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"You have requested our advice pursuant to Section 105(c) of the Atomic Energy Act of 1954, as amended, in connection with the purchase by North Carolina Electric Membership Corporation and Old Dominion Electric Cooperative of an ownership interest in the above captioned nuclear units.

"North Carolina Electric's proposed interest would be less than 20 MW per unit, and old Dominion's proposed interest would be between 20 MW and 60 MW per unit. The participation of these two entities in the nuclear units is the culmination of discussions beginning in 1972. Our earlier recommendation that it was not necessary for the Commission to conduct a hearing on the application by Virginia Electric and Power Company to construct certain units at the two plants was based, in part, on these discussions.[1]

"Our review of the information submitted in connection with the present application, as well as other relevant information, has disclosed no evidence that the proposed participation by Old Dominion and North Carolina Electric in the North Anna and Surry Units would either create or maintain a situation inconsistent with the antitrust laws under section 105(c). We do not, therefore, believe it is necessary for the Commission to hold an antitrust hearing on this matter."

Any person whose interest may be affected by this proceeding may, pursuant to § 2.714 of the Commission's "Rules of Practice," 10 CFR Part 2, file a petition for leave to intervene and request a hearing on the antitrust aspects of the application. Petitions for leave to intervene and requests for hearing shall be filed by May 3, 1979 either (1) by delivery to the NRC Docketing and Service Branch at 1717 H Street, NW, Washington, DC, or (2) by mail or telegram addressed to the Secretary, US Nuclear Regulatory Commission, Washington, DC 20555, ATTN: Docketing and Service Branch.

For the Nuclear Regulatory Commission.

Jerome Saltzman,

Chief, Antitrust and Indemnity Group, Office of Nuclear Reactor Regulation.

[Docket Nos. 50-280A, 50-281A, 50-338A, 50-339A, 50-404A, and 50-405A]

[FR Doc. 79-9859 Filed 4-2-79; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Advisory Committee on reactor Safeguards; Revised Notice of Meeting

Regarding the previous Federal Register Notice (published on March 21, 1979, Volume 44, p. 17237-8, as revised) for the meeting of the Advisory Committee on Reactor Safeguards to be held on April 5-7, 1979, in Washington, D.C., changes in schedule have been made as reflected below.

The agenda for the subject meeting will be as follows:

Thursday, April 5, 1979

8:30 a.m.-9:00 a.m.: Executive Session (Open)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities.

The Committee will discuss candidates proposed for appointment to the Committee, as appropriate. Portions of this session will be closed as necessary to protect information the release of which would represent a clearly unwarranted invasion of personal privacy.

9:00 a.m.-12:00 Noon: Meeting with NRC Staff (Open)—The Committee will hear and discuss reports by the Staff regarding the basis for shutting down five nuclear plants to resolve piping questions and a recent incident at the Three Mile Island Nuclear Station Unit 2 which released primary coolant into the containment.

12:00 a.m.-1:00 p.m.: Executive Session (Open)— The Committee will discuss matters proposed for discussion with the Commissioners including the timing and scope of the ACRS annual report on the NRC Safety Research Program: combination of dynamic loads, including those generated by seismic events, as a design basis for nuclear facilities: and a recent incident at the Three Mile Island Nuclear Station Unit 2 which released primary coolant into the containment.

1:30 p.m.-3:00 p.m.: Meeting with NRC Commissioners (Open)—The Committee will meet with the Commissioners to discuss items noted above.

3:00 p.m.-4:30 p.m.: Meeting with Department of Energy (Open)—The Committee will hear a report and hold discussions regarding the safety related aspects of the Tokamak Fusion Test Reactor.

4:30 p.m.-6:30 p.m.: Anticipated Transients Without Scram (Open)—The Committee will hear reports from and hold discussions with members of the NRC Staff and representatives of the nuclear industry as appropriate regarding alternative nuclear plant modifications to resolve this issue. Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter.

Friday, April 6, 1979

8:30 a.m.-9:00 a.m.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present regarding the request for a permit to construct the Palo Verde Nuclear Generating Station Units 4 and 5.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to this facility and provisions for the physical protection of this station.

9:00 a.m.-10:30 a.m.: Palo Verde Nuclear Generating Station Units 4 and 5 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC Staff and the applicant regarding the request for a permit to construct this facility.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to this facility and provisions for the physical protection of this station.

10:30 a.m.-12:00 Noon: Executive Session (Open)—The ACRS will discuss its proposed reports to NRC regarding the Palo Verde Nuclear Generating Station, and Anticipated Transients Without Scram. The Committee will hear the report of its subcommittee and consultants who may be present regarding proposed operation of the Sequoyah Nuclear Plant.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to these facilities, provisions for physical protection of the Palo Verde plant and matters involved in adjudicatory proceedings.

1:00 p.m.-4:30 p.m.: Sequoyah Nuclear Plant (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC Staff and the applicant regarding the request to operate this plant.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to this facility and provisions for the physical protection of this station.

4:30 p.m.—6:30 p.m.: Executive Session (Open)—The Committee will hear and discuss the reports of ACRS Subcommittees and members on items related to nuclear power plant safety, including evaluation of systems interactions, design of integrated protection systems, the ODYN Code, regulatory activities, and degradation of engineered safety features at a nuclear power plant.

The Committee will discuss its proposed reports to the Nuclear Regulation Commission regarding the Palo Verde Nuclear Generating Station, the Sequoyah Nuclear Plant, and Anticipated Transients Without Scram.

Portions of this session will be closed as necessary to discuss Proprietary Information, provisions for physical protection of these stations and matters involved in adjudicatory proceedings.

Saturday, April 7, 1979

8:30 a.m.—10:30 a.m.: Executive Session (Open)—The Committee will discuss its proposed reports to the NRC on the Palo Verde Nuclear Generating Station, the Sequoyah Nuclear Plant, and the proposed resolution of Anticipated Transients Without Scram. Portions of this session will be closed as necessary to discuss Proprietary Information, provisions for physical protection of these stations, and matters involved in adjudicatory proceedings.

10:30 a.m.-12:00 Noon: Meeting with NRC Staff (Open)—The Committee will hold discussions with members of the NRC Office of Inspection and Enforcement regarding policies and practices related to the imposition of civil penalties, and consideration of a proposed rule to reduce

[1] This recommendation was contained in a letter of August 1, 1972, with regard to units 3 and 4 at the North Anna Power Station and in a letter of November 14, 1973, with regard to units 3 and 4 at the Surry Power Station.

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the allowable limits on occupational radiation exposure.

The future schedule for ACRS activities will also be discussed.

12:00 Noon—12:30 p.m. and 1:30 p.m.—4:00 p.m.: Executive Session (Open)—The Committee will continue preparation of its reports to NRC on the Palo Verde Nuclear Generating Station, the Sequoyah Nuclear Plant, and Anticipated Transients Without Scram.

Portions of this session will be closed as necessary to discuss Proprietary Information, provisions for physical protection of these stations, and matters involved in adjudicatory proceedings.

The Committee will also discuss proposed comments and positions regarding other matters discussed during this meeting.

This meeting notice is being revised to include consideration of an unexpected incident at the Three Mile Island Nuclear Station Unit 2 on March 28, 1979 which resulted in a General Emergency being declared at this Station. Discussion by the ACRS will include reports by the NRC Staff regarding the status of the nuclear plant and interim measures taken to protect the public health and safety until final corrective action can be taken.

Further information regarding topics to e discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley (telephone 202/634/3265), between 8:15 a.m. and 5:00 p.m. EST.

March 29, 1979.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 79-10185 Filed 4-2-79; 8:45 am]

BILLING CODE 7590-01-M

Consolidated Edison Co. of New York, Inc., Issuance of Amendment To Facility and Termination of an Outstanding Order

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 52 to Facility Operating License No. DPR-26 issued to Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specification limits for total nuclear peaking factor (F_q), accumulator water volume and hot channel factor normalized operating envelope.

The Commission also terminated its Order for Modification of License dated April 27, 1978 having determined that, upon issuance of this amendment, the requirements of that Order had been satisfied.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 5, 1979, (2) Amendment No. 52 to License No. DPR-26, (3) the Commission's related Safety Evaluation and (4) the Commission's Order for Modification of License dated April 27, 1978. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 20th day of March, 1979.

For the Nuclear Regulatory Commission.

A. Schwencer, *Chief*

Operating Reactors Branch #1, Division of Operating Reactors.

[Docket 50-247]

[FR Doc. 79-10187 Filed 4-2-79; 8:45 am]

BILLING CODE 7590-01-M

Florida Power & Light Co.; Issuance of Amendment To Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Facility Operating License No. DPR-41, issued to Florida Power and Light Company, which revised Technical Specifications for Operation of the Turkey Point Nuclear Generating Station Unit No. 4, located in Dade County, Florida. The amendment is effective as of the date of issuance.

The amendment extends the current cycle 5 operating period before shutdown for steam generator inspections from six months to six months and ten days of equivalent operation (reactor coolant above 350°F).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 16, 1979, (2) Amendment No. 38 to License No. DPR-41, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental & Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 23rd day of March, 1979.

For the Nuclear Regulatory Commission.

A. Schwencer,

Chief, Operating Reactors Branch #1, Division of Operating Reactors.

[Docket 50-251]

[FR Doc. 79-10188 Filed 4-2-79; 8:45 am]

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

~~February 28~~, 1979

SCHEDULE AND OUTLINE
FOR DISCUSSION
228TH ACRS MEETING
APRIL 5-7, 1979
WASHINGTON, DC

Thursday, April 5, 1979, Room 1046, 1717 H Street, NW, Washington, DC

1)	8:30 A.M. - 9:00 A.M.	Executive Session (Open) A) Report of ACRS Chairman (Portions of this discussion will be closed as required to discuss information the release of which would represent an unwarranted invasion of personal privacy.)
2)	9:00 A.M. - 12:00 Noon	Meeting with NRC Staff (Open) A) 9:00 A.M.-11:00 A.M.- Seismic design of nuclear power plant piping B) 11:00 A.M.-12:00 Noon - Release of primary coolant at Three Mile Island Nuclear Station Unit 2
3)	12:00 Noon - 1:00 P.M.	Executive Session (Open) A) Discuss proposed topics for meeting with NRC Commissioners 1) Combination of dynamic loads as a design basis for nuclear power plants 2) Seismic design of piping for nuclear power plants (Preliminary discussion)

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3) Release of primary coolant and apparent core damage at Three Mile Island Nuclear Station Unit 2 (Preliminary discussion)
4) Timing and scope of ACRS Annual Report on the NRC Reactor Safety Research Program (tentative)
5) Use of Probabilistic Assessment in the licensing process (tentative)
6) ACRS use of foreign travel funds (tentative)
7) Participation of ACRS consultants in NRC hearings (tentative)

4) 1:00 P.M. - 1:30 P.M.

LUNCH (Lunch on the table will be provided if necessary to complete discussion of items noted above)

5) 1:30 P.M. - 3:00 P.M

Meeting with NRC Commissioners (Open) Room 1130-H

A) Items noted above will be discussed as appropriate

6) 3:00 P.M. - 4:30 P.M.

Meeting with Department of Energy (Open)

A) The Committee will hear and discuss a report by representatives of the Department of Energy regarding safety related aspects of the Tokamak Fusion Test Reactor

7) 4:30 P.M. - 6:30 P.M.

Anticipated Transients Without Scram (Open)

(Portions of this session will be closed as appropriate to discuss Proprietary Information related to this matter.)

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Friday, April 6, 1979, Room 1046, 1717 H Street, NW, Washington, DC

8) 8:30 A.M. - 9:00 A.M.

Executive Session (Open)

A) Report of ACRS Subcommittee on Palo Verde Nuclear Generating Station Units 4 and 5 (Portions of this session will be closed as required to discuss Proprietary Information applicable to this facility and provisions for the physical protection of this station.)

9) 9:00 A.M. - 10:30 A.M.

Palo Verde Nuclear Generating Station Units 4 and 5 (Open)

(Portions of this session will be closed as required to discuss Proprietary Information applicable to this facility and provisions for the physical protection of this station.)

10) 10:30 A.M. - 12:00 Noon

Executive Session (Open)

A) 10:30 A.M.-11:00 A.M.: Report of ACRS Subcommittee on the Sequoyah Nuclear Plant (Portions of this session will be closed as required to discuss Proprietary Information applicable to this plant, and provisions for physical protection of this facility.)

B) 11:00 A.M.-12:00 Noon: Discuss proposed ACRS reports to NRC regarding:

- Palo Verde Nuclear Station
- Anticipated Transients Without Scram

(Portions of this session will be closed as necessary to discuss Proprietary Information applicable to

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these matters and physical protection of the Palo Verde Nuclear Generating Station, and matters involved in adjudicatory proceedings.)

11) 12:00 Noon - 1:00 P.M. LUNCH

12) 1:00 P.M. - 4:30 P.M. Sequoyah Nuclear Plant (Open)

(Portions of this session will be closed as required to discuss Proprietary Information applicable to this plant, and provisions for physical protection of this facility.)

13) 4:30 P.M. - 6:30 P.M. Executive Session (Open)

A) Reports of ACRS Subcommittees on:
   1) Evaluation of systems interactions - Zion Nuclear Station
   2) Design of integrated protection system (RESAR-414)
   3) Use of the ODYN Code
   4) Regulatory Activities
   5) Degradation of engineered safety features at Arkansas Nuclear One Unit 2

B) Discuss proposed ACRS reports to NRC on:

   . Palo Verde Nuclear Generating Station
   . Sequoyah Nuclear Plant
   . Anticipated Transients Without Scram

(Portions of this session will be closed as required to discuss Proprietary Information related to

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these matters, arrangements for physical protection of the facilities noted and matters involved in adjudicatory proceedings.)

Saturday, Aril 7, 1979, Room 1046, 1717 H Street, NW, Washington, DC

14) 8:30 A.M. - 10:30 A.M.

Executive Session (Open)

A) The Committee will discuss its reports to the NRC on:

- Palo Verde Nuclear Generating Station
- Sequoyah Nuclear Plant
- Anticipated Transients Without Scram

(Portions of this session will be closed as required to discuss Proprietary Information related to these matters, arrangements for physical protction of the facilities noted and matters involved in adjudicatory proceedings.)

15) 10:30 A.M. - 12:00 Noon

Meeting with NRC Staff (Open)

A) Discussion with representatives of the Division of Inspection and Enforcement regarding procedures and policies related to the imposition of civil penalties
B) Report on proposed EPA action to reduce the allowable limits on occupational radiation exposure
C) Future Schedule
   1) Anticipated subcommittee activity
   2) Anticipated Committee activity

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16) 12:00 Noon - 12:30 P.M. & 1:30 P.M. - 4:00 P.M.

Executive Session (Open)

A) The Committee will complete preparation of its proposed reports to NRC on:

. Palo Verde Nuclear Generating Station

. Sequoyah Nuclear Plant

. Anticipated Transients Without Scram

(Portions of this session will be closed as required to discuss Proprietary Information related to these matters, arrangements for physical protection of the facilities noted and matters involved in adjudicatory proceedings.)

B) The Committee will complete discussion of proposed comments/positions regarding items discussed during this meeting.

oOo

Issue Date: AUG 23 1979

MINUTES OF THE
228TH ACRS MEETING
APRIL 5-7, 1979
WASHINGTON, D. C.

The 228th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N. W., Washington, DC, was convened at 8:30 a.m., Thursday, April 5, 1979.

[Note: For a list of attendees, see Appendix I.]

The Chairman noted the existence of the published agenda for this meeting, and the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been received from members of the public to present oral statements. He also noted that copies of the transcript of some of the public portions of the meeting would be available in the NRC's Public Document Room at 1717 H Street N. W., Washington, DC, within approximately 24 hours.

[Note: Copies of the transcript taken at this meeting are also available for purchase from ACE Federal Reporters, Inc., 444 North Capitol St N.W., Washington, DC 20001.]

I. Chairman's Report (Open to Public)
[Note: Raymond F. Fraley was the Designated Federal Employee for this portion of the meeting.]

A. Reviewers

The Chairman named Messrs. Plesset and Siess as reviewers, and Mr. Bender as alternate reviewer for the 228th ACRS Meeting.

B. Legal Support for ACRS Consultants under Subpoenas

The Committee approved, in principal, a letter from the Chairman to the Commissioners regarding the continued providing of legal counsel to ACRS Consultants subpoenaed before AS&LB proceedings (see Appendix XXVIII).

C. Meeting with Japanese Committee for Evaluation of Reactor Safety

The Committee agreed to postpone its planned meeting with the Japanese Committee for the Evaluation of Reactor Safety (CERS) (originally scheduled for April) because of the press of business arising from the Three Mile Island accident. It also agreed, however, that the trip would be made as soon as the workload permits it and arrangements could be made again with the Japanese.

D. Proposed Meeting with a Member of the Federal Republic of Germany Ministry of the Interior

The Committee agreed that it would be inconvenient to meet with Herr Schnurer of the Federal Republic of Germany, Federal Ministry of the Interior during his forthcoming trip to the U.S. in May.

E. Transcripts of Meetings on Three Mile Island -2

The Chairman noted that copies of the Committee's briefing by the NRC Staff on the Three Mile Island 2 (TMI-2) accident are available for those members who desire them.

Mr. Lawroski recommended that transcripts of all the meetings to be held relating to the TMI-2 accident be made available to all members.

F. Topics to be Discussed on Three Mile Island-2

The Committee agreed that the following topics relating to the TMI-2 accident should be discussed during this meeting:

1. necessary work to secure TMI-2,
2. effects of lessons learned from this accident on other B & W plants,
3. effects of what is learned from this accident on non-B&W PWRs.
4. basic philosophical questions raised by this accident regarding nuclear power,
5. should ACRS Members, Consultants, and Staff observe the activities currently being carried out at the TMI-2 site and in Bethesda,

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6. should the Committee write an interim report on this matter at this meeting, and

7. should the Committee become involved in changes to B&W plants derived from the accident.

The Committee agreed to hold a Special Meeting on April 16-17, 1979 to discuss the TMI-2 accident further.

