT PGA CAPACITY



PWR SENSITIVITY ANALYSIS OF COMPONENT PGA CAPACITY



PWR SENSITIVITY ANALYSIS OF COMPONENT PGA CAPACITY

ENCLOSURE

Operating Plants to be Reviewed to USI A-46 Requirement

The plant list was developed by determining from plant Safety Evaluation Reports whether or not the seismic qualification review was performed using IEEE-344/75. Plants for which there is no documentation of meeting the provisions of IEEE-344/75 are included on the list of affected plants.

Alabama

1. Browns Ferry, Unit 1
2. Browns Ferry, Unit 2
3. Browns Ferry, Unit 3
4. Joseph M. Farley, Unit 1

Arkansas

5. Arkansas Nuclear One, Unit 1
6. Arkansas Nuclear One, Unit 2

California

7. San Onofre, Unit 1
8. Rancho Seco, Unit 1

Colorado

9. Fort St. Vrain

Connecticut

10. Haddam Neck
11. Millstone, Unit 1
12. Millstone, Unit 2

Florida

13. Turkey Point, Unit 3
14. Turkey Point, Unit 4
15. Crystal River, Unit 3
16. St. Lucie, Unit 1

Georgia

17. Edwin I. Hatch, Unit 1
18. Edwin I. Hatch, Unit 2

Illinois

19. Dresden, Unit 2
20. Dresden, Unit 3
21. Zion, Unit 1
22. Zion, Unit 2
23. Quad-City, Unit 1
24. Quad-City, Unit 2

Iowa

25. Duane Arnold, Unit 1

Maine

26. Maine Yankee

Maryland

27. Calvert Cliffs, Unit 1
28. Calvert Cliffs, Unit 2

Massachusetts

29. Yankee Rowe
30. Pilgrim, Unit 1

Michigan

31. Big Rock Point
32. Palisades
33. Donald C. Cook, Unit 1
34. Donald C. Cook, Unit 2

Minnesota

35. Monticello
36. Prairie Island, Unit 1
37. Prairie Island, Unit 2

Nebraska

38. Fort Calhoun, Unit 1
39. Cooper

New Jersey

40. Oyster Creek, Unit 1
41. Salem, Unit 1
42. Salem, Unit 2

New York

43. Indian Point, Unit 2
44. Indian Point, Unit 3
45. Nine Mile Point, Unit 1
46. R. E. Ginna, Unit 1
47. James A. Fitzpatrick

North Carolina

48. Brunswick, Unit 1
49. Brunswick, Unit 2
50. W. B. McGuire, Unit 1
51. W. B. McGuire, Unit 2

Ohio

52. Davis-Besse, Unit 1

Oregon

53. Trojan, Unit 1

Pennsylvania

54. Peach Bottom, Unit 2
55. Peach Bottom, Unit 3
56. Beaver Valley, Unit 1
57. Three Mile Island, Unit 1

South Carolina

58. H. B. Robinson, Unit 2
59. Oconee, Unit 1
60. Oconee, Unit 2
61. Oconee, Unit 3

Tennessee

62. Sequoyah, Unit 1
63. Sequoyah, Unit 2

Vermont

64. Vermont Yankee

Virginia

65. Surry, Unit 1
66. Surry, Unit 2
67. North Anna, Unit 1
68. North Anna, Unit 2

Wisconsin

69. LaCrosse
70. Point Beach, Unit 1
71. Point Beach, Unit 2
72. Kewanee

APPENDIX A

DRAFT GENERIC LETTER
(Reference USI A-46)

TO: All Holders of Operating Licenses Not Reviewed to Current Licensing Criteria on Seismic Qualification of Equipment

SUBJECT: VERIFICATION OF SEISMIC ADEQUACY OF MECHANICAL AND ELECTRICAL EQUIPMENT IN OPERATING REACTORS UNRESOLVED SAFETY ISSUE (USI) A-46

Reference:
(1) EQE Report, "Pilot Program Report; Program for the Development of an Alternative Approach to Seismic Equipment Qualification," September 1982
(2) SSRAP Report, "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants," January 1985
(3) NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants (USI A-46)"

As a result of the technical resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants," the NRC has concluded that the seismic adequacy of certain equipment in operating nuclear power plants must be verified. The technical basis for this conclusion is documented in References 1, 2, and 3.

This requirement is based principally on work performed by the Seismic Qualification Utilities Group (SQUG). The SQUG collected seismic data from several non-nuclear facilities which had experienced strong motion earthquakes. This data is presented in Reference 1. In March 1983, the SQUG proposed to NRC management the formation of a Senior Seismic Review Advisory Panel (SSRAP) to provide expert opinion and advice on the applicability and use of seismic experience data in evaluating the seismic adequacy of equipment in nuclear plants. The NRC endorsed the idea and the SSRAP was formed in June 1983. SSRAP's scope was limited to seven classes of equipment. This was later extended to eight classes. The SSRAP investigation and conclusions are documented in Reference 2.

The SQUG and SSRAP investigations were closely monitored by the NRC staff. The NRC review of these investigations and its recommendations apppears in Reference 3. The staff has concluded that certain verification steps must be taken by each licensee to ensure that all equipment is adequately anchored and not mounted or configured in a manner which would make it susceptible to seismic damage. In certain cases where seismic experience data is unable to provide information to assess equipment seismic adequacy, collection of test data will be necessary. Seismic verification may be accomplished generically as specified in the enclosure. Utilities participating in a generic program should so state in their reply to this letter identifying the utility group and the schedule for completion of the effort. Guidelines for performing this verification are presented in the enclosure.

Implementation of this requirement will be accomplished in two stages. The seismic verification of all anchorages and all equipment other than those required to collect test data will be conducted first. The assessment of seismic adequacy and/or functional capability of equipment and components (including relays) for which collection of test data is required can be deferred until the test data base being developed by EPRI/RES (Office of Nuclear Regulatory Research, NRC) is completed.

The implementation schedule will be negotiated with the Generic Group or with individual utilities in accordance with the NRC policy on integrated schedules for plant modifications stated in generic letter 83-20 dated May 9, 1983.

In order to make a final determination on this issue, we request, pursuant to 10 CFR 50.54(f), that you provide the NRC, within 45 days of your receipt of this letter, a schedule for implementation of requirements requested in the enclosure and the basis for the schedule. This information should be submitted to the NRC, signed under oath and affirmation, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked.

