GA-C16708

GENERAL ATOMIC

INDEPENDENT VERIFICATION OF SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3 SEISMIC DESIGN AND QUALITY ASSURANCE PROGRAM EFFECTIVENESS

VOLUME 2
PROGRAM RESULTS

PREPARED FOR
SCE *Southern California Edison Company*

APRIL 5, 1982

TORREY
PINES
TECHNOLOGY

A DIVISION OF GENERAL ATOMIC COMPANY

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FOREWORD

This report is presented in three volumes:

Volume 1 - Executive Summary
Volume 2 - Program Results
Volume 3 - Potential Finding Reports
    Book 1 - Reports 0001 to 0058 and F001 to F040
    Book 2 - Reports F041 to F112 and Corrective Action Plans

Volume 1, Executive Summary, is a complete overview of the program, the work performed, and the major conclusions drawn.

Volume 2, Program Results, gives a more complete description of the program, particularly of the actual work performed, the questions raised during the review, the resolution of these questions, and the final conclusions associated with each part of the program. This volume is designed to give a thorough overview of the complete program.

Volume 3, Potential Finding Reports, is a compilation of all of the questions raised during the review and the Corrective Action Plans together with the review of those Corrective Action Plans. For convenience of handling, this volume has been divided into two books. Book 1 contains PFR's 0001 through 0058, the reports that were filed before the Interim Report was issued, and PFR's F001 through F040. Book 2 contains the remaining PFR's, F041 through F112, and the Corrective Action Plans that were prepared by Southern California Edison in response to the seven Findings. This document does not include program discussions, description of the work, or any conclusions.

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1. INTRODUCTION

General Atomic Company was engaged by Southern California Edison Company (SCE) to conduct an independent review of the seismic design of its San Onofre Nuclear Generating Station Units 2 and 3, including an assessment of the effectiveness of the quality assurance program for design. This is the final report for that program, which began in late November, 1981.

1.1. GENERAL PROGRAM STRUCTURE

The program was structured to verify that the design process adequately converted the seismic design bases specified in the Final Safety Analysis Report (FSAR) into design documents that were transmitted to the constructor or the fabricator. The major tasks (A, B, and C) were designed to do this verification by first reviewing all the procedures used in the design process to determine the adequacy of the basic process, then reviewing a sample of the points where the procedures should have been implemented to assure that they were in fact practiced, and finally technically reviewing the final design documents that were the products of the design process. The plan taken in its entirety provided a discerning basis on which the adequacy of the seismic design could be judged. The program effort applied to Tasks A, B, and C was 8, 18, and 85 man-months.

Calculational aspects of the design process were reviewed in Task H. Independent calculations using alternative analytical techniques were performed on two features that were reviewed in Task C and that had a relatively low design margin. Three man-months were expended on this task.

Two other tasks (D and G) reviewed aspects of the construction process. Task D reviewed the plan for field audits and Task G reviewed the as-built configuration of a segment of pipe. Seven man-months were expended on Task D and one man-month on Task G.

Task E covered processing of the PFR's and Task F pertained to preparation of the program reports. These two tasks, together with program management, required 55 man-months. The total program effort was 177 man-months.

This volume presents the details of the work that was performed in this review. The work performed in each of the six review tasks (A, B, C, D, G, and H) is discussed in Sections 2 through 7. Section 8 provides an overview of the Potential Finding Reports (PFR's) that were filed in the review program, and presents the overall conclusions of the program. Tasks E and F are not discussed because they were administrative tasks.

1.2. GENERAL ATOMIC QUALIFICATIONS AND INDEPENDENCE

General Atomic Company, through its Torrey Pines Technology (TPT) Division, brought significant qualifications to its task of evaluation for SCE. General Atomic Company has been in the nuclear power plant industry for more than 20 years and has a large staff of capable, experienced, technically trained personnel. In addition, General Atomic operates under the first NRC-approved Quality Assurance Progam and has acknowledged expertise in quality assurance. This seismic design evaluation for SCE was conducted under the provisions of this Quality Assurance Program.

General Atomic Company and all its personnel on this program are independent of SCE and San Diego Gas & Electric Company (SDGE), the major owners of San Onofre Units 2 and 3. Revenues from SCE and SDGE are not a significant portion of General Atomic's revenues. No person working on this program has a significant financial interest in SCE or SDGE, nor does any person have any family member who is presently employed by SCE or SDGE or who is engaged directly or indirectly in the design or construction of San Onofre Units 2 and 3.

1.3. EVALUATION PROCESS

One of the key elements in this program was the individual reviewer. Technical and Quality Assurance personnel were assigned specific items to review. It was their responsibility to request the proper documents from the original design organizations [SCE, Bechtel Power Corporation (BPC), and Combustion Engineering, Inc. (CE)] and to perform the necessary analysis to satisfy the requirements of the review. When an apparent deviation or discrepancy was uncovered, the reviewer filed a PFR and carried the report through to its final disposition.

Throughout the review and the processing of PFR's, emphasis was placed on the independence of the reviewer. A reviewer required no approval to file a PFR, and once filed the PFR was required to be fully processed unless the subject was determined to be outside the scope of the evaluation program.

To guide the reviewer in his work, a set of review procedures was written for each task and for the processing of PFR's. Table 1-1 lists these procedures. The procedures established the scope of the work and provided guidelines for conducting the review. Copies of the appropriate procedures were provided to all personnel who were designated as task reviewers Meetings were held by the task leaders to discuss these procedures with the reviewers. Also, training sessions were held to familiarize reviewers with the procedures for filing and processing PFR's.

A QA Program Document (QAPD) was prepared to describe the QA requirements governing work under this project. The QAPD was distributed to project personnel. The QA program included internal audits performed by the General Atomic QA division to evaluate compliance with the QAPD, with project procedures, and with the program plan. The audits indicated that compliance was satisfactory.

TABLE 1-1
PROJECT PROCEDURES

Procedure No.	Name
1	Review of Design Procedures (Task A)
2	Review of Design Procedure Implementation (Task B)
3	Processing of Findings (Task E)
4	Review of Audit Plans and Schedules (Task D)
5	Procedure for Field Auditing Piping, Isometrics, and Support Drawings (Task G)
6	Procedure for Seismic Design Technical Review (Task C)
7	Procedure for Performing Independent Calculations (Task H)

1.4. PROCESSING OF POTENTIAL FINDING REPORTS

The main features of the procedure for processing PFR's are shown in Fig. 1-1. Reviewers for tasks A, B, C, D, G, and H filled out a PFR when they encountered an apparent deviation that met the definition of a Potential Finding contained in their review procedures. All PFR's were maintained as permanent records and were reviewed by the task leader and the original design organization (SCE, BPC, or CE) to determine if the PFR was valid (i.e., if it was accurate, well defined, and traceable to the requirement). Independence of the initiator was maintained by giving the initiator the sole right to reject or incorporate comments by either the task leader or the original design organization (ODO). Records of each revision to a PFR were maintained and all comments, whether incorporated or rejected, were documented.

After review by the task leader and the ODO, the PFR was sent (together with an Impact Assessment for valid PFR's) to the Findings Review Committee for evaluation and classification. The Impact Assessment was the initiator's appraisal of the seriousness of the Potential Finding.

A Potential Finding was classified as invalid if after the above-described review, the initiator, the task leader, and the ODO agreed that the Potential Finding was inaccurate. In addition, Potential Findings could be classified as invalid if two of the above-identified three reviewers concluded that the Potential Finding was invalid and the Findings Review Committee also decided it lacked validity.

The review procedure contained criteria for classifying a valid Potential Finding as either a Finding or an Observation. Basically, if a Potential Finding was a deviation that could result in a substantial safety hazard, or if there was an indication of a repetitive or generic deviation that could create a substantial safety hazard, the Potential Finding was classified as a Finding. Potential Findings that were valid, but that did not satisfy the above criteria for a Finding, were classified as Observations.

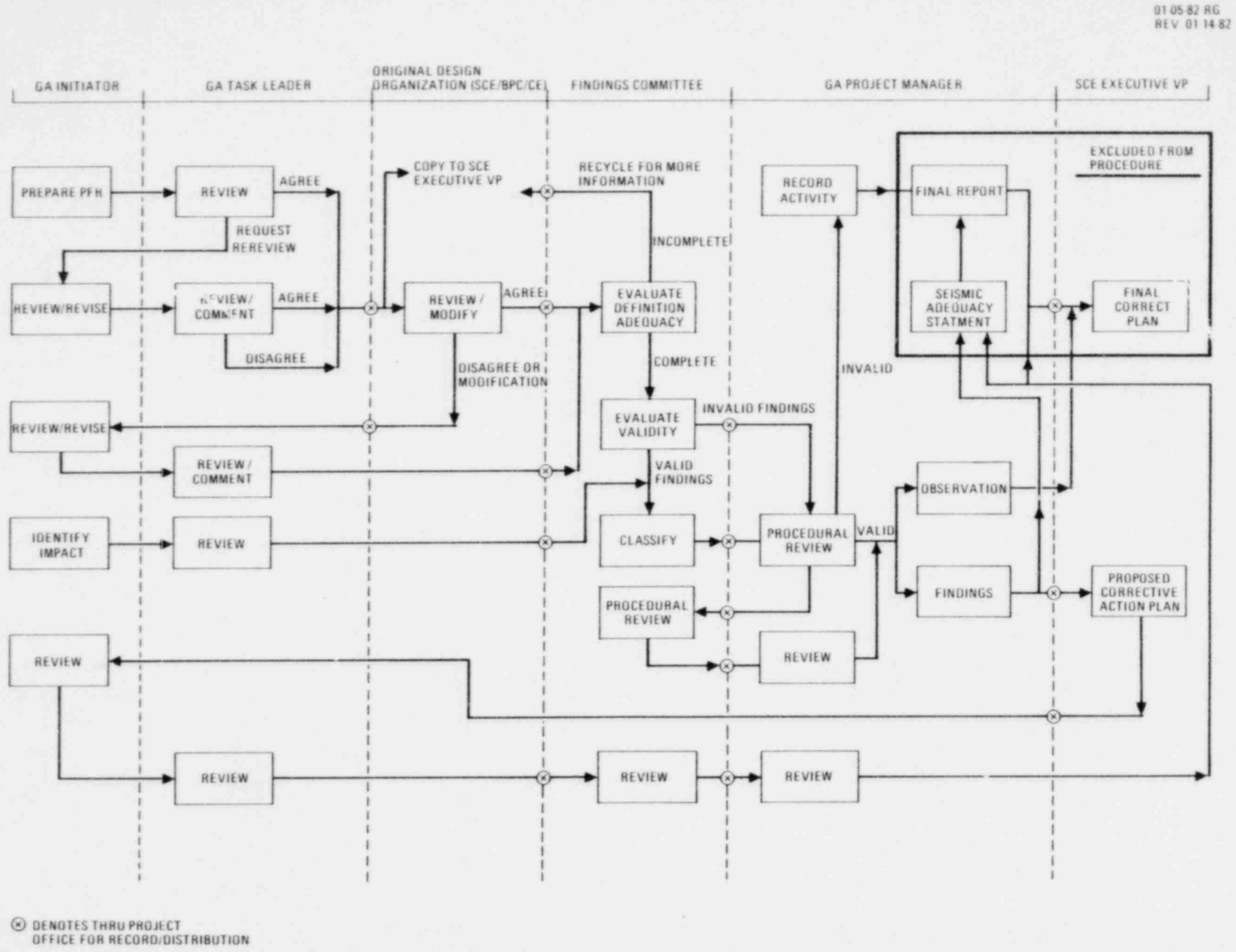

Fig. 1-1. Procedure for processing Potential Finding Reports

The classification of the Potential Finding was reviewed by the Project Manager to determine if the correct procedures had been followed. Subsequently, the Observations and Findings were sent to the Executive Vice-President of SCE for resolution. In the case of Findings, a Corrective Action Plan was prepared by SCE and returned for review. The review determined if the Corrective Action Plan satisfied the concern expressed in the Finding.

2. DESIGN PROCEDURE REVIEW, TASK A

This task was designed to determine if the design process used by SCE CE, and BPC was adequate. This was accomplished by determining if each organization had design control procedures in place during the design phase, determining if these procedures addressed commitments in the NRC-approved QA programs, and identifying the specific manuals and procedures that applied to the design activities at each organization. General guidance from Title 10, Code of Federal Regulations, Part 50 (10CFR50) and American National Standards Institute (ANSI) N45.2 was used to interpret and supplement these programs.

The scope of this task included the following:

- Preparation of procedures to provide detail working instructions for the review.
- Preparation of a description of the design control procedure structure.
- Performance of the review of current procedures for conformance to program commitments.
- Summarization of the design control process used by each organization.
- Evaluation of selected design control procedure revisions for compliance with the PSAR.

2.1. DESIGN CONTROL PROCEDURE STRUCTURE

A detailed description was developed of the structure of the design control procedures applicable to seismic design work performed by SCE, CE, and BPC. The approach taken was to examine the design control procedures, since these procedures included the seismic design work and since there are generally no procedures exclusively devoted to seismic design. Therefore, although this report is devoted to seismic design, this task addresses the design process in general.

These procedure structure descriptions were developed by visits to the SCE, CE, and BPC design offices, through interviews with staff members, by telephone contact with cognizant SCE, CE, and BPC personnel, and by reading Appendix A of the Preliminary Safety Analysis Report (PSAR) and the relevant manuals and procedures of SCE, CE, and BPC.

Figures 2-1 through 2-5 are summaries of the procedure structures for each of these organizations. Combustion Engineering centralized its design control procedures on May 3, 1976, and consolidated the design control QA requirements into one manual, the Quality Assurance of Design Manual (QADM). Before this date, each department maintained its own procedures. Thus, in the case of CE, the procedures in effect prior to May 3, 1976, were considered to represent the system in effect up to that time, and the QADM was considered to represent the system in effect after that time. These two systems are shown schematically in Figs. 2-2 through 2-4. Southern California Edison and BPC each had one system in place throughout the design period, as shown in Figs. 2-1 and 2-5, respectively.

The basis for acceptability of this review was Appendix A of the PSAR and (in the case of CE) the CE QA Topical Report (CENPD-210-A). These sources were supplemented by 10CFR50 Appendix B and ANSI N45.2.11.