II. Meeting on Palo Verde Nuclear Generating Station, Units 4 and 5 (CP) (Open to Public)

[Note: Gary R. Quittschreiber was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

Mr. Shewmon, Subcommittee Chairman, discussed the application for a construction permit for Palo Verde Nuclear Generating Station, Units 4 and 5, noting that these proposed plants are replicates of Units 1 through 3 as defined in NUREG-0427. (For details, see Appendix IV). He briefly discussed the site and design parameters, and noted the outstanding issues as identified by the NRC Staff:

- review of the constructor's Quality Assurance Program,
- review of the applicant's financial qualifications,
- review of the seepage analysis to determine the design basis groundwater levels, and
- review of the revised CE Emergency Core Cooling Systems (ECCS) evaluation model.

He briefly discussed the status of the generic issues that apply to the CESSAR-80 Standard Plant.

[Note: E. J. Van Brunt, Jr., coordinated presentations for the applicant; M. Licitra, for the NRC Staff.]

B. Applicant's Overview

E. J. Van Brunt, Jr., discussed the licensees schedule for the five Palo Verde Units, the important milestones in the develoment

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of the Nuclear Station, a brief description of the site, the participating utilities in the project, the administrative organizatgion of the Arizona Public Service Company, proposed operators of the station, and the construction schedule (see Appendix V).

C. Status of the NRC Review

1. Review

M. Licitra noted that the CP application for Palo Verde Units 4 and 5 describes replicates of Units 1-3, in accordance with the requirements of NUREG-0427. Units 1 through 3 previously were reviewed by both the NRC Staff and the Committee, a construction permit has been granted, and the three Units are currently under construction. He noted that in replication applications, the site and the utility applicant may vary, and in fact, on three previous replication applications, the site and the utilities differed from those of the basic plants:

Jamesport, which replicates Millstone 3; Marble Hill, which replicates Byron; and New England, which replicates Seabrook. All of these plants referenced the Westinghouse RESAR-3 Nuclear Steam Supply System (NSSS). The Palo Verde 4 and 5 replication application is the first for a Combustion Engineering designed NSSS. This application is unique in that the lead utility applicant and the site will be the same for all the replicated plants.

M. Licitra said that the scope of the review included items in the following categories:

- matters relating to the site-specific location, e.g., the new site geological investigations,
- changes made to the base plant design since the issuance of the construction permits, e.g., changes made in the main steam support structure to accommodate pipe breaks,
- changes in regulations, e.g., the cost-benefit analysis required by Appendix I, and
- other significant safety issues.

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These items are listed in Table 1.2 of the Safety Evaluation Report (SER) as Category 2, 3, and 4 items, and constituted the bulk of the NRC Staff safety review.

Matters relating to unresolved generic items are presented in Appendixes D and H of the SER. A cross reference index of ACRS generic items with NRC Staff generic issues is presented on pages D-13 and -14 of the Appendix D. He noted that four outstanding issues have been identified in the SER:

- since the applicants have not designated the constructor for Palo Verde 4 and 5, the NRC Staff has not been able to review the Quality Assurance Program of the constructor.
- The NRC Staff has not completed its review of the applicant's financial qualifications.
- The NRC Staff is currently reviewing a modified Combustion Engineering ECCS model.
- Agreement has not been reached with the applicants regarding design basis groundwater levels (the NRC Staff is not satisfied with the degree of conservatism proposed by the applicant).

As a result of the NRC Staff's continuing evaluation, an additional outstanding issue has been identified: a requirement for a temperature monitoring capability of the room containing the two steam turbine-driven auxiliary feedwater pumps.

2. Standardization Program

R. Boyd, NRC Staff, discussed the NRC Staff's standardization program, specifically as it relates to the Palo Verde Station. He said that the qualification review of Units 4 and 5 is probably the best one to date. The NRC Staff focused on all the Category 2, 3, and 4 matters that have come before the Regulatory Requirements Qualifications Committee; these matters represent the changing regulatory requirements over the past few years. He noted that these changes will also apply to Units 1, 2, and 3. During the operating license review, these matters may be specifically reviewed for Units 1, 2 and 3.

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In discussing the number of years involved between the proposed commissioning of Unit 1 and Unit 5, he noted that in the last year or two, the NRC Staff has become relatively systematic in categorizing new requirements. In addition, the NRC Staff is evolving a systematic process for considering new requirements. He noted that in standardization reviews, new considerations significant to safety are applied in the original plants as well as the replicates. He noted that he doesn't foresee any particular problems, even though the operations of these five plants cover a seven or eight year period.

In answer to a question, R. Boyd indicated that the difference between duplicate plants and replicate plants is that duplicate plants must be reviewed during a specific period of time, where replicate plant reviews can cover a longer period.

In answer to a question, D. Vassallo, NRC Staff, said that Regulatory Guide 1.97, Instrumentation to Follow the Course of An Accident, will apply to these standard plants.

3. Review of Palo Verde with Respect to the Arkansas Nuclear One, Unit 2 Incident

F. Rosa, NRC Staff, discussed the incident at Arkansas Nuclear One, Unit 2 (ANO-2) during which dedicated startup transformers were tripped. He concluded that the problem at ANO was a combination of undersizing of the transformers, and independent automatic switching arrangements which permitted both Units at ANO to have their electrical loads switched to the same transformer, and a situation where the overload relay was not set to handle both loads. He noted that the deficiencies at ANO have been identified, and that corrective actions are being undertaken. He noted that at Palo Verde, the switchyard is laid out in a manner that such an incident will not take place.

Mr. Okrent recommended that the Power and Electrical Systems Subcommittee review Regulatory Guides 1.47 and 1.68, and review several units to determine the adequacy of safety aspects of off-site electrical systems.

Mr. Ray suggested further that the fundamental philosophy underlying the design of the above systems should also be reviewed.

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Mr. Okrent noted that in the review of the ANO plants, there appeared to be too many oversights with respect to the off-site power system. He was of the opinion that the applicant could have done better analyses, and recommended that the entire review procedure be reviewed.

F. Rosa said that the NRC Staff's Quality Assurance Branch has improved its audit of procedures for pre-operational checks of the off-site power system, particularly since the ANO-2 incident, and is continuing to review the matter. (For circuit schematics of the switchyards at ANO and Palo Verde, see Appendix VI.)

4. Small Break LOCA Analyses

In answer to a question regarding the capability of the NRC Staff to analyze the ability of ECCS systems to handle small break LOCAs, W. Hodges, NRC Staff, said that the NRC Staff has the RELAP Code to use, which they are working through the semiscale tests.

Mr. Plesset indicated that he does not believe that RELAP is adequate. He said that the problem exists in that there is a coolant loss through a small leak, but the heat loss through this leak is not adequate to cool the core. He suggested that the semiscale test is difficult to translate to full scale equipment.

W. Hodges said that the NRC Staff plans to use the data obtained from the TMI-2 accident to try to verify the analytical tools available to the NRC Staff. He said that the Staff is aware that there is a problem.

D. Applicant's Response to the NRC Staff Report

E. J. Van Brunt said that the applicant is in agreement with the NRC Staff's conclusions regarding the four open items in the SER. He said that the applicant was not prepared to respond at this time regarding the question of temperature monitoring in the auxiliary feed pump room.

E. Technical Presentations

1. Exception to CESSAR-80 Design

E. J. Van Brunt said that there are no design differences among Palo Verde Units 1 through 5. He said that the only

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difference between Palo Verde units and the standard CESSAR-80 design relates to the refueling water temperature 27.5 hours after shutdown. The Palo Verde design calls for 125° F, while CESSAR-80 calls for 135° F. The reason for the reduced temperature is to minimize the airborne tritium concentration during refueling. (See Appendix VII.)

J. Allen stated that F. Rosa's presentation of the ANO-2 problem as it applies to Palo Verde covered the situation. He noted that safety equipment receives its power from the 480 volt vital bus through a battery charger onto a 125 volt d-c bus. There are four inverters on this bus, and the inverter output to the 120 volt vital a-c power is through a manual transfer switch. In the event the inverter is lost, the applicant does not utilize an automatic transfer. The applicant believes the use of the 125 volt d-c bus as a primary source of power gives a very stable source, not subject to regulation problems. (For switchyards schematic diagrams, see Appendix VIII).

2. Load Sequencing

D. Karner, Arizona Public Service Company, discussed the energy safety features load sequencer, the purpose of which is to start engineering safety feature equipment sequentially, thereby preventing an undervoltage on the engineering safety features bus that would occur if all the equipment started simultaneously (see Appendix IX).

Several members questioned the reliability claim made by the applicant.

3. Emergency Planning

B. Karner, Arizona Public Service Company, said that the applicant is in the process of developing a station emergency plan to be submitted to the State of Arizona (see Appendix X). He said that in accordance with Arizona law, Maricopa County must have an emergency plan and has the responsibility for off-site emergency response.

4. General Questions

In answer to a question, J. Allen said that the electrical systems are adequately protected from electrical transients, including lightning.

In answer to a question relating to the analyses for very small break LOCAs, E. Scherer, CE, said that his company analyzes breaks from the minimum at which the make-up system can no longer replenish the losses to the double-ended pipe break. In response to a further question regarding analyses for situations where the heat lost through a small break is less than decay heat generation, he said that CE normally relies on steam generators to dissipate decay heat. If the steam generators were not available, there might be other ways of removing this heat.

F. Caucus

Members were polled, and agreed that they could write a report on the application for a construction permit for the Palo Verde Nuclear Generating Station, Units 4 and 5. Members identified the items that they believed should be included in the report. The Chairman informed the applicant, however, that in view of the recent accident at Three Mile Island, the Committee might defer completion of the report until a better understanding of this accident can be developed.

III. Meeting on Sequoyah Nuclear Power Plant, Units 1 and 2 (OL) (Open to Public)

[Note: Richard P. Savio was the designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

Mr. Mark, Subcommittee Chairman, briefly described the design of the Sequoyah Nuclear Power Plant, Units 1 and 2, and its site, and discussed the major issues of the review (see Appendix XI.)

I. Catton, ACRS Consultant, raised the following additional matters:

- the need to analyze an expansion wave, and its potential effects, from a piping break back into the core.
- the need for an analysis of the response of the steam generator to a blowdown on the primary side.
- there is a need for documentation on the adequacy of the 1-D and 2-D codes.

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[Note: J. Gilliland, Tennessee Valley Authority (TVA), coordinated presentations for the applicant; H. Silver for the NRC Staff.]

B. Status of the NRC Staff View

H. Silver discussed the outstanding issues, the confirmatory issues, and the generic issues that were addressed in the SER (see Appendix XII). He noted that additional information has been requested from the applicant on one additional issue, foundation engineering. A response is expected by mid-April.

C. Applicant's Presentations

J. Gilliland discussed the TVA organization, the organization of its Office of Power, and the emergency plan interfaces between the various state, local, national organizations and TVA (see Appendix XIII).

D. Plant Location and Site

D. Lambert, TVA, described the Sequoyah location and some of the site features (see Appendix XIV).

In answer to a question regarding the configuration of the pressurizer piping, S. Varga, NRC Staff, said that the staff understands the problem, and will be reviewing piping configurations carefully. R. Sero, Westinghouse (W), said that the pressurizer piping connection does not go below the hot leg level, and the piping therefore would not act as a manometer. E. G. Beasley, TVA, said that the pressurizer connection joint is at the top of a horizontal section of the hot leg pipe and cannot form a loop seal.

E. Thermal and Hydraulic Design Parameters

D. Lambert discussed the similarities and differences in the thermal and hydraulic design parameters between the Sequoyah plants and the McGuire plant (see Appendix XV). In answer to a question, D. Lambert said that containment isolation occurs either on high containment pressure or initiation of high pressure injection systems. There is no automatic containment isolation for high containment radiation, however, the containment ventilation system does isolate on high radiation signal.

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H. Silver noted that isolation on high radiation signals is not requi.ed by the NRC Staff, and that many other plants also do not isolate on this signal.

S. Varga, NRC Staff, offered the opinion that the event which produces high .adiation in the containment will provide initiation for isolation of the containment.

Mr. Okrent t..ught it strange that the condition that the containment is provided to protect against does not actuate the containment isolation.

F. Construction Status

W. Popp, TVA, noted the status of construction of the Sequoyah Plant as follows:

- Unit 1 is 97% complete,
- Unit 1 is well into its preoperational test program, having completed or started 90 of 150 tests, with 40 more starting within the next few weeks,
- hot functional testing was scheduled to begin on April 7,
- the plant operating staff is aboard and trained,
- the radiological health staff is aboard, and
- security will be established for Unit 1 commencing with the hot functional testing program.

In answer to a question H. E. McConnell, TVA, said that the switch gear will be tested through normal practices, but that there is no special program for this testing.

Z. Zudans, ACRS Consultant, suggested that it might be useful if, in its prestartup program, TVA measured the non-condensable gases present in the primary cooling system. He also recommended that instruments be available to chart physical conditions in the primary system, such as the location of water and steam.

In answer to a question, D. Lambert said that the applicant will have a loose parts monitor in place in Sequoyah Unit 1 prior to ascension to power at the latest.

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G. Operator Training Program

W. Popp described the operator training program at Sequoyah. He said that seven years ago the first group of licensed operator trainees, a group of 15 people recruited from fossil fuel plants and Browns Ferry, began their training. Some of these potential operators had been trained for the EGCR at Oak Ridge. These trainees each had 10 to 20 years hands-on plant operating experience, and they were being trained to become principal supervisors. About a year later, a second group of younger men with experience ranging from 5 to 10 years in large plants, were given a 12-week basic nuclear course. Then they were sent to the Oak Ridge National Laboratory for their reactor operating experience and to get the reactor startups that they needed documented for their licensing. Following that, they received a W co-license training program, which consisted of 12 weeks' observation training at either Point Beach or Zion. Following that, they received a 12-week simulator certification course at Zion.

Mr. Kerr requested that the NRC Staff check the requirements in operator training regarding experience in starting up nuclear power plants.

Following offsite training, the trainees received a 400-hour W lecture series. They were audited for the NRC operator examination by W, and given a license review course. In January 1979, they took the NRC cold license written examination. Last month they took the oral examinations.

In addition to the 22 trainees submitted for the cold license examination, there are four more cold license candidates taking observation training at D. C. Cook. When Sequoyah reaches 20% power, 12 operators will take the hot license examination.

In answer to questions regarding the capability of the simulator to handle both anticipated and anomalous transient problems, R. J. Johnson said that the simulator is dynamically modeled to 140 pre-programed malfunctions. The model itself compensates for the operator behavior. He said that the readout on the simulator is equivalent to that in the control room of an operating reactor, and that additional information is not provided by the simulator. The simulator provides a real time integration solution to differential equations.

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W. F. Popp noted that once an operator is licensed, he must be re-qualified periodically, and receive a minimum of 86 hours per year of supervised training that includes 32 hours of simulation training. Further, because of attrition, promotions and transfer, new operators must constantly be trained. As a result, TVA has a very detailed operator training program.

In answer to a question, R. J. Johnson said that the minimum requirement to qualify for operator training is a high school education, but that, in fact, approximately 25% of the trainees may be college educated. He inferred that the TVA training program is comparable to a 2 year college technical program.

H. Seismic Design Criteria and Qualification Program

1. Overview

H. Silver noted that for purposes of determining the safe shutdown requirements for the Sequoyah plant, the historical earthquake in the Ridge Tectonic Province, in which Sequoyah is located, the Giles County earthquake of 1897, was assumed to recur anywhere in the province. This earthquake has been described by a Modified Housner Spectrum as 0.18g. No evidence has been found indicating faulting or other unsafe geological features. There are no known geological structures that would cause surface displacement or tend to localize earthquakes.

While the NRC Staff's evaluation of the controlling earthquake has not changed since the CP review, the characterization of ground motion has changed. The Standard Review Plan now requires a plant in this region to be designed to Reg. Guide 1.60. Therefore, the NRC Staff requested the Applicant to provide information that would confirm the adequacy of the Sequoyah seismic design. The NRC Staff has examined the available data, and concludes that the current design of Sequoyah is adequate to withstand the effect of the assumed earthquake without loss of capability, and to perform the required safety functions. However, because the Sequoyah design spectrum is lower than the selected 84th percentile site specific spectrum at frequencies of interest, and because the consideration of structural margins involves engineering judgment, the NRC Staff initiated a program to quantify margins of structures and components.

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2. Comparison of Sequoyah, Watts Bar, and Bellefonte Nuclear Plants Top-of-Rock Design Spectra for Reinforced Concrete Structures

L. Reiter, NRC Staff, compared the top-of-rock design spectra for reinforced concrete structures among the Sequoyah, Watts Bar, and Bellefonte plants (see Appendix XVI).

M. Trifunac, ACRS Consultant, noted that his calculations, assuming different intensities at the site, indicate that

- for a Modified Mercalli (MM) VI earthquake, there is a 20% chance that 0.18g would be exceeded, and a 5% chance that 0.25g would be exceeded,
- for a MM VII earthquake, there is a probability of 50% that 0.18g would be exceeded, and a probability of 35% that 0.25g would be exceeded, and
- for a MM VIII earthquake, there is a 75% chance that 0.18g would be exceeded, and a 60% chance that 0.25g would be exceeded.

The above calculations assume that the plant foundations are on rock, and that horizontal ground motion only is being considered. He further stated that he believes that the 7% critical damping factor assumed by the NRC Staff is too optimistic. He questioned the correlations between the assumed earthquake intensity and the magnitude. He concluded by questioning the assumed value for a SSE of 0.18g.

In answer to a question regarding the NRC Staff's conclusion regarding the relative values of the average risk of exceeding the SSE for Sequoyah and Phipps Bend, M. Trifunac said that intuitively these values seemed reasonable.

In answer to a question regarding the reliability of calculated values for the probability of a serious accident caused by an earthquake at the site, L. Reiter said that in general, the probability numbers seem to be in the proper range for the 1000 to 10,000-year earthquake. The NRC Staff obtained confidence from the fact that these relative numbers are stable for fluctuations of several orders of magnitude of absolute risk. The absolute risk is not known, but the NRC Staff believes that the seismic hazard alone is something of the order of 10^{-3} to 10^{-4}.