Sincerely,

Hugh L. Thompson, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
Procedure for Verifying Seismic Adequacy of Equipment in Nuclear Power Plants

APPENDIX A

Enclosure

Seismic Adequacy Verification Procedure

The proposed procedure for verifying seismic adequacy of equipment is addressed in the following paragraphs. Each licensee will be required to perform the verification steps and submit a report to the NRC including an affidavit that the verification has been completed and all equipment within the scope defined below has been found to be acceptable. A generic resolution will be accepted in lieu of a plant-specific verification review subject to the guidance presented herein.

1. Scope of Seismic Adequacy Review

Each licensee will determine the systems, subsystems, components, instrumentation and controls required during and following a design basis seismic event using the following assumptions.

(1) The seismic event does not cause a LOCA and a LOCA does not occur simultaneously with or during a seismic event. However the effects of transients that may result from ground shaking should be considered.

(2) Offsite power will be lost during or following a seismic event; and

(3) Plant must be capable of being brought to a safe shutdown condition following a design basis seismic event.

The equipment to be included is limited to active mechanical and electrical components. Piping, tanks and heat exchangers are not included except that those tanks and heat exchangers that are required to achieve and maintain safe shutdown must be reviewed for adequate anchorage.

Lessons learned from studies of nuclear and nonnuclear facilities under earthquake loading indicate that the effect of failure of certain items, such as suspended ceilings and light fixtures could influence the operability of equipment within the scope of reviews. This concern is addressed in USI A-17, "Systems Interaction," and is therefore not further considered in implemeting A-46. The failure of masonry wall that could affect the operability of nearby safety-related equipment is also of concern. However, this concern has been addressed by IE Bulletin 80-11, which requires that all such masonry walls be identified and re-evaluated to confirm their design adequacy under postulated loads and load combinations. This concern is therefore not considered as part of A-46 implementation.

For some pressurized water reactor plants, the seismic adequacy of Auxiliary Feed Water Systems (AFW) has been verified by licensee actions taken in response to generic letter 81-14 dated February 10, 1981. Review of the AFW may be deleted from consideration under A-46 if staff acceptance has been documented in an SER, or if the licensee has committed to meet the requirements of the generic letter.

The definition of safe shutdown is not well defined. For the purpose of seismic adequacy verification, the following guidance is given. Each licensee should identify equipment necessary to bring the plant to a hot shutdown condition and maintain it there for a minimum of 72 hours. The 72 hour time period is sufficient for inspection of equipment and minor repairs if necessary following an SSE or to provide additional source(s) of water for decay heat removal if needed to extend the time at hot shutdown. Equipment required includes that necessary to maintain required supporting functions for safe shutdown. For all equipment within the defined scope, the verification should closely follow the procedure outlined in paragraph 2 below.

Each licensee must show practical means of staying at hot shutdown for a minimum of 72 hours. In the event that maintaining safe shutdown is dependent on a single (not redundant) component whose failure, either due to seismic loads or random failure, would preclude decay heat removal by the identified means, the licensee should show that at least one practical alternative for achieving and maintaining safe shutdown exists which is not dependent on that component.

Each licensee will develop an equipment list. This list will include all equipment within the required scope. This list will be classified into three categories of equipment. They are: (1) equipment belonging to the eight types in the seismic experience data base, (2) equipment of type not included in the eight types in the seismic experience data base but which exists in the data base plants, (3) equipment unique to nuclear plants.

The equipment to be considered depends on the functions required to be performed. Typical plant functions would include:

(1) bring the plant to a hot shutdown condition and establish heat removal;
(2) maintain support systems necessary to establish and maintain hot shutdown;
(3) maintain control room functions and instrumentation and controls necessary to monitor hot shutdown;
(4) provide alternating current and direct current emergency power.

2. General Verification Procedure for Plant-Specific Review

The licensee will conduct a plant walk-through and visual inspection of all identified equipment items necessary to perform the functions related to plant shutdown. The inspection team should consist as a minimum of:

(1) plant operations supervisor or a licensed Senior Reactor Operator;
(2) an experienced structural engineer familiar with seismic anchorage requirements;
(3) an experienced mechanical engineer familiar with plant mechanical equipment; and
(4) an experienced electrical engineer familiar with plant electrical equipment.

As an alternative, licensees may use consultants instead of their staff for (2), (3), and (4) above.

Prior to the walk-through inspection the licensee will verify that the appropriate data base spectra envelope the site free field spectra at the ground surface defined for the plant. He will identify all equipment on his equipment list which is located at an elevation higher than forty feet above ground level. For equipment above forty feet, one and one half times the appropriate data base bounding spectra (defined in Paragraph 6 below) must envelope the floor response spectra for the equipment. For those cases where floor response spectra are needed, NUREG/CR-3266 entitled, "Seismic and Dynamic Qualification of Safety Related Equipment in Operating Nuclear Power Plants-Development of a Method to Generate Generic Floor Response Spectra" may be used as one alternative to develop the necessary floor response spectra on a case-specific basis. The appropriate bounding spectra for equipment belonging to the eight types in the data base are defined in paragraph 6 below. For equipment types not included in the eight types in the data base but which exist in the data base plants, and for equipment unique to nuclear plants, the appropriate bounding spectra are defined in paragraph 7 below.

The walk-through inspection should cover anchorage review and identification of potential "deficiencies" and "outliers," as outlined below. Deficiency in this context means equipment, components, and their anchorages/supports which is identified to be inadequate by the A-46 criteria during plant-specific walk-through reviews. Outlier in this context means equipment items that are subject to the caveats and exclusions defined in this generic letter, or are otherwise not covered by the experience data. The treatement of deficiencies is further described in paragraphs 4 and 5 below.

(1) for all equipment within scope, verify equipment anchorage (including required tanks and heat exchangers) using guidance provided in paragraph 3 below, and identify potential deficiencies.

(2) for equipment belonging to the eight types in the data base, identify data base exclusions and caveats (outliers) from guidance provided in paragraph 6.

(3) for equipment types which exist in the data base plants but not included in the eight types in the data base, guidelines provided in paragraph 7 below should be used for identification and review of "outliers" during the walk-through inspection for this equipment.

The licensee must specify all equipment items which are required to function during the period of strong shaking. The operability of these items must be demonstrated by means other than comparison with the experience data base, otherwise the licensee must determine that any change of state will not compromise plant safety. The period of strong shaking is defined to be the first 30 seconds of the seismic event and should be considered in conjunction with the loss of off-site power.