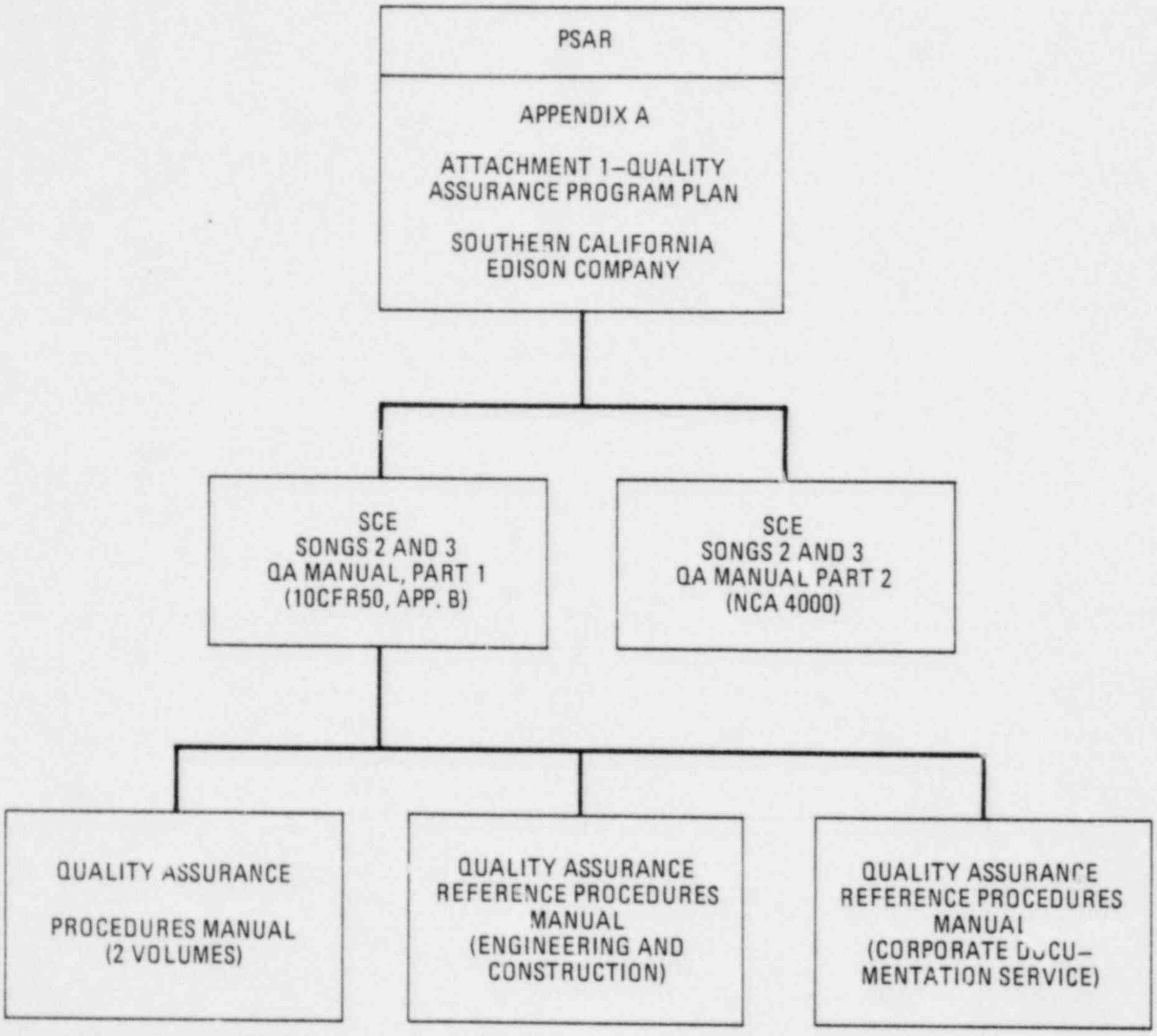

Fig. 2-1. SCE design control procedure hierarchy

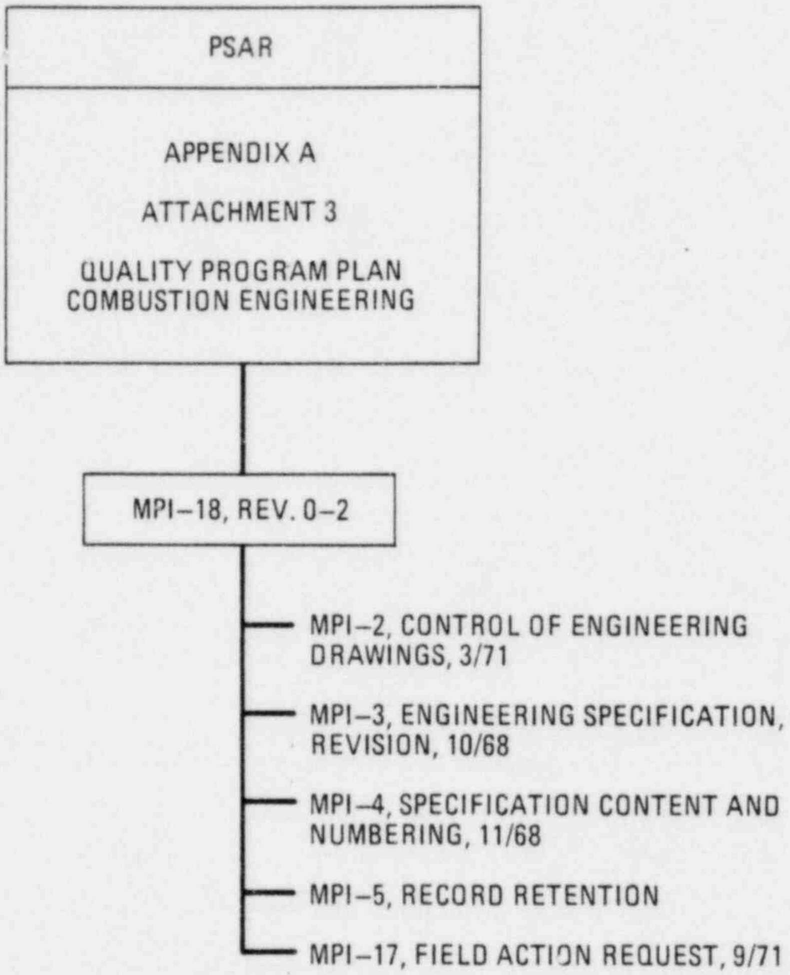

Fig. 2-2. CE design control procedures - 12/69 to 5/74

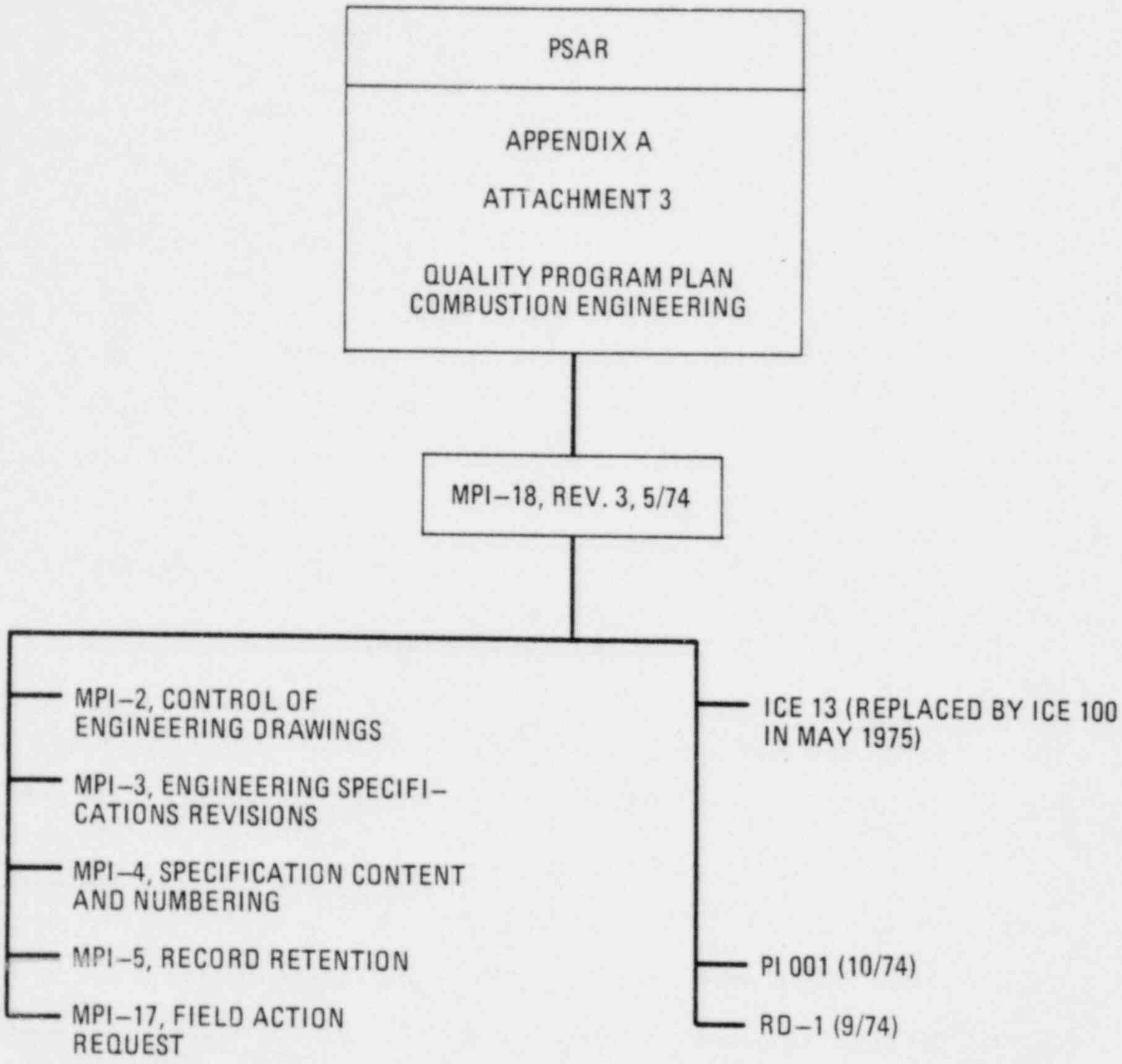

Fig. 2-3. CE design control procedure hierarchy - 5/74 to 5/3/76

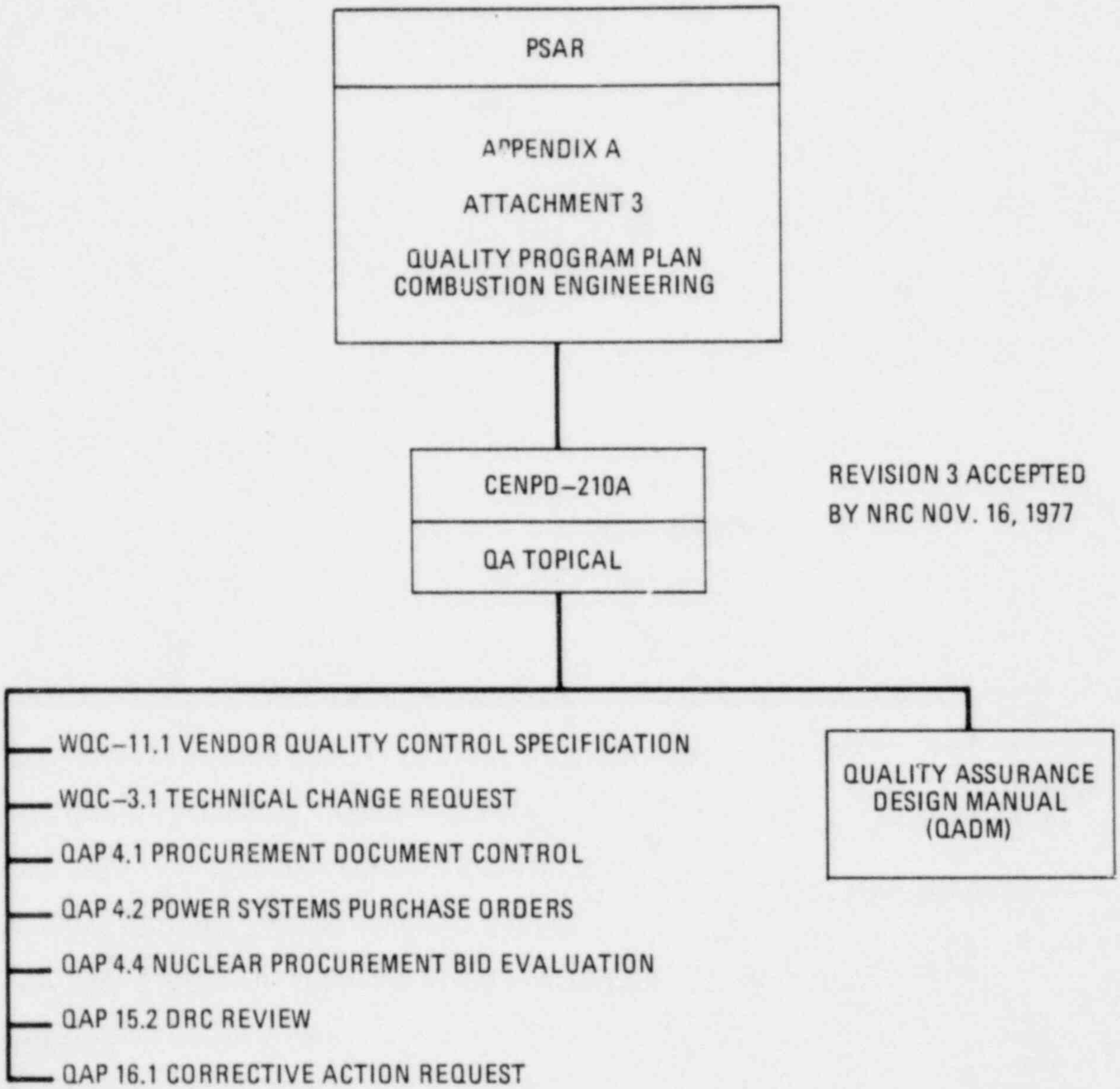

Fig. 2-4. CE design control procedures heirarchy - 5/3/76 to present

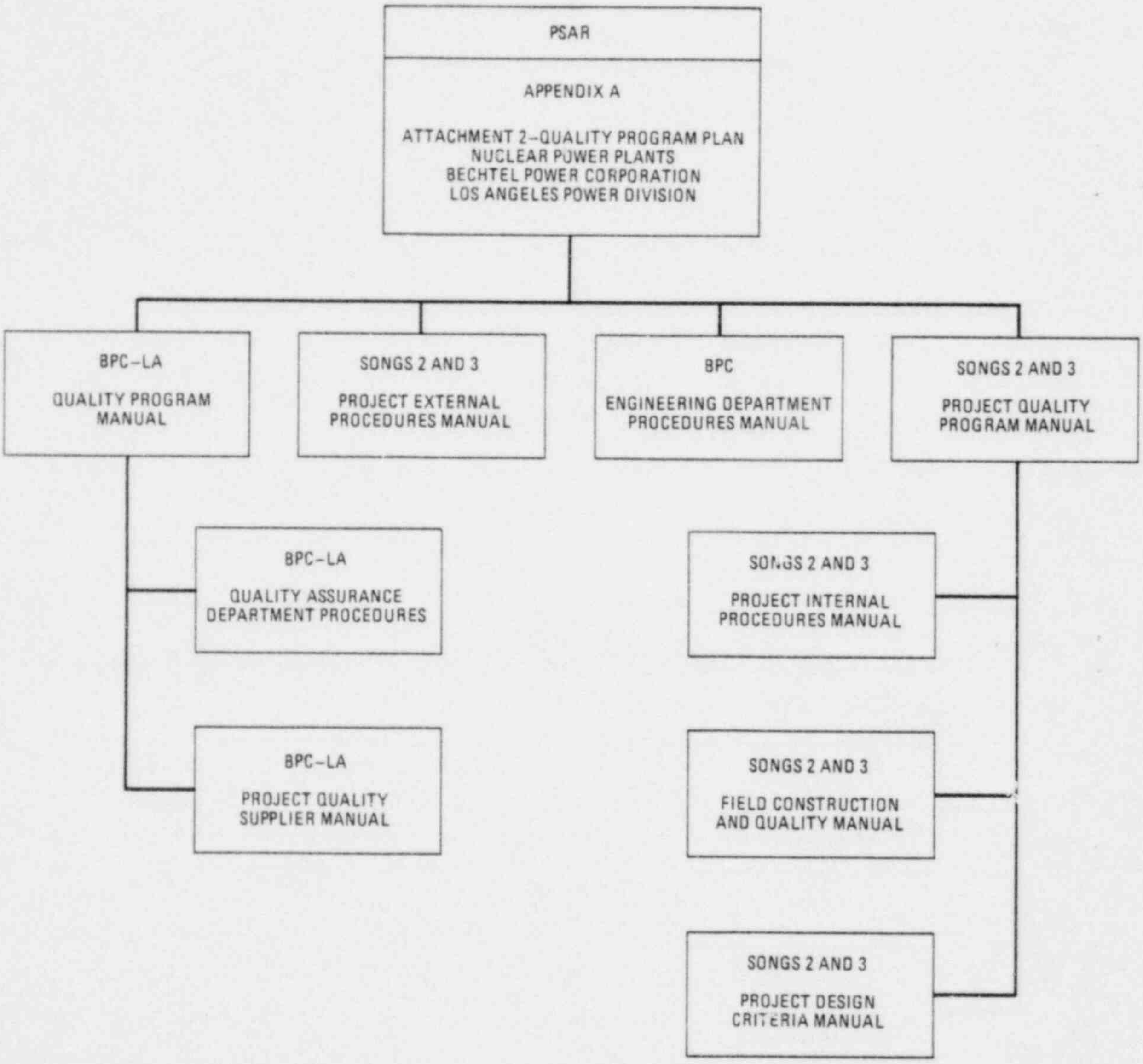

Fig. 2-5. Bechtel design control procedures hierarchy

2.2. PROCEDURE REVIEW

The relevant design control procedures were obtained. Tables 2-1 through 2-4 list the 32 procedures and manuals obtained from SCE, CE, and BPC.

These procedures were reviewed to determine if the QA program commitments listed in the PSAR and in the CE Topical Report were adequately addressed in the working-level procedures.

The first step was to prepare a procedure to provide working instructions to the Task A personnel.

The review consisted of the following:

1. Identifying and extracting the design control commitments made by SCE, CE, and BPC in Appendix A of the PSAR (and also by CE in its Topical Report).
2. Entering the commitments as checklist questions on a specially prepared checklist form.
3. Examining the manuals in detail and recording on the checklist the specific section(s) in which each commitment was addressed.
4. Indicating on the checklist whether or not the commitment was adequately addressed. In those cases in which judgement had to be exercised as to adequacy (e.g., the manuals contained wording or phrasing similar to, but not exactly the same as, that used in the PSAR), comments were added to justify the reviewer's decision as to adequacy. The documentation of the review is in the form of completed checklists and the procedure that describes the review process.

TABLE 2-1
SONGS UNITS 2 & 3 QA PROGRAM BASE DOCUMENTS
(Used as Basis for Task A Review)

PSAR - Appendix A, Amendment 20 (3/24/74)

Attachment 1 - QA Program Plan, SCE

Attachment 2 - Quality Program Plan, Nuclear Power Plants, BPC, LA Power Division

Attachment 3 - Quality Program Plan, CE

PSAR Deviation Numbers E19, E22-E23, E26-E32, E34-44, B97, B102, B11-B113

CE QA Topical Report, "Quality Assurance Program," CENPD-210A, Rev. 3, November 1977

TABLE 2-2
SCE DOCUMENTS FOR TASK A REVIEW

SONGS[a] 2&3 QA Manual, Part 1
(Chapter 3, all issues since 1977)

SONGS 2&3 QA Manual, Part 1

SONGS 2&3 QA Manual, Part 2

QA Reference Procedures Manual
(Engineering and Construction)

QA Reference Procedures Manual
(Corporate Documentation Services)

QA Reference Procedures Manual
(QA Sections N3.01, N3.03, N18.07, N18.08)

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[a] San Onofre Nuclear Generating Station

TABLE 2-3
BPC DOCUMENTS FOR TASK A REVIEW

Quality Program Manual, LA Power Division

QA Department Procedures Manual, LA Power Division

SONGS 2&3 Project Quality Program Manual, LA Power Division

Project Internal Procedures Manual, Volumes 1 and 2

Field Construction and QC Manual, Volumes 1, 2, 3, and 4

Project Design Criteria Manual

Project External Procedures Manual

TABLE 2-4
CE DOCUMENTS FOR TASK A REVIEW

QA of Design Manual

Preparation and Maintenance of MPIs, NPS-MPI-1

Control of Engineering Drawing, NPS-MPI-2

Specification Preparation and Revision, NPS-MPI-4

Technical Change Request (TCR), NPS-MPI-7

Deviation of Contract Requirements (DCR), NPS-MPI-8

Purchase and Manufacturing Prerequisitions and Supplements, NPS-MPI-10

Safety Analysis Report Preparations Procedures, NPS-MPI-19

QA of Design, NPS-MPI-18

Design QA Procedures - Reactor Design Department RD-1

Design QA Procedures Applicable to Plant Engineering NSSS Safety Related Design Activities, PE-QA-001

- Group QA Manual
  - QAP 4.1
  - QAP 4.2
  - QAP 4.4
  - QAP 15.2
  - QAP 16.1

Design Procedure, I&CE Procedure 100

Design Development and Review, I&CE Procedure 13

Quality Assurance of Design, I&CE Procedure 12

Vendor Quality Control Program Specification, WQC 11.1,

Submittal Instructions for Technical Change Request (TCR), WQC 3.1

This review determined that SCE, BPC, and CE each had design control procedures in effect for the plant design period. In the case of BPC and SCE, these procedures were found to satisfy the design control commitments made in the PSAR.

In the case of CE, the review uncovered several instances in which it appeared that a PSAR or a Topical Report commitment was not addressed in the CE internal procedures reviewed. Nine PFR's were written to cover these instances. Processing was suspended on three of these because they addressed procurement activities that were outside the program scope, and one was classified as invalid. Of the five valid PFR's two were classified as Observations and three as Findings. Further details of these PFR's are presented in Section 2.5.

2.3. DESIGN CONTROL PROCESSES

Summaries of the design control processes used by SCE, CE, and BPC were prepared. The summaries provided descriptions of the design control process within each organization, and were used to help train personnel working on the project under Task B. Using the background provided by these summaries, the Task B reviewers were able to perform their jobs more efficiently.

2.4. PROCEDURE REVISIONS

All procedure reviews described above were performed using the current revisions of procedures and, for CE, the procedures in effect immediately prior to the QADM (May 3, 1976).

Selected design control procedure revisions that applied during other time periods were reviewed for compliance with the applicable PSAR revision (and with the Topical Report in the case of CE). This work verified that the PSAR (and the Topical Report) commitments were implemented in working procedures and manuals throughout the design activity period, except as noted in the PFR's described in Section 2.2 above.

No additional PFR's were generated in this review.

2.5. CONCLUSIONS - TASK A

Five valid PFR's were issued under Task A. All related to CE activities. Table 2-5 lists the PFR's and shows the contractor, the subject, and the classification associated with each report.

PFR 0049 was classified as an Observation. The concern identified on the PFR was the possibility that an inadequate checklist was used to review specifications. However, it was determined that other procedures that were in effect minimized the significance of this possibility, as well as minimizing or eliminating any impact on design adequacy.

PFR F004 was classified an an Observation. The concern identified in the PFR was the lack of training requirements in a specific CE design department procedure. However, evidence was presented which demonstrated that adequate training was provided, even though there was no specific procedural requirement.

PFR 0038, 0047, and 0052 were classified as Findings. These PFR's all related to the lack of formal procedures to meet PSAR commitments pertaining to the project management coordination and committee review functions. PFR's 0038 and 0052 concern the lack of procedures to describe the interface and design coordination functions attributed to the Project Manager in the PSAR. PFR 0047 involves the lack of procedures to describe the design review function that the PSAR attributes to the Nuclear Safety Committee and the Chief Scientist.

All three PFR's were satisfactorily resolved by confirming that the required safety-related activities were properly and adequately carried out, even though they were not, in all cases, covered by formal QA procedures. The corrective action plans submitted by SCE in response to these findings demonstrated that many of the required safety-related activities described

TABLE 2-5
VALID PFR's ISSUED IN TASK A

PFR No.	Contractor	Subject	Classification
0038	CE	Project office - interface control	Finding
0047	CE	Design reviews	Finding
0049	CE	Specification review	Observation
0052	CE	Project Office - design coordination	Finding
F004	CE	Design control - training	Observation

in the PFR's were carried out, even though they were not in all cases covered by QA procedures. In addition, the review performed in Task B sampled implementation of issues covered by the PFR and verified that the work was performed satisfactorily. Finally, no evidence was found during the Task C review that would adversely impact on the design as a result of these procedural deficiencies. Current procedures cover these issues.

Based on the review performed in Task A, it is concluded that SCE, CE, and BPC each had design control procedures in place during the design process. The SCE and BPC procedures were adequate. The CE procedures were adequate except for the area of concern identified in the above-discussed PFR's. Although the CE procedures were not adequate in those areas, it was confirmed through the Corrective Action Plan and the Task B and C reviews that the functions in question were carried out, thus resolving the concern regarding safety impact of the procedural deficiency.

3. DESIGN PROCEDURE IMPLEMENTATION REVIEW, TASK B

Task B was designed to determine if the design control procedures in effect at SCE, CE, and BPC (as identified in Task A), were implemented in the design documents related to seismic design work.

The scope of this task included the following:

- Preparation of procedures to provide detail working instructions for the review.
- Identification of review items.
- Identification and location of pertinent design documents associated with these review items.
- Performance of the review for compliance with the governing procedures.

3.1. IDENTIFICATION OF REVIEW ITEMS

The points and steps in the design process that were to be checked for compliance with design control procedures were identified. Each point and step represented a document or documents (e.g., drawings, specifications, or calculations) associated with a specific component, structure, or system. The selection of points and steps was made using the information described in Section 2.4 and 4.1 of the report.

The following criteria were used to select the points and steps:

1. All points and steps associated with the features reviewed in Task C were to be included.

2. Additional safety-related points and steps were to be included to bring the total to about 200.

3. SCE/BPC/CE interfaces were to be included.

4. Work spanning the entire calendar period of the seismic design effort was to be included.

5. Work in all phases of the project was to be included.

6. All types of design documents were to be included.

7. Work within BPC and CE was to be included.

The total number of points and steps actually reviewed was 321.

3.2. DOCUMENT IDENTIFICATION

The pertinent design documents and governing procedures required to review the 321 steps described above were identified and located. This was accomplished in the course of the review by visits to the cognizant design offices and by requests for specific documents to be sent to the reviewer. Approximately 1,280 documents were identified.

3.3. IMPLEMENTATION REVIEW

The first step in this review was to develop a procedure of detailed working instructions for the review of design documents. This procedure included the use of checklists of procedural requirements for each type of design document. These checklists were used by the reviewers to ensure a thorough review of each document and to provide a record of the review. The review evaluated the compliance of seismic design activities, processes, and documents with the design control requirements called for by various manuals, procedures, and instructions.

The review was carried out by examining documents sent to the reviewers and by visits to the cognizant design offices to review documents, files, and records that supported the design process for each step or point in question. In addition, during these visits, interviews with design office personnel helped to support the review and to identify documents relevant to the review.

Each of the points and steps reviewed is documented by one or more completed checklists describing each check made for that point or step and describing the results of that check. Each document reviewed for each point or step (approximately 1,280 documents) is identified on the checklist. The types of documents examined are shown in Table 3-1. The total number of individual checks made in the course of the review exceeded 33,000.

The review resulted in the initiation of 52 PFR's. Sixteen of these were determined to be invalid and 36 to be valid. Of the valid PFR's 35 were classified as Observations and 1 as a Finding. Details of the PFR's are presented in Section 3.4.

3.4. CONCLUSIONS - TASK B

Thirty-six valid PFR's were issued under Task B. These are listed in Tables 3-2 and 3-3, organized by component and category, respectively. One PFR was classified as a Finding and 35 as Observations.

The single Finding (PFR F015) resulted from the accumulation of seven valid PFR's that were initiated against only two SCE design documents relating to lack of strict compliance with procedures in the design of the auxiliary intake structure. Individually, each of the seven PFR's were judged Observations, but collectively they were judged a Finding because they were repetitive similar procedural violations. PFR F015 served as the vehicle for transmitting this Finding. SCE provided acceptable corrective action responses for each of the PFR's that comprised the Finding. A review of all documentation in the SCE Corrective Action Plan demonstrated an understanding of the scope and importance of the problem, and provided a resolution or justification for each of the problems identified in the PFR's.

All PFR's associated with Finding PFR F015 were related to the auxiliary intake structure. This was the only seismic safety item designed by SCE, and the SCE design was reviewed for technical adequacy under Task C. That review showed the work to be satisfactory, eliminating any concern that the lack of compliance with procedures identified here had a safety impact.

Thirty-five PFR's were classified as Observations. Eight dealt with the SCE design process, including the seven mentioned above. These Observations comprised procedural violations that included deficiencies in documented design input requirements and design review, lack of documented reviews and approvals, and distribution of incomplete or unreleased design documents. These were classified as Observations because there was evidence that, although a procedural violation did occur, there was no adverse impact on plant safety.

Twenty-two of these Observations dealt with the BPC design process. These Observations comprised procedural violations in the following areas:

1. Calculations: format, insufficient reference to computer code validation information, references to applicable codes, standards, design criteria review and approval, timely completion.

2. Drawings: review and approval, issuance prior to completion of supporting calculations.

3. Design change notices: review and approval, timeliness of incorporation into design documents.

4. Design change review: input and authorization of SCE.

These were all classified as Observations because there was evidence to support the conclusion that there was no adverse impact on safety, and the nature of the PFR's was such that there was no trend indicated regarding lack of procedural compliance (i.e., PFR's were written against a wide variety of documents, components, or subjects.)

Five Observations pertained to the CE design process. Deficiencies included a lack of proper documentation of design input requirements for several components, a calculation that did not identify the method of verification, and design document approval forms that were not filed with the appropriate organization.

These were all classified as Observations because there was evidence to support the conclusion that there was no adverse impact on safety.

Based on the review peformed under this task, it was concluded that the design activities were carried out substantially in accordance with approved procedures. The deviations (Observations) found (and reported on PFR's) were within the limits of what can normally be expected in any major engineering project. That is, occasional procedural violations were identified, but they were not of the type that would have an adverse impact on plant safety. The deficiency in each of the valid PFR's was specifically evaluated for potential impact on plant safety. In each instance, documented justification demonstrated that there was no adverse impact on plant safety. The Finding against SCE was satisfactorily resolved since the item in question was the only seismic safety item designed by SCE, and the design was reviewed for technical adequacy in Task C.

TABLE 3-1
TYPES OF DOCUMENTS EXAMINED IN TASK B

Type	Description
1. Drawings	Sketches and preliminary drawings
	Design drawings
	Pipe support and hanger drawings
	Electrical drawings
	General arrangement drawings
	Piping & Instrumentation Diagrams (P&ID's)
	PLN's (planning logic networks) (CE)
2. Logs	Calculation control logs
	Drawing control logs
	Transmittal letter logs
	Specification control logs (including SCN's)
	Supplier deviation disposition request (SDDR) logs
	Field change request (FCR) manual logs
	FCR computer logs
	FCR delinquent lists
	Document distribution lists
	Pipe support logs
	Field change notice logs
3. Specifications	Purchase specifications
	Construction specifications
	ASME design specifications
	Design specifications
	Bills of material
	General specifications (CE)
	Project specifications (CE)

TABLE 3-1 (Continued)

Type	Description
4. Memoranda	Purchase memos Inspection memos Transmittal letters
5. Calculations	Computer code certifications (CE)
6. Files	FCR/FCN files Procurement files SCN Microfilm files Drawing (microfilm) files Personnel files Drawing control stick files Calculation files Purchase order files
7. Computer data	Computer printout and computer terminal display for information
8. Criteria	Design criteria Balance-of-plant design criteria
9. Procedures	Engineering Department procedures
10. Individual Documents	DRN's - Document Revision Notice SCN's - Specification Change Notice DCN's - Design Change Notice DCP's - Design Change Package SDDR's - Suppliers Deviation Disposition Request

TABLE 3-1 (Continued)

Type		Description
10.	Individual Documents (continued)	TCR's - Technical Change Requests (CE)
		DCR's - Deviation from Contract Requirement (CE)
		RAR's - Requests for Review and Approval (CE)
		FAR's - Field Change Requests (CE)
		Purchase orders
		Test requests (CE)
		Test procedure (CE)
		Vendor qualification reports
		Vendor surveys/audits
		NRC audits
11.	Computer Programs	Computer program validation lists
		Computer program verification reports
12.	Correspondence	BPC and SCE
		CE and BPC

TABLE 3-2
VALID PFR's ISSUED IN TASK B
(BY COMPONENT)

NOTE: All PFR's in this table were classified as Observations with the exception of the one Finding identified by an asterisk.

Component	PFR No.	Contractor	Subject
Auxiliary intake structure	F010	SCE	Purchase specification
	F011	SCE	Design document reviewer comments
	F012	SCE	Professional Engineer's approval of specification or calculation
	F013	SCE	Specification distribution
	F014	SCE	Review of revised calculation
	F015*	SCE	Computer program validation
Piping	F020	BPC	Calculations
	F021	BPC	Calculations
	F022	BPC	Calculations
	F023	BPC	Calculations
	F024	BPC	Calculations
	F027	BPC	Calculations
	F077	BPC	Computer codes
Pipe support	F056	BPC	Field change notice
	F066	BPC	Calculations
	F097	BPC	Calculations
Cable tray hangers	F029	BPC	Drawing review and approval
	F031	BPC	Drawing

TABLE 3-2 (Continued)

Component	PFR No.	Contractor	Subject
Containment	F034	BPC	Calculation
	F036	BPC	Calculation
	F044	BPC	Calculation
Tanks	F018	BPC	Specification
Boric acid makeup tank	F089	CE	Calculation, method of verification
Alarms	F075	BPC	Specification revision
Pump	F042	BPC	Specification change notice
Electrical penetrations	F060	SCE	Specification
Valves	F061	BPC	Specification change notice
Control board	F062	BPC	Supplier design change request
Generic	F016	SCE	Review and approval requirement for calculations
	F017	SCE	Quarterly listing of new and revised standards
	F035	BPC	Bechtel Design Criteria Manual review
	F079	CE	Design input requirements
	F080	CE	BOP design criteria
	F081	CE	Design basis information
	F086	BPC	Authorization design changes
	F088	CE	Design distribution/approval form

TABLE 3-3
VALID PFR's ISSUED IN TASK B
(BY CATEGORY)

NOTE: All PFR's in this table were classified as Observations with the exception of the one Finding identified by an asterisk.

Contractor	PFR No.	Subject	Component
Document Review, Verification, and Control			
SCE	F011	Design document	Auxiliary intake structure
	F012	Professional Engineer's approval of specification or calculation	Auxiliary intake structure
	F014	Review of revised calculation	Auxiliary intake structure
	F015*	Computer program validation	Auxiliary intake structure
	F016	Review and approval requirements for calculations	Generic
BPC	F022	Calculations	Piping
	F023	Calculations	Piping
	F027A	Calculations	Piping
	F029	Drawing review and approval	Cable tray hanger
	F031	Drawing	Cable tray hanger
	F035	Bechtel Design Criteria Manual review	Generic
	F042	Specification change	Feedwater pump
	F056	Field change notice	Pipe support
	F066	Calculations	Pipe support
	F077	Computer codes	Piping analysis
CE	F080	BOP design criteria	Generic
	F089	Calculation, method of verification	Boric acid makeup tank

TABLE 3-3 (Continued)

Contractor	PFR No.	Subject	Component
Document Maintenance			
SCE	F013	Specification distribution	Auxiliary intake structure
	F017	Quarterly listing of new and revised standards	Generic
	F060	Specification	Electrical penetrations
BPC	F018	Specification change notice	Storage tank
	F024	Calculations	Piping
	F061	Specification change notice	Valves
	F062	Supplier design change request	Control panel
	F075	Specification revision	Evacuation alarm
	F086	Authorization of design changes	Generic
	F097	Calculations	Pipe support
CE	F088	Document distribution/ approval form	Generic
Document Input Assumptions			
SCE	F010	Purchase specification	Auxiliary intake structure
BPC	F020	Calculations	Piping
	F021	Calculations	Piping
	F034	Calculations	Containment
	F044	Calculations	Containment
CE	F079	Design input requirements	Generic
	F081	Design basis information	Generic
Document Format			
BPC	F036	Calculation	Containment

4. SEISMIC DESIGN TECHNICAL REVIEW, TASK C

The objective of this task was to review the seismic design of selected safety-related structures, components, and systems of San Onofre Units 2 and 3 for compliance with the NRC-approved design basis and methodology specified in FSAR Sections 3.7 and 3.8.

The scope of this task included the following:

- Preparation of the seismic design chain networks for major safety systems of San Onofre Units 2 and 3.
- Preparation of the selection plan for use in choosing the features to be reviewed.
- Selection of the features to be reviewed.
- Preparation of the procedure to be used for performing the technical review.
- Performance of the detailed technical design review of the selected features.

4.1. SEISMIC DESIGN CHAIN NETWORKS

The seismic design chain networks (or the equivalent of the seismic interface chart described in Appendix B of ANSI N45.2.11-1974) illustrate the seismic-related design process associated with structures, components, and systems for San Onofre Units 2 and 3. The networks include the flow (input/output) of interface information between distinct design activities. The networks also identify the principal design organizations involved (SCE, BPC, and CE), including identification of design groups within these principal organizations at the design activity level. Major engineering service subcontractors involved are also identified.