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In answer to a question regarding the adequacy of the 7% critical damping factor, J. Knight, NRC Staff, said that the advice received by the NRC Staff from its consultants indicate that the 7% factor is appropriate.

3. Structural Margins

F. Rinaldi, NRC Staff, discussed the TVA structural seismic re-evaluation, and their recalculation of seismic margins (see Appendix XVII.)

4. Components Review

J.R. N. Rajan, NRC Staff, discussed the piping and mechanical equipment review for seismic design margin (see Appendix XVII).

In answer to a question, D. Denton, TVA, noted that TVA did not review all of the piping and components, but rather reviewed a characteristic sample of the equipment and piping.

Mr. Okrent recommended that the NRC Staff review all the plants in the eastern United States regarding seismic design to assure that for an event that has a probability on the order of 10^{-4} per year, there is assurance of safe shutdown.

5. TVA Response

F. F. Hand, TVA, responded to the NRC comments. When the structure was reanalyzed, TVA used a spectrum bounding the 84th percentile of the earthquake records. The structures were analyzed using the response spectrum analysis technique.

In order to perform the calculations, floor response spectra were needed. The easiest and conservative way to obtain the floor response spectra was to take the time histories used for Sequoyah, and to determine what factor they had to be multiplied by so that the Sequoyah spectrum was raised to adequately envelope the 84th percentile spectrum. The resulting number turned out to be, for the horizontal motion, 1.53. The vertical motions that were used were 2/3 of the old horizontal motions. The multiplying number for the vertical motion was 1.07.

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I. Ice Condenser Loading

Mr. Mark noted that TVA profited from information received from predecessors using ice condenser systems, in that the ice loading at Sequoyah is uniform.

J. Emergency Core Cooling Systems

D. Docherty, TVA, provided handouts for the upper head injection system, and the analytical model used to evaluate it (see Appendix XIX). He discussed the method used to pressurize the accumulators, and the precautions taken to keep nitrogen and other non-condensable gases from the coolant.

In answer to a question, P. J. Docherty, W, noted that there is no way to vent non-condensable gases from the reactor vessel head. He said that the small breaks are analyzed from one sq. in. to larger breaks. For 3/8 inch diameter and smaller breaks, the charging system makes up for lost inventory. For 1/2 inch breaks, even with loss of the charging pumps, intermediate-head safety injection prevents core uncovery. He said that if there were some way for nitrogen to get into the system, it could interfere with flow.

K. Caucus

Members were polled, and agreed that they would try to write a report on the Sequoyah Nuclear Power Station, Units 1 and 2. Members identified items they believed should be included in the report. The applicant was informed that the Committee might defer completion of its report until a better understanding can be developed of the implications of the March 28, 1979 accident at TMI-2.

J. Gilliland noted that the Sequoyah plant completion is late already, and that additional fuel costs for replacement power amount to $400,000 per day. He said that the Sequoyah plant is both vital and needed. He also said that the plant is well designed, well built, well reviewed and the operating personnel are well trained. Fuel loading is scheduled for mid-June. He voiced the hope that the ACRS report would be completed before the operating schedule for Sequoyah is impacted.

D. Vassallo said that the NRC Staff needs at least a month after receipt of a Committee report before it can issue an operating license.

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IV. Meeting on Safety-Related Aspects of the Tokamak Fusion Test Reactor (Open to Public)

[Note: James M. Jacobs was the designated Federal Employee for this portion of the meeting.]

A. Magnetic Fusion Program Overview

J. E. Baublitz, Department of Energy (DOE) provided an overview of DOE's magnetic fusion program, including the program's organization, magnetic field configuration studied, the objectives of the program, goals for fusion power reactors, new devices recently completed, the technical progress outlook, operating characteristics of fusion devices, and a future time table for fusion development (see Appendix XX).

B. Tokamak Fusion Test Reactor (TFTR) Program

W. Marton, DOE, discussed the TFTR program, including objectives of the program, the current project status, a description of the Princeton Plasma Physic Laboratory, a description of the TFTR complex in that laboratory, the TFTR energy flow, radiation dose criteria, safety concerns, gas flow, potential safety differences between PWRs and the TFTR, fire safety criteria, electrical safety, criteria, control system philosophy, tritium handling philosophy, flood hazards, tornado criteria, earthquake criteria, quality assurance plans, operations philosophy, test cell building design, tritium supply system design, tritium cleanup system design parameters, primary power systems, standby power, computer parameters, waste systems, contents of the preliminary safety analysis report, technical specifications and plans to upgrade the facility in the future (see Appendix XXI).

In answer to a question, J. E. Baublitz indicated that DOE is currently developing review procedures and appropriate requirements for NRC review of fusion facilities.

V. Preliminary Investigation of the March 28, 1979 Accident at Three Mile Island Nuclear Station Unit 2 (Open to Public)

[Note: Raymond F. Fraley was the designated Federal employee for this portion of the meeting.]

A. ACRS Consultant's Report

C. Michelson, ACRS Consultant, noted the sequence of events (as believed at this early date) during the March 28, 1979 accident at Three Mile Island Nuclear Station, Unit 2 (TMI-2). He then postulated a possible scenario, identifying plant conditions, matched to the sequence of events. He identified the key components of the reactor primary system and the steam generator, and noted their relative locations. From preliminary data that he had received, he postulated the conditions and causes of the events that appear to have occurred. In describing the postulated details of the transient, C. Michelson noted his opinion that the pressurizer level indicator was probably reading reasonably accurately, and that in such a system, where there is the possibility of a loop seal between the pressurizer and the primary circulation system, it is not unusual to have adequate level in the pressurizer and inadequate coolant in the cooling system. He suggested that with the drop of pressure in the system, the saturation temperature of the water was reached, and boiling could not be prevented. Once boiling began, vapor bubbles developed at the high points in the piping of the system and, if forced circulation was lost, natural circulation could not be achieved. With the shutdown of the main coolant pumps, forced circulation was lost.

B. NRC Staff Report

D. Eisenhut, NRC Staff, discussed the preliminary reports of the sequence of events at TMI-2 (see Appendix XXII).

D. Eisenhut noted that as a result of this accident, the NRC Staff is requiring all of the utilities who operate Babcock and Wilcox reactors to take the following measures:

- re-verify that the emergency feed water block valves are open,
- caution operators to observe all instrumentation during severe transients, and not to rely solely on pressurizer level indication,
- after high pressure injection actuation, permit this system to operate until either two low pressure injection pumps are running, or the high pressure pump has operated for at least 20 minutes, to the point where the hot leg and cold leg tempeatures have dropped to the point that they are at least 50° below saturation temperature,

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- at least one pump in each primary coolant loop should be continued in operation, and
- if the ECC systems actuate, the containment should be isolated.

D. Eisenhut described the current status of the plant, noting that pressure is being maintained at approximately 1,000 psig, the average inlet and outlet temperatures are approximately 280°F, heat is being removed through steam generator A, which has a secondary pressure of 33 psig, and that the peak temperature being recorded is about 400°F. One hydrogen recombiner is operating, a second is in standby. Hydrogen concentration in containment is approximately 1.9%.

Members noted their opinions of the inappropriateness of in-core thermal couples not being able to be read to their full scale, but rather being cut off at approximately 700° by the plant computer. They noted surprise that, during the course of this accident, no measures were taken to be able to read the temperatures that could be indicated by these thermocouples.

D. Eisenhut noted that the National Laboratories and other consultants are working with the NRC Staff to analyze this accident.

In answer to a question, D. Eisenhut said that the NRC Staff does not have the capability to simulate a wide range of transients in reactors; it can only simulate preset transients.

D. Eisenhut said that Staff plans to analyze the event to get a clear understanding of the transients, to review the safety analysis of bounding feedwater transients that have been formulated and to compare these with the experience at TMI-2 and other B&W plants.

P. Check, NRC Staff, said that operator training is also under review.

L. Higginbotham, NRC Staff, discussed the releases of radioactive materials from the TMI-2 accident, and also noted that the Region I inspectors arrived on site at approximately 10:45 a.m. on March 28, 1979. This team was equipped to do some offsite measurements, and they proceeded to implement a procedure under

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which they would operate. By afternoon, a DOE helicopter was available, and aerial surveys were begun, and the plume was tracked and defined. The NRC received cooperation and coordination from the State of Pennsylvania, the Bureau of Radiological Health, HEW, and EPA.

C. Michelson noted that the operation of the pressurizer heaters must be maintained in order to keep pressures above saturation condition during the cooldown of the damaged plant. If these heaters are lost, it would be necessary to revert to natural circulation to cool the plant.

C. Executive Session

The Commitee discussed methods by which to proceed with its investigation of the TMI-2 accident.

1. Schedule for Committee Activity in Harrisburg

The Committee agreed that Members and Consultants, as available, should go to Harrisburg to observe recovery operations at TMI-2 and gather information as appropriate for review by the Committee. ACRS Staff members would accompany the Committee Members as necessary. Members are to use their discretion regarding the length of time they remain in the Harrisburg area. The following schedule was set up:

Date	Member(s)	Consultant(s)	ACRS Staff
April 6	Bender Lawroski		Wright
April 7	Bender Lawroski		Wright
April 8	Lawroski		Wright
April 9	Etherington		McCreless
April 10	Etherington Ray	Michelson	McCreless

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Date	Member(s)	Consultant(s)	ACRS Staff
April 11			McCreless
April 12		Catton	McCreless
April 13		Catton	McCreless
April 14		Catton	McCreless
April 18-20	Lawroski (if needed)		as needed

2. Areas for Consideration

C. Michelson suggested that the Committee should consider two specific areas in the wake of the TMI-2 accident:

- shutdown and recovery operations at TMI-2, and
- changes in equipment and operation at other operating B&W-designed plants.

The Committee agreed that the short-term matters, including both the securing of the plant and the immediate implications of the TMI-2 accident will continue to be studied by the full Committee.

3. Subcommittee Appointments

The Chairman appointed an ad hoc subcommittee to study the long-term implication of the TMI-2 accident, with Mr. Okrent, Chairman, and Messrs. Carbon, Mark, Plesset, and Siess. The Chairman also appointed additional members to the TMI-2 subcommittee. The new makeup of this subcommittee is Mr. Etherington, Chairman, and Messrs. Bender, Kerr, Lawroski, Moeller, Okrent, Siess, and Shewmon.

4. Special April 16-17, 1979 ACRS Meeting

The Committee agreed to schedule a special ACRS meeting to be held in Washington, D. C., on April 16-17, 1979. The main business to be considered at this meeting will concern the TMI-2 accident and its implications to other nuclear power plants. Members requested that the ACRS Staff obtain copies of the replies to IE Bulletin 79-05 to operators of B&W reactors, dated April 1, 1979. Replies to this bulletin are due on April 10.

The replies and the resulting NRC actions should be available for the Committee's consideration at the special meeting.

Members discussed and set up a tentative list of topics to be discussed at the April 16-17 special ACRS meeting.

- An analysis of reactor response to small-break LOCAs is necessary.
- Current codes do not adequately model the reactor response to small-break LOCAs; the codes cannot be interpolated accurately.
- Current codes do not model the location of a break.
- If a break changes in size, the analyses become invalid.

5. Problems Identified so far Relating to PWRs

Members identified a number of problems or lessons learned from the preliminary studies of the TMI-2 accident, and prioritized them into four lists A-D (see Appendix XXIII).

Analyses similar to those recommended for B&W plants should also be made on Combustion Engineering Plants. The problems may not be the same, but plant performance with respect to small-break LOCAs is now not known.

W plants without upper head injection systems are similar to Combustion Engineering plants. The upper head injection systems may make difference in plant performance, but that is not now fully understood.

Members identified the following immediate problems with regard to other Babcock and Wilcox reactors:

- It is necessary to understand the TMI-2 accident as soon as possible and to provide good operating instructions to operators.

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- If the above can be done in a few days, other B&W plants could run safely on the basis of accident probability.
- If above measures cannot be accomplished in a very short time, B&W reactors probably should be shut down.
- Indication of coolant level in the core is a minimum requirement.

a. Recent Information from Harrisburg

C. Michelson, ACRS Consultant, upon his return from the TMI-2 site, reported the following:

- The primary coolant pump 1A was shut down, pump 1B was started, and as a result the temperature profile of the damaged TMI-2 core shifted significantly (see Appendix XXIV).
- Most NRC work is being conducted in the Harrisburg area; the Bethesda Emergency Response Center is being reduced to an information center relating to this accident.
- A cadre of 250 to 300 engineers and technicians from private industry have been assembled to work on the problems.

- Current efforts relate to the removal of hydrogen from the primary system.
- Additional effort is being expended to remove radioactive materials from the auxiliary building.
- Primary system instrumentation is operating.
- Danger from core melt is believed to be passed.
- Members of the NRC Staff seem to have good working relations with the industrial cadre.

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VI. Executive Session (Open to Public)

[Note: James M. Jacobs was the designated Federal employee for this portion of the meeting.]

A. Meeting with NRC Commissioners

The Committee agreed on topics that they wished to discuss with the NRC Commissioners at the Joint NRC-ACRS Meeting held on Thursday afternoon, April 5, 1979.

B. Future Schedule

Members agreed on issues and projects to be reviewed at the 229th and subsequent ACRS meetings (see Appendix II).

C. Subcommittee Activities

A schedule of future subcommittee activities was provided to the Members (see Appendix III).

D. Subcommittee Reports

1. Regulatory Activities Subcommittee

The Committee concurred in the Regulatory position on Regulatory Guide 1.140 (Rev. 1), Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (see Appendix XXV).

2. Power and Electrical Systems Subcommittee

Mr. Ray recapped the incident that occurred at Arkansas Nuclear 1 (ANO) regarding degraded voltage and loss of off-site power (see Item IIC3 preceeding). He noted the manner in which safety systems were jeopardized by the decrease in voltage. He noted that matters regarding off-site power are considered in different manners at different plants.

The Power and Electrical Systems Subcommittee, as a follow-on to the review of the power supply failure at ANO, will consider the current regulatory requirements for auxiliary power supplies and will evaluate the adequacy of onsite

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power systems and also determine whether Regulatory Guide 1.68, Preoperational Testing, provides for adequate testing of the above systems to assure safety.

Mr. Ray will provide the LER Subcommittee with an evaluation of the incident at ANO.

3. Reactor Safety Research Subcommittee

Mr. Siess, Coordinator for the annual safety research report, proposed a schedule for preparation of the report (see Appendix XXIV).

E. NUREG-0531

Following a brief discussion of the NRC Staff Report, NUREG-0531, Investigation and Evaluation of Stress-Corrosion-Cracking in Piping of Light-Water Reactors, the Metal Component Subcommittee agreed to review the report and determine if proposed corrective actions are appropriate.

In view of the record of considerable stress corrosion cracking, the Reliability and Probabilistic Assessment Subcommittee agreed to re-evaluate the reliability of ECCS in nuclear plants.

F. ACRS Reports and Letters

1. Three Mile Island Nulcear Station Unit 2

The Committee prepared an interim report on the Three Mile Island accident providing ACRS interim recommendations regarding the March 28, 1979 accident (see Appendix XXVII).

2. Sequoyah Nuclear Power Plant Units 1 and 2

The Committee considered a draft report on its review for an operating license for the Sequoyah Nuclear Power Plant, Unit 1 and 2, but deferred completion of this report to the Commissioners until a better understanding can be developed of the implications of the March 28, 1979 accident at TMI-2.

The ACRS Staff was requested to ascertain whether it would be useful for the Committee to recommend temporary operation of Sequoyah 1 at zero power for testing purposes, without a report from the Committee recommending issuance of an operating license.

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Appendix I

APPENDIXES
TO
MINUTES OF THE 228TH ACRS MEETING
APRIL 5-7, 1979

3. <u>Palo Verde Nuclear Generating Station, Units 4 and 5</u>

The Committee considered a draft report to the Commissioners on its review of the application for a Construction Permit for the Palo Verde Nuclear Generating Station, Units 4 and 5, but deferred completion of this report until a better understanding can be developed of the implications of the March 28, 1979 accident at TMI-2.

The 228th ACRS Meeting was adjourned at 3:00 p.m., Saturday, April 7, 1979.

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228TH ACRS Meeting

Meeting Date:
April 5-7, 1979

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Max W. Carbon, Chairman
Milton S. Plesset, Vice-Chairman
Myer Bender
Harold Etherington
William Kerr
Stephen Lawroski
J. Carson Mark
William Mathis
Dade Moeller
David Okrent
Jeremiah Ray
Chester Siess
Paul Shewmon

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
James M. Jacobs, Technical Secretary
Herman Alderman
John H. Austin
Andres L. Bates
Paul A. Boehnert
Sam Duraiswamy
Elpidio G. Igne
Morton W. Libarkin
Richard K. Major
Thomas G. McCreless
John C. McKinley
Robert E. McKinney
Ragnwald Muller
Gary R. Quittschreiber
Jean A. Robinette
Richard P. Savio
Hugh E. Voress
Robert L. WRight

CONSULTANTS

C. Michelson
I. Catton
Z. Zudans
M. Trifunac
M. White

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APPENDIX I

PUBLIC ATTENDEES

228TH ACRS MTG.