All relays, which could potentially change state during an SSE due to contact chatter and preclude use of equipment needed after the SSE to place the plant in safe shutdown must be identified by the licensee. These relays must be qualified by test, by comparison with test data (considering the point of attachment of such devices) or replaced by relays qualified to current licensing requirements. As an alternative, the licensee may show that chattering or change of state of the relays does not affect system performance or preclude subsequent equipment or system functions.

For components included in the data base by type but outside the limits of experience data, or of a type not included in the eight types in the data base but which exist in the data base plants, or is unique to nuclear plants, or required to function during the first 30 seconds of earthquake strong motion, the seismic verification can be deferred until the EPRI/RES (Office of Nuclear Regulatory Research, NRC) test data base is fully developed and endorsed by SSRAP and approved by the NRC staff, provided that the seismic verification be completed no later than 28 months from the date of issuance of the USI A-46 final resolution.

In the event that components are replaced by the utility as a result of A-46 review, each replacement (assembly, subassembly, device) must be verified for seismic adequacy either by using A-46 criteria and methods or as an option, qualification by current licensing criteria. Component in this context means equipment and assemblies such as pumps and motor control centers, and subassemblies and devices such as motors and relays which are part of assemblies.

3. Verification of Anchorage

To verify acceptable seismic performance, adequate engineered anchorage must be provided. There are numerous examples of equipment sliding or overturning under seismic loading due to lack of anchorage or inadequate anchorage. Inadequate anchorage can include short, loose or poorly installed bolts or expansion anchors, inadequate torque on bolts, and improper welding or bending of sheet metal frames at anchors. Torque on bolts can normally be ensured by a preventive maintenance and inspection program.

In general, checking of equipment anchorages requires one to estimate the equipment weight and its approximate center of gravity. Also, one will have to either estimate the equipment fundamental frequency so as to obtain the spectral acceleration at this frequency or else use the highest spectral acceleration for all frequencies. When horizontal floor spectra exist, these spectra may be used to obtain the equipment spectral acceleration. Alternatively, for equipment mounted less than about 40 feet above grade, 1.5 times the free-field horizontal design ground spectrum may be used to conservatively estimate the equipment spectral acceleration. For equipment mounted more than about 40 feet above grade, floor spectra must be used.

Equipment anchorage must not only be strong enough to resist the anticipated forces but also be stiff enough to prevent excessive movement of the equipment and potential resonant response with the supporting structure. Review of anchorages should include consideration of both strength and stiffness of the anchorage and its component parts.

Additional discussions on seismic motion bounds and equipment supports and anchorage for each of the eight classes of equipment in the experience data base is included in Paragraph 6 below. This guidance supplements the general guidance above.

During the walk-through inspection, anchors and supports of all equipment within the scope of review will be carefully inspected using the detailed guidance provided.[-1] If adequacy of supports and anchors cannot be determined by inspection, an engineering review of the anchorage or support will be made. This engineering review will include review of design calculations or performance of new calculations and/or verification of fundamental frequency of equipment to ensure adequate restraint and stiffness. Physical modifications may be necessary if engineering review determined the anchorage or support to be inadequate.

4. Generic Resolution

The NRC will endorse and encourage a generic resolution of USI A-46 provided the guidelines presented below are followed.

(1) All member utilities of the SQUG would be eligible to participate.

(2) The Generic Group would be responsible to (a) develop procedures to identify relays to be evaluated, (b) to define functionality requirements and develop evaluation procedures for relays. This procedure will be reviewed and endorsed by SSRAP and the NRC staff.

[-1] The detailed guidance will be developed jointly by SQUG/SSRAP, EPRI and the NRC staff and will be available prior to implementation.

(3) The Generic Group would submit to the NRC a generic schedule for the implementation of the A-46 requirements within 90 days of receipt of the A-46 generic letter. The schedule should apply to all participating utilities. The Generic Group would prepare walk-through procedures and checkslists based on guidance provided in paragraphs 2 and 3 above. It is expected that a pilot walk-through would be conducted on a few selected plants to test the procedure. Afterwards, workshops would be held by the Generic Group for participating utilities to assure uniformity in approach. Individual utilities would then perform the plant-specific implementation review.

(4) Each individual utility should submit to the NRC an inspection report which should include: certification of completion of the review, identification of deficiencies and outliers, justification for continued operation (JCO) for identified deficiencies, modifications and replacements of equipment/anchorages (and supports) made as a result of the reviews, and proposed schedule for future modifications and replacements.

The objective of this requirement is to provide assurance that the plant can continue to be operated without endangering the health and safety of the public during the time period required to correct the identified deficiency.

The JCO may consider arguments such as imposition of administrative controls or limiting conditions for operation (LCO) or consideration of the importance of the safety function involved and/or identification of alternate means to perform that function.

(5) Consultants to the Generic Group would perform audits of plant-specific reviews. All plants would be audited. The NRC staff will participate in plant audits on a selective basis. The Generic Group must submit a generic implementation review report to the NRC certifying that the walk-through inspection has been completed by the individual utility and that the audit has been completed. This report covers all

participating utilities, and must be endorsed by the SSRAP. The NRC staff involved in plant audits should have appropriate background and experience. As a minimum they will participate in the Generic Group workshop.

(6) The SSRAP and the NRC staff would perform a limited review of the Generic Group audit process to evaluate effectiveness.

(7) Final approval of the implementation will be made by the NRC following receipt of a final report from individual utilities certifying completion of implementation reviews and equipment/anchorage modifications and replacements.

(8) The Generic Group must provide for the continuation of the SSRAP as an independent review body. The SSRAP would be consulted during development of the generic program and walk-through procedure, and audit the implementation.

(9) NRC staff members would be invited to participate in all meetings between the Generic Group and the SSRAP.

5. Provisions for Resolution for Individual Utilities

The Generic Resolution described in paragraph 4 above is the method preferred by the NRC for the resolution of A-46. This paragraph offers provisions for resolution of A-46 for individual utilities not participating in the Generic Group.

Each utility is required to perform plant-specific verification reviews according to guidance in paragraphs 2 and 3. He is also required to maintain an auditable record of implementation of USI A-46.

Within 45 days of receipt of the A-46 generic letter, the utility should submit to the NRC a schedule for implementation of the A-46 requirements. An inspection report should be submitted by the utility to the NRC following the plant-specific walk-through inspection. It should consist of the following:

(1) Certification of completion of the walk-through inspection and a description of procedures used.
(2) List of equipment included in the review scope. Equipment required to function during the strong shaking period should be identified.
(3) Identified deficiencies.
(4) Identified outliers.
(5) Modifications and replacements of equipment/anchorages (and supports) made as a result of the inspection.
(6) Proposed schedule for future modifications and replacements.
(7) A justification for continued operation (JCO) for identified deficiencies.