For this program, seismic design chain networks were generated for nine safety-related systems as follows:

1. Safety Injection System (SIS).
2. Reactor coolant system and reactor internals.
3. Shutdown cooling system.
4. Component cooling water system.
5. Ultimate heat sink.
6. Containment spray system.
7. Chemical and volume control systems.
8. Reactor protection system.
9. On-site electric power systems.

The network titled "Site seismicity and soil-structure interaction" (Fig. 4-1) is the common starting point for all safety system networks. The seismic design chain network for each of the nine safety-related systems is shown in Figs. 4-2 through 4-10. These ten seismic design chain networks cover all the seismic design work performed by SCE, BPC, and CE.

These networks, in conjunction with the selection plan (Section 4.2), were utilized in choosing the features of San Onofre Units 2 and 3 to be subjected to a detailed design review. The networks were also used to identify the design process points and steps that were checked for compliance with the design control procedures discussed in Section 3.

4.2. SELECTION PLAN FOR PLANT FEATURES

The selection plan was prepared for use in choosing the features (i.e., plant structures, systems, segments of systems, components, and other equipment) of San Onofre Units 2 and 3 to be subjected to a detailed seismic review. The selection plan satisfied the criteria shown in Table 4-1 for

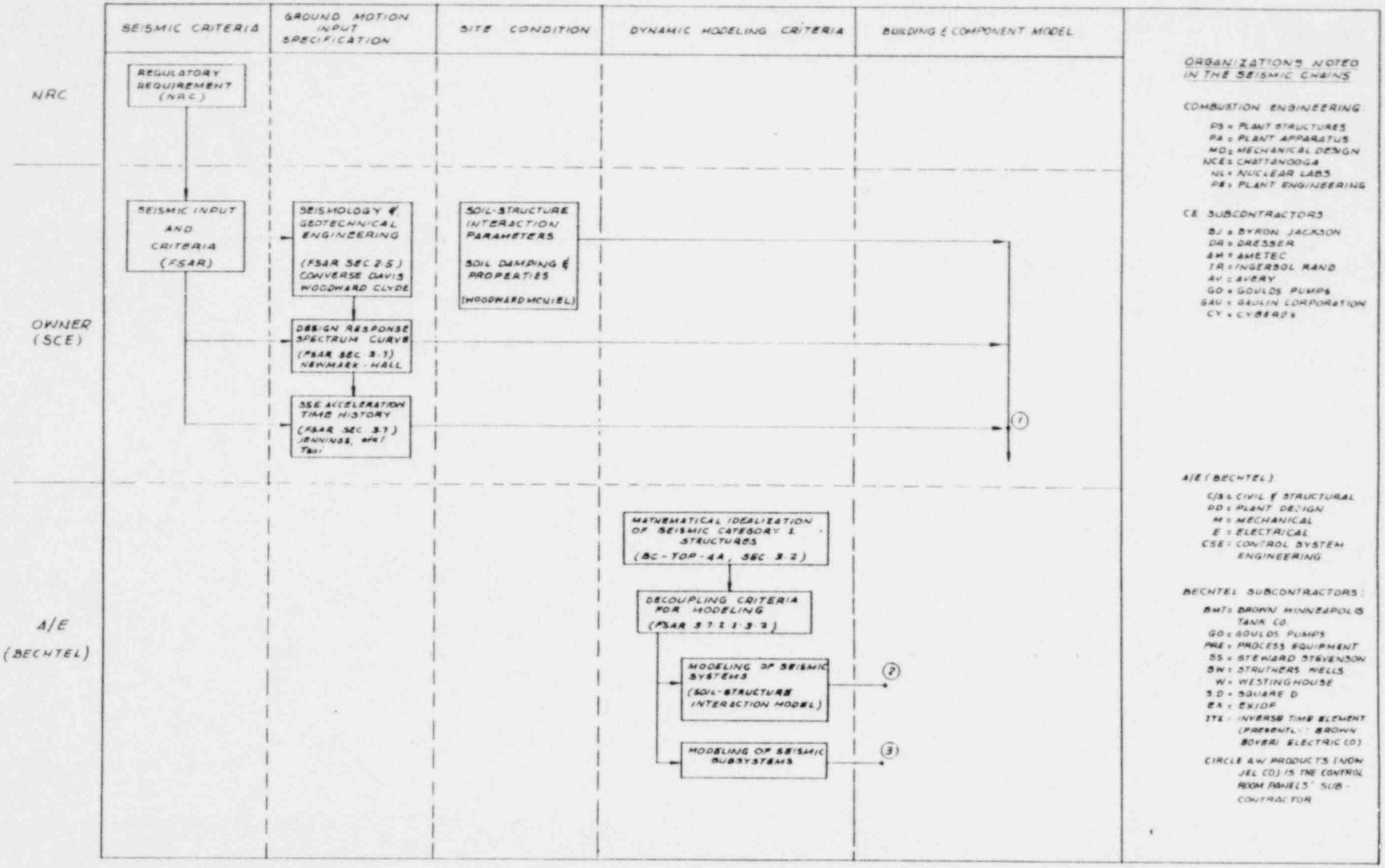

Fig. 4-1. Site seismicity and soil-structure interaction

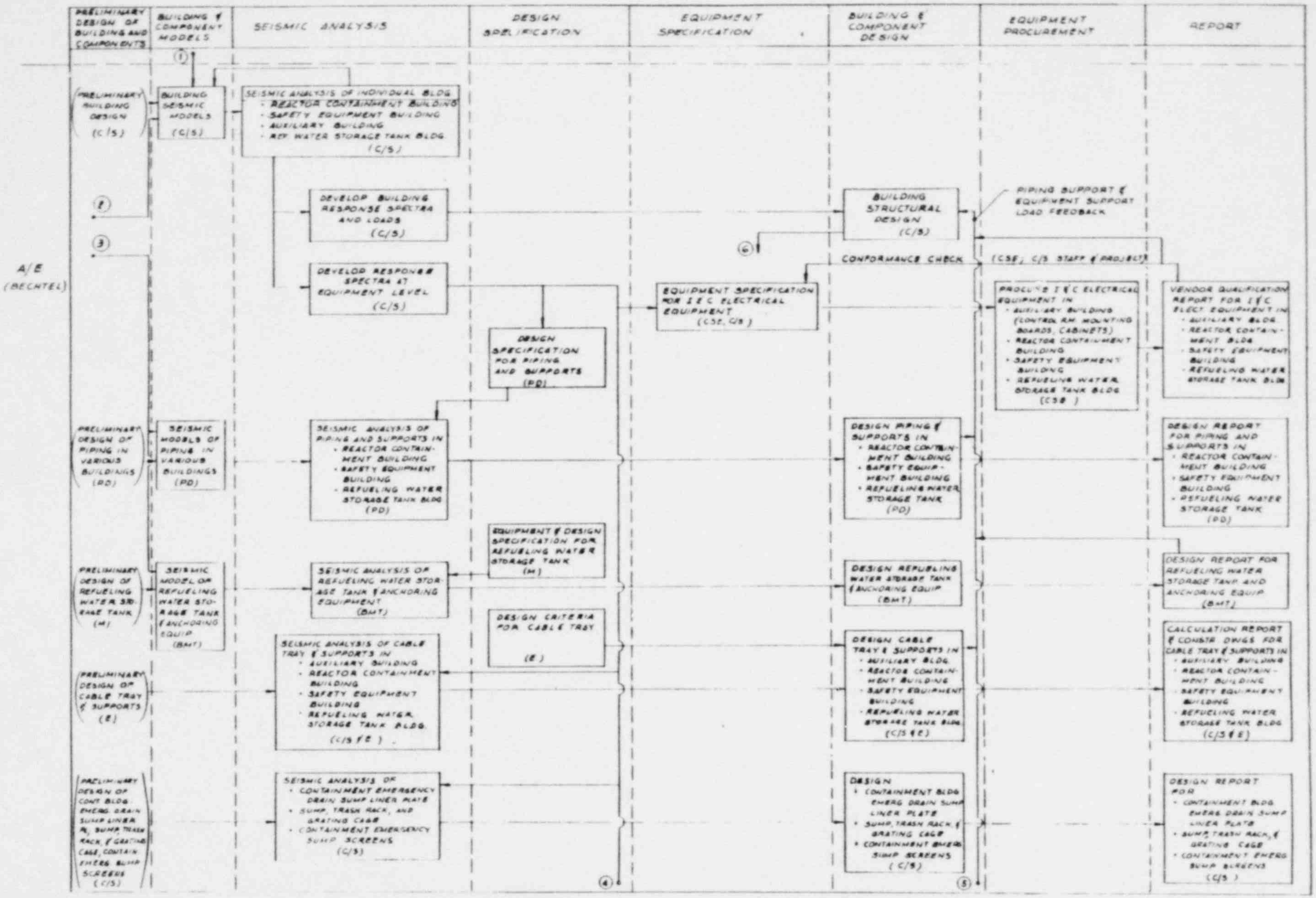

Fig. 4-2. Safety injection system (sheet 1 of 2)

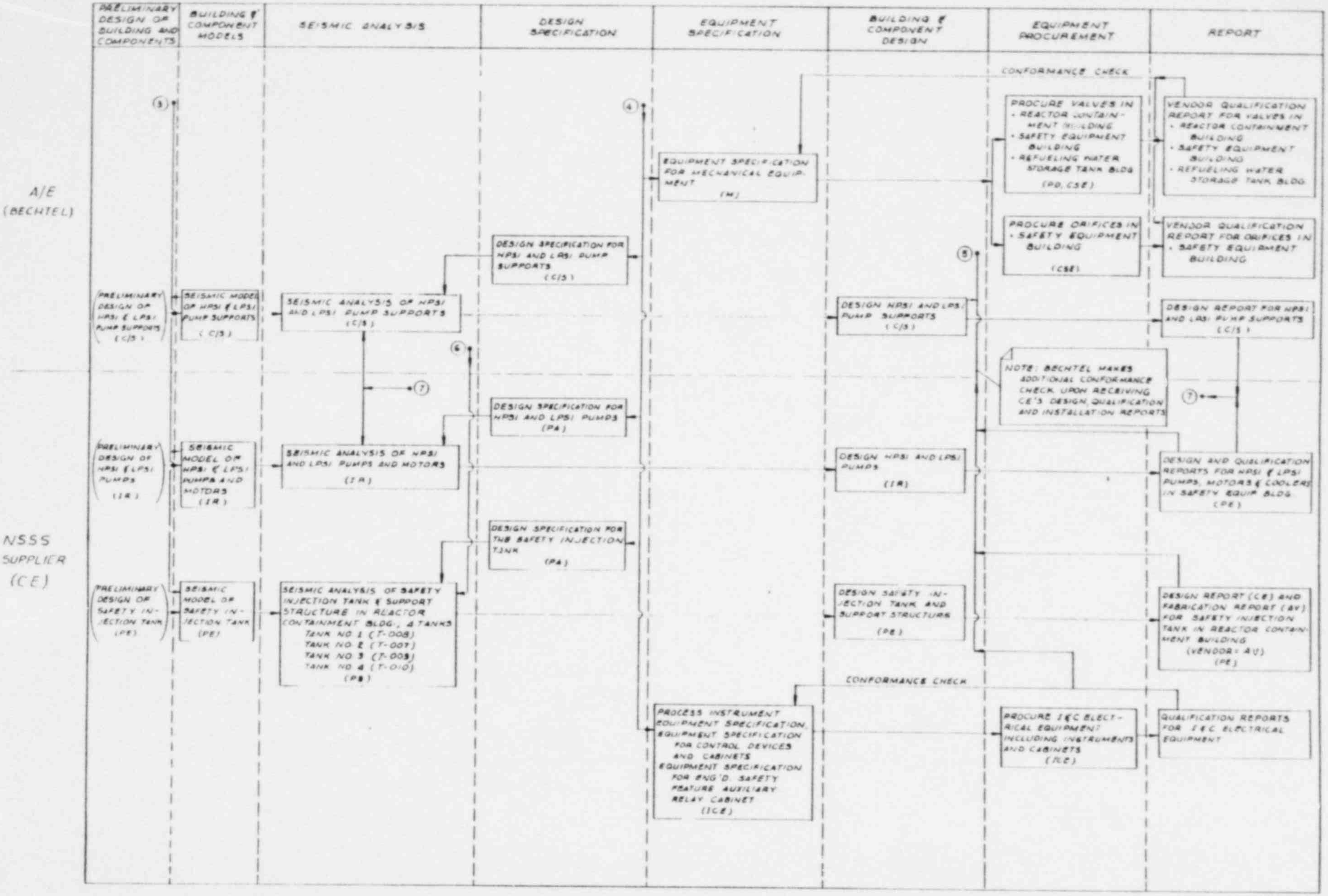

Fig. 4-2. Safety injection system (sheet 2 of 2)