Thursday, April 5, 1979

Amelinskx, Severin, CEN/SCK - Mol (Donk) Belgium
Ernett Blake, DPPT, 2000 Geibbs Chaney/Silver Spring, MD
R. Borsum, B&W, Derwood, MD
Bruce W. Churchill, Metropolitan Edison Co., Washington, DC
DOPCHIE, Vincotte, Rhode St. Genese, Belgium
Paul A. Dozinel, Fraction & Elechicite, Brusself, Belgium
John J. Fialkin, Wash. Star, Wash., DC
J. Michael Griesmeyer, ACRS Fellow, Santa Monica, CA
Craig Grochmal, Stone and Webster, Bethesda, MD
Hiroyoshi, Hamada, Tokyo Electric Power Co., 1901 L., NW, Wash., DC
Thomas R. Hencey, NUS Corp., Gaithersburg, MD
H. C. Huang, Westinghouse, Pittsburgh, PA
Dresse Hubert, Electrobel, Bruxelles, Belgique
S. B. Jacobs, Stone & Webster., Boston, MA
W. S. L. Kennedy, Stone & Webster, Boston, MA
A. Kimmins, Wash. Public Power Supply Sys., Richland, WA
A. Kranish, Trends Publishing, 3611 Taylor St, Chevy Chase, MD
V. Mackenzie, State Of. Calif. (PVC.), San Francisco, CA 94102
M. McGarry, Debevdise & Liberman, Wash., DC
Gregory Minor, MHB, San Jose, Calif.
R. C. L. Olson, BG&E Co., Lutherville, MD
Richard J. Paccione, Power Authority, 10 Columbus Circle, NY, NY
Paul C. Parshley, House Interior Committee, Alex, VA
James A. Quinn, IEAL, Alexandria, VA
N. S. Reynolds, Debeuoise & Liberman, Alexandria, VA
Noel Shirley, GE, Gaithersburg, MD
Kieoshi, Suzaki, Toshiba, Corp., San Jose, CA
Margaret Thomas, Wash. Post, Wash., DC
Thomas F. Timmons, Westinghouse, Pittsburgh, PA
Doug Todd, Amer. Nuclear Energy, 1750 K St., NW #300 Wash., DC
Marvin D. Tower, Jr., VEPCO, IVOR, VA
Clifford Webb, State of Calif. Sacramento, CA
Steve Wynkoop, McGraw-Hill Publications Co., Arlington, VA
Kathryne, M. Bruner, General Atomic Co., Wash., DC
P. B. Haga, Westinghouse, OPS
D. C. Cab, OPS
T. D. Martin, NUTECH

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NRC STAFF ATTENDEES

228TH ACRS MEETING

Thursday, April 5, 1979

Division of Project Management

Jim Meyer
John K. Long
L. P. Crocker

Division of Operating Reactors

D. Eisenhut
P. Check

Inspection and Enforcement

L. Higginbotham

Office of Reactor Regulatory Research

S. Levine
A. Budnitz

Div. of Site Safety & Systems Eval.

R. Denise

DEPARTMENT OF ENERGY

John Clarke
Warren Marton
John E. Baublitz
Gene Nardella

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NRC ATTENDEES

228TH ACRS MTG.

Friday, April 6, 1979

Div. of Project Management

L. P. Crocker
H. Silver
D. Vassalla
G. Heltemes
J. Villalera
R. Boyd
H. Clayton
O. D. Parr
S. Varga
E. Licitra

Div. of System Safety

D. A. Powers
J. P. Knight
V. Leung
M. Dunenfeld
J. R. Rajan
Om Chopra
J. Wermiel
F. Rosa
R. Meyer
R. Aermann
D. R. Lasher
F. Rinaldi

Nuclear Reactor Regulation

J. Greeves
C. Arkin

Nuclear Material Safety & Safeguards

J. Martin
J. Malaro

Div. of Systems Safety

M. W. Hodges

Div. of Site Safety & Systems Eval.

L. Reiter
J. Kane
R. Jackson
R. Priebe
R. Gonzales
W. Bivins

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APPLICANT ATTENDEES

228TH ACRS MEETING

Friday, April 6, 1979

Airzona Public Service Co.
Edwin E. Van Brunt, Jr.
Rex W. Kramer
D. B. Karner
John Allen

Snell & Wilmer
C. A. Bischoff
Arthur C. Gehr

Westinghouse
R. J. Sero
M. R. Harding
G. L. Augustine
Ray Skwarek
W. J. Johnson
P. J. Docherty

Bechtel Power Corp.½
L. G. Hinkelman
Dennis Keith
William G. Bingham

Combustion Engineering
J. Goldberg
A. E. Scheun
G. A. Davis
C. B. Brinkman
Charles Fuguson

Fugro, Inc.
John Scott

Tennessee Valley Authority

John Lobdell
R. Joe Hunt
R. D. Guthrie
Frank R. Hand
David Lambert
Mark R.Wisenburg
R. Joe Johnson
C. H. Noe
D. R. Denton
H. E. McConnell
E. G. Beasley
D. W. Wilson
C. R. Morgan
J. E. Gilleland
L. M. Mills
W. M. Seay
A. W. Crevasse
Wang Lau
Howard Crisler
W. F. Popp
V. S. Jephenson
Charles A. Myers
Walter I. Dothard
Richard J. Holt
George C. Klimkiewicz
A. Cornell

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PUBLIC ATTENDEES

228th ACRS MTG.

April 6, 1979

R. L. STright, Consultant to NRC
M. I. Goldman, NUS Corp.
V. MacKenzie, State of California
M. B. Whitaker, SCE&G
O. W. Dixon, SCE&G
J. E. McEwen, KMC, Inc.

Gene R. Plesset

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PUBLIC ATTENDEES

228TH ACRS MTG.

April 7, 1979

J. Michael Griesmeyer, ACRS, Santa Monica
M. M. Mlynczak, Teknekron, McLean, VA
C. R. Morgan, Tennessee Valley Authority, Knoxville, TN
Henry Myers, H. Int. Comm. Wash., DC
Isabel. R. Plesset, Self, San Marino, CA
Michael Stern, 1020 Cornelius Drive, Green Bay, Wis.

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Appendix II

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APPENDIX II

ACRS FUTURE AGENDA 4/2/79

ACRS MEETING PROJECT	TYPE OF REVIEW	REACTOR VENDOR	SER ISSUE DATE
MAY			
MILLSTONE 2	STRETCH POWER	CE	4/9/79
ATWS			
COMBINED LOADS			
JUNE			
NONE			
JULY			
SHOREHAM	OL	GE	6/1/79
LASALLE 1&2	OL	GE	6/1/79
FNP 1-8	ML	W	6/1/79
AUGUST			
WATTS BAR 1&2	OL	W	7/2/79
SEPTEMBER			
SAN ONOFRE 2&3	OL	CE	8/1/79
SUMMER 1	OL	W	8/1/79
HAVEN 1	CP	W	8/1/79

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Appendix III

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

APPENDIX III

April 7, 1979

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS AND TOURS

The following is a list of tours and Subcommittee meetings currently scheduled, subject to the approval of the Advisory Committee Management Officer. If you are listed and cannot attend a meeting, or if you are not listed but would like to attend, please advise the ACRS Office as soon as possible.

Most hotels currently being used by ACRS Members in the downtown Washington and Bethesda areas require a guaranteed reservation if arrival is scheduled after 6:00 p.m. Failure to use a room under these conditions involves forfeiture of the cost. Please advise the ACRS Office as soon as possible if you cannot attend a meeting for which you are scheduled so that reservations can be cancelled in time to avoid this.

MW Libarkin

M. W. Libarkin
Assistant Executive Director
for Project Review

cc: ACRS Technical Staff
M. E. Vanderholt
B. Dundr
R. F. Fraley
M. C. Gaske

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Appendix IV

PROJECT STATUS REPORT
PALO VERDE, UNITS 4 AND 5
ACRS CONSTRUCTION PERMIT REVIEW
APRIL 6, 1979
WASHINGTON, DC

PROJECT: Palo Verde, Units 4 and 5 (Construction Permit Review)

DESCRIPTION:

Palo Verde, Units 4 and 5 are replicates of Palo Verde, Units 1-3, located at the same site. Units 1-3, in turn, are CESSAR-80 Standard Design Plants. The plants are located on a 3800 acre desert site in Maricopa County, Arizona, about 36 miles west of Phoenix. Arizona Public Service Company is the largest percentage owner and has full authority and responsibility to design, engineer, construct, operate and maintain the plant. Combustion Engineering (CE) will provide the NSSS, and Bechtel Power Corporation is the architect-engineer. Highlights of the plant and plant-related design are:

NSSS: Two-loop CESSAR-80 Standard Design (3817 MWt) - 1235 MWe) consisting of the following base systems.

(1) NSSS
(2) NSSS Control System
(3) Reactor Protection System
(4) Engineered Safety Features Actuation System
(5) Chemical and Volume Control System
(6) Shutdown Cooling System
(7) Safety Injection System
(8) Fuel Handling System

Containment: Cylindrical, steel-lined, reinforced, post-tensioned, concrete structure with a free volume of about 2.7 million cubic feet. Design pressure and temperature are 60 psi gauge and 300^{o}F, respectively.

Site: Exclusion area -- within the boundaries of the site
Low population zone -- 4 miles in radius

Seismic Design: SSE -- 0.20 g
OBE -- 0.10 g

Nearby Industrial, Transportation, and Military Facilities:
There is, at present, no military, industrial, or airport facilities within 5 miles of the site. The Applicant has noted that the Maricopa County Planning Commission has a proposal under consideration for construction of a petroleum refinery and energy-related research facilities, 5.5 miles and 3.7 miles, respectively, from the site. If these facilities are to be constructed, the NRC will evaluate their impact on plant safety at the OL stage of the review.

Plant Cooling Water:
The plants will use a closed cycle cooling water system with 3 mechanical draft cooling towers per unit. Makeup water will be provided from an on-site storage reservoir that receives water from the city of Phoenix Water Reclamation Project. The ultimate heat sinks for each unit are Seismic Category I Spray Ponds. The ponds will store enough water for 30 days cooling supply.

Project Schedule:

PSAR Docketed	3/2/78	Operation:	5/88 - Unit 4
SER Issued	2/21/79		5/90 - Unit 5
CP Decision Date	12/14/79		(Units 1-3 operation is scheduled for 5/82, 5/84, and 5/86, respectively)
Construction Begins	Early 81 (both Units)		

OUTSTANDING ISSUES:
NRC has identifed four outstanding issues requiring resolution. They are:

(1) Review of the constructor's Quality Assurance Program (identification of the constructor has not yet been made by the Applicant).

(2) Review of the Applicant's financial qualifications (to be conducted at a later stage of review).

(3) Review of the seepage analysis to determine the design basis groundwater levels. Preliminary review indicates the safety margin between design and predicted groundwater levels is less for Units 4 and 5 than for Units 1-3, due to uncertainties in the seepage from the unlined storage reservoirs, and particularly the evaporation ponds near Unit 5. NRC's position is that the design basis groundwater levels for Units 4 and 5 be the same as that for the evaporation ponds (930 ft. MSL). The Applicant wants a design basis groundwater level that is 20 feet below plant grade (923 and 920) for Units 4 and 5, respectively.

(4) Review of the revised CE ECCS evaluation model. Since issuance of the CESSAR SER, NRC has determined that the strain-rupture curve in the flow-blockage submodel may be nonconservative. In response to a NRC request, CE submitted an alternate flow-blockage model. NRC has concluded that sufficient margin exists in the presently approved flow-blockage model to offset the possible nonconservative aspects of the strain-rupture curve. NRC believes, however, that upon review of the alternate flow-blockage model, changes to the ECCS evaluation model may be necessary.

NRC REVIEW:

This construction permit review is being conducted under NRC's "Streamlined Plant Review Process" as an experiment in expediting Staff plant reviews. As noted, Palo Verde, Units 4 and 5 replicate Palo Verde, Units 1-3. NRC issued a CP for Units 1-3 on May 25, 1976. Units 1-3 are CESSAR-80 Standard Design Plants. A Preliminary Design Approval (PDA) for CESSAR was issued on December 31, 1975. NRC discussed this expedited review process at the Subcommittee meeting.

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ACRS REVIEW:

The ACRS reviewed and approved a PDA for CESSAR-80 at its 185th meeting (September 1975). Approval of the CP for Palo Verde, Units 1-3 was granted at the 187th meeting (November 1975). Copies of the Committee letters are attached.

GENERIC ITEMS:

Following is a discussion of the status of action on ACRS generic items as they affect Palo Verde, Units 4 and 5. It should be noted that the Committee has resolved four items at the March 1979 meeting. These items are asterisked and noted in the discussion:

II-1 Turbine Missiles - Resolved due to peninsular orientation and turbine overspeed protection.

II-2 Effective Operation of Containment Sprays in a LOCA - This item is under generic review by the Staff. NRC has a Reg. Guide in preparation designed to address a portion of this issue.

II-3 Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock - Resolved for CESSAR by conformance to Appendix G.

II-4 Instruments to Detect (Severe) Fuel Failures - This item is under generic review by Staff. The CESSAR Standard Design does include instruments to detect fuel failures. Use of these instruments is addressed in Section 4.2 of the NRC Standard Review Plan.

*II-5A Monitoring for Loose Parts Inside the Reactor Pressure Vessel - Resolved for Palo Verde by the Applicant's commitment to install a loose parts monitoring system.

II-5B Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel - NRC is developing a task action plan for this item and considers it a Category B Task.

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*This generic item was resolved by the Committee at the March 1979 Meeting.

II-6A Common Mode Failures: Reactor Scram Systems - This item is under generic review by the Staff. NRC is attempting to resolve this issue as part of the ATWS resolution now under review by the Committee.

II-6B Common Mode Failures: Alternating Current Sources Onsite and Offsite - This item is currently under generic review by the Staff. The review is be conducted under Technical Activity No. A-35, "Adequacy of Offsite Power Systems." A NUREG report addressing this item is scheduled for completion by July 15, 1980. NRC also has a study underway to improve the reliability of diesel generators. This study is included in the scope of Technical Activity No. B-56, "Diesel Reliability."

II-6C Common Mode Failures: Direct Current Systems - This item is under generic review by the Staff. Task Action Plan A-30 will address this problem. Following completion of this program, a NUREG report will be issued, and a Staff position regarding the adequacy of existing acceptance criteria for D.C. power systems will be developed. Completion is scheduled for mid-1979.

II-7 Behavior of Reactor Fuel Under Abnormal Conditions - NRC believes this item should no longer be carried as an unresolved generic item.

II-8 BWR Recirculation Pump Overspeed During a LOCA - Not applicable to Palo Verde.

II-9 The Advisability of Seismic Scram - This item is under generic review by the Staff. The NRC had proposed resolution of this item to the ACRS in 1977, stating that the Staff does not propose to require installation of seismic trip systems on commercial nuclear power plants. The ACRS suggested that the seismic scram should be set at about 1/2 the SSE value; the Committee also expressed interest in what the Japanese are doing in regard to seismic scrams. NRC has learned that the Japanese do install seismic scrams in their reactors with trip levels set 1/2 to 2/3 the SSE design level. NRC now carries this generic items as a Category D Task Action.

II-10 ECCS Capability for Future Plants - This item is now included in the research topics of the Commission's long-range safety research plan for improved safety system concepts.

IIA-1 Ice Condenser Containments - This item is not applicable to Palo Verde.

IIA-2 PWR Pump Overspeed During a LOCA - This item is under generic review by the Staff. CE has submitted a topical report on pump overspeed which is under review by the NRC Staff. NRC is also performing independent pump overspeed calculations during a LOCA. Results of this study will be available during this year. The Staff has this item under the scope of Task Action Plan B-68.

IIA-3 Steam Generator Tube Leakage - This item is under generic review by the Staff. Both NRC and NSSS vendors are conducting studies on steam generator tube degradation mechanisms. NRC also has an experimental research program underway at Battelle PNL to verify burst and cyclic strengths of steam generator tubes and to obtain leakage rate data. The results of these efforts will be used to develop tube leakage rate limits and plugging criteria which will be incorporated into Reg. Guides and standard Technical Specifications. NRC is also reviewing the probability and consequences of the main steam line break and LOCA, concurrent with steam generator tube failures. The Staff is also evaluating the inservice inspection programs for steam generator tubes. Task Action Plan A-4 addresses the above activities for CE steam generators.

IIA-4 ACRS/NRC Periodic 10-Year Review of All Power Reactors - Since this item does not apply to facilities for which NRC review for an OL has not been completed, this matter is not applicable to Palo Verde.

IIB-1 Computer Reactor Protection System - This item was erroneously reported in the SER to be not applicable to Palo Verde. However, the CESSAR-80 design will use the CPC (Core Protection Calculator) system. This item will be clarified at the March 29, 1979 Subcommittee meeting.

*IIB-2 Qualification of New Fuel Geometries - This item is resolved for Palo Verde at the CP stage of review.

IIB-3 Behavior of BWR Mark III Containments - Not applicable to Palo Verde.

IIB-4 Stress Corrosion Cracking in BWR Piping - Not applicable to Palo Verde.

IIC-1 Locking Out of ECCS Power-Operated Valves - This item is under generic review by the Staff and has been assigned Task Action Plan B-8; this plan is currently under development.

IIC-2 Design Features to Control Sabotage - This item is under generic review by the Staff, and is resolved for Palo Verde at the CP stage of review by compliance with current Staff requirements.

IIC-3A Decontamination of Reactors - This item is under generic review by the Staff and is included under the scope of Task Action Plan A-15. The Staff notes that, to date, there has been little experience with primary system decontamination in operating U.S. commercial power reactors. EPRI has initiated research programs on decontamination, and NRC will study the results of the Dresden Unit 1 primary system decontamination now underway.

IIC-3B Decommissioning of Reactors - This item is under generic review by the Staff and is included under the scope of Task Action Plan B-64. NRC noted that AIF and Battelle PNL have studies underway on reactor decommissioning alternatives.

IIC-4 Vessel Support Structures - This item is under generic review by the Staff and is included under the scope of Task Action Plan A-2. CE had submitted a topical report that argued that a break at the cold leg nozzle of a reactor vessel has such a low probability that no further analysis is necessary. NRC has rejected that argument and informed all PWR applicants that this analysis must be undertaken.

*This generic item was resolved by the Committee at the March 1979 Meeting.

IIC-5 Water Hammer - This item is under generic review by the Staff under the scope of Task Action Plan A-1. A series of subtasks are being actively pursued in this area.

*IIC-6 Maintenance and Inspection of Plants - This item is resolved for Palo Verde at the CP stage of review by compliance with NRC's current requirements.

IIC-7 Behavior of Mark I Containments - Not applicable to Palo Verde.

*IID-1A Safety-Related Interfaces Between Reactor Island and Balance-of-Plant - This item is resolved for Palo Verde at the CP stage of review.

IID-1B Systems Interactions in Nuclear Power Plants - This item is under generic review by the Staff and is included in the scope of Task Action Plan A-17. NRC has determined that contract assistance is necessary to complete this task.

IID-2 Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment - This item is under generic review by the Staff and is included in the scope of Task Action Plan C-1. A plan of action has been established, pending management approval. Such areas as field experience, adequacy of current designs and quality assurance practices, the practicability of testable designs, and the need for the development of guidance criteria will be reviewed under this task.