Following the completion of implementation reviews and all necessary modifications and replacements of equipment/anchorages, a final report should be submitted to the NRC. A description of the procedures used for the implementation reviews, and modifications and replacements should be included.

The NRC will review both the inspection report and the final report and will audit <u>all</u> plant-specific reviews prior to final NRC approval.

6. Guidance on Use of Seismic Experience Data for the Eight Equipment Types in the Experience Data Base *

(1) SEISMIC MOTION BOUNDS

In order to compare the potential performance of equipment at a given nuclear power plant with the actual performance of similar equipment in the data base plants in recorded earthquakes, SSRAP has developed Seismic Motion Bounding Spectra to facilitate comparison. The purpose of these Bounding Spectra is to compare the potential seismic exposure of equipment in a nuclear power plant with the estimated ground motion that similar equipment actually resisted in earthquakes described in the data base. For convenience, the Bounding Spectra are expressed in terms of ground response at the nuclear site rather than floor response or equipment response. These bounding spectra represent approximately two-thirds of the free-field ground motion to which the data base equipment was actually exposed.

Three different seismic motion bounds (Type A, B, and C) are used. Different bounding spectra were developed, not to infer different ruggedness of equipment, but to represent the actual exposure of significant numbers of each class of equipment within the data base to ground motion. These bounds are defined in terms of the 5% damped horizontal ground response spectra shown in Figure A-1. The seismic motion bounds may be used for the equipment class as defined below.

Equipment Class	Bound
Motor control centers Low-voltage (480-V) switchgear Metal-clad (2.4 to 4-kV) switchgear Unit substation transformers	Type B
Motor-operated valves with large eccentric operator lengths to pipe diameter ratios	Type C
Motor-operated valves (exclusive of those with large eccentric operator lengths to pipe diameter ratios) Air-Operated valves Horizontal pumps and their motors Vertical pumps and their motors	Type A

* Guidance in this paragraph is excerpted from the Senior Seismic Review and Advisory Panel report dated January 1985.

These spectra bounds are intended for comparison with the 5% damped design horizontal ground response spectrum at a given nuclear power plant. In other words, if the horizontal ground response spectrum for the nuclear plant site is less than a Bounding Spectra at the approximate frequency of vibration of the equipment and at all greater frequencies (also referred to as the frequency range of interest), then the equipment class associated with that spectra is considered to be included within the scope of this method. Alternately, one may compare 1.5 times these spectra with a given 5% damped horizontal floor spectrum in the nuclear plant.

The comparison of these seismic bounds with design horizontal ground response spectra is judged to be acceptable for equipment mounted less than about 40 feet* above grade (the top of the ground surrounding the building) and for moderately stiff structures. For equipment mounted more than about 40 feet above grade, comparisons of 1.5 times these spectra with horizontal floor spectra is necessary. In all cases such a comparison with floor spectra is also acceptable. It is felt that the vertical component will not be any more significant relative to the horizontal components for nuclear plants than it was for the data base plants. Therefore, it was decided that seismic bounds could be defined purely in terms of horizontal motion levels.

The criteria are met so long as the 5% damped design horizontal spectrum lies below the appropriate bounding spectrum at frequencies greater than or equal to the fundamental frequency range of the equipment. It is felt this estimate can be made judgmentally by experienced engineers without the need for analysis or testing.

The above recommendation that the seismic bounding spectra can be compared with the design horizontal ground response spectra for equipment mounted less than about 40 feet above grade is based upon various judgements concerning how structures respond in earthquakes.
It is felt that this 40 foot above grade criteria must be applied with some judgement, as some structures may respond in a different manner.

*In most cases where numerical values are given in this section they should be considered as either "approximate" or "about" and a tolerance about the stated value is implied.