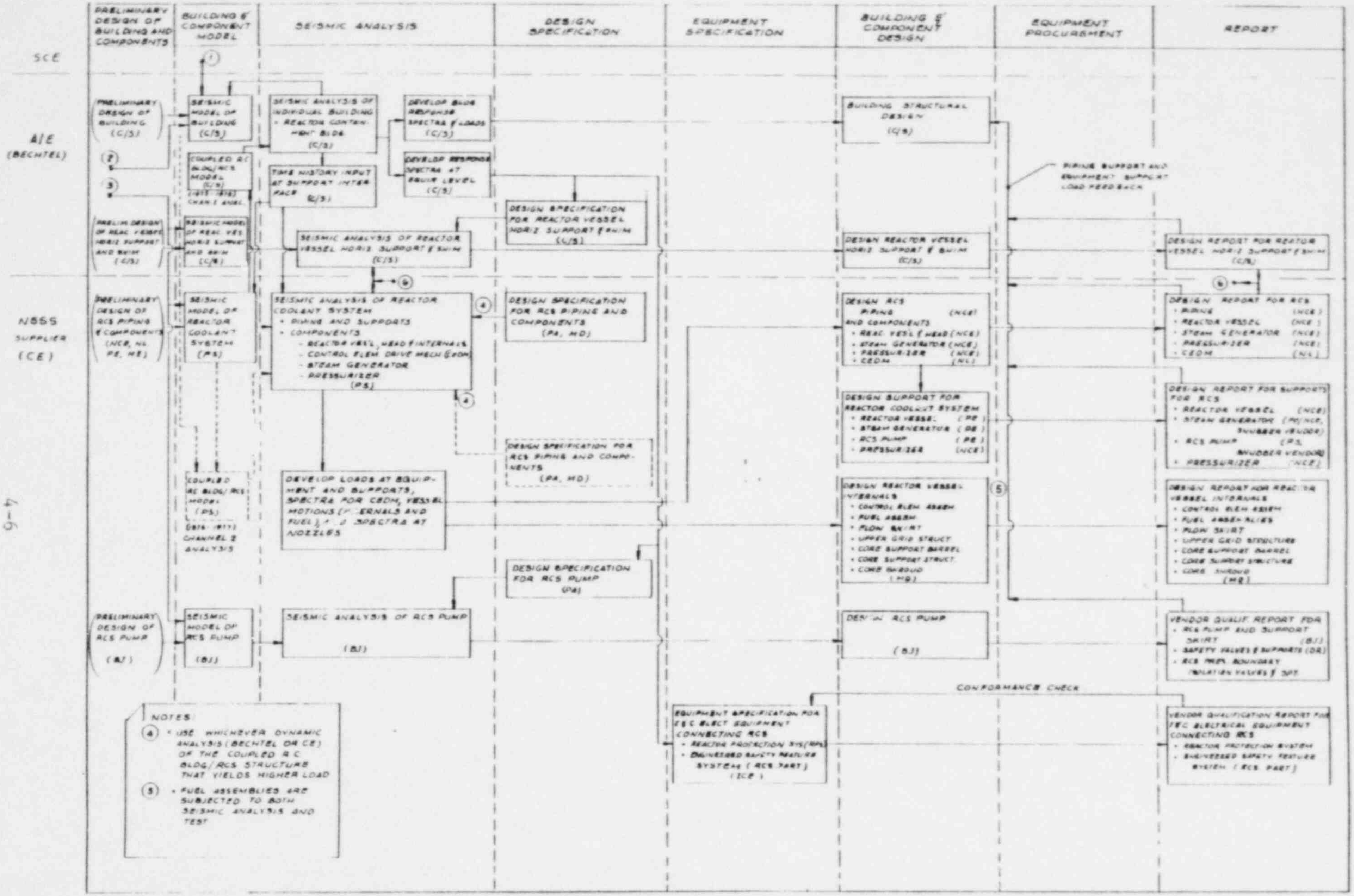

Fig. 4-3. Reactor coolant system and reactor vessel internals

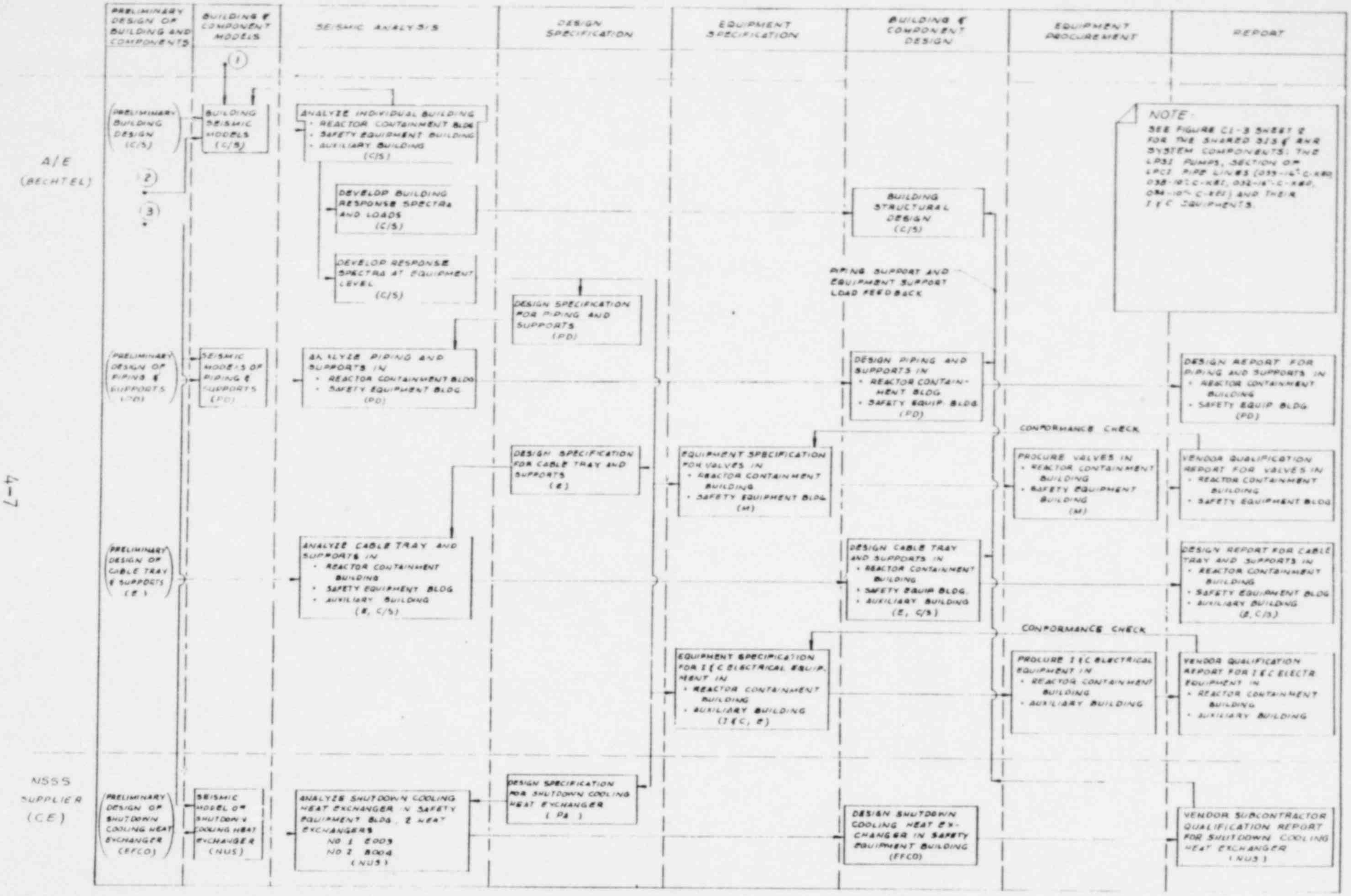

Fig. 4-4. Shutdown cooling system

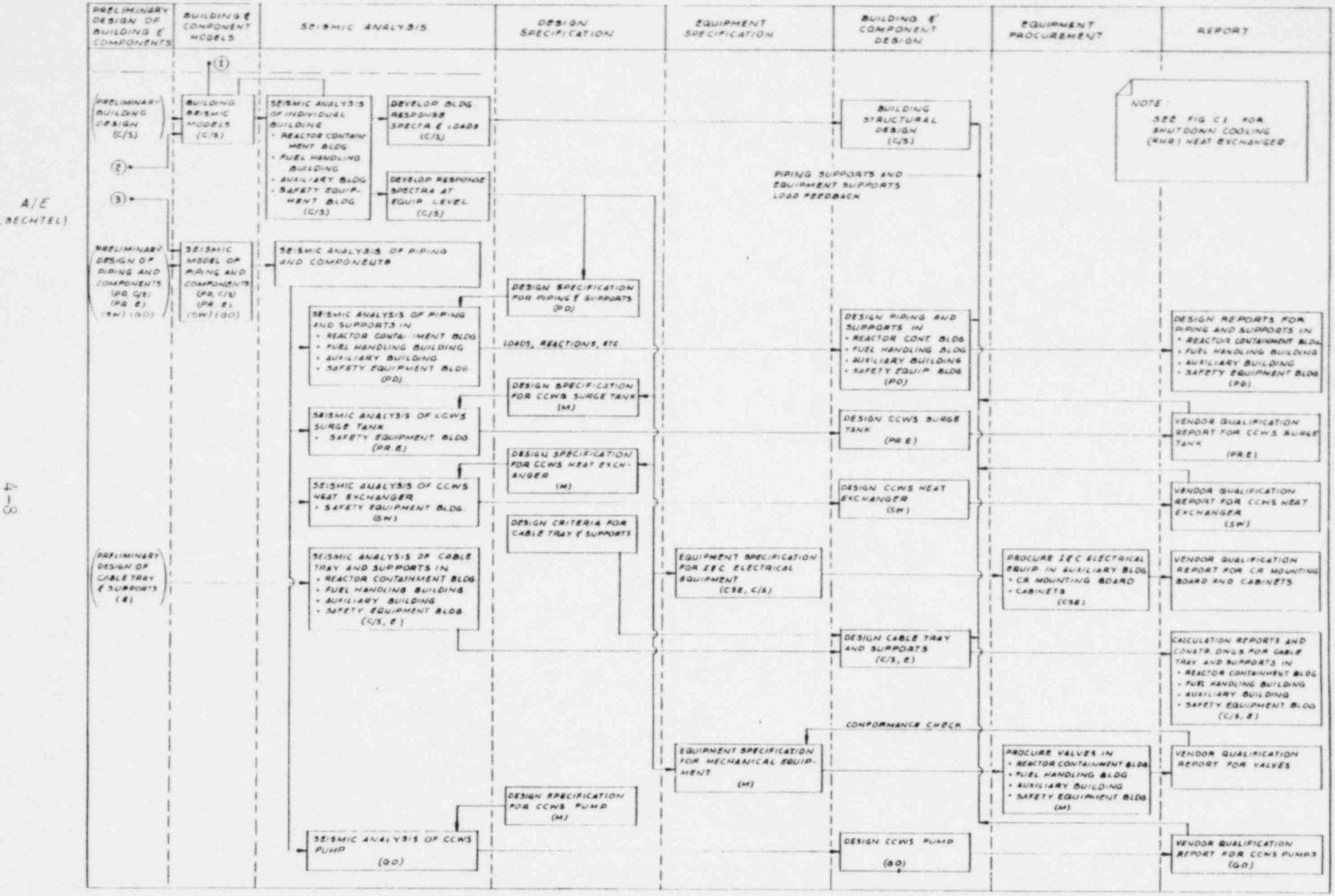

Fig. 4-5. Component cooling water system

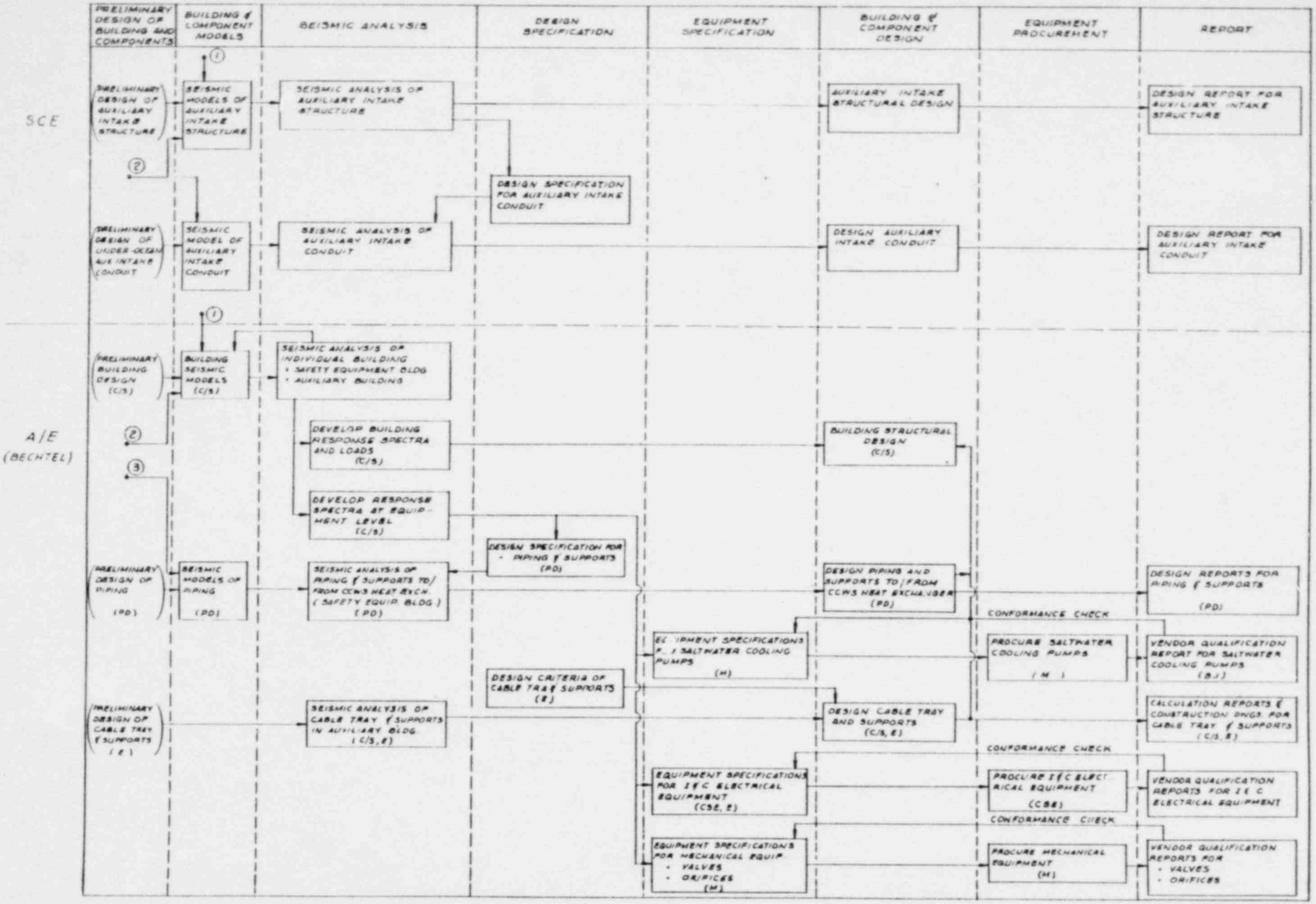

Fig. 4-6. Ultimate heat sink system

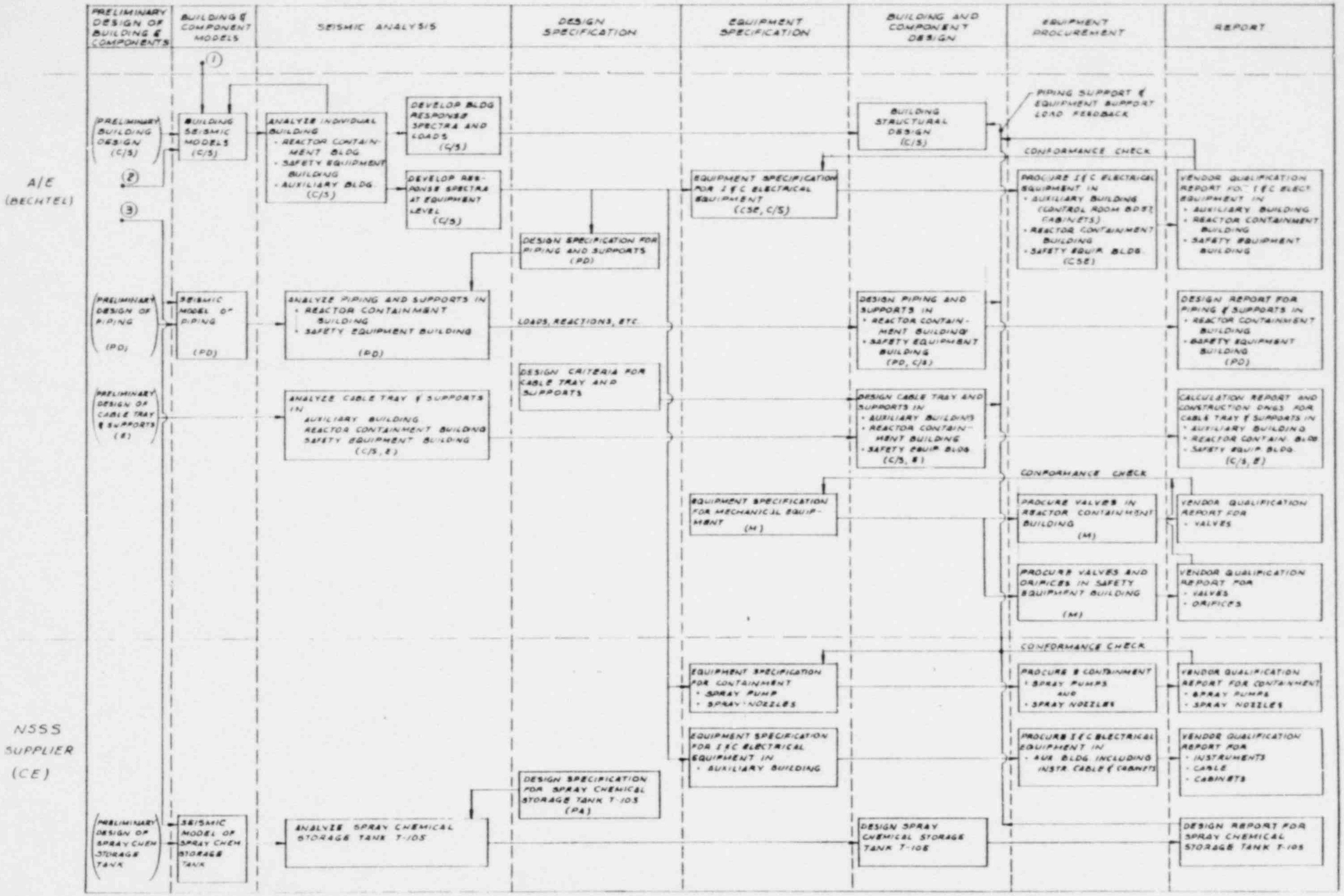

Fig. 4-7. Containment spray system

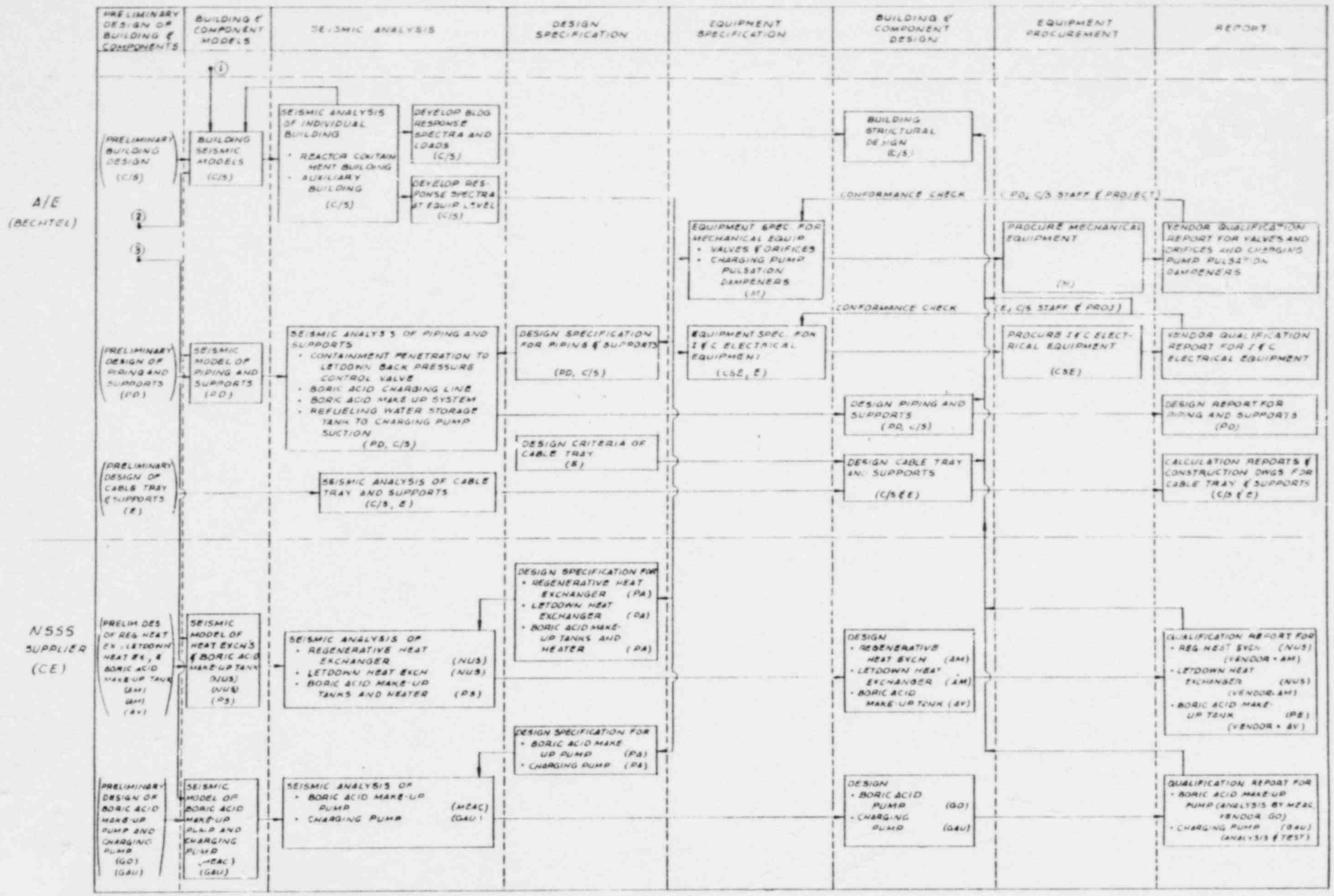

Fig. 4-8. Chemical and volume control system

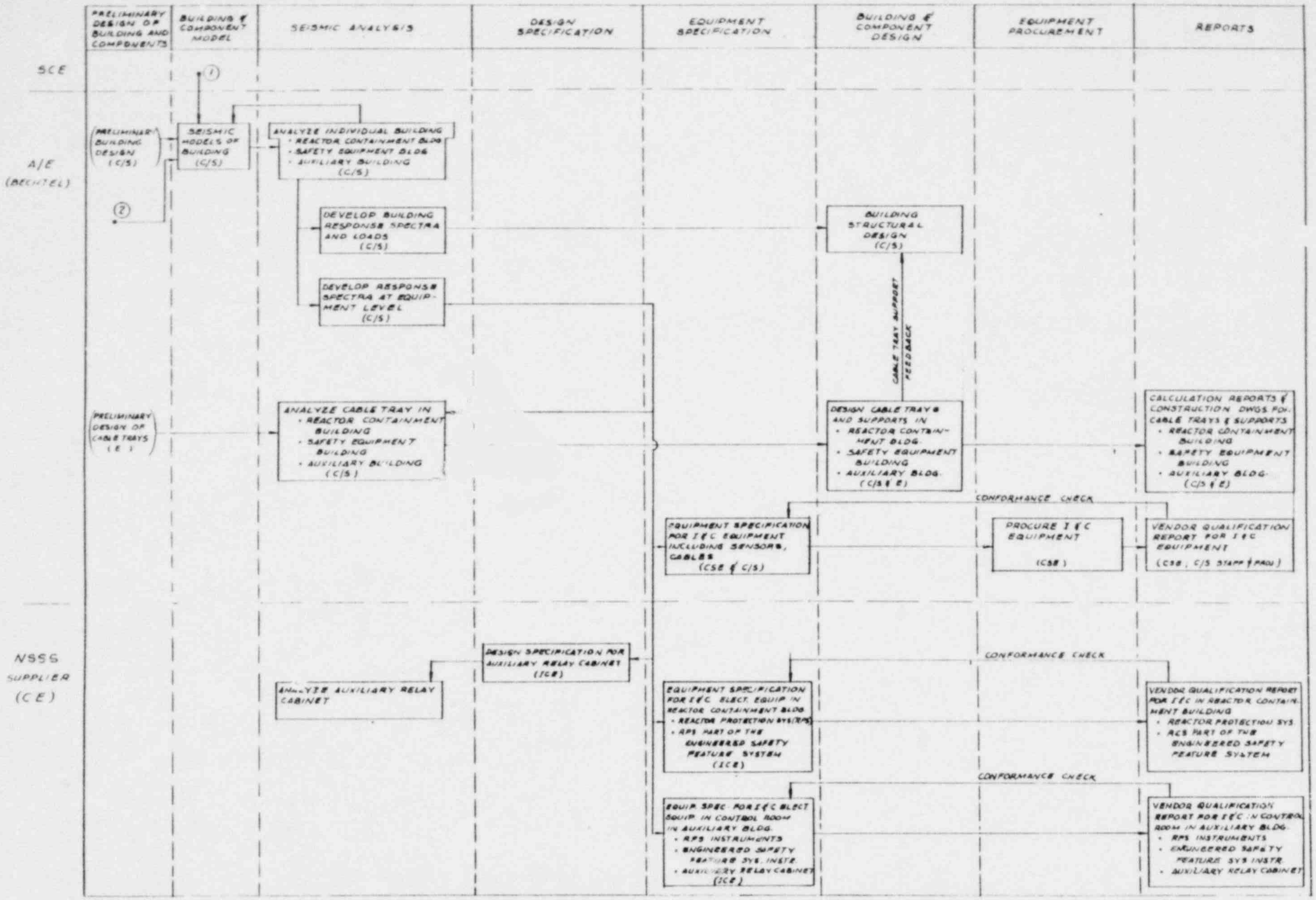

Fig. 4-9. Reactor protection system

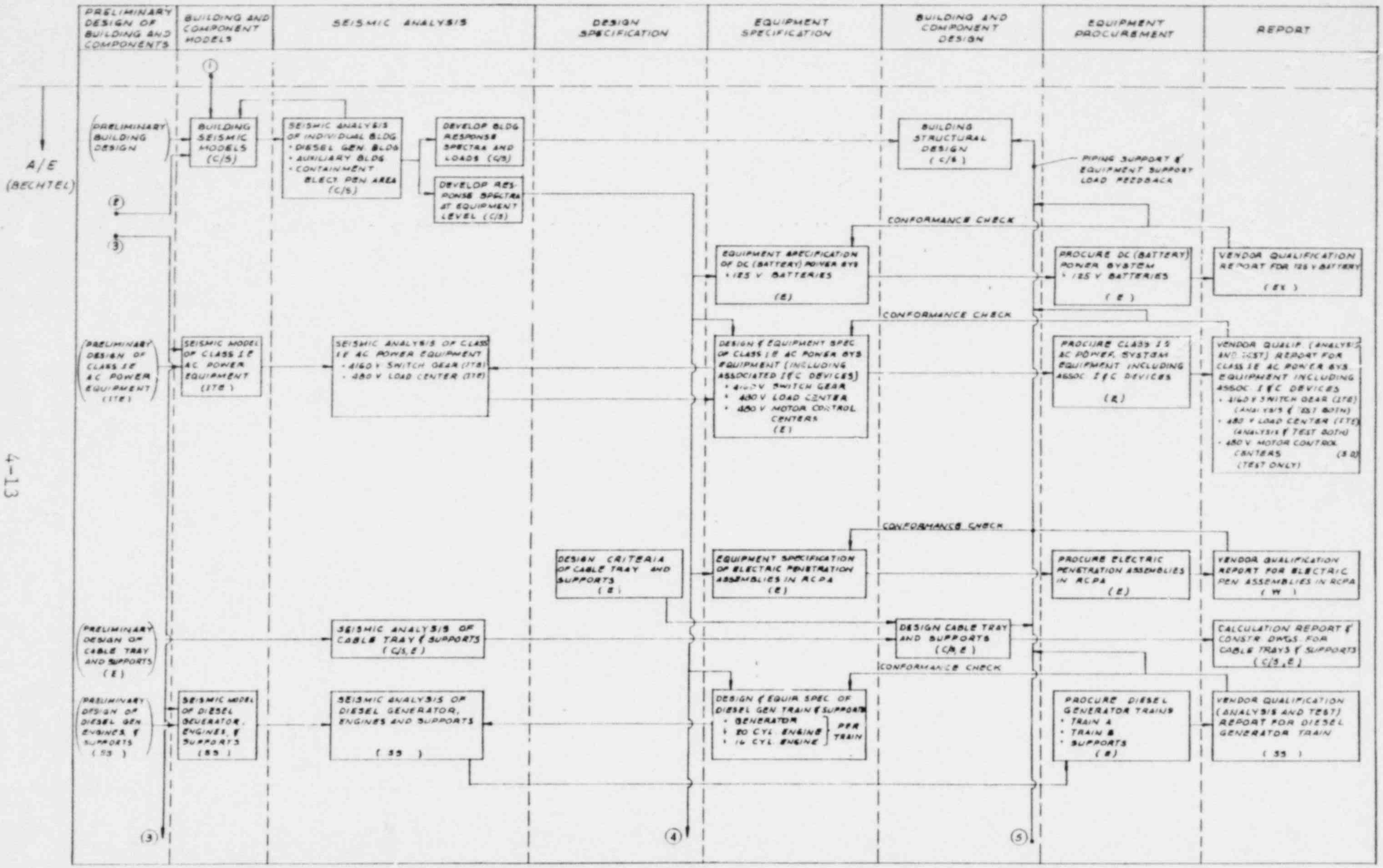

Fig. 4-10. On-site electric power system (sheet 1 of 2)

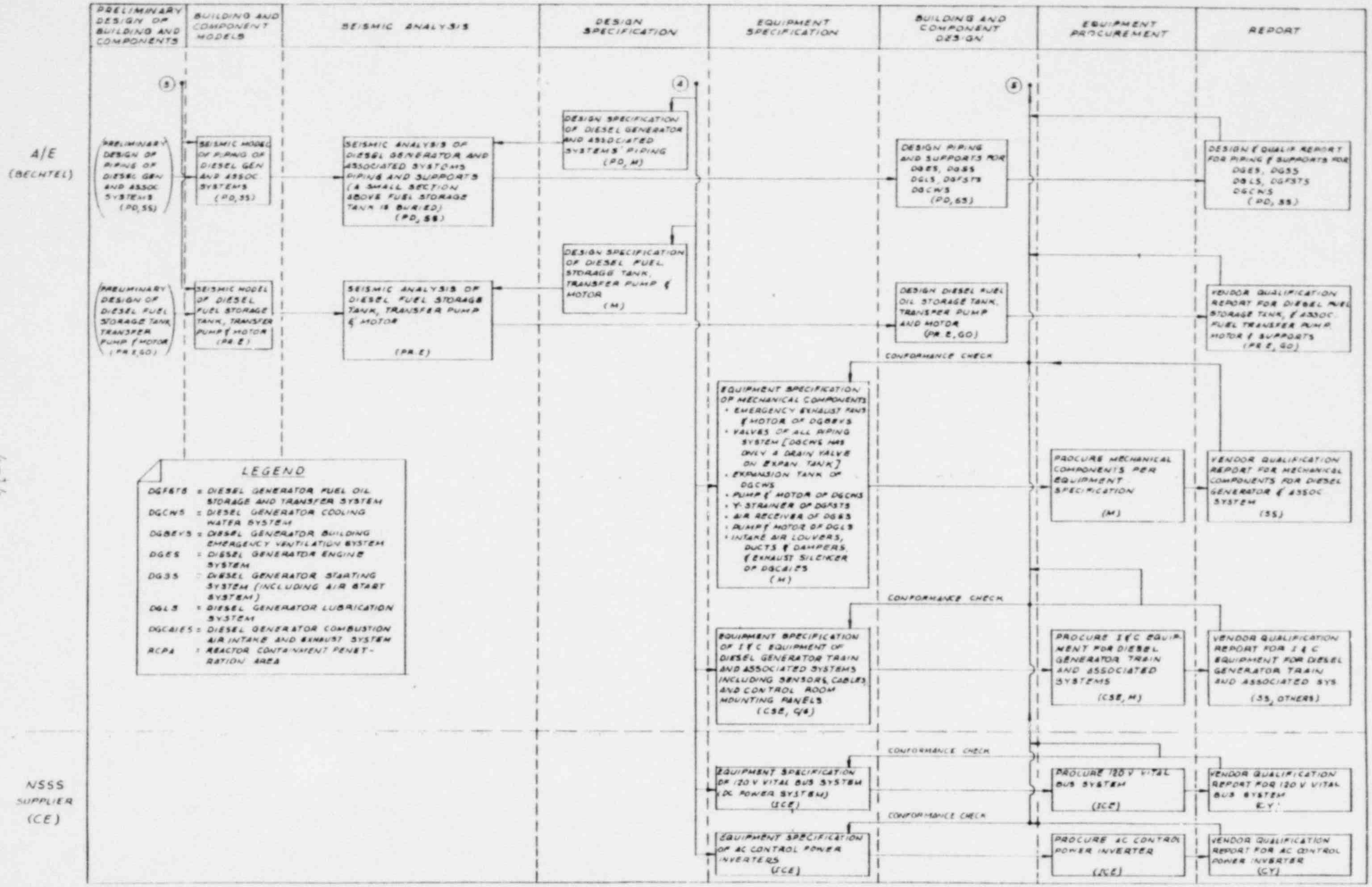

Fig. 4-10. On-site electric power system (sheet 2 of 2)

TABLE 4-1
CRITERIA FOR SELECTING FEATURES FOR SEISMIC DESIGN REVIEW

1. Most of the features selected shall be important to safe shutdown and cooldown of the reactor in the event of a safe shutdown earthquake [or the equivalent, the design basis earthquake (DBE)].

2. Features selected shall be representative of safety-related portions of the plant, including:

   a. At least one safety-related structure.
   b. At least one major NSSS component.

3. Components selected shall be at different elevations.

4. The majority of components selected shall be in the selected safety-related structure(s).

5. The complete range of sophistication in seismic design methods shall be included in the review.

6. Features with design interfaces between SCE, BPC, and CE shall be included. Other subcontractors will be included, if significant.

7. The system(s) selected shall contain safety-related mechanical components, controls, electrical, piping, and cabling.

selecting representative features. In addition to these criteria, other factors were considered in developing the selection plan.

The first factor considered was previous seismic reviews of other nuclear power plants, such as the PWR plants included in NRC's systematic evaluation program (SEP).* The reassessment of seismic design under the SEP was based on the review of selected structures, components, and systems of the nuclear plants. The basis for, and selection of, features reviewed provided background data for use in the selection plan.

The second factor considered was the margin designed into the features to assure continued functioning during a seismic event. The results of the preliminary determination of failure modes associated with a seismic event for safety-related structures and components of the reference plant (Zion 1) for NRC's seismic safety margins research program (SSMRP) have been reported by Campbell and Wesley.** The selection of features for the San Onofre seismic design review utilized this report as guidance, especially in focusing on the seismic-sensitive areas of the selected feature to be subjected to a detailed structural evaluation.

The third factor considered was the results of previous audits of San Onofre conducted by SCE. Some features that had been audited previously were selected for review with emphasis on any open seismic design issues.

Conformance of the selection plan with established criteria is demonstrated in Table 4-2, where elements of the plan are cross-referenced to relevant selection criteria.

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*Nelson, T. A., R. C. Murray, D. A. Wesley, and J. D. Stevenson, "Seismic Review of the Palisades Nuclear Power Plant Unit 1 as Part of the Systematic Evaluation Program," NUREG/CR-1833 UCRL-53015, January 1981.

**Campbell, R. D., and D. A. Wesley, "Preliminary Failure Mode Predictions for the SMRP Reference Plant (Zion 1)," NUREG/CR-017303, UCRL-15042, January 1981.

TABLE 4-2
SELECTION PLAN FOR IDENTIFYING FEATURES FOR SEISMIC DESIGN REVIEW

Element	Relevant Criteria (Table 4-1)
1. Review dynamic analysis and structural design of seismic-sensitive areas of two major structures. The selected structures must contain portions of the selected systems in Items 2 and 3.	2,5
2. Review seismic design of a well defined segment of a major safety system. Include:	1,3,4,5,6,7
a. Large- and small-bore piping at low and high elevations covering various ASME piping classes.	3,4,5,7
b. At least 10 pipe supports and snubbers.	3,4,5
c. At least one major piece or component supplied by BPC; one supplied by CE and installed by BPC. Components to be considered are tanks, pumps, and valves.	6
d. Instruments and electric-equipment-associated cabling, panels, racks, and supports at low and high elevations. Choose at least 2 instruments supplied by CE and installed by BPC.	3,5,6
e. Electrical raceways and at least 10 raceway supports at low and high elevations.	3,5
f. At least 5 seismic-sensitive items.	(a)
3. Review features within other systems, primarily the reactor coolant system, in CE's scope of supply with BPC design interfaces. Include:	1,2,4,5,6,7
a. Reactor vessel, internals, and supports.	1,2,5,6
b. At least one major mechanical component, e.g., primary coolant pump.	1,2
c. Class 1 piping supplied by CE.	1,4
d. Items with design interface between BPC and CE.	6
e. At least 5 seismic-sensitive items.	(a)
4. If the seismic design of a major safety-related feature other than equipment (e.g., piping, structures) was subcontracted by BPC, CE, or SCE, review at least one feature to represent each chain.	1,6
5. Review at least one feature that had been previously audited by SCE and left open or recommended for further review.	(a)
6. If significant differences in the design of Seismic Category 1 features are found between San Onofre Units 2 and 3, review at least one feature representative of the differences.	1,6

(a) Additional factors.

4.3. SELECTED FEATURES FOR REVIEW

Twenty-two features were selected for the detailed seismic design review in accordance with the selection plan (Section 4.2) and in conjunction with the seismic design chain networks (Section 4.1). Table 4-3 lists the selected features and shows how they comply with the elements of the selection plan.

The major structures selected for detailed review were the reactor containment building and the auxiliary intake structure (features 1 and 2 in Table 4-3). The review included all dynamic analyses necessary to show reasonableness of in-structure response spectra used for seismic design of components and systems located in the reactor containment building. Structural design of several component and equipment supports were also reviewed to verify that imposed loadings and responses were correctly reflected in the structural design. Of primary interest was the internal structure that supports major pieces of equipment in the reactor containment building. Stress analyses of seismic-sensitive areas of the reactor containment building and auxiliary intake structure were reviewed to verify that the structure included adequate resistance to DBE loads.