IIE-1 Soil-Structure Interactions - This item is under generic review by the Staff and is included under the scope of Task Action Plan A-40. An in-depth study will evaluate, from an analytical point of view, the various techniques, including deconvolution analyses, being performed. Attention will be given to requirements concerning variation of soil properties, enveloping the response spectra at the foundation level, and fixing a minimum value of the response spectra at the foundation level.

*This generic item was resolved by the Committee at the March 1979 Meeting.

Attached is a cross-index of ACRS generic items vs. the NRR generic tasks. It should be noted that with the March 1979 ACRS generic items letter, the numbering system we have used in the past is being dropped in favor of a Arabic numbering system. This new system will simply number the generic items 1-52 for the resolved items, and the first unresolved item will begin with number 53.

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TABLE D-1

CROSS INDEX OF ACRS GENERIC ITEMS VS NRR GENERIC TASKS

ACRS GENERIC ITEM		NRR GENERIC ITEM	
II-1	Turbine Missiles	A-32 A-37	Missile Effects Turbine Missiles
II-2	Effective Operation of Containment Sprays in a LOCA	C-10	Effective Operation of Containment Sprays in a LOCA
II-3	Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock	A-11	Reactor Vessel Materials Toughness
II-4	Instruments to Detect (severe) Fuel Failures	Not yet considered by NRR. Will be considered as a Category C proposal.	
II-5A	Loose Parts Monitoring	B-60	Loose Parts Monitoring Systems
II-5B	Monitoring for Excessive Vibration	B-73	Monitoring for Excessive Vibration
II-6	Common Mode Failures	C-13	Non-Random Failures
II-6A	Scram Systems	A-9	ATWS
II-6B	Alternating Current Systems	A-24 A-25 A-35 A-44 B-56	Qualification of Class IE Safety Related Equipment Non-Safety Loads on Class IE Power Sources Adequacy of Offsite Power Systems Station Blackout Diesel Reliability
II-6C	Direct Current Systems	A-24 A-25 A-30 A-44	Same as above Same as above Adequacy of Safety Related DC Power Supplies Same as above
II-7	Behavior of Reactor Fuel Under Abnormal Conditions	B-22	LWR Fuel
II-8	BWR Recirculation Pump Overspeed During LOCA	B-68	Pump Overspeed during a LOCA
II-9	The Advisability of Seismic Scram	D-1	Advisability of Seismic Scram
II-10	ECCS Capability for Future Plants	D-2	ECCS Capability for Future Plants
II A-1	Ice Condenser Containments	B-54	Ice Condenser Containments
II A-2	PWR Pump Overspeed During a LOCA	B-68	PWR Pump Overspeed During a LOCA
II A-3	Steam Generator Tube Leakage	A-3 W A-4 CE A-5 B&W	Steam Generator Tube Integrity
II A-4	ACRS/NRC Periodic 10-year Review of All Power Reactors	Not a generic technical task. Is being treated as a policy matter.	

TABLE D-1 (Continued)

ACRS GENERIC ITEM		NRR GENERIC ITEM	
II B-1	Computer Reactor Protection System	A-19	Digital Computer Protection System
II B-2	Qualification of New Fuel Geometries	B-22	LWR Fuel
II B-3	Behavior of BWR Mark III Containments	B-10	Behavior of BWR Mark III Containments
II B-4	Stress Corrosion Cracking in BWR Piping	A-42	Pipe Cracks in Boiling Water Reactors
II C-1	Locking Out of ECCS Power Operated Valves	B-8	Locking Out of ECCS Power Operated Valves
II C-2	Design Features to Control Sabotage	A-29	Design Features to Control Sabotage
II C-3A	Decontamination of Reactors	A-15	Chemical Decontamination
II C-3B	Decommissioning of Reactors	B-64	Decommissioning of Reactors
II C-4	Vessel Support Structures	A-2	Asymmetric Blowdown Loads on the Reactor Vessel
II C-5	Water Hammer	A-1	Water Hammer
II C-6	Maintenance and Inspection of Plants	B-34	Occupational Radiation Exposure Reduction
II C-7	Behavior of BWR Mark I Containments	A-6 A-7	Mark I Short Term Program Mark I Long Term Program
II D-1A	Safety Related Interfaces Between Reactor Island and Balance-of-Plant	Not a generic technical task. Is being treated as a policy matter.	
II D-1B	Systems Interactions in Nuclear Power Plants	A-17	Systems Interactions in Nuclear Power Plants
II D-2	Assurance of Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	C-1	Assurance of Continuous Long-Term of Seals on Instrumentation and Electrical Equipment
I E-1	Control Rod Drop Accident (BWRs)	D-3	Control Rod Drop Accident (BWRs)
I E-2	Rupture of High Pressure Lines Outside Containment	B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
I E-3	Isolation of Low Pressure From High Pressure Systems	B-63	Isolation of Low Pressure Systems Connected to RCPB

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Appendix V

PVNGS LICENSING SCHEDULE 123/45

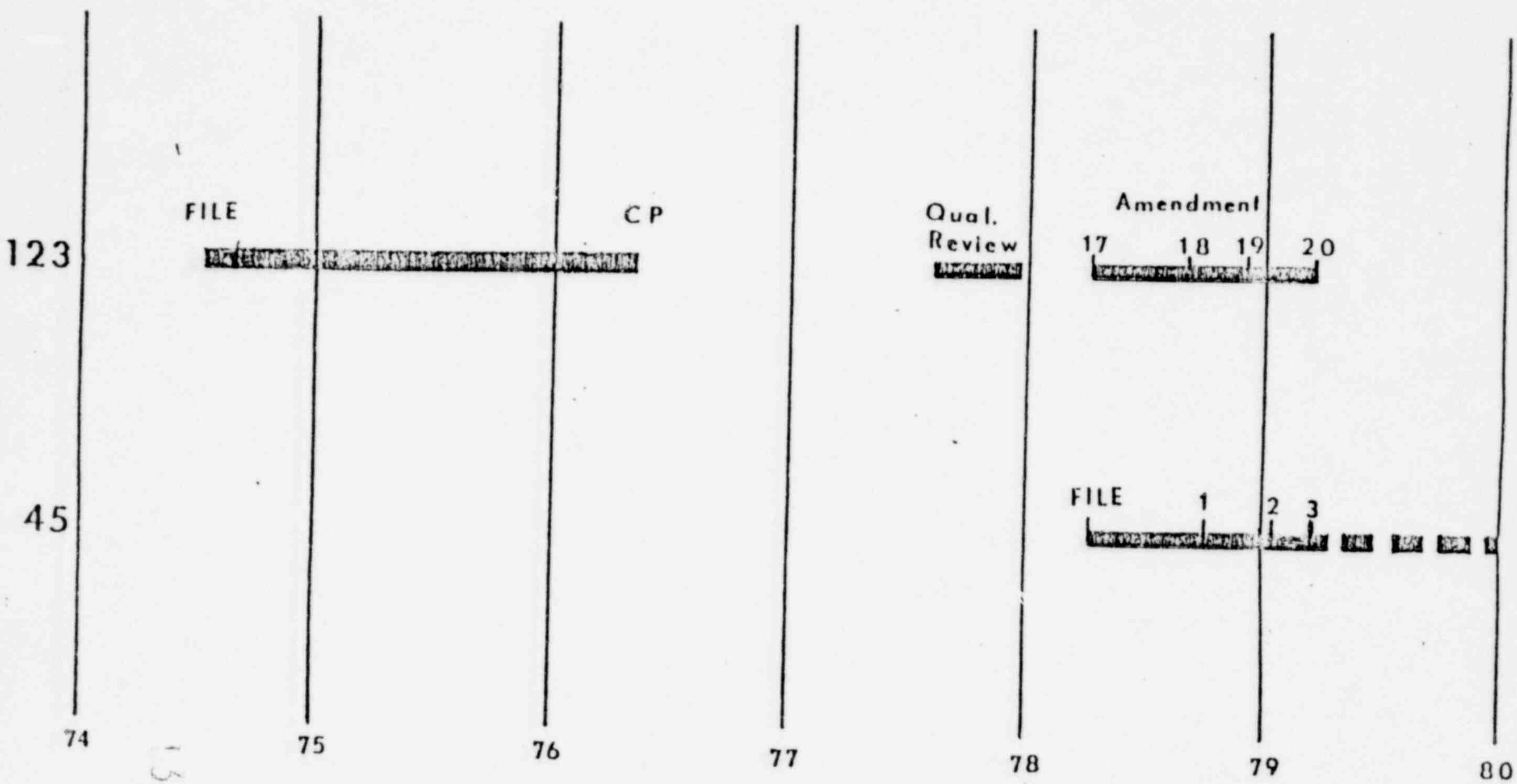

FIGURE 2

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PVNGS LICENSING MILESTONES

6/74	FILE PVNGS 1, 2 & 3 PSAR
5/76	PVNGS 1, 2 & 3 CP
4/77	NRC NOTIFIED OF INTENT TO REPLICATE PVNGS 1, 2 & 3
8/77	REQUEST FOR QUALIFICATION REVIEW
12/77	QUALIFICATION REVIEW LETTER ISSUED (34 REGULATORY GUIDES AND STAFF POSITIONS TO BE ADDRESSED FOR ALL 5 PVNGS UNITS)
3/78	PVNGS 1, 2 & 3 PSAR AMENDMENT # 17 FILED PVNGS 4 & 5 PSAR FILED
9/78	PVNGS 1, 2 & 3 PSAR AMENDMENT # 18 FILED PVNGS 4 & 5 PSAR AMENDMENT # 1 FILED
1/79	PVNGS 1, 2 & 3 PSAR AMENDMENT # 19 FILED PVNGS 4 & 5 PSAR AMENDMENT # 2 FILED
2/79	PVNGS 1, 2 & 3 PSAR AMENDMENT # 20 FILED PVNGS 4 & 5 PSAR AMENEMENT # 3 FILED
3/79	PVNGS 4 & 5 SER
10/79	PVNGS 1, 2, 3, 4 & 5 FSAR TO BE FILED

FIGURE 3

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INTRODUCTION

PVNGS 4 & 5

LOCATION: APPROXIMATELY 36 MILES WEST OF THE NEAREST BOUNDARY OF THE CITY OF PHOENIX, MARICOPA COUNTY, STATE OF ARIZONA

NSSS: COMBUSTION ENGINEERING, INC.
SYSTEM 80 PWR

TURBINE: GENERAL ELECTRIC COMPANY

SOURCE OF COOLING WATER:

CONDENSER COOLING : SEWAGE EFFLUENT CONTRACTED FROM THE CITY OF PHOENIX

OTHER : WELLS

ARCHITECT ENGINEER:

BECHTEL POWER CORPORATION, NORWALK, CALIFORNIA

CONSTRUCTOR:

TO BE IDENTIFIED

ENVIRONMENTAL CONSULTANT:

NUS CORPORATION, ROCKVILLE, MARYLAND

GEOLOGICAL CONSULTANT:

FUGRO, INC., LONG BEACH, CALIFORNIA

FIGURE 4

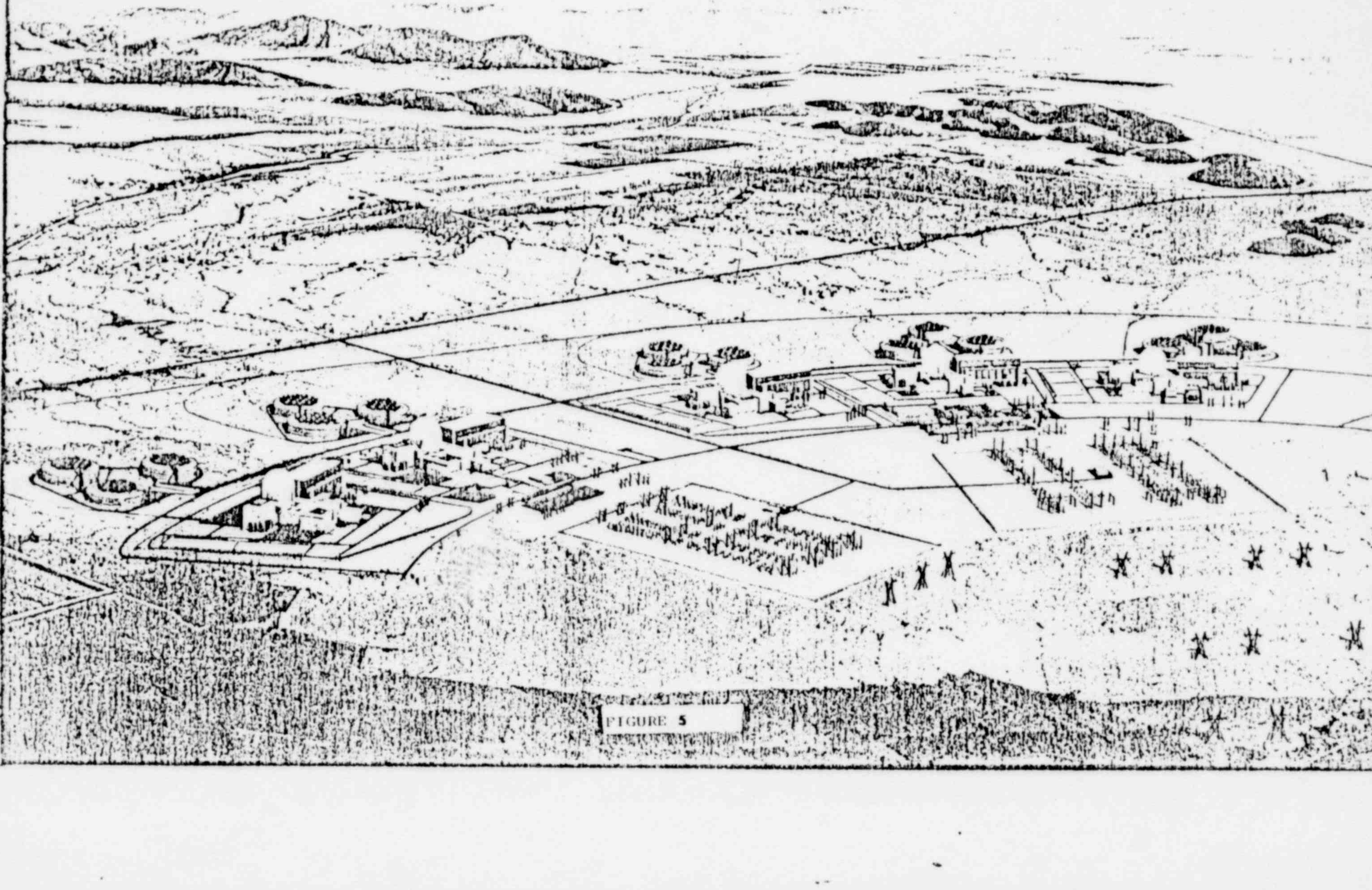

FIGURE 5

POOR ORIGINAL

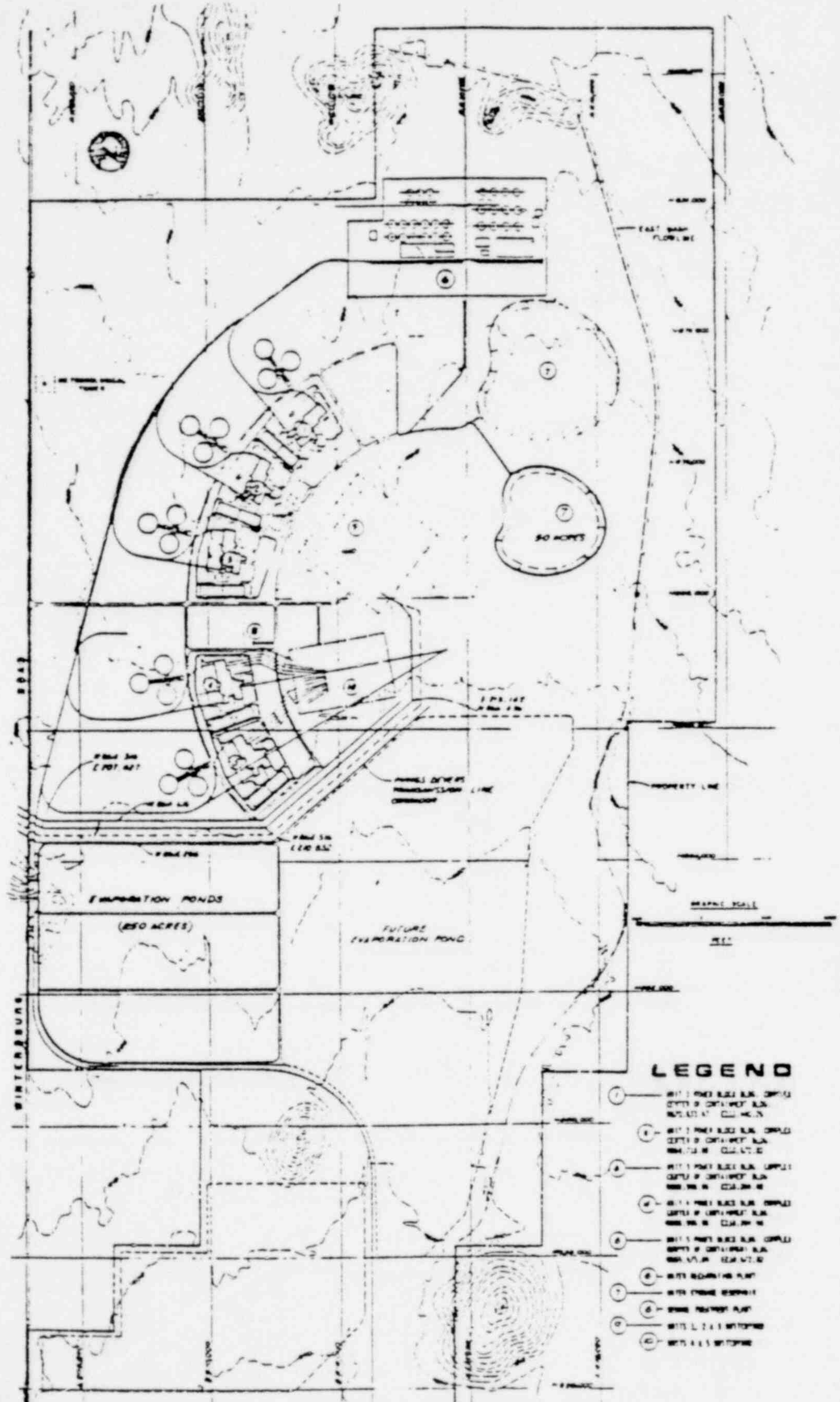

FIGURE 6

1374 140

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PVNGS 4 & 5 PARTICIPANTS

– ARIZONA PUBLIC SERVICE COMPANY	39.1%
– SOUTHERN CALIFORNIA EDISON COMPANY	32.3%
– DEPARTMENT OF WATER AND POWER, CITY OF LOS ANGELES	11.7%
– SAN DIEGO GAS AND ELECTRIC COMPANY	5.2%
– EL PASO ELECTRIC COMPANY	4.0%
– NEVADA POWER COMPANY	2.2%
– CITY OF ANAHEIM	1.5%
– CITY OF BURBANK	1.0%
– CITY OF GLENDALE	1.0%
– CITY OF PASADENA	1.0%
– CITY OF RIVERSIDE	1.0%