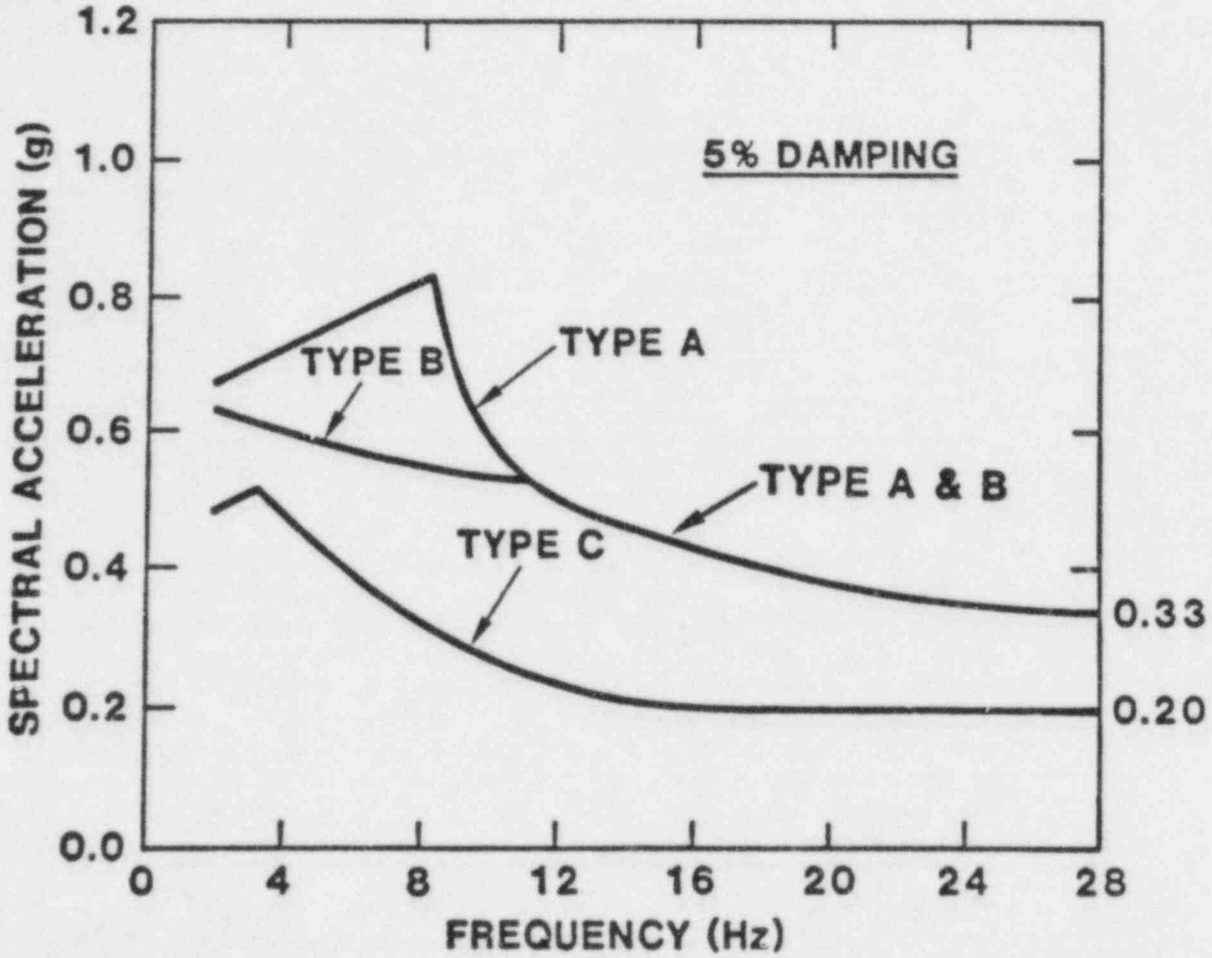


Figure A.1

Seismic Motion Bounding Spectra
Horizontal Ground Motion

(2) MOTOR CONTROL CENTERS

Motor control centers contain motor starters (contactors) and disconnect switches. They also provide over-current relays to protect the system from overheating. In addition, some units will contain small transformers and distribution panels for lighting and 120V utility service.

Motor control centers of the 600 volt class (actual voltage is 480V) are considered. The general configuration of the cabinets must be similar to those specified in NEMA Standards. This requirement is imposed to preclude unusual designs not covered in the data base. It is felt that cabinets which are configured similar to NEMA Standards will perform well if they are properly anchored. Cabinet dimensions and material gauges need not exactly match NEMA Standards.

Based on a review of the data base and anticipated variations in conditions, SSRAP is of the opinion that motor control centers are sufficiently rugged to survive a seismic event and remain operational thereafter provided the following conditions exist in the nuclear facility:

- The spectrum for the nuclear facility is less than the Type B bounding spectrum described in Fig. A. 1 for frequencies above the estimated fundamental frequency of the cabinet, and the motor control center is located less than 40 feet above exterior grade and has stiff anchorage as discussed below. If the motor control center is located higher than 40 feet above exterior grade or does not have stiff anchorage, the floor spectrum shall be compared to 1.5 times the Type B bounding spectrum. In all cases a comparison with floor spectra is also acceptable.

- The cabinets have stiff engineered anchorage. Both the strength and stiffness of the anchorage and its component parts must be considered. Stiffness can be evaluated by engineering judgment based on the cabinet construction and the location and type of anchorage, giving special attention to the potential flexibility between the tiedown anchorage and the walls of the cabinet. One concern is with the potential flexibility associated with bending of a sheet metal flange between the anchor and the cabinet wall. It is felt that stiffly anchored cabinets will have a fundamental frequency greater than about 8 Hz under significant shaking.

  The intent of this recommendation is to prevent excessive movement of the cabinet and to assure that under earthquake excitations the natural frequency of the installed cabinet will not be in resonance with both the frequency content of the earthquake and the fundamental frequency of the structure, thereby allowing comparison of the ground response spectra with the Type B bounding spectrum.

- Cabinets with sufficiently strong anchorage which do not have the stiff anchorage as recommended above, are still considered in the data base, however the floor response spectrum must be compared to 1.5 times the Type B bounding spectrum.

- Cutouts in the cabinet sheathing are less than about 6 inches wide and 12 inches high including side sheathing between multi-bay cabinets.

- All internal subassemblies are securely attached to the motor control cabinets which contain them.

- Adjacent sections of multi-bay cabinet assemblies are bolted together.

- Equipment and their enclosures mounted externally to motor control center cabinets and supported by them have a total weight of less than one hundred pounds.

The functionality, that is, inadvertent change of state or failure to change state on command of relays during an earthquake is not considered here. The functionality must be established by other means. The structural integrity of relays contained in the motor control centers and their ability to function properly after earthquakes, as defined in Fig. A.1, has been demonstrated.

(3) LOW-VOLTAGE SWITCHGEAR

Low-voltage switchgear consists of low voltage, that is, 600V or less, distribution busses, circuit breakers, fuses, and disconnect switches.

Low-voltage switchgear of the 600V class (actual voltage is 480V) is considered. The general configuration of cabinets must be similar to those specified in ANSI C37.20. This requirement is imposed to preclude unusual designs not covered in the data base. It is felt that cabinets which are configured similar to ANSI Standards will perform well if they are properly anchored. Cabinet dimensions and material gauge need not exactly match the ANSI Standard.

All the conclusions, limitations and bounding spectrum for motor control centers are applicable to low-voltage switchgear.

(4) METAL-CLAD SWITCHGEAR

Metal clad switchgear consists primarily of circuit breakers and associated relays (such as over-current relays or ground fault protection relays), interlocks, and other devices to provide protection to the equipment that it services.

Metal clad switchgear of 2.4kV and 4.16kV is considered. The general configuration of cabinets must be similar to those specified in ANSI C37.20 Standards. This requirement is imposed to preclude unusual designs not covered in the data base. The SSRAP feels that cabinets which are configured similar to ANSI Standards will perform well if they are properly anchored. Cabinet dimensions and material gauges need not exactly match ANSI Standards.

All the conclusions, limitations and bounding spectrum for motor control centers are applicable to metal-clad switchgear, except that the cutouts in the cabinet sheathing shall be less than about 12 inches by 12 inches.

(5) MOTOR-OPERATED VALVES

Motor-operated valves consist of an electric motor and gear box cantilevered from the valve body by a yoke and interconnected by a drive shaft. The motor and gear box serve as an actuator to operate the valve.

Based on a review of the data base and anticipated variations in conditions, it is felt that motor-operated valves are sufficiently rugged to survive a seismic event and remain operational thereafter provided the following conditions exist in the nuclear facility:

- The spectra for the nuclear facility are less than the appropriate bounding spectrum described in Fig. A.1 , for frequencies above the estimated fundamental frequency of the piping-valve system.

- The valve is located less than 40 feet above exterior grade. If the valve is located higher than 40 feet above exterior grade, the floor spectra shall be compared with 1.5 times the appropriate bounding spectrum.