The major safety system selected for detailed review was the Safety Injection System (SIS). The segment of the SIS reviewed (features 3 to 12 in Table 4-3) extends from the refueling water storage tank T-006 (which is part of the fuel pool cooling system) to the nozzle in the cold leg loop 1A of the NSS piping. The major piping for this segment runs from tank T-006 in the yard to the low-pressure safety injection (LPSI) pump P-016 in the safety eqiupment building, to the low-pressure header which is also in the safety equipment building, through containment penetration number 48, past the safety injection tank T-008 in the reactor containment building, and thence to the cold leg piping nozzle. The branch line to the safety injection tank was also included within the major piping. Small-bore piping included within the SIS segment consisted of one-inch lines for safety injection tank T-008 and one-and two-inch lines between the major piping in the vicinity of the tank. All valves on this segment of the SIS system, and

TABLE 4-3
FEATURES SELECTED FOR SEISMIC DESIGN TECHNICAL REVIEW

Review Feature	Selection Plan Element (Table 4-2)
Major Structures	1
1. Reactor containment building	1
2. Auxiliary intake structure	1,4
Segment of the Safety Injection System	2
3. Refueling water storage tank	2c, 2f
4. Low pressure spray injection pump	2c, 2f
5. Safety injection tank	2c, 2f
6. Major piping	2a, 5
7. Small-bore piping	2a
8. Pipe supports and snubbers	2b
9. Valves	2d
10. Instruments, racks, and panels	2d, 2f
11. Switchgear and power panels	2d, 2f
12. Electrical and control cables	2d
Features Within Other Systems	3
13. Dynamic analysis of reactor coolant system	3a, 3d
14. Reactor coolant pump and supports	3b, 3d, 3e
15. Reactor vessel support	3a, 3d, 3e
16. Fuel element grid spacers	3a, 3e
17. Reactor coolant system cold leg (piping)	3c, 3d, 3e
18. Diesel generator oil storage tank	3c
19. Two locally mounted instruments	2d
Other Structures, Components, or Features	
20. Cable raceways	2e, 2f
21. Control panel CR57	4
22. Segment of reactor containment building internal structure and supported equipment	5, 6

motor operators where they occurred, were included in the review. Major mechanical equipment reviewed within this system consisted of tanks T-006 and T-008, and LPSI pump P-016. All 11 power control panels associated within this segment of the system were reviewed. All major instruments and some subtier instruments, including electrical cables, associated with this segment of the SIS system were also reviewed. Selected pipe supports and snubbers, equipment support, and cable tray supports within the SIS segment were reviewed for seismic design adequacy.

The following features associated with the reactor coolant system were selected for review. These features are generally in the CE scope of supply, but they also contain BPC interfaces.

	Table 4-3 Feature No.
• Dynamic analysis of the RCS major components (reactor vessel, steam generators, primary coolant pumps, and pressurizer).	13
• Seismic-sensitive areas of the reactor vessel (vessel support).	15
• Fuel assembly clip grid spacers.	16
• Reactor coolant pump and support.	14
• RCS cold leg piping.	17

Additional selected features having aspects satisfying other requirements of the Selection Plan were:

	Table 4-3 Feature No.
• Diesel generator oil storage tank (underground).	18
• Control room panel design subcontracted by BPC.	21
• Segment of reactor containment building internal structure and supported equipment.	22
• Cable raceways.	20
• Two locally mounted instruments.	19

4.4. REVIEW PROCEDURES

The purpose of the review procedure was to establish a uniform and comprehensive method of performing the seismic design technical review of selected safety-related structures, components, and systems of San Onofre Units 2 and 3. The objective of the review was to ascertain that the seismic design of the selected features is consistent with the NRC-approved design basis and methodology specified in FSAR Sections 3.7 and 3.8.

Listed below is a series of specific questions related to seismic design. These questions have been extracted from Section 6.3.1 of ANSI Standard N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants," and have been used as a guideline in the development of these procedures.

- Were the inputs correctly selected and incorporated into design?
- Are those assumptions that are necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
- Are the appropriate quality and quality assurance requirements specified?
- Are the applicable codes, standards, and regulatory requirements including issue and addenda properly identified and are their design requirements met?
- Have the design interface requirements been satisfied?
- Was an appropriate design method used?
- Is the output reasonable compared with inputs?

- Are the specified parts, equipment, and processes suitable for the required application?

- Are the acceptance criteria that are incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?

The technical review was performed following the steps shown in the flow diagram in Fig. 4-11. The reviewer addressed the questions from Section 6.3.1 of ANSI N.45.2.11 as they applied to each step of the review. These questions were augmented by lists of factors considered important in a technical review of a seismic design. The lists were provided for the review of structures, piping and support, cable raceway and supports, components, equipment, and cables.

Where deemed essential, the technical review was supplemented by independent calculations performed by the reviewer. These calculations ranged from simple calculations verifying structural section properties and load combinations to simplified computer models. The computer models were used to independently check the dynamic response of structures where complex dynamic analyses were utilized in the original analysis.

The technical review was documented. The documentation included a listing of the documents reviewed, a brief description of the review process, a checklist, and calculation files, where generated, for each item reviewed.

4.5. DETAILED TECHNICAL DESIGN REVIEW

The summary of the review conducted for each feature follows.

4.5.1. Reactor Containment Building, Feature 1

The reactor containment building is a cylindrical concrete shell 150 ft in diameter and 170 ft high, topped by an integral hemispherical dome. The

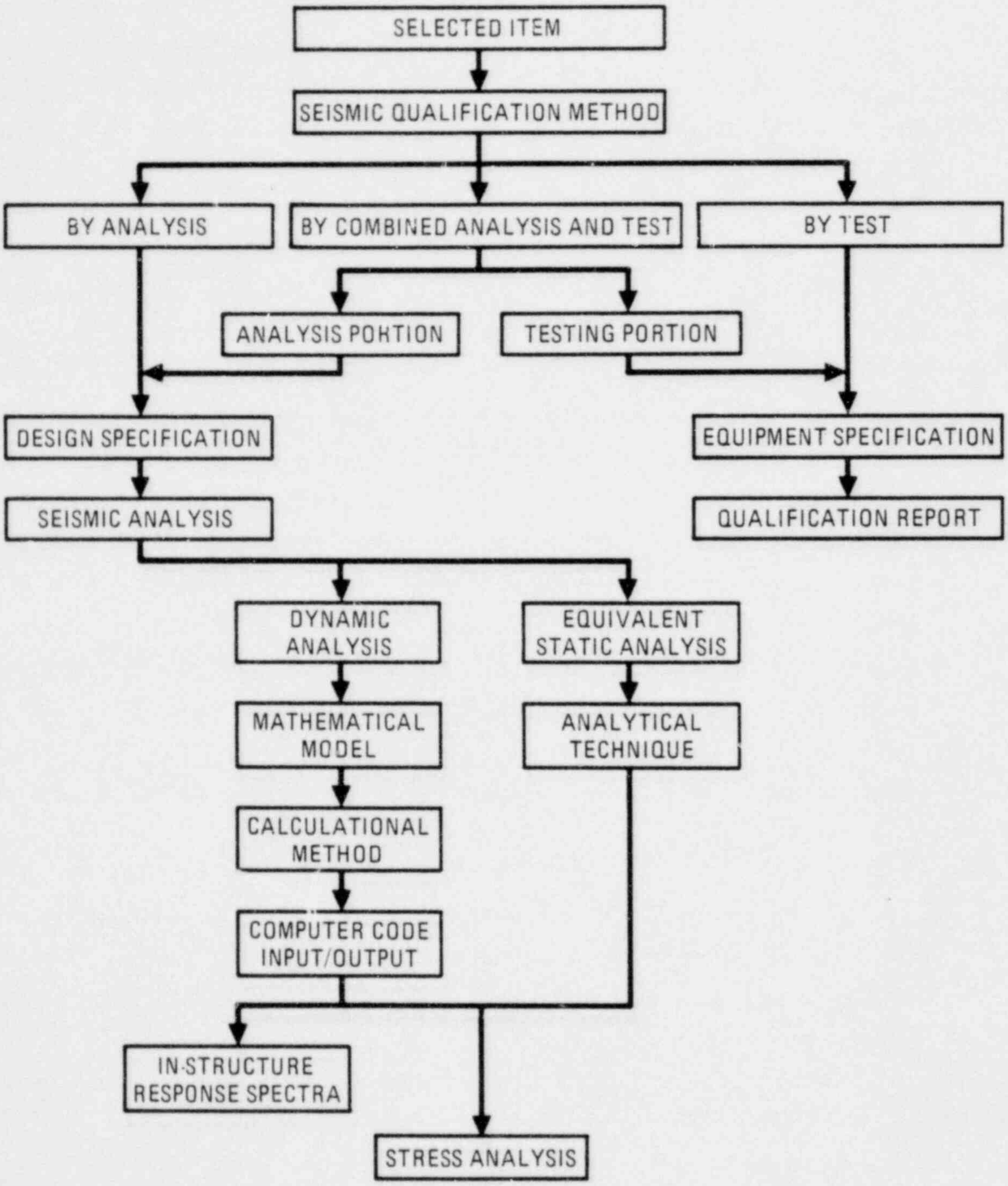

Fig. 4-11. Flow diagram for the seismic design technical review

containment is a prestressed concrete structure with walls approximately 4 ft thick, and is supported on an integral concrete basemat 9 ft thick. This structure is equipped with approximately 150 penetrations for piping, electrical cabling, and personnel and equipment access. The containment building is physically separated from surrounding structures to avoid any significant seismic interaction. BPC is responsible for the building design.

Thirty-one documents were reviewed while evaluating the seismic analysis of the reactor containment building and the effects of its interior structure and reactor coolant system. The seismic analysis of the containment building with its soil foundation must consider the soil-structure interaction. Areas of seismic review concentrated on the following items: the time history of seismic acceleration input traces and design response spectra; soil parameters (damping, spring constants); basemat and superstructure dynamic property input values; modeling features and computer programs used; output quantities, which include natural frequencies and mode shapes; maximum acceleration response values; in-structure response spectra; and methods of combining response data. For all of these areas of review, a checklist of selected items was maintained to ensure that inputs were properly incorporated and that assumptions, codes, and standards were correctly interpreted.

The containment building was analyzed by BPC as a lumped parameter model or finite element model using computer codes such as SMIS and ASHSD to determine seismic response.

Natural frequencies of the containment building were checked by developing idealized models that were solved by analytical solutions. Calculated natural frequencies obtained from these models, using either single or two-degree-of-freedom systems, indicated good agreement with the SMIS model used by BPC for the lower modes. Information from this study was then used during the development of a more accurate analysis based on a multi-degree-of-freedom computer model.

The MODSAP computer program, developed by General Atomic, was used to construct a model that would verify the accuracy of BPC's acceleration values and in-structure response spectra curves. This model, which is dynamically equivalent to SMIS, couples the properties of soil springs, basemat, exterior shell, and interior structure, including effects due to the mass and flexibility of the reactor coolant system. Following the incorporation of a basemat cracking feature in the MODSAP program, the maximum horizontal acceleration figures for the various building elevations were in excellent agreement with the SMIS model. The horizontal in-structure response spectra were also in good agreement. The vertical in-structure response spectra comparison, however, showed differences between the two models. This was attributed to the additional response resulting from nodal rotations due to horizontal input in SMIS, whereas such response combinations were not incorporated in the simplified MODSAP analysis. The in-structure resonse spectra generated by the SMIS model are considered valid for seismic design.

Four PFR's, three of which were classified as Observations, were issued under this review of the reactor containment building. One PFR was invalid.

One PFR (0057) pertained to an incorrect value in the input data for the SMIS code. Inspection of the data by BPC revealed that the error found during the seismic review was in fact a typographical error in the input tabulation. It was further established that the actual computer input data was correct, and that the calculated results were therefore valid. This PFR was classified as an Observation.

The second PFR (0058) questioned the orientation of beam elements that were defined by local coordinates. The moment of inertia associated with these elements was provided as input to the SMIS code, and because of orientation it appeared that the values used may have been incorrect. A further review, however, showed that the actual input data were indeed correct and that only the moment of inertia designations were inconsistent. This PFR was classified as an Observation.

The third PFR (F105) questioned certain deviations noted in the response values obtained from the alternate analysis using the MODSAP code as compared with the values computed by the SMIS code. These deviations resulted from differences in damping simulation and the incorporation of a basemat tracking feature in the SMIS model. It was noted, however, that the free-field ground motion tracking technique used by BPC for time history analysis is a valid but not a widely used method for seismic analysis of embedded basemats. Also, it was not specifically mentioned in the methodology referenced in the FSAR. Comparison of the results from the two codes was acceptable following inclusion of the basemat tracking feature in the MODSAP analysis. This PFR was classified as an Observation.

No major problems were found during the review of the dynamic analysis of the reactor containment building. The output of the dynamic analysis in terms of the loads, displacements, and in-structure response spectra is valid for seismic design and qualification of internal structures, components, and equipment located in the reactor containment building.

4.5.2. Auxiliary Intake Structure, Feature 2

The offshore circulating water system auxiliary intake structure is a submerged reinforced concrete structure mostly embedded in the ocean floor. It is located approximately 3200 ft offshore, approximately 100 ft shoreward of the primary intake structure. The auxiliary intake structure was designed by SCE.

Five documents were reviewed, including the design calculations and drawings. From these documents, the seismic design criteria and assumptions were reviewed (including the g-levels and hydrodynamic loads used). The review also included the structural analysis, design, and stability analysis of the structure itself. Review checklists were maintained to ensure that all of the prescribed areas of the design review had been properly addressed.

During the course of this design review, information became available concerning a repair by SCE of damage to the base block structure of the auxiliary intake. The procedure and specifications supplied by SCE to effect repairs on the base block were reviewed to determine the extent of the damage and to evaluate the effectiveness of the repair procedure. The epoxy-aggregate system used was judged to be capable of providing effective rebar cover and of sustaining design loads. It was concluded that the repair was satisfactory and that the structural capability of the base block was adequately restored.

Three PFR's were written during this review. Two of these were classified as Observations, and one PFR was invalid.

One PFR (F108) refers to an erroneous parameter used during the application of a moment distribution method to calculate moments and loads in the velocity cap and columns of the intake structure, and to a numerical error made while calculating moments in the riser. Both deviations were minor and did not affect the design margins significantly. This PFR was classified as an Observation.

The second PFR (F106) refers to the seismic loads used in the design of the conduit section of the auxiliary intake structure. The loads used in the original calculation did not include the transverse seismic effects contributed by the upper part of the structure. Further analysis during the design review, however, indicated that the design can conservatively accommodate all the loads expected in the conduit. This PFR was classified as an Observation.

Based on this review, no significant problems were discovered with respect to the overall seismic design of the auxiliary intake structure.

4.5.3. Refueling Water Storage Tank, Feature 3

The refueling water storage tank is a closed vessel 36 ft in diameter and 40 ft high, located at grade level in the tank building adjacent to the containment. This tank (one of two similar tanks) has a capacity of 245,000 gallons, and is operated under normal atmospheric pressure. The refueling water storage tanks provide a source of water for the low-pressure safety injection system. BPC supplied the design specifications for the tank; Brown-Minneapolis Tank performed the engineering and fabrication.

Five design-related documents were reviewed during the design evaluation of this tank. The BPC specifications were examined to verify that proper seismic criteria were defined and that applicable seismic loads could be traced to source documents. The design analysis documents developed by the tank fabricator were also reviewed to verify that the proper input data had been incorporated into the tank design and to evaluate certain assumptions that had been made in the analysis. These documents showed that a fairly low natural frequency had been calculated for the tank, and that a correspondingly low seismic acceleration value had therefore been used in the analysis. Further, a check was made to determine if a proper accounting had been made for the hydrodynamic effects associated with seismic vibration (slosh), and to verify that the resulting loadings would not overstress the tank structure. Throughout this effort, checklists were maintained to assure that the review was correctly performed.

Three PFR's were issued and each was classified as an Observation.

The first PFR (0008) concerned the accuracy of nozzle loads included in the design specification and whether the use of these load values would result in a nonconservative nozzle design. Subsequent evaluation of the calculations from which these loads were developed indicated an error of about 10%. This was insufficient to materially affect the design margins, and the PFR was therefore classified as an Observation.

Two related PFR's (F071 and F078) were written on the refueling water (RW) storage tanks.

PFR F071 concerned the procedures and calculations used to determine allowable buckling stresses of the tank structure. This PFR noted that the design specification and the structural analysis, in this instance, did not appropriately apply Section III of the ASME code. BPC concurred and performed new calculations, based on the proper application of the code, which showed that the design stresses still remained within acceptable values. Extensive calculations were also performed by TPT to evaluate the results of hydrodynamic buckling loads applied to this tank. TPT found the stress at a point in the top course of the storage tank (as calculated by BMT) to slightly exceed the allowable stress, but it was well below the minimum buckling stress. The allowable buckling stresses referenced above are obtained from nonmandatory Appendix F, Subsection F-1325, of the ASME code, which includes a margin. The difference in calculated allowable stresses between BPC and TPT arose from a difference in interpretation of the curve defining the minimum of the data on buckling stress as a function of tank radius-to-thickness ratio.

PFR F078 involved the method for calculating hydrodynamic loads on the tank during a DBE. BMT used a technique for rigid tanks developed by Housner.* Tank flexibility may lead to increased hydrodynamic loads. In addition, fluid sloshing was predicted to cause significant interaction with the domed roof of the tank. A series of calculations was performed to assess the added effect of hydrodynamic loading on the buckling and tensile stresses in the tank structure. The results indicated that although the allowable buckling stresses are exceeded at the base of the tank, these stresses still remain below the minimum buckling values, based on evidence from tests and analysis of actual seismic data on tanks. It was also recognized that the buckling phenomenon is basically a stability consideration and is not a mode of failure that would result in loss of fluid from the tank. In the review of these two PFRs, no inadequacies were observed that would prevent the tank from performing its safety function. Thus, these two PFR's (F071 and F078) were classified as Observations.

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*TID-7024, "Nuclear Reactors and Earthquakes, "August 1963, as modified in BPC Design Report 407-13-110.

In the review of the refueling water tank, no inadequacies were observed which would prevent the tank from performing its safety function.

4.5.4. Low Pressure Safety Injection Pump, Feature 4

The low-pressure safety injection (LPSI) pump (P-016) is a motor-driven vertically oriented single-stage centrifugal pump, used to transfer water into the reactor coolant system under emergency conditions and to provide for circulation during subsequent plant cooldown. CE was responsible for the pump specification, while BPC provided engineering for the pump/support interface plus the suction and discharge piping.

Ten documents were examined during the seismic design review of this pump. These documents included general and project specifications (CE), pump operational analysis studies, piping analyses, and design drawings. Other documents were also reviewed, including shaft and motor frame stress analyses, critical frequency studies, and bearing loads based on manufacturers' limits. These documents were used to evaluate calculated natural frequencies of the overall installation.

The procurement specifications were reviewed carefully to establish that appropriate seismic requirements had been provided to the vendor (Ingersoll-Rand). Similarly, the motor specifications (Westinghouse) were also reviewed. Seismic acceleration levels and response spectra presented in the FSAR were thoroughly researched to verify that the procurement specifications developed for this pump were responsive to the procedure and methodology commitments listed for this type of plant equipment. A checklist was maintained during this review to ensure that all required areas of seismic design were adequately addressed.

Implicit in the design of the LPSI pump suction piping system is a requirement to maintain adequate net positive suction head. It was postulated that seismically induced sloshing in the refueling water tank, which supplies feed for the LPSI pump, coupled with the seismic accelerations could result in periodic pressure fluctuations at the pump that could cause the net positive suction head to momentarily decrease below the minimum prescribed value. Extensive calculations and analyses were performed during this seismic review, and it was determined that these fluctuations do not affect the ability of the pump to operate satisfactorily.

Four PFR's were written. One was classified as an Observation and the remaining three were invalid.

PFR 0011 questioned the use of a 1.0-g seismic acceleration level during component testing, as opposed to an apparent value of 1.2 g based on interpretation of an FSAR figure depicting response spectra. Although the precise value for the acceleration was not established, a reevaluation of the pump stresses was performed to verify that 1.2 g would not result in exceeding allowable stresses. Since this was verified, PFR 0011 was classified as an Observation.

No major problems were found during the review of the seismic design of the LPSI pump. (See Task H, Section 7.0, for further discussion on this pump.)

4.5.5. Safety Injection Tank (T-008), Feature 5

The safety injection tank is a 42-ft-high by 9-ft-diameter assembly installed at the 45-ft level in the reactor containment building. There are four such tanks in Unit 2, and four more in Unit 3. These tanks store borated water under pressure for use during emergency injection into the primary coolant loops. CE specified the tank design and performed the dynamic analysis on the tank, including the effects of slosh, evaluation of interface loads at the nozzles (based on BPC input) plus the design of the vertical and lateral supports. P. F. Avery (a division of CE) performed the engineering and fabrication.

Eleven design-related documents were reviewed to establish the seismic adequacy of this tank. Verified computer codes were used in the tank design; and material properties, load combinations, and stress allowables were all determined to be in conformance with the ASME code. Review checklists were maintained to ensure that all prescribed areas of the seismic design had been included.

Two PFR's were written in this review of the safety injection tank. One PFR (0037) was an Observation, and one was invalid.

PFR 0037 identified a discrepancy in the calculated moment of inertia of the concrete floor slab. The discrepancy pertained to a difference between values calculated by BPC and those used by CE in the dynamic analysis of the tank. Review of the calculations revealed that the discrepancy involved an incorrect conversion of units by CE. In response to this PFR CE established that use of the correct moment of inertia in the calculation did not result in stresses exceeding design allowable values.

This review of the seismic design of the T-008 safety injection tank found no major problems concerning the adequacy of the design.

4.5.6. Major Piping, Feature 6

A seismic review of major piping (typically 8 in. or greater in diameter), associated with a prescribed segment of the SIS, was completed. The piping that was reviewed included those runs between the refueling water tank and the LPSI pump (24-in. and 16-in.), and from the pump to the 1A reactor coolant loop (8-in., 10-in., 12-in., and 14-in.). Also included was the discharge piping between the safety injection tank and the injection system (12-in.). BPC was responsible for the design and analysis of this ASME Class 1, 2, and 3 piping. Six separate analysis packages were examined during this review.

Fifty-four individual documents were reviewed, including design specifications, isometric drawings, calculations, piping area drawings, valve data sheets, input/output computer data, and piping and instrument (P&I) diagrams.

For all of the prescribed areas of review associated with major piping, a checklist of selected criteria was maintained to ensure that inputs were properly incorporated into the design and that assumptions, codes, and standards were correctly interpreted.

In developing pipe support loads, BPC uses the square root of the sum of the squares (SRSS) method for combining seismic inertia and seismic anchor movement (SAM) loads, which are then added algebraically with the weight and thermal loads. The issue of whether the SRSS or the absolute sum be used for combining the seismic loads was addressed. However, the SAM and thermal loads are secondary-type effects for pipe support design under faulted conditions and need not be included. Therefore, the manner of combining DBE-induced loads is not an issue of concern.

Twenty-one PFR's were issued for the major piping design review. Twelve of these were classified as Observations; the remaining nine were invalid.

One PFR (0035) involved a failure to incorporate the effects of a change notice into the piping analysis, two PFR's (0036 and F074) addressed the omission of piping specialty weights or the use of incorrect values for piping weights, and two other PFR's (0003 and 0024) involved problems with the basis and traceability of seismic anchor movement loads used in the analysis. Three PFR's (0023, 0040, and 0056) pointed out inconsistencies between the input isometric sketches and the resulting computer model for the pipe runs represented by those sketches, including incorporation of the valve center-of-gravity data. Three PFR's (0001, 0006, and 0007) involved documentation errors and the fact that one pipe support analysis did not correspond exactly to the actual installed support. All of these PFR's were classified as Observations.

One PFR (F043) questioned the techniques used by BPC in calculating the seismic g-loading on valves to account for the contribution of the higher frequency modes which are excluded from the computer output due to program limitation. In reviewing the piping analysis packages, it was noted that various techniques were used by BPC. The basic approach was to add the zero period acceleration (ZPA) component under certain conditions. No mathematical justification was provided to substantiate the techniques used. In response, BPC presented the correct procedure and showed that the various techniques previously used were appropriate simplifications. In essence, the ZPA is combined with the computer-calculated g-load by the SRSS method only when the calculated g-load is less than the ZPA in a specific direction. The resulting g-load in the three orthogonal directions are then combined by the SRSS method to obtain the effective valve g-loads. BPC presented justification of this procedure which was evaluated and confirmed. This PFR was classified as an Observation.

The effect on seismic design due to each of the above-discussed PFR's does not materially change the calculated pipe stresses or pipe support loads associated with the major piping.

No major problems were found during this review concerning the adequacy of the design of the major piping.

As a result of the review of major piping, a trend was investigated concerning the incomplete analysis of Unit 3 piping in areas where it was not identical or the mirror image of Unit 2 piping as noted in several PFR's. In response to TPT's inquiry, BPC stated that two independent checks were provided to assure that all unique Unit 3 piping was analyzed. First, BPC reviewed all design documents and tabulated the Unit 3 piping runs where they were not identical nor the mirror image of Unit 2. This tabulation, prepared prior to this program, has been reviewed and the piping runs associated with the above-mentioned PFR's are included in the tabulation. BPC is in the process of completing the analysis of these piping runs. The second check is the comprehensive as-built verification program. The procedure used for this verification and the format for the tabulation has been reviewed and found to be acceptable.

4.5.7. Small Bore Piping, Feature 7

Certain small-bore piping runs (typically 2-in. diameter and under), installed in the SIS of Unit 2, were selected for seismic review. This piping provides for fill, drain, relief, and pressure equalizing functions associated with the T-008 safety injection tank and its main 12-in. discharge line. The seismic design of approximately 100 linear feet of piping was reviewed, including the effects contributed by several in-line valves. BPC was responsible for the design, analysis, and installation of this small-bore piping.

Six documents were examined during this review, including piping stress analysis packages and appropriate hand calculations. The major stress package, PSG-245, contains a complete dynamic analysis of the modeled piping system, and provides the analytical justification for the overall analysis.

The seismic review was conducted by comparing isometric sketches, P&I diagrams, and area drawings to verify consistency, and by carefully checking techniques and assumptions used in the hand calculations. For the computer analyses, input data from the isometrics and valve drawings were verified, and the resulting output was analyzed to assure the validity of support reactions, pipe stresses, and anchor loads. The seismic response spectra used for these studies was also checked.