PROJECT MANAGER AND OPERATING AGENT:

ARIZONA PUBLIC SERVICE COMPANY

Figure 7

ARIZONA PUBLIC SERVICE COMPANY
MASTER ORGANIZATION

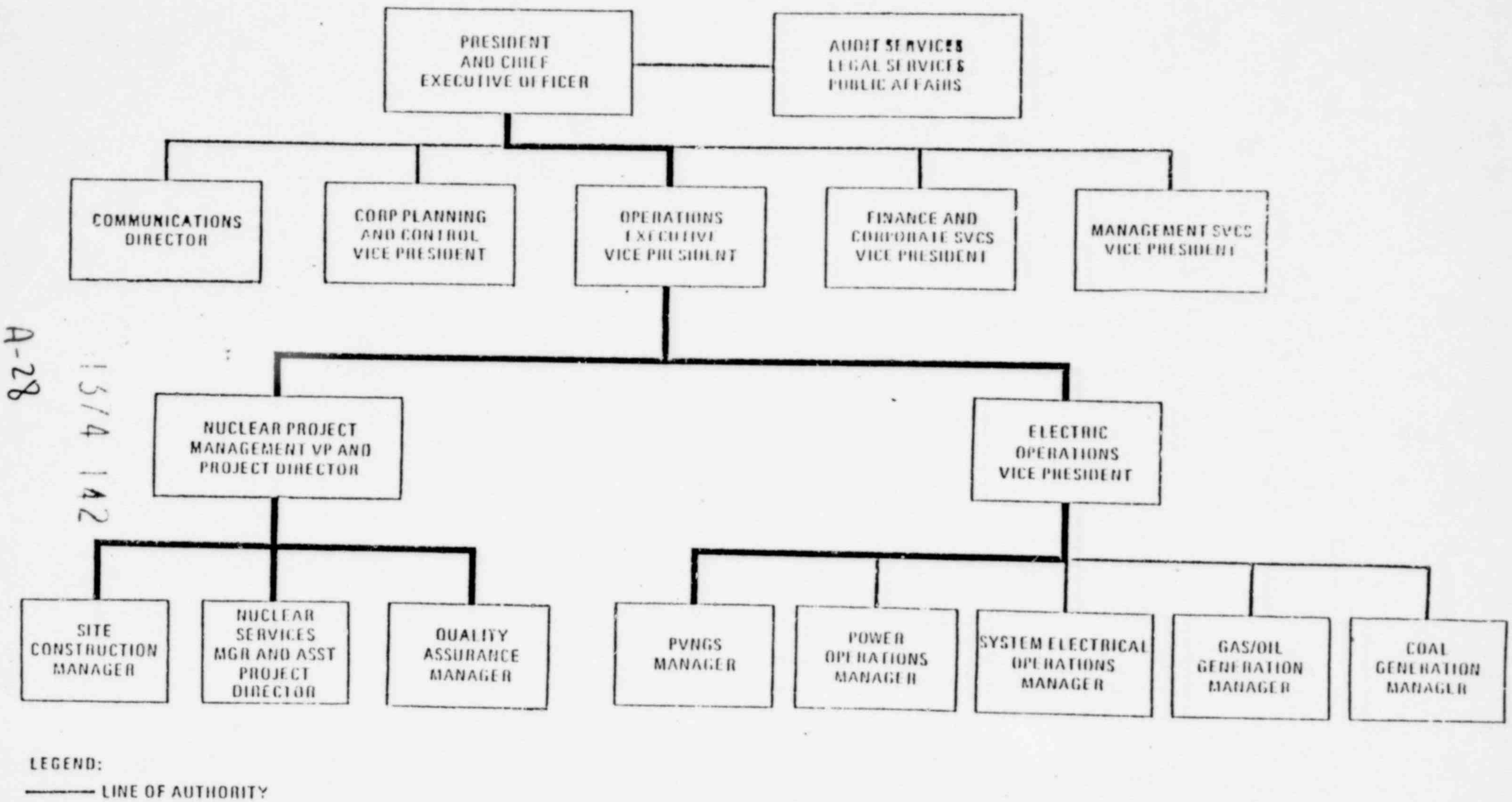

FIGURE 8

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PVNGS

CONSTRUCTION SCHEDULE

BEGIN CONSTRUCTION PVNGS-4&5.	EARLY 1981
FUEL LOAD UNIT 1	11/81
FUEL LOAD UNIT 2	11/83
FUEL LOAD UNIT 3	11/85
FUEL LOAD UNIT 4	11/87
FUEL LOAD UNIT 5	11/89

FIGURE 9

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Appendix VI

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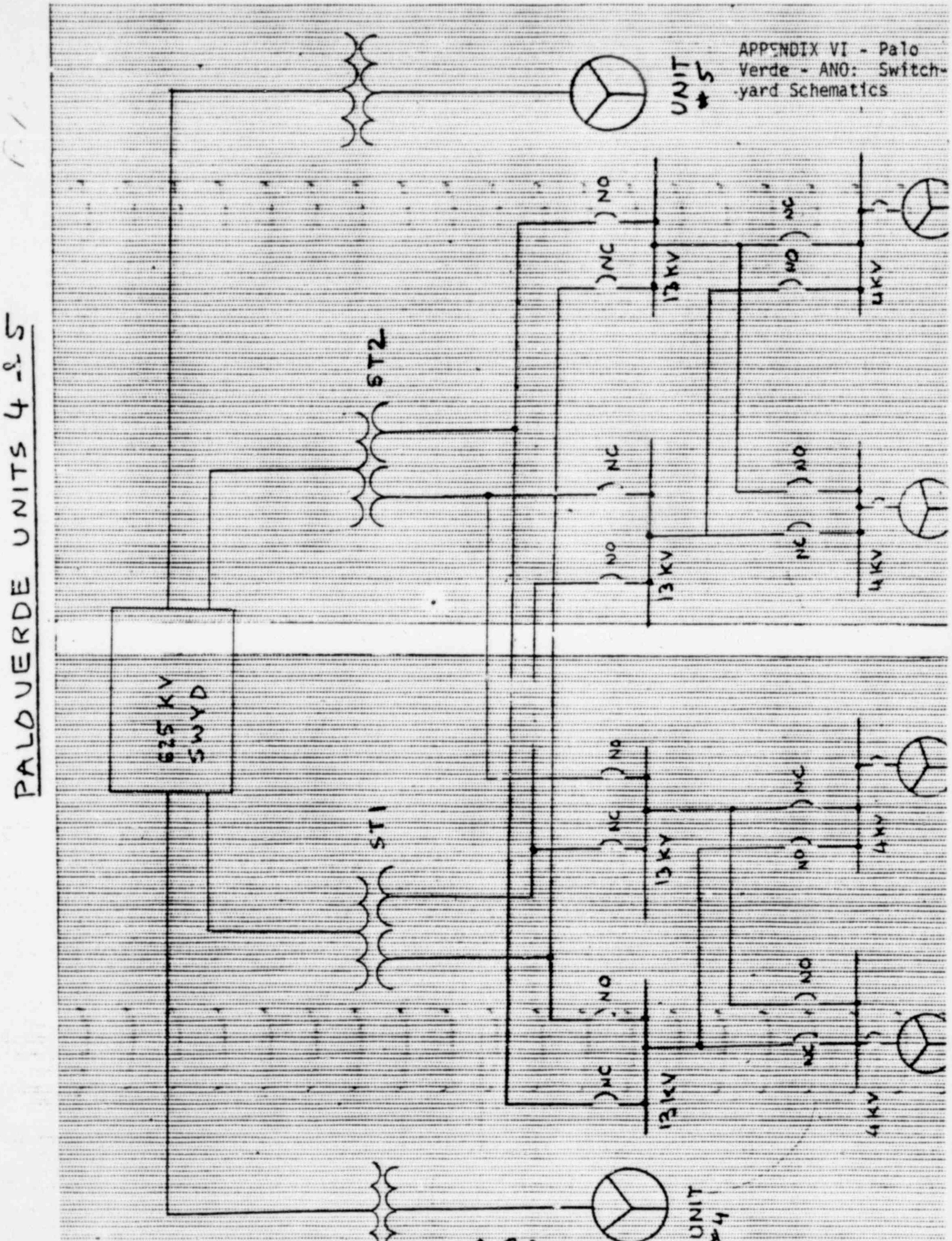
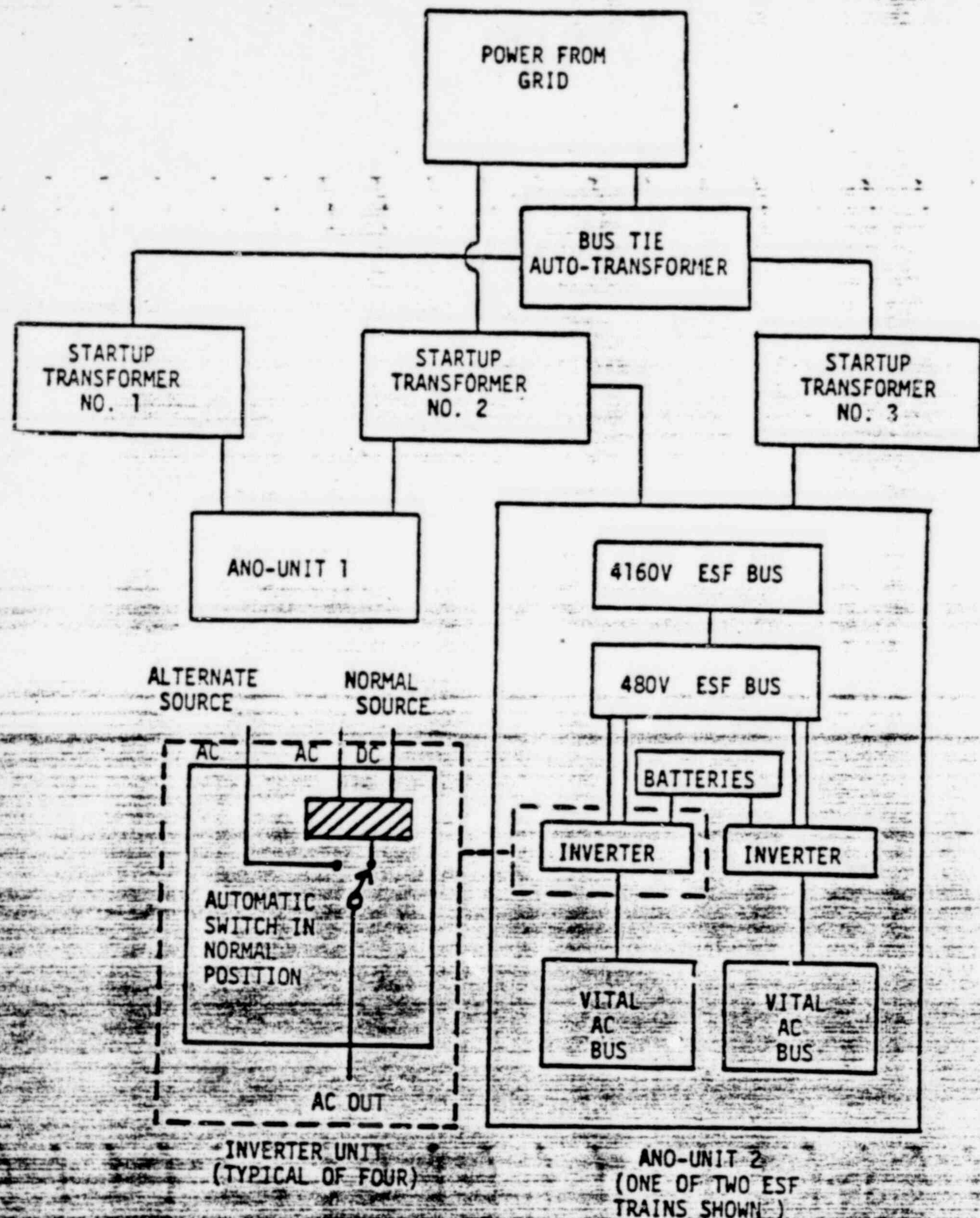
APPENDIX VI - Palo Verde - ANO: Switch-yard Schematics



A-30

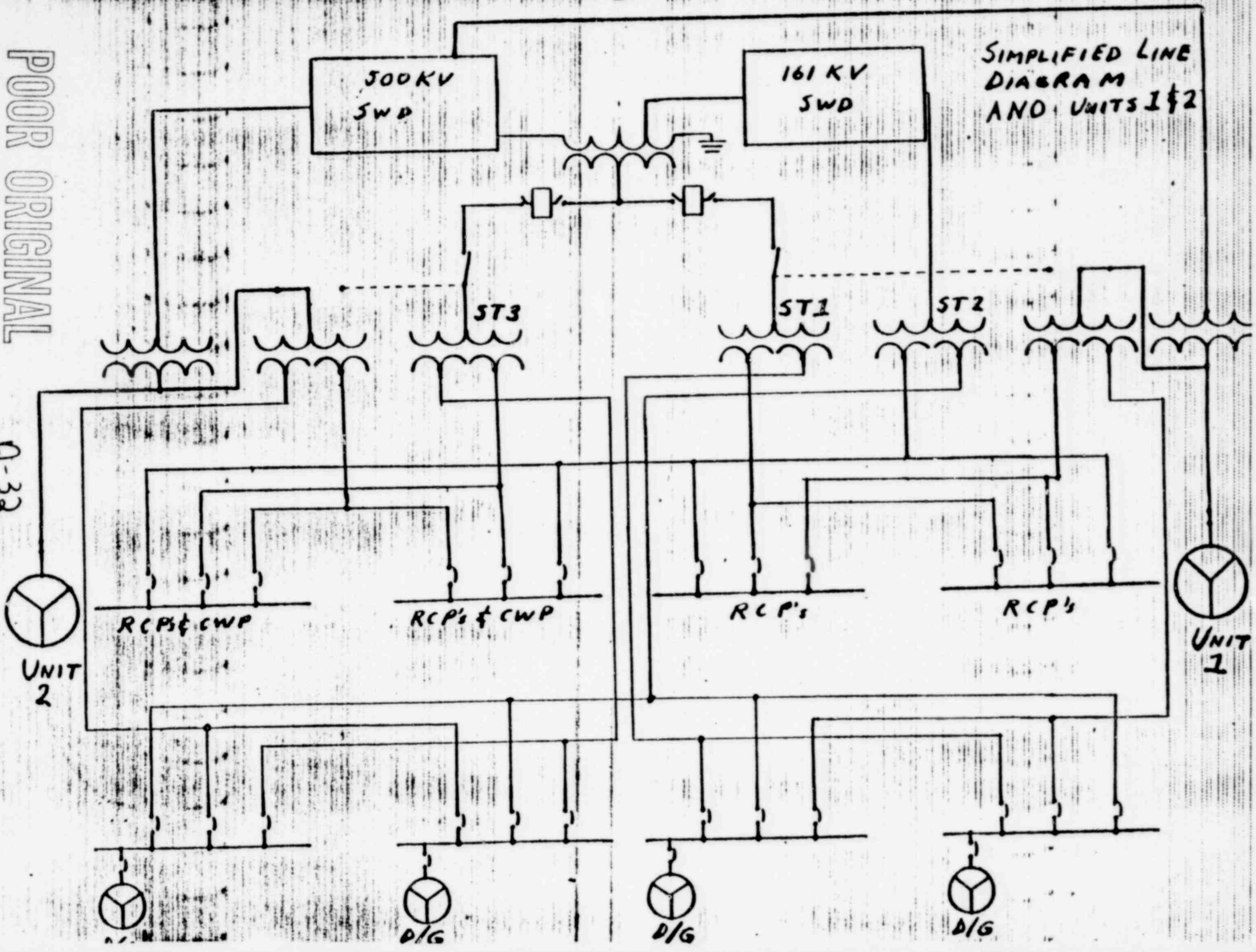


SIMPLIFIED BLOCK DIAGRAM - ELECTRIC DISTRIBUTION

FIGURE 1

POOR ORIGINAL

A-32



1574 147

Appendix VII

1374 148

DIFFERENCES BETWEEN

PVNGS 1, 2 & 3 AND PVNGS 4 & 5

- There are no differences in the power block design between all five units.