- The valve body and yoke construction is not of cast iron.

- The valve is mounted on at least a 2 inch pipe.

- The actuator is supported by the pipe and not independently braced to or supported by the structure unless the pipe is also braced immediately adjacent to the valve to a common structure.

The following limitations on operator weight and eccentric length relative to pipe diameter are derived from the data base for motor-operated valves that was provided by SQUG. The data base contains relatively few heavy operators and small pipe diameters subjected to severe ground shaking. These limitations could be less restrictive if more motor-operated valves had been located and documented in the areas of higher shaking. It is felt that additional data, either from other earthquake experience or seismic qualification tests, can expand the scope of these recommendations.

- Type A bounding spectrum shall be used for the following cases: (See Figure A.2)

  Valves mounted on 12-inch diameter or larger pipes with a 60 inch or less distance from the pipe centerline to the top of the motor actuator and the approximate actuator weight is less than 400 pounds.

Valves mounted on 24-inch diameter or larger pipes with a 100 inch or less distance from the pipe centerline to the top of the motor actuator and the approximate actuator weight is less than 300 pounds.

- Type C bounding spectrum shall be used for the following cases: (See Figure A.3).

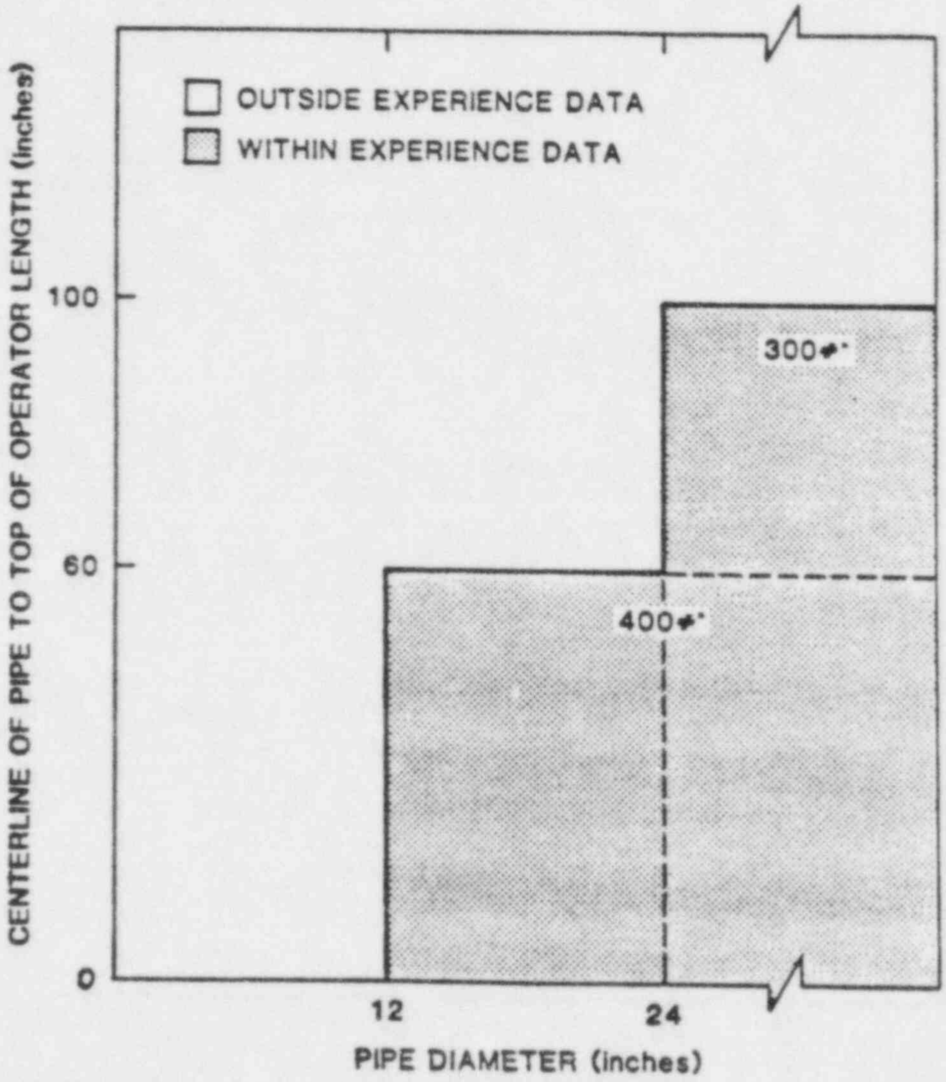
  Valves mounted on a pipe diameter of at least 2 inches but less than 6 inches, with a 30 inch or less distance from the pipe centerline to the top of the motor actuator, and the approximate actuator weight is less than 100 pounds.

  Valves mounted on a pipe diameter of at least 6 inches but less than 8 inches, with a 40 inch or less distance from the pipe centerline to the top of the motor actuator, and the approximate actuator weight is less than 300 pounds.

  Valves mounted on a pipe diameter of at least 8 inches but less than 10 inches, with a 50 inch or less distance from the pipe centerline to the top of the motor actuator, and the approximate actuator weight is less than 400 pounds.

  Valves mounted on a pipe diameter of at least 10 inches with a 70 inch or less distance from the centerline of the pipe to the top of the motor actuator, and the approximate actuator weight is less than 640 pounds; or weigh more than 300 pounds for cases where the distance from the centerline of the pipe to the top of the motor actuator is not greater than 100 inches.