In general, it was observed that the small-bore piping stress analysis packages lacked adequate documentation and traceability of the various assumptions used when combining loads and evaluating the effect of axial restraints at support points. Although this information was not directly available, it was determined during this review that no significant effect on the seismic design resulted from these deficiencies. To verify that all aspects of the seismic analysis were properly addressed, tabulated checklists of the significant requirements were maintained throughout the review.

Only one PFR was written for the small-bore piping, and it was classified invalid.

No problems were found with respect to this small-bore piping, and it was concluded that the seismic design of the piping, including the location of its supports, is adequate.

4.5.8. Pipe Supports and Snubbers, Feature 8

A representative group of 13 pipe supports was selected for a detailed seismic review. These supports are directly associated with piping that comprises the SIS of Unit 2. Four piping analysis packages were examined: PSG Nos. 78, 82, 245, and 56-4. Within these packages, the pipe support structure was reviewed in detail, along with both the physical orientation and the means of attachment to the building structure. The supports are designed as planar or three-dimensional frames composed of structural steel members welded or bolted together. These frames bear on the building structure by means of rock-bolt anchors, by welding to steel plates embedded in concrete, or by some combination of these methods.

The pipes are supported on these frames either by direct contact or by integral attachments. Snubbers and/or shock arrestors may also be connected between the integral attachments and the adjacent load bearing structure. BPC is responsible for the overall design of these supports.

One hundred and one documents were examined during this seismic design review. These documents included design specifications, criteria and procedures, calculations (both hand and computer), pipe support drawings, and applicable field change requests and design change notices.

The seismic review involved load verifications to assure consistency between design calculations and piping analyses, review of the support drawings to establish that configuration details (dimensions, connections, member and weld sizes, etc.) were properly reflected in the design calculations, and a verification check was made of the calculations and procedures used to develop the final design of the support.

In the course of this overall review, a checklist of seismic analysis requirements was maintained to ensure that all appropriate areas were adequately addressed.

Fourteen PFR's were written for the pipe supports and snubbers. Five of these were classified as Observations and the remaining nine were invalid.

The first PFR (0014) dealt with a mislabeled piping node in the computer calculation; however, there was no impact on the design due to this error. A second PFR (F065) discussed inclusion of shear deformation effects in the piping support calculation. A recalculation was performed during the design review to assess the impact of considering such deformation. The result showed that the calculated stresses were not materially affected and that the design remained conservative. A third PFR (F082) addressed utilization of an incorrect pipe wall thickness plus incorporation of inaccurate parameters in the stress calculations. Consideration of the correct values, however, still resulted in stresses below the allowables. A fourth PFR (F091) determined that incorrect procedures had been used to evaluate tension and shear loadings in the rock-bolt anchors. However, the reevaluated loadings developed during the design review indicate that stresses still remain conservative. The fifth PFR (F092) described several calculations in which stresses in the structural elements and welds were not adequately documented. BPC's analysis in response to this PFR, however, showed that the calculated stresses are within allowable values.

As a result of this review no problems were found that showed the seismic design of the pipe supports and snubbers to be inadequate.

The review of BPC's pipe support design, performed under features 6, 7, and 8 resulted in a trend being investigated concerning the possible implications of node point disagreement between pipe stress isometrics and pipe support drawings as noted in PFR's 0006, 0014, and 0056. In answer to TPT's inquiry, BPC stated that this discrepancy could not lead to any design inadequacies. The first line of defense against any such inadequacy is the design procedure followed by BPC engineers. BPC engineers do not use the node points to correlate pipe supports and piping system stresses after the initial preparation of the documentation. This practice is substantiated by TPT's review in that no inadequacies resulted from the observed mislabeled nodes. The second level of defense is the BPC comprehensive verification that the pipe support capability is greater than the stress placed on the support by the piping system. The procedure used and typical checklists developed were reviewed and found acceptable.

4.5.9. Valves, Feature 9

A representative group of 28 valves was selected for review from the major piping runs in the Safety Injection System. Certain design documents associated with these valves were reviewed to determine the extent to which the specifications issued for the valves were responsive to the bases and methodology prescribed in the FSAR, and to verify that the seismic design requirements applicable to the valves and their installation had been adequately addressed.

Of the 28 valves reviewed, 24 were specified by BPC; the remaining four were the responsibility of CE. These valves range in size from a 3/4-in. 600-lb manually operated globe valve, to a 24-in. diameter clamshell type of check valve. There were three electric-motor-driven valves, three pneumatically operated valves, two solenoid valves, two relief valves, and 18 miscellaneous manual or process operated valves (check, stop check, etc.).

Forty-three design documents were examined during this review. For each of the valves, the calculated seismic acceleration level was tabulated and compared to allowables, and several representative calculations were carefully reviewed to verify input information, assumptions, methods, and accuracy. During the design review of these valves, checklists were maintained to ensure that all appropriate areas of the review requirements had been met.

Only one PFR was written for the valves and it was classified as an Observation. This PFR (0051) noted that, for the CE-procured valves, a seismic acceleration limit of 3.0 g is specified in the design documents. This value is less than the 5.0-g figure specified for the BPC-supplied valves. This qualification difference was accommodated by checking the piping design calculations at the points where CE valves are located and verifying that these valves are installed and supported in such a way that their 3.0-g qualification limit is not exceeded. The impact of this difference produces no significant effect on the seismic capability of the valves.

The overall review of these 28 valves revealed no problems of consequence, and as a result the seismic design requirements established in the appropriate valve documents were considered to be adequate.

4.5.10. Instruments, Racks, and Panels, Feature 10

A representative cross section of control and instrumentation equipment was selected for seismic review. This equipment controls and monitors the SIS for Unit 2. Included in this review were the following equipment items:

1. Auxiliary relay panels 2L-34 and 2L-35 in the engineered safety features actuation system (these features control operation of valves, pumps, fans, and dampers, etc., based on input signals from the plant protection system).

2. Local control room panels 2L-071 and 3L-071 which, among other functions, provide for operation of valves HV-9345 and HV-9341.

3. Local panels 2L-123, 2L-127, and 2L-147, which contain plant control instrumentation.

4. Auxiliary relay panel 2L-413, which contains a series of relays and handswitches.

5. Electrical penetration assemblies.

6. A series of nine instruments, including temperature sensors, transmitters, and recorders (the locally mounted transmitters are evaluated as part of Feature No. 19).

7. Two position-indicating limit switches associated with valve HV-9341.

8. Two panel-mounted handswitches, HS-9301-2 and HS-9391-2.

9. Three position-indicating lights, two for valve HV-9340 and one for the FV-0306 manual bypass valve. This equipment was furnished by either BPC, CE, or SCE, depending on the function involved.

Twenty-five documents were reviewed during this evaluation. The various procurement specifications supplied to the vendors were examined to determine the adequacy of the prescribed seismic qualifications with respect to instrument panel mounting details. This examination included the operability of the panels under seismic conditions, as well as their ability to remain intact during and after the DBE event. Procedures and seismic qualification methodology delineated in these specifications were also reviewed for completeness and applicability, and the listed seismic response spectra were checked for consistency with instrument and equipment location. These procedures were also checked to verify consistency with source documents such as IEEE-344 and the FSAR. A review checklist was maintained for each of the above-listed equipment items to verify that design inputs were properly incorporated and that assumptions and applicable codes and standards were correctly interpreted.

Twelve PFR's were issued, only one of which was classified as an Observation. The remaining 11 were invalid.

The valid PFR (0027) concerned a lack of reference to the relay contact chatter phenomenon as part of the procurement specifications. However, a review of in-situ test data indicated that the relays did function properly during testing. This PFR was therefore classified as an Observation.

No significant issues were discovered during this review, and it is concluded that the seismic qualification of the selected instruments, racks, and panels is satisfactory.

4.5.11. Switchgear and Power Panels, Feature 11

A typical grouping of safety-related electrical control equipment and the related panels was selected for this seismic review. The mechanical equipment that operates on power distributed by means of the subject electrical equipment is part of the SIS of Unit 2. This seismic review involved three representative types of electrical equipment: 4160-v switchgear, 408-v motor control centers, and circuit breakers. BPC was responsible for supplying this equipment.

The 4160-v switchgear supplies power to the LPSI pump (P-016), and the motor control centers supply power to motor-operated valves HV-9301, HV-9322, and HV-9340. The panels for this equipment are located at elevation 50 ft in the control area of the auxiliary building. The circuit breaker services a back-up function to protect the integrity of a containment electrical penetration and is located at elevation 63 ft-6 in. in the north electrical penetration room.

Three design documents were examined during this seismic review, with emphasis on the following three primary questions:

1. Were the proper seismic response spectra provided to the equipment ment supplier, and was the correct response properly incorporated during subsequent equipment testing?

2. Has the equipment mounting interface been conservatively simulated during equipment testing, and were the equipment mounting seismic loads submitted to BPC for incorporation in the details of the anchor design?

3. Have the basic criteria for Seismic Category I equipment been included in the specification documents and have those criteria been met?

During the course of this review, particular note was taken to assure that the effects of heavy cabling, bus-bars, and panel interconnections had been considered in the seismic design. In addition, documented locations and floor elevations for the various panels were verified so that it could be confirmed that proper seismic response spectra had been used in the specifications.

The interim report questioned the design associated with the floor mounting configuration for the panels, and the floor embedment constraints. Based on additional information supplied by BPC, and the vendors, it was determined that this interface was adequate. (See also Section 4.5.21.)

Throughout the seismic review of switchgear and power panels, checklists were maintained to assure that all required aspects of the evaluation were being accomplished.

Five PFR's, three of which were classified as Observations, were issued for these switchgear and power panels. The other two were invalid.

The first PFR (0031) noted that the seismic response spectra contained in the procurement specifications for motor control centers was marked "Preliminary." Subsequent correspondence, however, indicated that the proper response spectra were used during equipment testing but that the "Final" response spectra still required formalization in the procurement specification. This was classified as an Observation.

A second PFR (0053) discussed an error in an elevation callout for the 4160-v switchgear. Following the design review, it was established that the correct elevation spectral response data were used during equipment testing, but the fact that an incorrect elevation listing was shown in the specification was determined to be valid. The PFR was classified as an Observation.

The last PFR (F003) involved questions regarding the validity of extrapolating seismic data for one-and five-panel arrays to larger numbers of arrays (6 to 13 panels). Data from tests of cantilevered rectangular plates showed that additional panels in general produce a still more rigid installation, although it is recognized (and supported by test data) that very long arrays of panels could respond less rigidly under seismic excitation. Since it was determined that the maximum number of Unit 2 panels was 13, and that this number was still in the rigid range, the original test conclusions remain valid. This PFR was classified as an Observation.

No major problems were found during this review concerning the adequacy of the seismic qualification requirements of this equipment.

4.5.12. Electrical and Control Cables, Feature 12

One design document was reviewed for this feature. The objective of the review was to verify that safety-related cables associated with instruments and controls (Feature No. 10) in the segment of the SIS are routed in Seismic Category I raceways. Review of BPC's design criteria for Seismic Category I cable tray supports and design criteria for Seismic Category I electrical conduit supports was completed as part of Feature No. 20. In these design criteria, BPC specifies that all tray and conduit supports in Seismic Category I buildings must be designed as Category I supports irrespective of electric cable classification or tray type. The cables for the instruments and controls of the SIS are located in the safety equipment building, reactor containment building, and electrical tunnels, all of which are Seismic Category I structures. According to BPC's design criteria, all cable tray and conduit supports in these buildings are Seismic Category I items; this then assures that electrical cables in these buildings are located in Seismic Category I trays and conduits.

No PFR's were written for Feature 12.

It was concluded that the electrical and control cables that were reviewed are routed in the proper trays and raceways. This assures that their installation is seismically adequate.

4.5.13. Dynamic Analysis of Reactor Coolant System, Feature 13

The major components of the reactor coolant system include the reactor vessel, two steam generators, four reactor coolant pumps, supporting structures for each component, plus the interconnecting piping system (two hot legs and four cold legs).

CE was responsible for developing the design of the nuclear steam supply system, including the requirement to provide a seismic analysis of the reactor coolant system.

A simplified three-dimensional mathematical model of the reactor containment building was generated by BPC and was incorporated into the CE seismic analysis.

The reactor coolant system dynamic analysis review was conducted in three successive steps:

1. A review was made of the FSAR seismic analysis procedure (Section 3.7) and of its implementation as displayed by CE in the "Flow Diagram of a Computer Code for STRUDL Model."
2. A detailed review was made of the reactor coolant system mathematical model. The review included an independent calculation of the majority of the elastic and inertia properties. This was followed by a review of the dynamic behavior of the mathematical

model of the reactor coolant system coupled with the reactor containment building. Dynamic behavior includes eigenvalues, eigenvectors, participation factors, and modal damping values.

3. An evaluation was made of a set of in-structure response spectra and seismic loads based on experience and engineering judgment.

Thirty-four documents pertinent to the seismic analysis of the reactor cooling system were reviewed. During analysis of the reactor coolant pump support structure elastic properties, the vertical and horizontal support stiffnesses were discovered to be underestimated. The pertinent calculations were reviewed by CE to determine if any increase in the reactor coolant pump support stiffness (the seismic review had determined such stiffness to be the case) would produce an increase in motion decoupling between the reactor containment building and the reactor coolant pump. The net effect of this decoupling increase is a lowering of calculated seismic loads resulting in an increase in design margin.

During the course of this review, checklists were maintained to assure that all aspects of the seismic design were properly addressed.

One PFR was written for this analysis, but it was determined to be invalid.

No major questions were found based on the results of this review, and the seismic analysis of the reactor coolant system was therefore concluded to be satisfactory. The output in terms of loads, displacements, and accelerations is valid for use in the seismic design of the reactor coolant system components.

4.5.14. Reactor Coolant Pump and Supports, Feature 14

The primary coolant system associated with Unit 2 contains four large pumps, two in each of two loops. These pumps, each of which is designed to provide a normal operating flow rate of 100,000 gpm, were manufactured by

Byron Jackson, and are driven by 9700-hp electric motors. This vertically oriented pump/motor installation is 12 ft in diameter and 32 ft high. The seismic design review was limited to the analysis of the 1A pump installation along with its various supports. CE was responsible for specifying the pump and the hydraulic snubber attached to the motor, plus the vertical and horizontal support columns for both the pump and the motor. Twenty-two design documents were reviewed during this seismic analysis, and for each of the areas reviewed (pump, snubber, and supports), proper checklists were maintained to ensure that all required aspects of the seismic evaluation were being investigated.

The pump specifications and the Byron-Jackson performance reports were reviewed to determine that these documents were consistent with applicable FSAR statements, that seismic load tables were correctly referenced, and that the faulted conditions (DBE/LOCA) were being considered. These data were found to be consistent, and nozzle loads plus g-loadings on the pump were also found to be satisfactorily referenced.

The large horizontal and vertical support columns (5 ft to 7 ft long, 9 in. in diameter) were reviewed to verify the structural analysis. This review included the eight Byron-Jackson anchor clevises on the pump skirt, plus the connecting bearings and pins at both ends of the columns. Actual bolt anchoring into the nearby walls and floor are a BPC responsibility. The computer program MARC was used by CE to define the pin load distribution and to check the vertical columns for buckling.

The large hydraulic snubber assembly (1.5 ft dia, 5 ft long, 2000 lb) is mounted between the upper section of the motor and a nearby concrete wall. This device allows the slow movement associated with thermal expansion, but locks up and becomes a rigid support to resist OBE (90,000 lb) and DBE (826,000 lb) seismic loadings. The snubber stress report was reviewed and reconciled to the FSAR and the ASME code requirements and was found to be adequate.

Three PFR's were written against the pump and its supports. Two of these are classified as Observations, and one was invalid.

PFR F098 was written against the pump snubber assembly. It was determined that the procurement specifications did not delineate seismic qualification requirements; neither procedures nor criteria were mentioned from which a qualification program could be established. Subsequent information supplied by CE indicated that the snubber itself was adequately qualified by analysis and that the snubber appurtenances were not required to be operable during or after the DBE event. The impact of the deficiency in the procurement specifications, to waive qualification of the appurtenances, was not considered significant. The PFR was therefore classified as an Observation.

The second PFR (F102) involved calculation of stresses for the column anchor clevises on the pump skirt. It was determined that a nonconservative analysis technique was used to evaluate these stresses, and that design allowables might be exceeded. However, it was later shown by an independent calculation by TPT that the original clevis design was satisfactory. Checks were also made by TPT for other such clevis designs and these were also found to be adequate. This PFR was classified as an Observation.

No major problems were discovered during the review of the 1A reactor coolant pump specification and the design of its associated supports and snubber, that would indicate the seismic design to be inadequate.

4.5.15. Reactor Vessel Support, Feature 15

This seismic design review pertained to the vertical and lateral support structure which bears the physical weight of the reactor vessel plus its internals. The design support load under normal operating conditions is 2092 tons and is carried by four 21 ft-long steel columns, 11 in. by 30 in. in cross-section. The overall support assembly consists of the columns, the reactor inlet piping nozzles, upper support flanges, and column expansion plate subassemblies. Each of the four inlet nozzles is fabricated with the upper column support flange as an integral part. The column support bolts to the upper flange of the column, and the columns in turn are supported on the expansion plates that bear on the building foundation.

The reactor vessel supports were designed and fabricated by CE, and are intended to accommodate the reactor vessel under all normal loading conditions, plus the effects of combined loads induced by the DBE.

Fifteen documents were reviewed to establish the seismic design adequacy of the vessel support structure. These included five design specifications, two stress reports, and eight design and installation drawings. The review included verification of the seismic loadings specified in the project specification and a comparison with the DBE loading values listed in the FSAR. In addition, the procedures and calculations associated with the stress analysis for this support structure were examined to verify the adequacy of the design. Seismic review checklists were maintained as required to ensure that all aspects of the seismic design had been covered.

The design review of the reactor vessel support included an evaluation of the technique employed by CE in combining seismic-induced loads with other loads (Normal Operation and LOCA for faulted conditions). An equivalent static analytical technique is used by CE wherein the DBE loads were calculated from seismic acceleration input obtained from a dynamic analysis and combined with Normal Operation and LOCA loads. The LOCA loads are input as jet forces for a specific pipe break case. Normal Operation includes dead weight and normal thermally induced loads. The specification required the absolute sum technique for combining LOCA and DBE loads be used. The equivalent static model compares well with the model developed for the time-history dynamic analysis. The equivalent static analytical technique for combining design loads is considered valid and produces reactions at the reactor vessel support columns which are combined in an absolute sum.

Three PFR's were issued for the reactor vessel supports. One of these was classified as an Observation and two were invalid.

PFR F096 pointed out a discrepancy in geometry and material specifications associated with the reactor vessel inlet nozzle forging. Although the allowable value cited was incorrect, the resulting effect on calculated stresses did not materially affect the conservative design margins. This PFR was classified as an Observation.

No significant problems were found during this design review, and it is therefore concluded that the seismic design of the reactor vessel support structure is adequate.

4.5.16. Fuel Element Grid Spacers, Feature 16

The fuel element grid spacers are an integral part of the reactor core, and are intended to provide necessary separation between fuel rods. These spacers are fabricated of various materials (Zircaloy, Inconel, etc.) and are located at appropriate elevations along the vertical length of the fuel asemblies.

Eleven design documents were examined during the course of this seismic review. Seismic qualification of these spacers was accomplished by a combination of analysis and testing. The input seismic forces were based on a time-history analysis, and were applied to the reactor vessel at its supports. The analytical core model was then developed and analyzed for acceptable response of the internals, including the grid spacers. During the review, a checklist was maintained to ensure that all areas of the seismic design review were adequately addressed.

Combustion Engineering acceptance criteria for the spacer design were based on a series of licensing reports and safety studies prepared by various organizations for the NRC. Based on these documents, several tests on the spacers were performed by CE. The tests involved static lateral deflection and stiffness, natural frequency, and critical damping ratio. The test criteria stated that no significant permanent set would be allowed to occur that might result in reduction of the coolant flow area. The amount of allowable permanent set within the spacer grid was subsequently established as an amount equal to or less than manufacturing tolerances. These criteria were met during test.

No PFR's were written for the fuel rod grid spacers.

Based on this review it is concluded that the seismic design of these fuel rod grid spacers is adequate.

4.5.17. Reactor Coolant System Cold Leg (Piping), Feature 17

The reactor coolant system cold leg includes a segment of 30-in. dia high-pressure piping, which forms that portion of the primary coolant system between the outlet of the steam generator, through a reactor coolant pump (one of four identical pumps), and into a typical reactor vessel inlet nozzle. The piping and fittings that are seismically analyzed in this review include the following items: main line piping and elbow fittings, the safety injection nozzle, the charging inlet nozzle, the pressurizer spray nozzle, and the letdown and drain nozzles. Specifically excluded from consideration in this review were the steam generator and reactor inlet nozzle interfaces, plus any piping beyond the various branch nozzles connected to the main cold-leg piping.

Six documents were reviewed during this effort, including a CE report that describes the analytical approach used to evaluate this piping segment. This seismic review consisted primarily of checking the methodology and consistency of the procedure assumptions and input information used in the cold-leg piping analysis. The results were then evaluated to determine the combined stresses that developed, based on consideration of all potential loading conditions. For the piping, simple hand calculations were used for the initial wall sizing as suggested by the ASME code. For nozzles, stress conditions were established by using a photoelastic approach plus computer modeling. These models include ring/thick-shell theory, finite element analysis, and a stress indices approach as described in the ASME Code, Section III. Checklists of seismic design criteria were maintained during this review to ensure that all applicable aspects of codes and standards were addressed.

The mathematical models used in the various calculations are approximations of the actual geometry; furthermore, no specific justification (more detailed models or experimental results) were provided with which to determine the degree of conservatism associated with the overall cold leg analysis. Therefore, selected calculations and analyses were performed during this review to establish adequacy of the models. No significant problems were found in either the analytical approach used for the piping and nozzle analyses, or with the predicted stress levels.

Two PFR's were issued for the cold-leg piping, one of which was an Observation. The other PFR was invalid.

PFR F095 questioned the method used to combine stresses during the piping analysis. Further documentation supplied by CE indicated that certain stresses were incorrectly combined but that the design margins remained adequate. This PFR was classified as an Observation.

As a result of this review no problems were found that showed the seismic design of the cold-leg piping to be inadequate.

4.5.18. Diesel Generator Fuel Oil Storage Tank, Feature 18

The diesel generator fuel oil storage tank is a 60-ft-long, 11-ft-diameter steel tank buried in the ground adjacent to the emergency diesel generator building. This tank is a BPC-procured item manufactured by Process Equipment Co., Boston, MA. Five specification and drawing documents were reviewed, and an additional five references were consulted to obtain further information and to establish seismic design guidelines pertinent to the design review. Checklists of seismic design criteria were maintained during this review to ensure that all pertinent items were addressed.

Following BPC-supplied design specifications and seismic response data, Process Equipment Co. designed, manufactured, tested, and delivered this tank to the site. A formal stress report was not prepared, since it was not required by the procurement specification. However, a detailed review was made with respect to the stress analysis associated with this tank and its various nozzles and connections, including dynamic and static loads and the effects of sloshing. This analysis reconciles the manufacturers' calculations to the applicable sections of the ASME code (1971). Minor discrepancies were discovered with respect to local stresses at nozzle penetrations. When these local stresses are combined with membrane stresses due to external soil pressure, allowable stresses for Class 3 components are exceeded. However, combinational stress limits appropriate to Class 1 equipment are not exceeded, and therefore the overall nozzle penetration design was considered satisfactory.

No PFR's were issued for the fuel oil storage tank.

No significant problems were found during this design review, and it is concluded that the seismic design of the this tank is adequate.

4.5.19. Two Locally Mounted Instruments, Feature 19

Two types of representative locally installed instruments (located near the process equipment, but not in panels) within the SIS were selected for seismic review. The safety injection tank T-008 level transmitter instrument plus an integral series of three system temperature transmitters were chosen. These instruments are specified and supplied by CE; however, their actual installation in the plant is performed following BPC procedures. The instruments are identified as 2LT-0313 (level transmitter) and TT-0352-2, TT-0351-1, and TT-0351Y (temperature transmitters). Twenty-two design documents were examined during this review.

The seismic review of these locally mounted instruments included a determination that the applicable procurement specifications properly represent commitments in the FSAR. In addition, appropriate installation and mounting design calculations were reviewed to confirm the adequacy of configuration and to verify the results presented. It was also noted that specific reference to IEEE-344 test criteria was included in the seismic documentation. Checklists were maintained during this review to ensure that all aspects of the seismic design criteria were included.

Three PFR's were written for these locally mounted instruments, however, all of them were classified as invalid.

No problems were found with respect to the seismic design of any of these locally mounted instruments or of their supports.

4.5.20. Cable Tray and Conduit Raceway Support Systems, Feature 20

This seismic review involves the design, analysis, and installation of Category I cable tray and conduit raceway support systems. Fifteen cable tray hangers and one typical conduit support were reviewed. These supports are located within the SIS boundary, and the cabling contained therein feeds power to the low-pressure safety injection pump (P-016) and to the refueling water storage tank outlet valve, HV-9301. Bechtel Power Corporation is responsible for the overall design of these raceway support systems.

Thirty-two documents were reviewed to establish the seismic adequacy of these support systems. The design criteria were reviewed to ensure that the proper seismic loads were being referenced and that these values were consistently utilized in the calculations.

The seismic review involved verification of assumptions and design loads, with special emphasis on seismic load derivation. The natural frequencies (periods) of the various support systems were verified, and load combinations were checked and compared with allowable stresses. Drawing details were examined to assure proper representation, and structural member connections and bracing were carefully reviewed. Checklists of seismic review criteria were maintained to ensure that all aspects of the seismic design were covered.

Three PFR's were issued for the raceway supports. One of these PFR's was classified as a Finding, one was determined to be an Observation, and one was invalid.

The first PFR (F032) pointed out that an incorrect load combination was used in the analysis of a cable tray support. This PFR was classified as an Observation, since the existence of a large design margin in this area resulted in an insignificant impact on the support function.