EXCEPTIONS TO CESSAR

- Refueling water temperature
  ($27\frac{1}{2}$ hours after shut down)

PVNGS	CESSAR
125° F	135° F

FIGURE 10

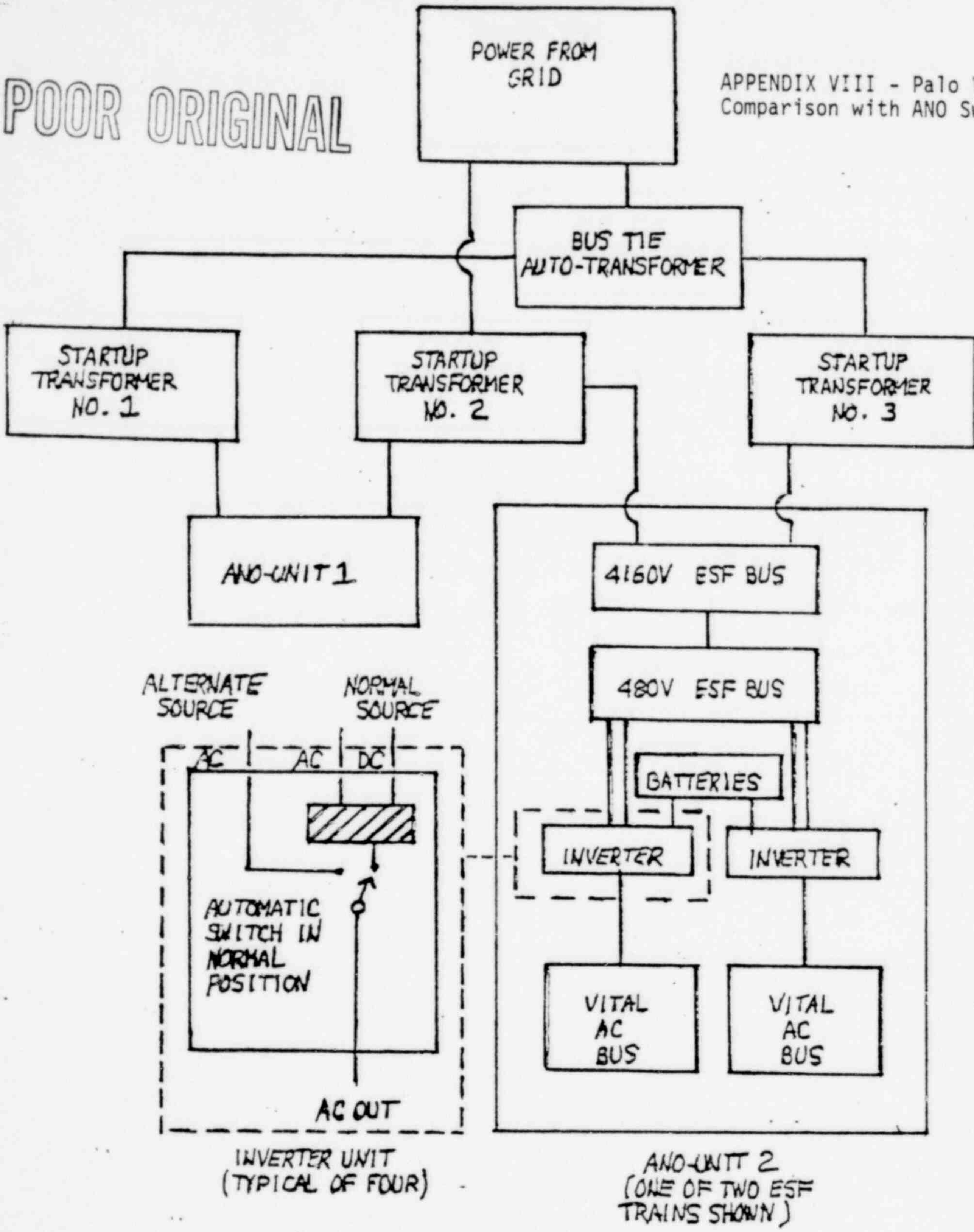
1374 149

1374 150

Appendix VIII

ANO POWER DISTRIBUTION

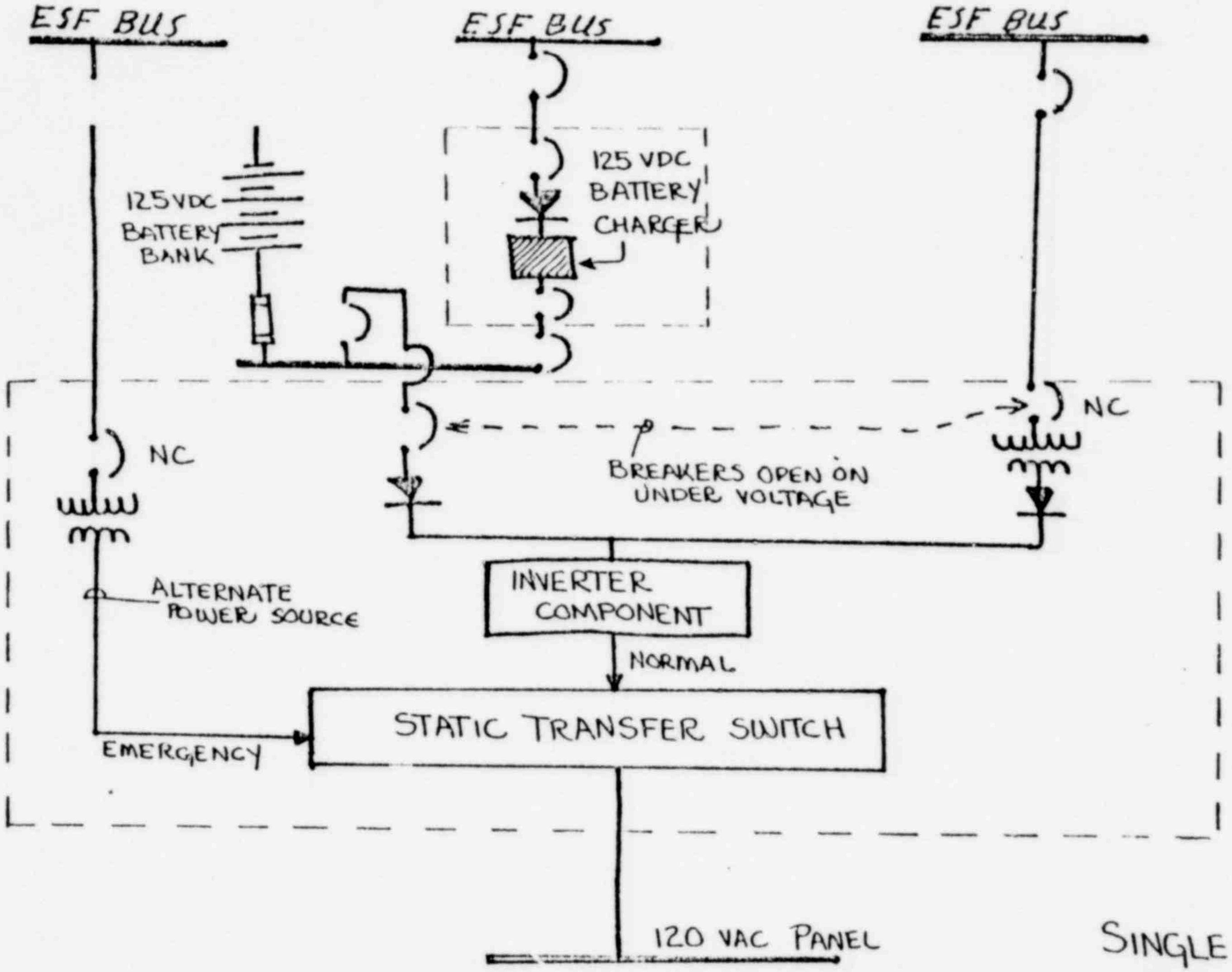
APPENDIX VIII - Palo Verde 4-5:
Comparison with ANO Switchyard



SIMPLIFIED BLOCK DIAGRAM - ELECTRIC DISTRIBUTION

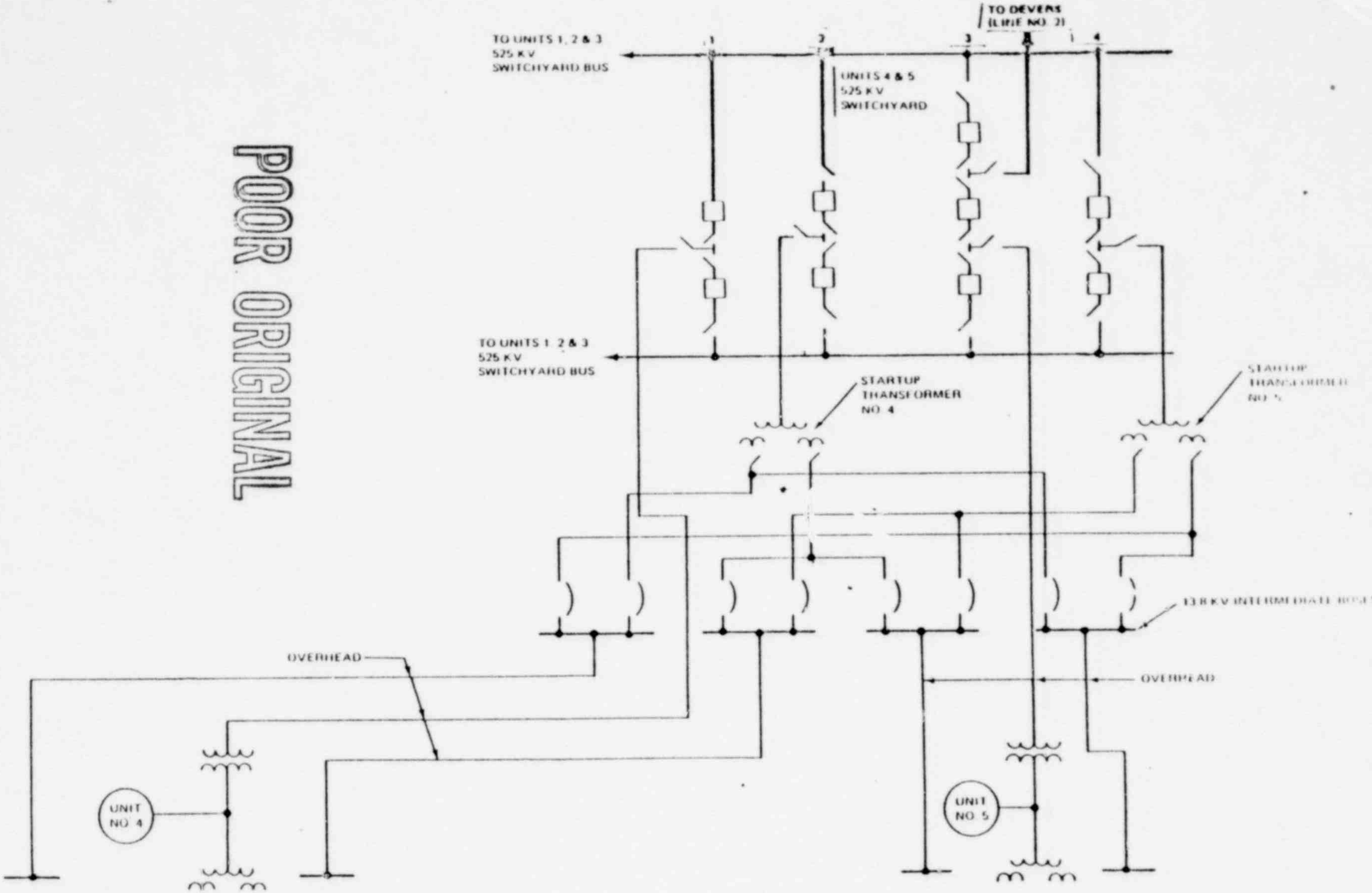
Figure 11

1374 151



SINGLE LINE DIAGRAM
120 VAC ESSENTIAL POWER
ARKANSAS NUCLEAR
{ANO}

Figure 12



POOR ORIGINAL

Palo Verde Nuclear Generating Station
Units 4 & 5

525 KV SWITCHYARD
SINGLE LINE DIAGRAM FOR UNITS 4 & 5

Figure 8.2-7

January 12, 1979

Amendment 2

FIGURE 13

A-36

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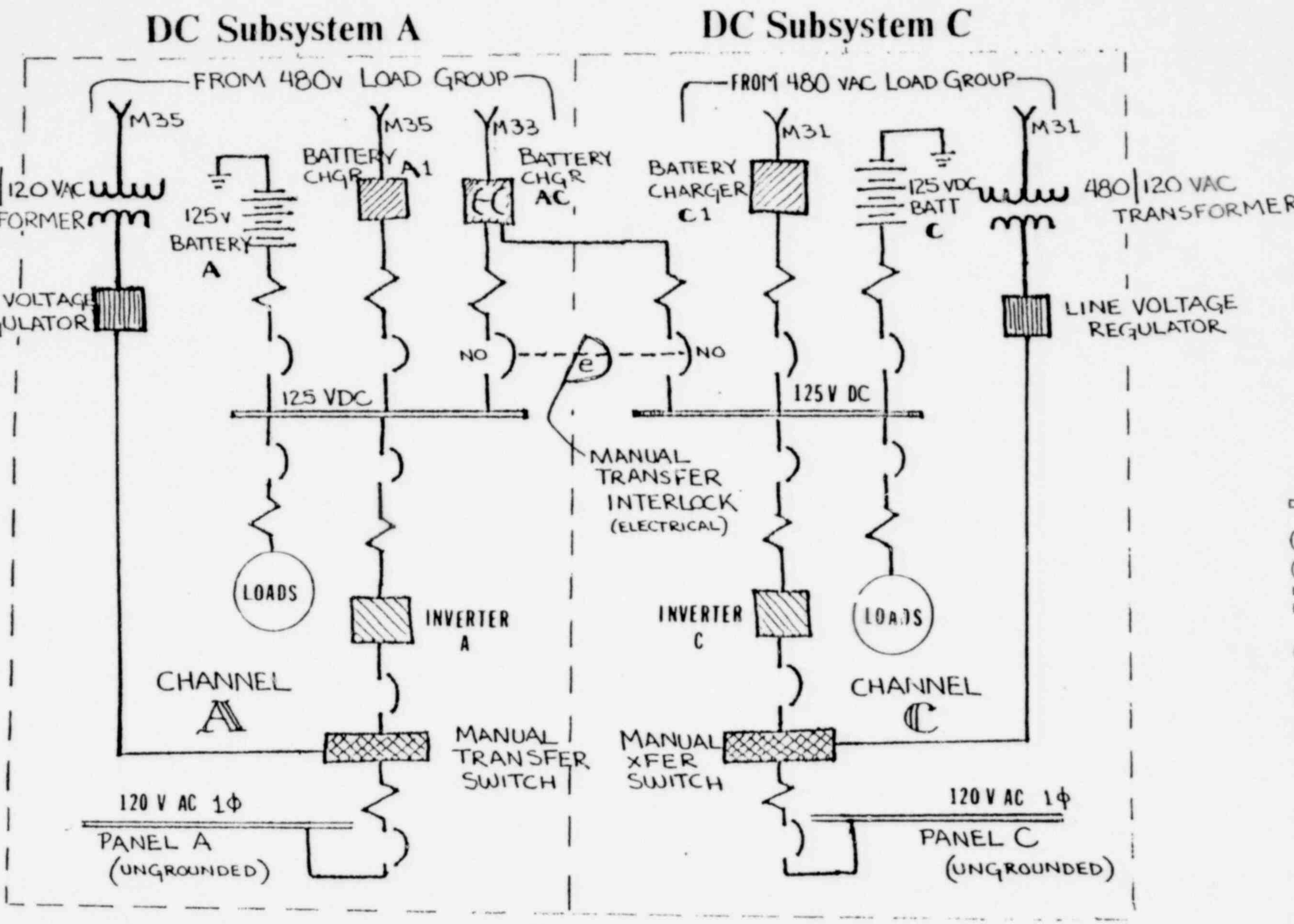

PVNGS
DC POWER SYSTEM
—LOAD GROUP 1—

Figure 14

1374 155

Appendix IX

APPENDIX IX - Palo Verde 4 and 5:
Sequencer Design Features

SEQUENCER DESIGN FEATURES

THE SEQUENCER IS DESIGNED TO RESPOND TO:

- LOCA WITH OFFSITE POWER AVAILABLE
- LOCA WITH SIMULTANEOUS LOSS OF OFFSITE POWER
- LOCA FOLLOWED AT A LATER TIME BY LOSS OF OFFSITE POWER
- LOSS OF OFFSITE POWER FOLLOWED AT A LATER TIME BY A LOCA
- LOSS OF OFFSITE POWER WITHOUT LOCA
- ACCIDENT OTHER THAN LOCA WITH OFFSITE POWER AVAILABLE
- ACCIDENT OTHER THAN LOCA WITHOUT OFFSITE POWER AVAILABLE

LOAD SHEDDING IS PERFORMED ONLY ON BUS UNDERVOLTAGE

ACCIDENT SIGNALS FOLLOWING INITIAL SEQUENCER OPERATION CAUSE SEQUENCER RESET AND SEQUENTIAL STARTING OF ANY ADDITIONAL REQUIRED EQUIPMENT

Figure 15

SEQUENCER DESIGN IMPLEMENTATION

- 100% REDUNDANT POWER SUPPLIES - ONE SUPPLIED BY AC, ONE SUPPLIED BY DC
- SOLID STATE COMPONENTS (EXCEPT ACTUATION RELAYS) OF PROVEN RELIABILITY
- SEQUENCER RESPONDS TO SIAS CONTACT OPENING (FAIL SAFE)
- SEQUENCER OUTPUT RELAYS ARE NORMALLY ENERGIZED, DE-ENERGIZE TO ACTUATE
- SEQUENCER HAS CONTINUOUS AUTO-TEST WITH CONTROL ROOM ALARMS ON DETECTION OF FAILURE
- SEQUENCER COMPONENTS ARE MODULARIZED PROVIDING QUICK AND EASY MAINTENANCE OR REPLACEMENT

Figure 16

DESIGN REVIEW SUMMARY

PVNGS DESIGN USING SINGLE SOLID-STATE ESF LOAD SEQUENCER PER SAFETY TRAIN VS.