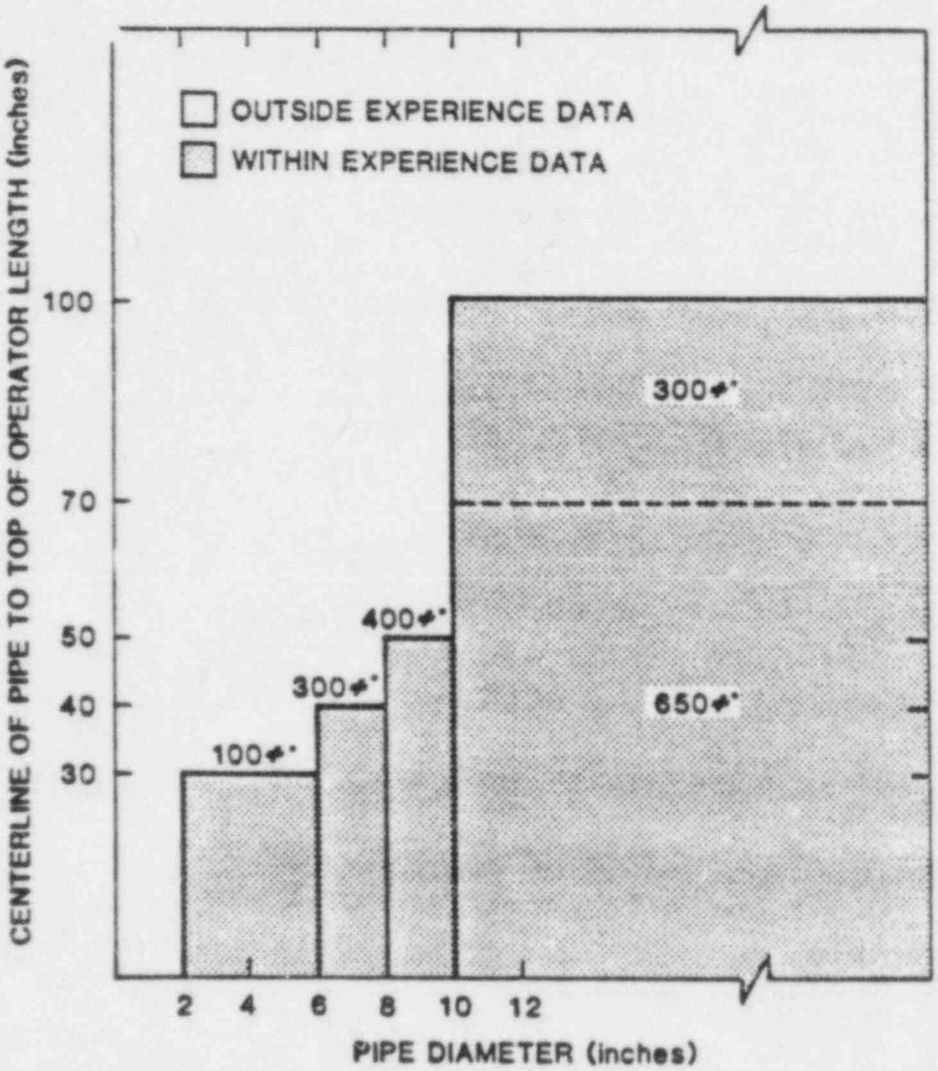
For motor-operated valves not complying with the above limitations, the seismic ruggedness for ground motion not exceeding the Type A bounding spectrum may be demonstrated by static tests. In these tests, a static force equal to three times the approximate operator weight shall be applied non-concurrently in each of the three orthogonal principal axes of the yoke. Such tests should include demonstration of operability following the application of the static load. The limitations other than those related to the operator weight and distance from the top of the operator to the centerline of the pipe, given above, shall remain in effect.



*APPROXIMATE MAXIMUM OPERATOR WEIGHT

Figure A.2

Motor Operated Valves
For Which Type A Spectrum Is To Be Used



*APPROXIMATE MAXIMUM OPERATOR WEIGHT

Figure A.3

Motor Operated Valves
For Which Type C Spectrum Is To Be Used

(6) UNIT SUBSTATION TRANSFORMERS

Unit substation transformers convert the distribution voltage to low voltage.

Unit substation transformers which convert 2.4kV or 4.16kV distribution voltages to 480V are considered.

Based on a review of the data base and anticipated variations, it is felt that unit substation transformers are sufficiently rugged to survive a seismic event and remain operational thereafter provided the following conditions exist in the nuclear facility:

- The spectrum for the nuclear facility is less than the Type B bounding spectrum described in Fig. A.1, for frequencies above the estimated fundamental frequency of this equipment, and the unit substation transformer is located less than 40 feet above exterior grade. If the unit substation transformer is located higher than 40 feet above exterior grade, the floor spectra shall be compared with 1.5 times the bounding spectrum. In all cases a comparison with floor spectra is also acceptable.

- Both unit substation transformer enclosures and the transformer itself must have engineered anchorage.

The functionality of properly anchored unit substation transformers during and after earthquakes, as defined above, has been demonstrated.

(7) AIR-OPERATED VALVES

Air-operated valves consist of a valve (controlled by a solenoid valve) operated by a rod actuated by air pressure against a diaphragm attached to the rod. The actuator is supported by the valve body through a cantilevered yoke.

Based on a review of the data base and anticipated variations in conditions, it is felt that air-operated valves are sufficiently rugged to survive a seismic event and remain operational thereafter provided the following conditions exist in the nuclear facility:

- The ground motion spectra for the nuclear facility are less than the Type A bounding spectrum for frequencies above the estimated fundamental frequency of the piping-valve system.

- The valve body is not of cast iron.

- The valve is mounted on a pipe of 1 inch diameter or greater.

- If the valve is mounted on a pipe of less than 4 inch diameter, the distance from the centerline of the pipe to the top of the operator shall not exceed 45 inches. If the valve is mounted on a pipe of 4 inch diameter or greater, the distance from the centerline of the pipe to the top of the operator shall not exceed 60 inches. See Figure A.4.

- The actuator and yoke is supported by the pipe and neither is independently braced to the structure or supported by the structure unless the pipe is also braced immediately adjacent to the valve to a common structure.

The air supply line is not included in this assessment.

For air-operated valves not complying with the above limitations the seismic ruggedness for ground motion not exceeding the Type A bounding spectrum may be demonstrated by static tests. In these tests, a static force equal to three times the approximate operator weight shall be applied non-concurrently in each of the three orthogonal principal axes of the yoke. Such tests should include demonstration of operability following the application of the static load. The limitations other than those related to the distance of the top of the operator to the centerline of the pipe, given above, shall remain in effect.