The second PFR (0009), classified as a Finding, pertains to the design adequacy of a cable tray support diagonal brace connection to the concrete slab. The primary concern was that the diagonal pullout design loads imposed on the concrete inserts exceeded allowable limits. In addition, the diagonal braces are designed to be installed at an angle of 45°, while the installation drawings allow these braces to be installed at 30°, 45°, and 60° angles. A field inspection revealed that the specific brace is installed at 67.3°, causing the dominant vertical load component to be at its worst case. In reviewing this Finding with BPC, it was also agreed that concrete insert pullout allowable values from the catalog should be discounted to account for the limited data base upon which the allowable values were derived.

A Corrective Action Plan has been prepared by BPC to resolve this PFR. This plan involves 1) a review of calculation packages to establish that proper methodology and procedure was used for design of bracing for cable tray supports, 2) calculation of maximum bracing angles for certain braces, plus use of actual maximum cable tray loadings rather than conservative design values, 3) use of qualified capacity reduction factors for connections, where the load capability data are referenced to a limited data base, and if necessary, reduction in the tray load to the actual electrical maximum rather than design tray loading figures, and 4) a field inspection sampling to verify that bracing angles and member sizes are consistent with calculations and drawings. This plan has been reviewed and is considered adequate to resolve the Finding.

The design review of the cable tray and conduit raceway supports indicate no area of concern as to the design adequacy once the Corrective Action Plan has been implemented.

4.5.21. Control Panel 2CR57, Feature 21

Control panel 2CR57 is located in the Unit 2 main control room area at elevation 30 ft. This is an engineered safety features panel that provides for the operation of safety-related equipment associated with the SIS and other equipment. This panel is installed at the end of the central horseshoe array in the control room. Bechtel Power Corporation is responsible for supplying this panel. BPC subcontracted the design and fabrication of this panel to Jelco, who, in turn, subcontracted the dynamic analysis and testing to Wyle Labs.

Six design documents were reviewed. In particular, BPC's procurement specification was thoroughly reviewed for adequacy of seismic requirements. A computer analysis involving 2- and 3-dimensional frame modal analysis was also reviewed. Four areas of interest were addressed: The dynamic interaction of coupled panels; the dynamics of a free-standing end-panel; the floor anchorage details; and the need for simulating the weight of wiring and associated items (clips, connectors, straps, etc.). Checklists were maintained to ensure that all the prescribed areas of the seismic review analysis had been addressed.

Three PFR'S were written for this control panel. Two of these were classified as Observations and one was invalid.

One PFR (0015) addressed the fact that the procurement specifications failed to require that panel wiring be considered during seismic testing. However, review of a seismic test report showed that the wiring was properly included during the in-situ tests; therefore, this PFR was classified as an Observation.

The second PFR (0017) related to the dynamic interaction of a panel with one free-standing face, plus panel interaction between bolted sections. Considerable analysis, plus review of existing seismic-related test documents, were required to resolve the dynamic interaction issue. The result indicated that the acceleration levels that would exist at panel 2CR57 during a DBE are conservatively less than the specified maximum of 3 g's, with the conclusion that the seismic qualification was met. This PFR was classified as an Observation.

Completion of this seismic review showed that no major problems exist, and that the seismic qualification of the control room panel 2CR57 is adequate.

4.5.22. Segment of Reactor Containment Building Internal Structure, Feature 22

Six general areas of the containment building internal structure were selected for seismic review. These areas included the concrete slab supporting the safety injection tank (T-009) at elevation 45 ft; a secondary shield wall between elevations 15 ft and 93 ft 6 in.; concrete floor slabs at elevation 63 ft 6 in.; vertical support columns between elevations 15 ft and 63 ft 6 in.; the safety injection tank upper lateral supports at elevation 63 ft 6 in.; and the reactor vessel and reactor coolant pump support areas of the containment building. Forty-seven documents were reviewed during the verification of seismic response for these items.

All of these structures were designed by BPC. In the case of the major component supports, the design input figures were referenced from CE load tables, and then a 15% contingency factor was applied to compensate for uncertainty associated with the preliminary status of the initial design loads.

The portion of the concrete slab that supports T-009 is situated between vertical support columns 12, 13, and 1, in an area which represents the severest floor loading conditions. In particular, the slab was checked to assure that the anchor bolts for T-009 would not pull out of the slab during a seismic event. Although the anchor bolt installation appears to be adequate, the design calculations did not include a check for the punching shear capacity of the concrete. However, an independent hand calculation showed that this shear capacity is not a controlling variable in the design.

The secondary shield wall analyzed in this review is located in the south and southeast area of the structure, near safety injection tank T-008. The design calculations for this wall utilized thin plate elements to determine forces and moments from which the reinforcing steel in the concrete could be sized. Since this wall is four feet thick, an independent investigation was made to establish the validity of using thin plate elements to model the configuration. The results showed that the use of thin shells does not significantly affect the design conservatism.

The concrete floor slabs at elevation 63 ft 6 in. are located between columns 6 and 8, outside the secondary shield walls, near safety injection tanks T-008 and T-009. This design review includes not only the slab, but also the supporting steel beams and girders. Analysis of this floor structure is done by assuming that the beams and girders do not exist and then dividing the slab into representative beam components upon which the total design load is supported. This arrangement conservatively accounts for the various large openings in the slab. The analytical results demonstrated that the overall seismic design is appropriately conservative.

The containment building internal structure also includes 13 vertical columns located outside the secondary shield walls, extending from the foundation at elevation 15 ft to the operating deck at elevation 63 ft 6 in. Three representative columns (6, 7, and 8) were selected for seismic design review. Loads and dimensions were verified, and the design calculations and details were examined. The floor slabs supported by the columns were observed to be appropriately proportioned between the columns, and the resulting structural loads were used to complete the column design. This evaluation indicated that the design is seismically adequate.

The seismic design review of the lateral supports for the safety injection tank included the following: design of the shear key and steel bracket, the bolts in the bracket, the bearing area against the plate, the side plates, the bearing of the bolt sleeves on the concrete, and the pull-out resistance of the anchor bolts. This support is located at elevation 63 ft 6 in. in the containment building. The design review verified loading values used in the references, and noted that an increase in actual bolt size to 1-1/4 in. had been made, from the original design value of 1-1/8 in., a change which further increased the design margin.

The seismic design loads calculated for the major component supports (concrete embedments) represent maximum forces and moments based on computer studies. These loads, which develop from a consideration of normal, earthquake, and LOCA events, are comparatively smaller than the design load actually used to seismically qualify the internal structure associated with this review.

Several computer programs were utilized during the design of the above-discussed structures. The codes were SAP 1.9, RESCO, and OPTCON. These codes were reviewed to establish that the appropriate input seismic response spectrum was utilized to generate the maximum design loads for the structural members. In addition, basic assumptions, load combinations, static analyses and seismic analyses were also checked. In general, the computer results appeared satisfactory.

During the course of the above review, a checklist of selected items was maintained to ensure that all aspects of the seismic design were adequately covered.

Two valid PFR's, both of which were classified as Observations, were issued for this internal structure.

The first PFR (F100) pointed out an error in the moment of inertia calculation for a concrete beam element. The error was valid; however, the floor beam was subsequently deleted from the building design. The finite element computer run in which the numerical error was found involved the basic stress analysis for the internal structure. The deleted beam was determined to be a nonessential element that afforded no lateral support, nor did it significantly interact with adjoining walls or slabs. The PFR was classified as an Observation.

The second PFR (F104) involved a failure to reflect the final reactor coolant pump support load figures in the completed calculations for these supports. It was acknowledged that a document revision is required by BPC to update the design calculations; however, the seismic review demonstrated that utilization of the final values resulted in stresses within allowable limits, and that the building supports for the pump are adequately sized. This PFR was classified as an Observation.

No major problems were found during the review of the seismic design of the reactor building internal structure, thus indicating that reasonable design margins exist, and that the structure is adequate.

This design feature (No. 22) was also selected with the intent of reviewing differences between Units 2 and 3. However, because of the right-hand/left-hand symmetry associated with the two units, it was subsequently established that there were no significant differences within the framework of seismic design specifications. This conclusion is based on a consideration of the applicability of seismic analysis to the structural elements involved and recognition that the procedures and calculations employed are not affected by the right-hand/left-hand orientation. The results, therefore, are relevant to either unit.

4.6. CONCLUSIONS

The seismic design review of 22 selected features of San Onofre Units 2 and 3, involving review of approximately 500 technical documents, resulted in filing of 89 Potential Finding Reports (PFRs). Forty-seven of these PFRs were invalid and of the 42 valid PFRs, 41 were classified as Observations and one as a Finding (see Table 4-4 for a list of valid PFRs).

TABLE 4-4
VALID PFR's

NOTE: All PFR's in this table were classified as Observations with the exception of the one Finding identified by an asterisk.

Review Feature	PFR	Design Organization	PFR Description
1. Dynamic analysis of reactor containment building	0057	BPC	Soil spring stiffness values are inconsistent in computer input
	0058	BPC	Deviation in computer input for beam element moments of inertia
	F105	BPC	Use of basemat tracking feature in the SMIS code
2. Auxiliary intake structure	F106	SCE	Combination of loads and moments in the structural calculations
	F108	SCE	Incorrect factor used in the moment distribution calculations
3. Refueling water storage tank	0008	BPC	Incorrect nozzle loads
	F071	BPC	Buckling criteria for tank design
	F078	BPC	Effects of hydrodynamic loading associated with sloshing
4. LPSI pump	0011	CE	Response spectra g-levels used for design
5. Safety injection tank	0037	CE	Inconsistencies in the moment of inertia values used in the floor support slab
6. Major piping	0001	BPC	Line incorrectly identified
	0003	BPC	Documentation of seismic anchor movement loads
	0006	BPC	Inconsistencies in support description. Wrong type of support analyzed
	0007	BPC	Inconsistency in use of response spectra

TABLE 4-4 (Continued)

Design Feature	PRF	Design Organization	PFR Description
	0023	BPC	Improper modeling of valve center of gravity
	0024	BPC	Improper documentation of seismic anchor movement data
	0035	BPC	Failure to incorporate changes in analysis following an equipment change
	0036	BPC	Valve and flange weights not considered during analysis
	0056	BPC	Inconsistencies between computer model and isometric drawing
	F040	BPC	Improper valve modeling in computer input
	F043	BPC	Inconsistencies in applying ZPA accelerations at valve locations
	F074	BPC	Incorrect pipe weight used in calculations
7. Small-bore piping		BPC	No valid PFR's
8. Pipe supports and snubbers	0014	BPC	Mislabeled node point on piping assembly drawing
	F065	BPC	Shear deformation effects on forces and moments
	F082	BPC	Weld stress calculations
	F091	BPC	Improper utilization of rock-bolt load criteria
	F092	BPC	Inadequate documentation of stresses
9. Valves	0051	CE	Inconsistencies in specification of valve acceleration levels
10. Instruments Racks and Panels	0027	BPC	Contact chatter in relays during seismic test

TABLE 4-4 (Continued)

Design Feature	PFR	Design Organization	PFR Description
11. Switchgear and power panels	0031	BPC	Use of "preliminary" response spectra in "final" specification
	0053	BPC	Elevation and location inconsistencies for switchgear
	F003	BPC	Extrapolation of seismic test data to multi-array motor control centers
12. Electrical and control cables		BPC	No PFR's
13. Dynamic analysis of the reactor coolant system		CE	No valid PFR's
14. Reactor coolant pump and supports	F098	CE	Seismic qualification criteria for pump snubber
	F102	CE	Nonconservative design of pump support clevises
15. Reactor vessel supports	F096	CE	Inlet nozzle material allowable stresses
16. Fuel element grid spacers		CE	No PFR's
17. Reactor coolant system cold leg (piping)	F095	CEC	Stress analysis intensity values - charging inlet nozzle

TABLE 4-4 (Continued)

Review Feature	PFR	Design Organization	PFR Description
18. Diesel generator fuel oil storage tank		BPC	No PFR's
19. Two locally mounted instruments		BPC CE	No valid PFR's
20. Cable tray and conduit raceway support systems	0009*	BPC	Anchor loads and allowable pull-out loads on support brace element
	F032	BPC	Load combinations for cable tray supports
21. Control panel 2CR57	0015	BPC	Specification omitted requirement for including wiring weight in tests
	0017	BPC	Dynamic interaction of coupled panels during seismic excitation
22. Segment of reactor containment internal structure	F100	BPC	Incorrect moment of inertia calculation
	F104	BPC	Failure to update load table data

Thirty-four valid PFRs were written on BPC's design resulting from the review of approximately 340 technical documents on 14 design features. The majority of the valid PFRs (19 of 34) pertained to the review of 4 features which involve design and analysis of piping, pipe supports, and cable tray supports. The remaining 15 valid PFRs resulted from the review of 7 other design features. The deviations uncovered were generally attributed to inadequacies in documentation and in the subsequent checking and internal review of the design. The design and construction experience of BPC, when coupled with the large degree of conservatism found in essentially every part of the design, resulted in these deviations having no significant impact on the seismic adequacy of the features reviewed.

The one PFR (0009) classified as a Finding identified an inadequacy in the BPC design of a cable tray support brace connection to the concrete. The diagonal pullout design loads imposed on the concrete insert exceed the allowable limits. A field inspection revealed that the specific brace is installed at an angle resulting in the dominant vertical load component to be at its worst case. The review also established the need to discount catalog allowable values to account for the limited data base used to derive the allowable values.

SCE's Corrective Action Plan for the above Finding includes 1) review of existing cable tray support calculations; 2) re-calculation using actual maximum cable tray loads; 3) derivation of a justifiable capacity reduction factor to be applied to allowable values, and, if necessary, reduction in the tray load to the actual electrical maximum rather than design tray loading figures; and 4) field inspection sampling to verify installed brace angles as consistent with calculations and drawings. The plan was reviewed and is considered adequate to resolve the concerns of the Finding.

In addition to concerns identified in the PFRs, certain trends noted during the design review were investigated. Two of these trends pertained to BPC's design activities, specifically 1) analysis of Unit 3 piping where it was neither identical nor the mirror image of Unit 2 piping, and 2) possible implications of node point disagreement between pipe stress isometrics and pipe support drawings. In both instances, it was established that BPC has on-going design verification programs that were reviewed and resolved the concerns in the trends noted.

Six valid PFRs were written on CE's design resulting from the review of approximately 150 technical documents on 8 design features. The deviations identified in these PFRs were judged to not significantly impact the seismic design adequacy of the NSS-supplied components covered in these reviews; the PFRs were therefore classified as Observations.

Two valid PFRs were written on SCE's design resulting from the review of five technical documents relevant to the design of the auxiliary intake structure. The deviations had no significant impact on the design adequacy of the auxiliary intake structure since SCE's approach to the design is conservative.

Only one technical Finding resulted from this design review. Implementation of the Corrective Action Plan will resolve the design inadequacy concerning the cable tray support design.

In summary, based on the review performed under this task and the review performed under Tasks A and B, the seismic design of San Onofre Units 2 and 3 is adequate.

5. AUDIT PLAN REVIEW, TASK D

Task D was designed to determine if SCE and BPC performed audits at the site and at fabricators' shops in the area of implementation of seismic design documents, and to evaluate the effectiveness and results of such audits. (CE audits were not included in this review, since CE had no responsibility for site audits, and the CE fabricators were audited by SCE.) In carrying out this task, the work focused on a review of audits that addressed site or fabricator's activity on seismic-qualified components, systems, or structures.

The scope of this task included the following:

- Preparation of a procedure to provide detailed working instructions for the review.
- Performance of a review of audit plan and schedule preparation.
- Performance of a review of audit plan implementation.

5.1. AUDIT PROCEDURE

The SCE and BPC procedural requirements (from 1971 to the present) for audit planning and scheduling were identified and evaluated against the applicable regulatory requirements.

The work was carried out by visits to the SCE and BPC offices, by interviews with cognizant personnel, and by the review of relevant procedures and regulatory requirements. The procedures collected and reviewed under this subtask are listed in Table 5-1.

The review showed that SCE and BPC procedures for audit planning and scheduling were consistent with regulatory requirements with the exception

TABLE 5-1
PROCEDURES GOVERNING AUDIT PLANNING AND SCHEDULING
(Reviewed for Task D)

SCE

- QA Reference Procedures Manual N18.04
  (Rev. 18)
  (Ref. 17-6)

  (Ref. 401)

BPC

- SONGS 2 and 3 Project Quality Program Manual Procedure No. 18 - "Project Quality Audit," Rev. 0 (October 1974) to Rev. 6 (Oct. 1979).
- LAPD Quality Program Manual, procedure No. 18.1, "QA Audits," REf. 0 (10-74) and Rev. 1 (10-75).
- LAPD QA Department Procedures, No. 5.1, "Project QA Audits," Rev. 7 (1-15-76) to Rev. 15 (11-16-81).
- LAPD QA Standard, "Project Quality Audit," No. 12; Rev. 3 (12-28-73) and Rev. 4 (4-16-74).
- Procedure Supplier Quality Manual, 6th Edition, Rev. 0 (Nov. 1979); 5th Edition, Rev. 0-2 (October 1977-April 1978).
- Procurement Inspection Department Manual, 4th Edition, Section 3.1.2, "Audit Scheduling," Rev. 0 (May 1973) to Rev. 4 (Aug. 1976); 3rd Edition, Seciton 5.3.2.18, "Audits," Rev. 3 (March 2, 1970).
- Procurement Supplier Quality Department Procedure, PSQP-TS-4.2 Rev. 0 (4-81); "Audit Planning and Scheduling."

of one regulatory requirement that was not addressed in the SCE procedures. These procedures (prior to November 1981) did not require planned periodic audits to determine the effectiveness of the QA program as required in 10CFR50, Appendix B, Section XVIII. Procedures following November 1981 did include this requirement. PFR 0034 was issued on this deficiency.

PFR 0034 was the only PFR issued on this review item, and was classified as a Finding. The SCE Corrective Action Plan, SCE audit reports, and SCE supporting documents indicated that, although not specifically stated as an audit objective, audits did assess adequacy of controls beyond simply determining compliance or noncompliance with specific requirements. Thus, the PFR was satisfactorily resolved.

5.2. PREPARATION OF AUDIT PLANS AND SCHEDULES

The BPC and SCE records were reviewed to verify that audit schedules and plans were prepared in accordance with the procedural requirements identified in Section 5.1 above.

The results confirmed that audit schedules and plans were prepared as required. This work was completed by reviewing audit schedules and audit plans for BPC and SCE.

No PFR's were initiated.

5.3. IMPLEMENTATION OF AUDIT PLANS AND SCHEDULES

Audit plans were evaluated to determine if they included audits of the construction site and fabricator's shops to verify implementation of seismic design output.

This work was done in conjunction with the work identified in Section 5.2 above, and is documented by listings of specific audits performed at supplier locations and at the site in the area(s) of interest.

This work confirmed that the area of interest (implementation of seismic design documents) was, in fact, included in the SCE and BPC audit programs.

No PFR's were initiated.

5.4. EVALUATION OF AUDIT REPORTS

Audit reports identified in Section 5.3 above were examined for the following:

1. Determining if audit plans were carried out as required.

2. Identifying any deficiencies uncovered in the area of implementation of seismic design documents.

3. Evaluating the resolution of any such deficiencies to determine if they were satisfactorily resolved or corrected.

Audits were selected from each of the six categories, as shown in Table 5-2. A total of 279 audit reports were reviewed. For each audit, a checklist was completed to document the review.

The audits identified some deficiencies in the implementation of seismic design documents. While most of the deficiencies were satisfactorily resolved or corrected, several instances were noted in which this was judged not to be the case.

Sixteen PFR's were initiated. Ten of these PFR's were judged to be invalid after additional information was reviewed. The six valid PFR's are discussed in Section 5.5.

Overall, the review confirmed that a comprehensive audit program was implemented by SCE and BPC, and that audits were carried out as required.

TABLE 5-2
AUDIT REPORTS REVIEWED UNDER SUBTASK D4

Category		No. of Reports
1.	SCE audits of CE	37
2.	SCE audits of suppliers (CBI, Pullman-Kellogg, Westinghouse)	10
3.	SCE audits of BPC (site)	54
4.	SCE audits of SCE (site)	20
5.	BPC audits of suppliers	80
6.	BPC audits of BPC (site) and vendors (site)	78
	Total	279

5.5. CONCLUSIONS, TASK D

A total of seven valid Potential Finding Reports were initiated under Task D. These are listed in Table 5-3. PFR 0034 was initiated under the review described in Section 5.1; all other valid PFR's were initiated under the review described in Section 5.4. In summary, two PFR's were classified as Findings and five as Observations.

PFR 0034, classified as a Finding, was discussed earlier. It addressed a deficiency in the SCE audit procedures regarding lack of a procedural requirement for each audit to address effectiveness of the QA program. SCE corrected the procedural deficiency and provided evidence to demonstrate that the effectiveness evaluation was carried out, even though not explicitly required by procedure.

PFR F051, classified as a Finding, addressed the issue of BPC not maintaining a permanent record of audit nonconformances classified as "minor." The impact of this was that the permanent records appeared not to include all deficiencies or nonconformances uncovered in audits. This was judged to be a violation of ANSI N45.2 requirements regarding maintenance of records. SCE's Corrective Action Plan provided information showing that all deficiencies reported on BPC audit checklists were in fact maintained in the SCE permanent record system. Thus, the issue which led to the Finding classification was resolved and the substance of the PFR was, in essence, a perceived procedural violation by BPC that would not have been a Finding if the information in the Corrective Action Plan had been known at the time the PFR was classified.

Five PFR's were classified as Observations; one of these was against BPC, and four were against SCE. The Observation with regard to BPC concerned corrective action judged to be inappropriate. The BPC response and the nature of the corrective action was such that it was judged to have no adverse impact on safety.

TABLE 5-3
VALID PFR's ISSUED IN TASK D

PFR No.	Contractor	Subject	Classification
0034	SCE	Procedure inconsistent with Appendix B	Finding
F037	SCE	Procedure violation	Observation
F038	SCE	Procedure violation	Observation
F046	SCE	Procedure violation	Observation
F051	BPC	Report of deficiencies	Finding
F052	BPC	Corrective action	Observation
F054	SCE	Corrective action	Observation

The four Observations written concerning SCE included three on procedural violations and one on inappropriate or ineffective corrective actions.

The three procedural violations were judged to have no adverse impact on safety since it was shown that the subject in question (corrective action implementation and verification) was properly resolved, even though certain procedural steps were not followed.

The SCE Observation regarding ineffective corrective actions (PFR 0054) also led to a concern that, because of ineffective corrective actions and a continuing trend of deficiencies in implementation of Document Management Center procedures, there was a possibility that the document control system was not being implemented in accordance with procedures over an extended time. However, the activities performed in Task B and C of this review and the results of BPC and SCE audits of site activites indicated no evidence that the possibility expressed above actually led to the use of improper design documents.

Based on the work performed in this task, it is concluded that SCE and BPC did carry out an audit program of site and supplier activities in the area of implementation of seismic design documents. That program is documented by procedures responsive to Appendix B and by records of audit schedules, plans, audit reports, and follow-up reports. Further, those records do not indicate any significant unresolved deficiencies.

6. PIPE SEGMENT WALKDOWN, TASK G

The objective of this task was to verify, to the extent possible with nondestructive (e.g., visual) examination, the proper installation of a segment of pipe that was reviewed in Task C, Section 4.

The scope of this task included the following:

- Selection of a pipe segment that is representative of the piping reviewed in Task C in terms of length and complexity.
- Development of a specific procedure for conducting a field audit of the piping isometrics and support drawings.
- Performance of the on-site examination and evaluation of the results.

6.1. WALKDOWN RESULTS

A field audit of stress isometric drawings and a representative number of supports was performed on a portion of the safety-related piping and supports for the Safety Injection System (SIS) in San Onofre Unit 2. Approximately 300 ft of piping were examined.

Thirty-seven pipe support drawings and four piping isometric drawings were examined. A general audit that examined the attachment to the pipe and the attachment to the building structure was performed on 12 of the 37 pipe support drawings. A detailed examination, including measurements, was performed on one of those support drawings. An audit was also performed on the four piping isometric drawings that were referenced in these 12 pipe support drawings.

It was observed during this field audit that the piping was routed as defined on the stress isometric drawings and that the types of seismic restraints installed in the field agreed with the types shown on the stress isometric drawings. Some discrepancies were noted in dimensions, material sizes and shapes, and configuration between pipe support analysis drawings and the data acquired from the walkdown.

6.2. CONCLUSIONS

Two PFR's were issued and both were classified as Observations.

PFR 0042 concerned an as-built drawing error in the location of a clevis to which a strut was attached. BPC provided a calculation that verified that the as-built configuration was acceptable. This PFR also noted that clearance between the lugs on the pipe and the steps on the support were out of specification tolerance. The BPC response and a TPT calculation showed the allowable values were not exceeded.

PFR 0043 involved a drain line installation that did not conform to the BPC standard detail. BPC provided a calculation that verified that the as-built configuration was acceptable.

The walkdown showed the as-built configuration of this pipe segment to be adequate.

7. INDEPENDENT CALCULATIONS, TASK H

The objective of Task H was to perform independent calculations using alternate analytical techniques on selected features of San Onofre Units 2 and 3. Results were compared with original calculations for consistency. Features independently analyzed met the following criteria:

1. Features were reviewed in Section 4.5.
2. Features had clearly defined analytical interfaces.
3. Features are subject to seismic loads that are a significant fraction of total loads.
4. Features have relatively low design margin.

The features selected to be independently analyzed were 1) the low pressure safety injection pump support mount (Feature 4, Section 4.5.4) and 2) the safety injection tank (Feature 3, Section 4.5.3).

7.1. LOW-PRESSURE SAFETY INJECTION (LPSI) PUMP SUPPORT MOUNT

A modal analysis of the support mount of the low-pressure safety injection (LPSI) pump was performed with the use of the computer program MODSAP.* Data provided to BPC by CE/Ingersoll-Rand were used as input for this analysis, while other necessary information was calculated.