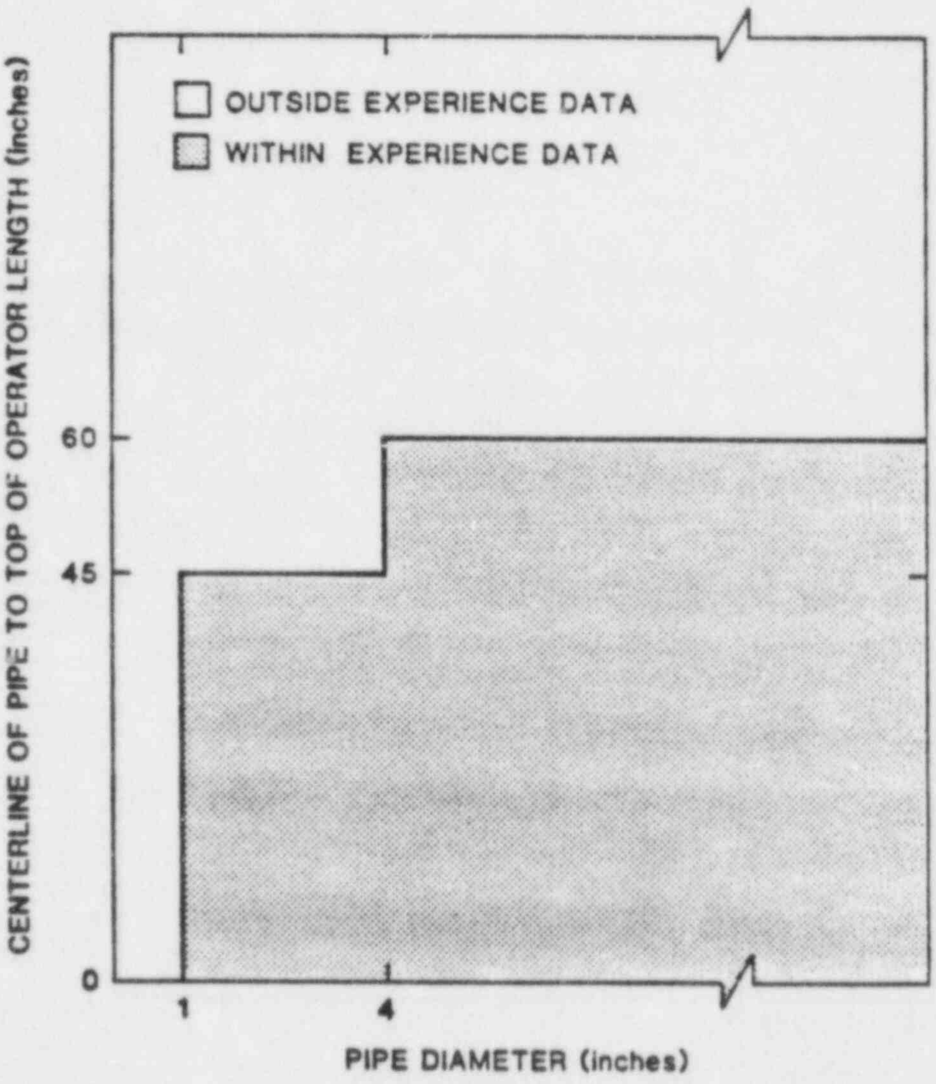


Figure A.4

Air-Operated Valves
For Which Type A Spectrum Is To Be Used

(8) HORIZONTAL AND VERTICAL PUMPS

It is felt that horizontal pumps in their entirety, and vertical pumps above their flange are relatively stiff and very rugged devices due to their inherent design and operating requirements. Motors for these pumps are also included. Subject to the limitations set forth below, all pumps meet the criteria for the Type A bounding spectrum.

For horizontal pumps, one must assure that the driver (electric motor, turbine, etc) and pump are rigidly connected through their base so as to prevent damaging relative motion. Of concern are intermediate flexible bases; these must be evaluated separately. Thrust restraint of the shaft must also be assured in both axial directions. The data base covers pumps up to 2500 hp. However, the SSRAP feels that the conclusions are equally valid for horizontal pumps of greater horsepower.

For vertical pumps, the data base has many entries up to 700 hp and several up to 6000 hp. However, it is felt that vertical pumps, above the flange, of any size at nuclear plants are sufficiently rugged to meet the Type A bounding spectrum.

The SSRAP feels that the variety of vertical pump configurations and shaft lengths, below the flange, and the relative small number of data base points in several categories, preclude the use of the data base to screen all vertical pumps. Vertical turbine pumps, i.e. deep well submerged pumps with cantilevered casings up to 20 feet in length and with bottom bearing support of the shaft to the casing are well enough represented to meet the bounding criteria below the flange as well. It is recommended either individual analysis or use of another method as a means of evaluating other vertical pumps below the flange. The chief concerns would be damage to bearings due to excessive loads, damage to the impeller due to excessive displacement, and damage due to inter-floor displacement on multi-floor supported pumps.

7. Guidance on Review of Equipment Which Exists in the Experience Data Base Plants but Which Are Not Included in the Eight Types in the Data Base

Based on the above experience and reviews conducted by the staff in the SEP Program and licensing activities (SQRT audit) and the observation of the behavior of equipment beyond the eight classes found in the data base plants, seismic adequacy of equipment other than the eight types can be achieved by (1) anchorage verification, (2) a careful review of caveats, outliers and exclusions observed during licensee reviews, SEP reviews and SQRT audits, and (3) documentation by SQUG of the basis for seismic adequacy of each equipment type. In addition, the SQUG with SSRAP assistance is compiling a checklist of caveats and exclusions. Typical caveats that should be reviewed during the walk-through are presented below. The final detailed walk-through guidelines and procedures including the list of caveats will be developed by the SQUG and reviewed and endorsed by SSRAP and the staff before implementation.

(1) Diesel generators and associated equipment: The airlines and oil lines in several of nuclear plants reviewed were identified to be excessively flexible. These lines should be supported in such a way that they will not be damaged during earthquake.

(2) Battery chargers and inverters: These items should be treated similarly to the motor control centers (MCC), namely, similar exclusions and caveats for MCC should apply here.

(3) Distribution panel (AC and DC): If the panel is a cabinet, it should be treated similarly to the MCC's. If the panel is a cantilevered frame rack, in addition to the anchorage requirement, the stiffness and displacement of the rack should be adequately assessed.

(4) Attachment of components inside other equipment: Equipment such as control panels, distribution panels and the like have numerous devices and components attached therein. During walk-through, attention should be paid to make sure that these devices or components are securely attached to the equipment. In other cases, these devices and component may be attached to trays, in this case attention should be paid to make sure that these trays are securely attached to the equipment.

(5) Interference between equipment: During walk-through, care should be taken to assure that there is enough space for seismic motion such that damage will not result due to impact between adjacent equipment.

Bounding spectra for equipment not included in the eight types in the data base but which exist in the data base plants, and equipment that is unique to nuclear plants will be defined as part of the detailed procedure to be developed by the SQUG with SSRAP review and approval by the NRC.

For individual utilities not participating in the Generic Group, the detailed procedures used to review the seismic adequacy of equipment not included in the eight types in the data base should be submitted to the NRC for review. Items such as equipment caveats and exclusions, bounding spectra to be used, and the like should be included in the submittal.

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