The finite element model used in the analysis simulated the support mount with 3-D beam elements. The LPSI pump was idealized as a rigid structure with a concentrated mass. Since the pump is bolted to the support

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*Johnson, J. J., "MODSAP, a Modified Version of the Structural Analysis Program SAP IV for the Static and Dynamic Response of Linear and Localized Non-Linear Structures," General Atomic Report GA-A14006 (Rev), June 1978.

mount, a pinned connection was assumed between the pump and support. The computed frequencies were compared with BPC's original calculation to verify that the pump mount was rigid as assumed in the original design.

The column base plates are bolted to the concrete slab with four bolts per base plate. These column base connections were modeled as both pinned and clamped conditions to bound the possible constraint by the foundation and base plate. For a pinned condition, the calculated fundamental natural frequency is 15.4 Hz, while for a clamped condition, the frequency is 31.3 Hz. These values differ substantially when compared with the 112 Hz originally calculated by BPC and are lower than the CE-specified minimum requirement of 33 Hz for the pump support. If the flexibility of the pump is considered, the resulting frequency may be further reduced. PFR-F101 was filed as a result of this discrepancy.

In response to the PFR, Bechtel agreed that the originally calculated fundamental frequency of 112 Hz for the pump support was too high and incorrect. Bechtel performed a more rigorous analysis (similar to TPT's) and derived fundamental frequencies of 15.5 Hz for the pinned condition and 32 Hz for the clamped condition. BPC conservatively estimated a fundamental frequency of 20 Hz for the support mount although the actual condition is closer to the clamped than the pinned condition. This frequency is outside the region of amplification of the floor response spectra at the pump location; the seismic response of the pump-support system as estimated from the floor response spectra at 20 Hz is less than the design basis of 1.5 g in the horizontal direction. Bechtel also checked to see that 20 Hz is not close to the fundamental frequencies of the piping connected to the pump, which ranged from 8.0 to 15.5 Hz. The fundamental frequency of 20 Hz results in minimal amplification with respect to the ZPA, which is consistent with the design requirement of a rigid support for the pump; therefore, the PFR was classified as an Observation.

7.2. SAFETY INJECTION TANK

A seismic analysis of the safety injection water tank was performed using the MODSAP computer program. The MODSAP model was developed based on a configuration drawing of the safety injection water tank and data supplied to CE by BPC. The data supplied to CE by BPC included the stiffnesses and the mass properties of the concrete floor, the stiffness of the seismic lug support, the response spectra for the seismic events at different elevations, and the relative horizontal displacements between the seismic restraint and the floor at the base of the tank for OBE and DBE conditions.

The MODSAP stick model, shown in Fig. 7-1, includes the dynamic effects of the water slosh. This model was used to evaluate the effects of the OBE and DBE response spectra. Hand calculations and the computer results of the MODSAP model were used to estimate the effects of seismic restraint-to-floor displacements. In all cases these results were consistent with the results of the CE analysis.

Table 7-1 compares the natural frequencies obtained from the two analyses. The comparison of the two bending frequencies is very close. The vertical frequencies are different, since CE used only one mass to model the tank where TPT used a set of distributed masses. The sloshing modes are different, since TPT used a more sophisticated sloshing model. Neither the vertical mode nor the sloshing mode has a significant effect on the design loads.

Table 7-2 compares the maximum forces and moments acting at the supports of the safety injection water tank as a result of a DBE. The critical stresses calculated using the CE loads are all more conservative than the stresses calculated using the TPT loads. The forces at the base of the support skirt and the loads acting at the seismic lugs are all higher for the CE model, as is the maximum moment in the support skirt. The moments acting at the base of the support skirt are more symmetrical for the MODSAP model than for the CE model.

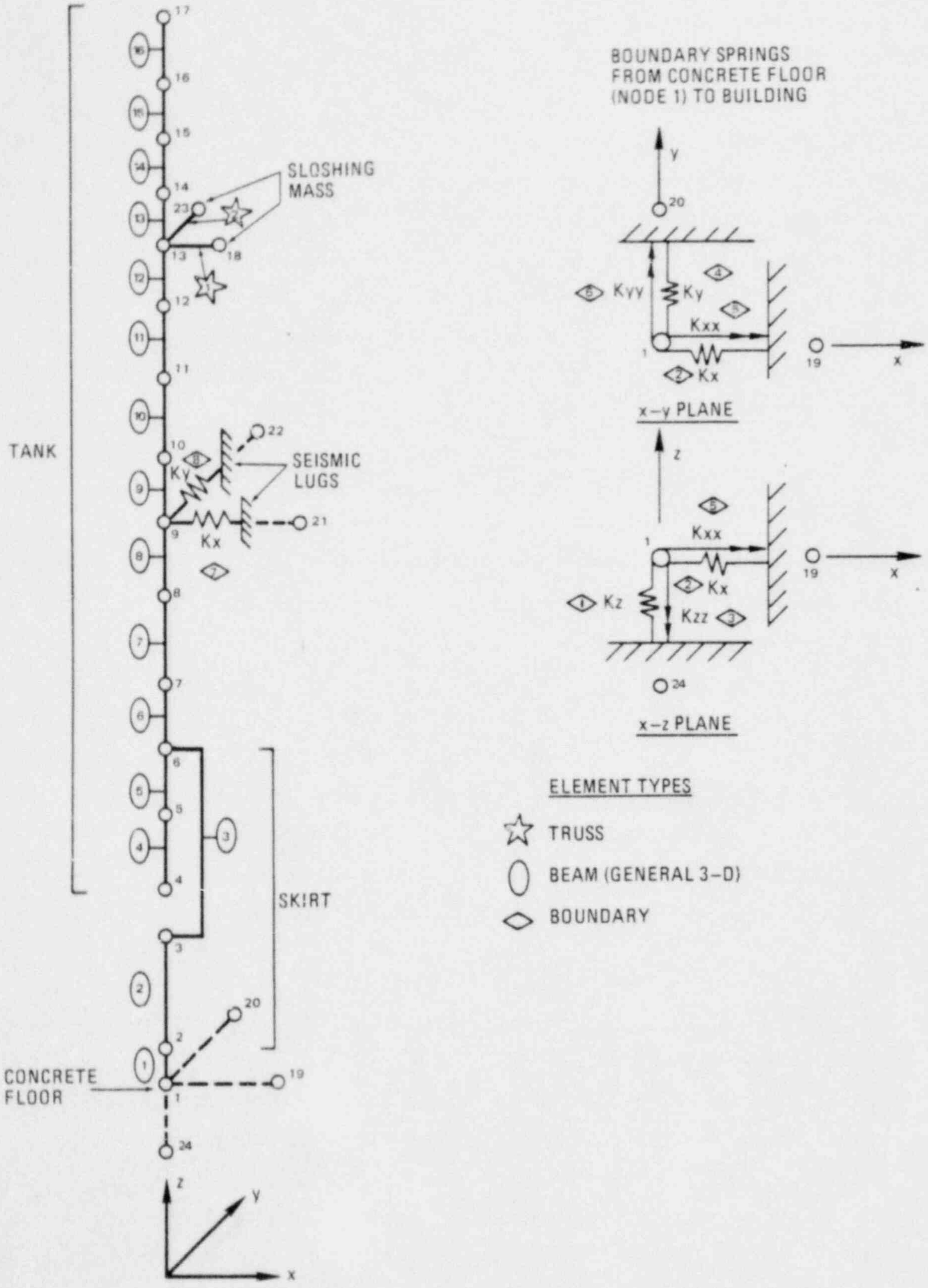

Fig. 7-1. MODSAP model of the safety injection water tank

TABLE 7-1
COMPARISON OF NATURAL FREQUENCIES FOR THE SAFETY INJECTION WATER TANK

Mode Description	Natural Frequencies (Hz)	
	CE	MODSAP
Sloshing - x-direction	0.581	1.044
Sloshing - y-direction	0.581	1.044
Bending - x-direction	20.41	20.32
Bending - y-direction	20.41	20.36
Vertical direction	24.97	28.33

TABLE 7-2

COMPARISON OF MAXIMUM FORCES, MOMENTS AND LOADS ACTING AT THE SUPPORTS OF THE SAFETY INJECTION WATER TANK FOR THE DBE RESPONSE SPECTRA

Result	CE Model	MODSAP Model
Dead Weight (lb)	208,396	204,259
Forces at base of the support skirt		
F_x (lb)	99,792	88,420
F_y (lb)	101,605	88,240
F_z (lb)	255,081	214,800
Moments at base of the support skirt		
M_{xx} (lb-in.)	3,510,717	4,097,000
M_{yy} (lb-in.)	4,036,067	3,935,000
Maximum moment acting on support skirt (at skirt-head interface)		
M_{xx} (lb-in.)	11,269,633	9,687,000
M_{yy} (lb-in.)	11,514,328	9,561,000
Loads acting at the seismic lugs		
Lug 1 (lb)	121,087	102,000
Lug 2 (lb)	119,232	101,450
Lug 3 (lb)	121,087	102,000
Lug 4 (lb)	119,232	101,450

The following are two differences between the TPT and CE computer models which contribute to the differences between the calculated forces and moments. (TPT's model is shown in Fig. 7-1 and CE's model is shown in Fig. 7-2.)

1. The mass moment of inertia used by CE for the concrete floor was too small. This was the subject of PFR 0037 resulting from the design review of this task under Features 3, Section 4.5.5.

2. The mass properties of CE's concrete floor were input at node 2 instead of node 1.

7.3. CONCLUSION

The results of the independent calculations using different analytical techniques performed by TPT confirmed the seismic design adequacy of the LPSI pump support mount designed by BPC and the safety injection tank designed by CE/Ingersoll Rand.

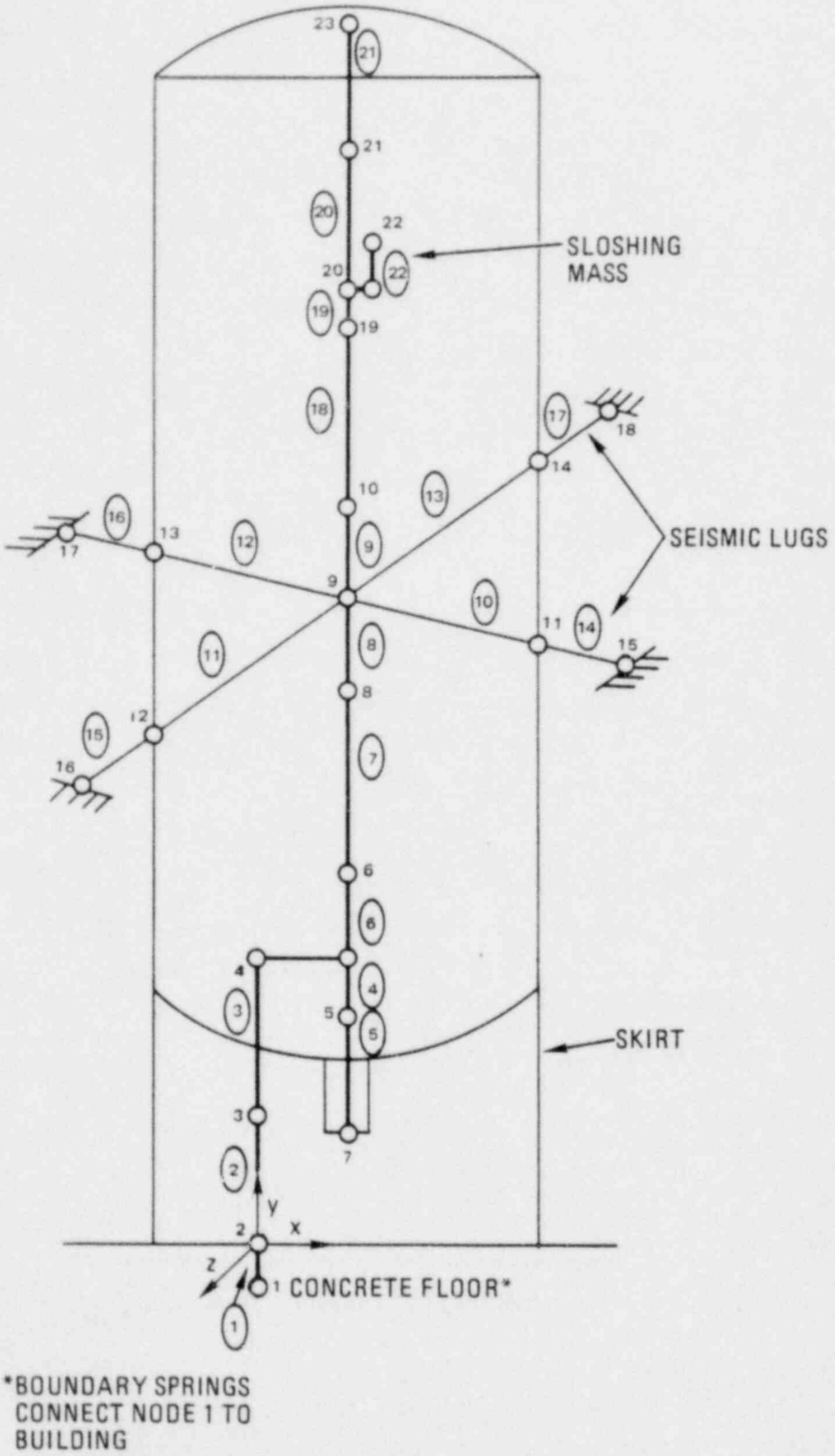

Fig. 7-2. STRUDL model of the safety injection water tank

8. TOTAL PROGRAM OVERVIEW

8.1. SUMMARY OF POTENTIAL FINDING REPORTS

PFR's are the mechanism that were used in this program to explore questions raised during the review process. This mechanism was highly formalized to assure that no pressure would sway the reviewers' technical judgment, thus allowing any potential comment or concern to be raised. As a result, many PFR's were created because of a lack of information or lack of adequate understanding of the process or approach used by SCE, BPC, or CE in the area of concern. Thus, many of the documented PFR's are satisfactorily answered and are declared invalid. During the course of the entire review, 170 PFR's were written, of which 77 were declared invalid or out of the scope of the review.

Of the valid PFR's (Table 8-1), 86 were Observations and 7 were Findings. These valid PFR's covered a wide range of deviations, including clerical violations, failure to establish adequate procedures, numerical errors, and errors in calculational logic. In all these valid concerns, however, none indicated any failure of the design to meet basic safety objectives. In some cases, some or all of the design margin was used, but since the basic design is very conservative, the ability to perform the required safety function was always preserved. This suggests that the combination of margin and the cognizant design organization's experience and judgment resulted in an adequate design in spite of minor errors in the design.

From another perspective, the number of deviations that are discovered in any examination depends on the "magnification" used. If insufficient magnification is used, no deviations will be discovered. If sufficient magnification is used, deviations will always be found. Torrey

TABLE 8-1
VALID PFR's ISSUED

NOTE: All PFR's in this table were classifed as Observations with the exception of the seven Findings which are identified with an asterisk.

TASK A

PFR No.	Contractor	Subject
0038*	CE	Project office - interface control
0047*	CE	Design reviews
0049	CE	Specification review
0052*	CE	Project Office - design coordination
F004	CE	Design control - training

TASK B

Component	PFR No.	Contractor	Subject
Auxiliary intake structure	F010	SCE	Purchase specification
	F011	SCE	Design document reviewer comments
	F012	SCE	Professional Engineer's approval of specification< or calculation
	F013	SCE	Specification distribution<
	F014	SCE	Review of revised calculation
	F015*	SCE	Computer program validation
Piping analysis	F020	BPC	Calculations
	F021	BPC	Calculations
	F022	BPC	Calculations
	F023	BPC	Calculations
	F024	BPC	Calculations

TABLE 8-1 (Continued)

TASK B (Continued)

Component	PFR No.	Contractor	Subject
	F027	BPC	Calculations
	F077	BPC	Computer codes
Pipe support	F056	BPC	Field change notice
	F066	BPC	Calculations
	F097	BPC	Calculations
Cable tray hangers	F029	BPC	Drawing review and approval
	F031	BPC	Drawing
Containment	F034	BPC	Calculation
	F036	BPC	Calculation
	F044	BPC	Calculation
Tanks	F018	BPC	Specification
Boric acid makeup	F089	CE	Calculation, method of verification
Alarms	F075	BPC	Specification revision
Pump	F042	BPC	Specification change notice
Electrical penetrations	F060	SCE	Specification
Valves	F061	BPC	Specification change notice
Control board	F062	BPC	Supplier design change request
Generic	F016	SCE	Review and approval requirement for calculations
	F017	SCE	Quarterly listing of new and revised standards
	F035	BPC	Bechtel Design Criteria Manual review
	F079	CE	Design input requirements
	F080	CF	BOP design criteria
	F081	CE	Design basis information
	F086	BPC	Authorization design changes
	F088	CE	Design distribution/approval form

TABLE 8-1 (Continued)

TASK C

Review Feature	PFR	Design Organization	PFR Description
1. Dynamic analysis of reactor containment building	0013	BPC	Deviation in computer input for mass values at a node
	0057	BPC	Soil spring stiffness values are inconsistent in computer input
	0058	BPC	Deviation in computer input for beam element moments of inertia
	F105	BPC	Use of basemat tracking feature in the SMIS code
2. Auxiliary intake structure	F106	SCE	Combination of loads and moments in the structural calculations
	F108	SCE	Incorrect factor used in the moment distribution calculations
3. Refueling water storage tank	0008	BPC	Incorrect nozzle loads
	F071	BPC	Buckling criteria for tank design
	F078	BPC	Effects of hydrodynamic loading associated with sloshing
4. LPSI pump	0011	CE	Response spectra g-levels used for design
5. Safety injection tank	0037	CE	Inconsistencies in the moment of inertia values used in the floor support slab
6. Major piping	0001	BPC	Line incorrectly identified
	0003	BPC	Documentation of seismic anchor movement loads
	0006	BPC	Inconsistencies in support description. Wrong type of support analyzed
	0007	BPC	Inconsistency in use of response spectra
	0023	BPC	Improper modeling of valve center of gravity
	0024	BPC	Improper documentation of seismic anchor movement data

TABLE 8-1 (Continued)

TASK C (Continued)

Review Feature	PFR	Design Organization	PFR Description
	0035	BPC	Failure to incorporate changes in analysis following an equipment change
	0036	BPC	Valve and flange weights not considered during analysis
	0056	BPC	Inconsistencies between computer model and isometric drawing
	F040	BPC	Improper valve modeling in computer input
	F043	BPC	Inconsistencies in applying ZPA accelerations at valve locations
	F074	BPC	Incorrect pipe weight used in calculations
7. Small-bore piping		BPC	No valid PFR's
8. Pipe supports and snubbers	0014	BPC	Mislabeled node point on piping assembly drawing
	F065	BPC	Shear deformation effects on forces and moments
	F082	BPC	Weld stress calculations
	F091	BPC	Improper utilization of rock-bolt load criteria
	F092	BPC	Inadequate documentation of stresses
9. Valves	0051	CE	Inconsistencies in specification of valve acceleration levels
10. Instruments Racks and Panels	0027	BPC	Contact chatter in relays during seismic test

TABLE 8-1 (Continued)

TASK C (Continued)

Design Feature	PFR	Design Organization	PFR Description
1 Switchgear and power panels	0031	BPC	Use of "preliminary" response spectra in "final" specification
	0053	BPC	Elevation and location inconsistencies for switchgear
	F003	BPC	Extrapolation of seismic test data to multi-array motor control centers
12. Electrical and control cables		BPC	No PFR's
13. Dynamic analysis of the reactor coolant system		CE	No valid PFR's
14. Reactor coolant pump and supports	F098	CE	Seismic qualification criteria for pump snubber
	F102	CE	Nonconservative design of pump support clevises
15. Reactor vessel supports	F096	CE	Inlet nozzle material allowable stresses
16. Fuel element grid spacers		CE	No PFR's
17. Reactor coolant system cold leg (piping)	F095	CE	Stress analysis intensity values - charging inlet nozzle

TABLE 8-1 (Continued)

TASK C (Continued)

Review Feature	PFR	Design Organization	PFR Description
18. Diesel generator fuel oil storage tank		BPC	No PFR's
19. Two locally mounted instruments		BPC CE	No valid PFR's
20. Cable tray and conduit raceway support systems	0009*	BPC	Anchor loads and allowable pull-out loads on support brace element
	F032	BPC	Load combinations for cable tray supports
21. Control panel 2CR57	0015	BPC	Specification omitted requirement for including wiring weight in tests
	0017	BPC	Dynamic interaction of coupled panels during seismic excitation
22. Segment of reactor containment building internal structure	F100	BPC	Incorrect moment of inertia calculation
	F104	BPC	Failure to update load table data

TABLE 8-1 (Continued)

TASK D

PFR No.	Contractor	Subject
0034*	SCE	Procedure inconsistent with Appendix B
F037	SCE	Procedure violation
F038	SCE	Procedure violation
F046	SCE	Procedure violation
F051*	BPC	Report of deficiencies
F052	BPC	Corrective action
F054	SCE	Corrective action

TASK G

PFR No.	Contractor	Subject
0042	BPC	Piping support installation not per drawing
0043	BPC	Installed piping does not conform to stress isometric

TASK H

PFR No.	Contractor	Subject
F101	BPC	LPSI pump support fundamental frequency

Pines Technology has used a high magnification in the review of San Onofre Units 2 and 3, and deviations have been found. In spite of this, no design inadequacies have been found that could endanger the public health and safety.

Each Observation and Finding is discussed in the appropriate section, together with its implication on the Task conclusion.

8.2. SUMMARY OF CORRECTIVE PLANS

A Corrective Action Plan was prepared by SCE for each Finding issued in this review program. The purpose of these plans was to describe the approach planned to correct any deviations identified in the Findings. These Plans were reviewed to assure that the deviation was properly understood and that the Plan when implemented would remove any concern the Finding may have revealed about the adequacy of the seismic design.

Seven Findings resulted from this review. These were addressed in six Corrective Action Plans.

All six Plans demonstrated that the deviations in the Finding was indeed properly understood and when implemented, the planned actions taken in concert with the rest of the Program would remove any concern that the Finding may have raised about the adequacy of the seismic design.

The Findings that resulted from Task A, PFR 0038 and 0052 were covered by a single Corrective Action Plan. In this Plan it was shown that the Finding was understood and generally treated satisfactorily. Any remaining concern was totally removed by the reviews done in Task B, which was specifically altered to identify any deficiencies resulting from this lack of procedures, and Task C. PFR 0047 was covered by a separate Corrective Action Plan. In this Plan, too, it was shown that the Finding was understood and generally treated satisfactorily. The remaining concern, that of the lack of evidence that some of the design reviews committed in the PSAR were carried out, was satisfactorily removed by the review in Tasks C and H. The results from these tasks indicated that there was no design inadequacy resulting from a lack of design review.

One Finding, PFR F015, resulted from Task B. The corresponding Corrective Action Plan, together with the review in Task C, satisfactorily remove any concern about design adequacy.

Another Finding, PFR 0009, resulted in Task C. The Corrective Action Plan demonstrates that the concern is understood and, when implemented will remove all concern on the cable tray support design.

Finding PFR F051 and 0034 resulted in Task D. The information submitted in the Corrective Action Plan for F051 indicated that the Finding was due to lack of information and thus inappropriate. The Corrective Action Plan submitted satisfactorily closed the Finding. The Corrective Action Plan for PFR 0034 indicated that the procedural deviation had been corrected and provided evidence to demonstrate that the effectiveness evaluation was carried out, even though not explicitly required by procedure. Thus, these Corrective Action Plans remove any concern about design adequacy.

In summary, all the Findings are either satisfactorily closed out or will be upon completion of the Corrective Action Plan. It should be noted that no physical changes were required as a result of the Findings.

8.3. CONCLUSIONS

The independent verification program for San Onofre Nuclear Generating Station Units 2 and 3 was structured to verify that the design process adequately converted the seismic design bases specified in the Final Safety Analysis Report into design documents that were transmitted to the constructor or fabricator. The major tasks, Tasks A, B, C, and H (see Sections 2, 3, 4, and 7) taken together have provided a discerning basis to judge the adequacy of the seismic design. The conclusions on the seismic design are as follows:

1. SCE, CE, and BPC each had design control procedures in place during the design process. In general, these procedures were adequate. The one exception was the lack of procedures at CE to assure management coordination and overall design review as committed in the PSAR. This lack, however, was judged to have no effect on the design adequacy based on additional information submitted in the Corrective Action Plan and the review performed in Tasks B, C, and H. Thus, it is concluded that the design system or process in effect for the entire life of the San Onofre Units 2 and 3 project is in accordance with commitments and can reasonably be expected to produce an adequate design.

2. The review of the implementation of this system shows that the system was, in fact, implemented during the life of the project. The one exception is the implementation of the design procedures by SCE. However, this lack was judged to have no effect on the design adequacy, since all of the seismic Category I design work performed by SCE was reviewed and found adequate. Thus, overall, the design system or process was, in fact, adequately implemented.

3. Based on the technical design review of 22 features, including the reanalysis of two features, the design system or process in place during the life of this project and its adequate implementation has, in fact, produced an adequate seismic design.

Therefore, the seismic design of San Onofre Units 2 and 3 is judged adequate.

Two aspects of the construction process were reviewed.

1. Task D, Section 5, reviewed the plan for field audits. Based on this review it is concluded tht SCE and BPC did carry out an audit program that was properly planned and scheduled in accordance with commitments and that this program was effective as indicated by the lack of significant unresolved deficiencies.

2. Task G, Section 6, reviewed the as-built configuration of a segment of pipe. Based on this review it is concluded that the installation of this pipe segment is adequate with regard to seismic requirements.

Although the program was structured to concentrate on Unit 2, Unit 3 review was included insofar as there are signficant unique features. Based on this review, the conclusions of the program are applicable to both Units.

The procedure review in Tasks A and D, and to some extent B, used the QA Program documents and PSAR commitments as a source of requirements. These requirements were interpreted in light of 10CFR50 Appendix B and N45.2. Although a comprehensive review of the QA Program using these documents as sources of requirements was not done, the current QA Program was, in general, responsive to 10CFR50 Appendix B and ANSI N45.2 and no deficiencies were noted.

Overall, the seismic aspects of the San Onofre Units 2 and 3 project are judged adequate and no reason has been found to prohibit issuance of the full power license for Units 2 and 3.

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