	ALTERNATE	RELIABILITY IMPACT	REASON
o	USE OF INDIVIDUAL TIME DELAY RELAYS	UN-RELIABILITY INCREASED BY 7×10^3	ADDITIONAL COMPONENTS OF REDUCED RELIABILITY
o	USE OF SEPARATE ON-SITE AND OFF-SITE SEQUENCERS	UN-RELIABILITY INCREASED BY 2	ADDITIONAL COMPONENTS TO CAUSE INADVERTENT SIMULTANEOUS LOADING
o	USE OF RELAY LOGIC TO START EQUIPMENT ON ESFAS AND SEQUENCING ONLY UNDER LOSS OF OFFSITE POWER CONDITION	UN-RELIABILITY INCREASED BY 6×10^2	ADDITIONAL COMPONENTS OF REDUCED RELIABILITY (RELAYS)

Figure 17

1374 158

1374 159

Appendix X

APPENDIX X - Palo Verde 4 and 5:
Emergency Plan

EMERGENCY PLANNING FOR PVNGS 4 and 5

- PROVISION TO EXPAND TO 5 UNIT STATION
- RESPONDS TO VARIOUS EMERGENCIES INCLUDING
  - MINOR PERSONAL INJURY WITH RADIOLOGICAL COMPLICATIONS
  - NUCLEAR ACCIDENT WITH OFFSITE CONSEQUENCES
- MARICOPA COUNTY IS RESPONSIBLE BY LAW FOR DEALING WITH OFF SITE CONSEQUENCES

1374 160

Figure 18

MARICOPA COUNTY, ARIZONA

NATURAL AND TECHNOLOGICAL DISASTER PLAN*

November 1977

*BASED ON CALIFORNIA STATE PLAN

1374 161

Figure 19

ANNEX J - NUCLEAR REACTOR INCIDENTS

1374 162

Figure 20

Appendix XI

1374 163

March 30, 1979

STATUS REPORT
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

The Sequoyah Nuclear Plant is located on the west bank of the Chickamuga Lake on the Tennessee River. The site is approximately 9.5 miles northeast of Chatanoogaa, TN and is located in a rural area heaving no unusual characteristics. The nuclear steam supply system (NSSS) and the initial core loading will be supplied by Westinghouse Electric Corporation. The reactor containment will be of the ice condenser type and the fuel will be the 17 X 17 "R" grid design. Sequoyah will utilize the upper head injection (UHI) in the ECCS and will be the second plant come before the Committee utilizing this design. The Sequoyah is similar to the McGuire Nuclear Station which was the lead plant with the UHI design. The ice condenser containment is similar to that used at McGuire and D. C. Cook, Units 1 and 2 (D. C. Cook, Unit 1 utilizes the 15 X 15 grid fuel design, while D. C. Cook, Unit 2 utilizes the 17 X 17 "R" grid fuel design). Tables comparing the design features of the Sequoyah to similar plants, along with some figures illustrating important features of design are included as Attachment 1. The Tennessee Valley Authority will act as both the architect-engineer and the construction contractor.

The construction permit for the Plant was issued on June 27, 1970. Construction was started on June 5, 1969 and is currently about 97% complete for Unit 1 and 80% complete for Unit 2. The Applicant's projected fuel load dates are June 1979 for Unit 1 and December 1979 for Unit 2. It is believed that these fuel load dates will be met.

The NRC Staff issued their Safety Evaluation Report on the Sequoyah plant on March 1, 1979. The NRC Staff's reevaluation of the seismic design base for the Sequoyah plant appears to be the most substantial issue in this review. During the construction permit review, the NRC Staff concluded that a modified Houser response spectrum anchored at 0.18g was acceptable as the safe shutdown earthquake. This conclusion was based on the assumption that the maximum historic earthquake (the 1897 Modified Mercalli Intensity VIII at Giles County, VA) might re-occur anywhere within the tectonic province in which Sequoyah is sited. The NRC Staff has since adopted the more conservative response spectra specified in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design Nuclear Power Plants," and would characterize a Modified Mercalli Intensity VIII earthquake with a peak acceleration of 0.25g (Trifunac and Brady, 1975). The NRC Staff discusses the results of their evaluation in Section 2.5 of the SER. The NRC Staff has required TVA develop a site-specific shutdown earthquake response spectra for the Sequoyah and to reevaluate the plant response using the site-specific response spectra. TVA was also required to compare the probability of the safe shutdown earthquake being exceeded at Sequoyah to other TVA plants that meet the Standard Review Plan. The NRC Staff has concluded that the seismic hazard for the Sequoyah plant was comparable to other plants in the eastern United States. The NRC Staff will also require that TVA quantify the seismic design margins in the safe shutdown and residual heat removal equipment. This process will similar to what is being done for the Davis Besse nuclear plant.

Outstanding Issues, Confirmatory Issues, Licensing Conditions - Items of Disagreement

The NRC Staff's summary of the Outstanding Issues, Confirmatory Issues, and Licensing Conditions is attached. At this point, TVA has complied with all of the NRC Staff's recommendations and there are no items of disagreement listed in the SER. TVA did disagree vigorously with the NRC Staff on the need for a study which would quantify the seismic design margins in the safe shutdown and residual heat removal equipment but has complied with the NRC Staff's recommendation. This issue is discussed in the introduction of this report.

ACRS Generic Items

The status of the NRC Staff and Applicant actions on the ACRS generic items is as follows: (*indicates items suggested for inclusion in generic item list in ACRS letter.)

1374 145

53 - Turbine Missiles

The Sequoyah facility has a peninsular turbine orientation and has a turbine overspeed protection system. With the exception of the essential raw water cooling intake structure, this configuration protects all systems important to safety from low trajectory turbine missile strikes. The Staff has concluded that the probability of a missile strike on the intake structure is less than 10^{-3} and has concluded that this is acceptable. The Staff considers this item to be resolved on this basis.

* 54 - Effective Operation of Containment Sprays in a LOCA

The NRC Staff considers this item to be resolved on the basis that no credit is taken in the accident analysis for fission product removal by the containment sprays. The ice condenser is designed to remove iodine from the post-accident atmosphere passing through the ice bed. Sodium tetreborate has been added to the ice to enhance the iodine absorption characteristics of the ice. The Technical Specifications will require a minimum ice pH whenever the reactor is critical. The NRC Staff feels that thisgeneric item is resolved for McGuire.

* 55 - Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock

The NRC Staff considers this item to be resolved on the basis of conformance to (or approved exceptions to) the Appendix G of 10 CFR 50.

* 56 - Instruments to Detect Severe Fuel Failures

The Sequoyah facility utilizes gamma monitors on a hot leg sampling line. The adequacy of the instrumentation to detect failures associated with very rapid fuel failure has not yet been established.

* 57 - Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel

The NRC Staff has indicated that this item is under generic review and is unresolved for this facility. The Applicant has made no commitment as to the installation of equipment in the event that the NRC considers that the usefulness of such devices has been established.

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* 58 - Non-Random Multiple Failures

This item is unresolved for this facility. These matters have been addressed to some extent by the NRC Staff.

58A - Scram Systems

The NRC has published reports on the anticipated transients without scram in December of 1975 in which they identified the portions of the reactor system that needed modifcations to improve the reliability. In addition, these reports provided guidelines on evaluation models, analysis assumptions, and system reliability requirements and acceptance limits. NUREG-0460, "Anticipated Transients Without Scram for Light-Water Reactors," is currently being reviewed by the ACRS.

58B - Alternating Current Sources

The Staff is addressing this under Technical Activity No. A-35, "Adequacy of Offsite Power Systems." The Staff is evaluating the need to upgrade the offsite power source and its interface with onsite power systems. A NUREG report is currently scheduled to be completed by July 15, 1980. Technical Activity No. B-56, addresses the need to improve the reliability of the diesel generators. The NRC Staff has contracted with the University of Dayton to: (1) perform a study of LERs related to diesel generator malfunctions, (2) make a limited number of visits to operating facilities, and (3) obtain manufacturers' recommendations regarding operations, maintenance, and repair of diesel equipment and to survey a comparable industry experience with standby emergency power supply.

58C - Director Current Sources

The Staff is addressing this under Task Action Plan A-30. To this date, Task Action Plan A-30 has addressed the issues of the data base, the recalculation of allowable time for manual actions, and the consequences of a total loss of d-c power. The next phase of this work will attempt to quantify the d-c power system reliability in relationship to assuring adequate decay heat removal capability. A NUREG report on this subject is scheduled for issuance in mid-1979.

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*59 - Behavior of Reactor Fuel Under Abnormal Conditions

The NRC Staff, in their December 4, 1978 Status Report, indicated that it was their belief that this item should no longer be considered an unresolved generic item. The ACRS has not yet concurred with the NRC Staff. Research directed toward the understanding of the behavior of reactor fuel under abnormal conditions is continuing.

*60 - BWR/PWR Pump Overspeed During a LOCA

This item is unresolved and is under generic review by the NRC Staff. The NRC Staff has asked each BWR vendor to submit its most recent prediction of pump overspeed during a LOCA in order to access the potential for pump flywheel failure and the validity of electrical breaking or other means of controlling pump speed. The NRC Staff is performing some independent reactor coolant pump overspeed calculations using the RELAP-4/MOD 5 computer code. It is expected that results will be obtained from this study during 1979 (Task Action Plan B-68).

61 - The Advisability of Seismic Scram

The Applicant has not proposed the use of the seismic scram for this facility and the Staff has indicated that they will not require such a scram. This matter has been discussed with the Committee on a number of occasions in the past. A Lawrence Livermore Laboratory report, UCRL-52156, "Advisability of Seismic Scram," was initially used by the NRC Staff as the basis for not requiring the installation of seismic trip systems on commercial power plants. This study, however, addressed only relatively low "g" value trip levels. Some Members have indicated that it would perhaps be more appropriate to consider trip levels that were much higher and suggested consideration of trip levels of about one-half of the SSE design level. The NRC Staff in its discussions with the Japanese have determined that the Japanese do require the installation of seismic scram systems. Trip levels are typically set about one-half to two-thirds of the SSE design level. The NRC Staff is continuing to address this matter under Task Action Plan No. D-1.

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62 - ECCS Capability of Future Plants

The NRC Staff has indicated that this item is unresolved for this facility and is under generic review and notes that it is included in one of the research topics in the Commission's long range safety research plan for improved safety system concepts. The McGuire design does, however, utilize the 17 x 17 "R" grid fuel and the upper head injection system.

* 63 - Ice Condenser Containments

The Sequoyah plant is the third station to come before the Committee for an operating license review with an ice condenser. Programs are in place for monitoring the performance of ice condenser containments. D. C. Cook Units 1 and 2 are, to this date, the only units which are operating with ice condenser containments. It is expected that both the McGuire and Sequoyah plants will load fuel in mid-1979. The Staff has, at this state, developed an independent analytical capability for analyzing the short-term ice condenser performance. The results are, to this date, compare favorably with Westinghouse calculations.

* 64 - Steam Generator Tube Leakage

The Staff has indicated that this item may be considered to be partially resolved by the requirement for inservice inspection. Steam generators used in the Sequoyah plant were manufactured prior to the implementation of the latest Westinghouse steam generator design fixes. Nuclear steam supply system vendors are currently conducting research programs to study the structural integrity of steam generator tubes that are subjected to various degradation mechanisms. The NRC Staff is funding a confirmatory experimental research program at the Pacific Northwest Laboratory to verify the burst and cyclic strengths of degraded steam generator tubes and to obtain leakage rate data. The Brookhaven National Laboratory is currently in the process of evaluating the impact of steam generator tube failures on the consequences of the main steamline break accident. The Idaho National Engineering Laboratory is developing a computer code which will aid in the evaluation of the effe[illegible] of tube plugging on the predicted peak clad temperatures and on emer[illegible] [illegible]ore cooling system performance following a postulated loss of [illegible] Statistical studies are being conducted at the Sandia National Laboratory to confirm the adequacy of existing inservice inspection criteria and to develop schemes for optimizing sampling techniques.

*65 - Periodic (10-year) Review of All Power Reactors

This item is unresolved and is under generic review.

66 - Computer Reactor Protection System

This item is not applicable to the Sequoyah plant. The license of this type are not being used at the Sequoyah plant. The NRC Staff considers this item resolved for the Sequoyah plant.

67 - Behavior of BWR Mark II Containments

This item is not applicable to the Sequoyah plant.

68 - Stress Corrosion in BWR Piping

This item is not applicable to the Sequoyah plant.

*69 - Locking Out of ECCS Proper Operated Valves

The NRC Staff has accepted valve lockout in the adminsitrative controls established by the Applicant at the Sequoyah and considers that this item is resolved on this basis. The generic aspects of this matter are being studied under the Task Action Plan B-8.

70 - Design Features to Control Sabotage

This item is unresolved and is under generic review by the NRC Staff. The Sequoyah facility is in compliance with the current NRC requirement.

*71 - Decontamination of Reactors

This item is unresolved and is under generic review by the NRC Staff within the scope of Task Action Plan A-15.

* 72 - Decommissioning of Reactors

This item is unresolved and is under review by the NRC Staff within the scope of Task Action Plan B-64. It is anticipated that this program will be completed in approximately two years.

73 - Vessel Support Structures

The load analysis has been performed for this facility and the structures have been found to be adequate. The NRC Staff has concurred in this analysis. The ACRS Fluid Dynamics Subcommittee has reviewed the Westinghouse analysis models and has concluded that the models are conservative. This item is considered to be resolved for the Sequoyah plant.

*74 - Waterhammer

This item is unresolved and is under generic review by the NRC Staff within the scope of Task Action Plan A-1.

75 - Behavior of BWR Mark I Containments

This item is not applicable to the Sequoyah plant.

*76 - Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

This item is unresolved and is under generic review by the NRC Staff within the scope of Task Action Plan C-1. The NRC Staff has established a plan of action and is waiting management approval. The plan includes a schedule for accomplishing the needed investigation into: field experience, the adequach of current designs and quality assurance practices, practicability of testable designs, and the need for the development of regulatory guide criteria.

77 - Soil- Structure Interaction

This item is considered by the NRC Staff to be not applicable to the Sequoyah plant since the principal seismic Category I structures are founded on rock. Category I structures not founded on rock are conservatively designed.

Intervenors Significant Differences of Opinion Among the NRC Staff

The Sequoyah application will not have a hearing and there are no intervenors in the case. We have received no requests for time to make oral statements at the Subcommittee or written statements from members of the public. No significant differences of opinion among the NRC Staff have been identified.

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TABLE 6.2-1

COMPARISON OF CONTAINMENT DESIGN PARAMETERS

	Catawba	McGuire	D. C. Cook	SEQUOYAH
Reactor Containment Volumes				
(net-free volume, cubic feet):				
Upper Compartment	720,000	717,000	746,000	698,000
Ice Condenser	127,000	111,000	127,000	110,500
Lower Compartment	374,000	368,000	368,000	383,000
Total Containment Volume	1,221,000	1,196,000	1,241,000	1,191,500
Reactor Containment Air Compression Ratio:	1.40	1.41	1.41	1.43
Reactor Power (megawatts, thermal):	3,582	3,579	3,394	3,582
Design Energy Release to Containment:				
Initial Blowdown Mass Release (pounds)	498,200	493,210	543,000	543,330
Initial Blowdown Mass Energy Release (Btu)	324.2×10^6	318.4×10^6	334.6×10^6	334.6×10^6
Ice Condenser Parameters:				
Weight of Ice Condenser (pounds)	2.55×10^6	2.45×10^6	2.45×10^6	2.45×10^6
Vent Flow Areas (lower Compartment, square feet)				
Vent Flow Area Past Steam Generators (total)	2,372	2,724	2,440	2,372
Vent Flow Area Past Pressurizer	632	679	740	232
Vent Flow Area Through Lower Inlet Doors	1,064	1,064	1,064	1,344
Containment Spray Flow (LOCA Analysis, gallons per minute):				
One Spray Train Inoperable				
Upper Compartment	3,400	3,432	2,000	4,750
Lower Compartment	0	0	900	0
One Residual Heat Removal Pump Inoperable				
Upper Compartment	2,000	1,623	2,000	2,000
Lower Compartment	0	0	0	0
Total Spray	5,400	5,055	4,900	6,750
Containment Design Pressure (pounds per square inch gauge)	15.0	15.0	12.0	

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Revised by Amendment 27, September 23, 1974

TABLE 1.3-1

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

Sequoyah Nuclear Plant Units 1 and 2 - Comparison with Donald C. Cook Units 1 and 2 and Trojan

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
3.0	Containment	Section 3.8.2	D. C. Cook Units 1 and 2	The use of freestanding steel primary containment vessel.
4.0	Reactor			
	Fuel	Section 4.2.1	Trojan	None.
	Reactor Vessel Internals	Section 4.2.2	D. C. Cook Units 1 and 2, Trojan	D. C. Cook Units 1 and 2 and Sequoyah Units 1 and 2 have thermal shields. Trojan has neutron pads. Sequoyah upper internals have been modified to incorporate UHI.
	Reactivity Control	Section 4.2.3	D. C. Cook Units 1 and 2, Trojan	None.
	Nuclear Design	Section 4.3	D. C. Cook Units 1 and 2, Trojan	None.
	Thermal-Hydraulic Design	Section 4.4	D. C. Cook Units 1 and 2, Trojan	The total primary heat output and coolant temperatures are higher for Sequoyah and Trojan than for the D. C. Cook Plant.
5.0	Reactor Coolant System	Sections 5.1, 5.2	D. C. Cook Units 1 and 2, Trojan	The following have been added or changed: • New requirements for fracture toughness testing. • New means of determining heat-up and cool-down rates.
	Reactor Vessel*	Section 5.4	D. C. Cook Units 1 and 2, Trojan	None.
	Reactor Coolant Pumps*	Section 5.5.1	D. C. Cook Units 1 and 2, Trojan	None.
	Steam Generators*	Section 5.5.2	D. C. Cook Units 1 and 2, Trojan	None.
	Piping*	Section 5.5.3	D. C. Cook Units 1 and 2, Trojan	None.

* All components designed and manufactured to Code edition in effect at date of purchase order.

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TABLE 1.3-1 (Continued)

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
5.0 (Cont'd)				
	Residual Heat Removal System	Section 5.5.7	D. C. Cook Units 1 and 2, Trojan	None.
	Pressurizer*	Section 5.5.10	D. C. Cook Units 1 and 2, Trojan	None.
6.0	Engineered Safety Features			
	Emergency Core Cooling System	Section 6.3	D. C. Cook Units 1 and 2, Trojan	D. C. Cook Units 1 and 2 and Trojan do not have an Upper Head Injection System
	Ice Condenser	Section 6.2	D. C. Cook Units 1 and 2.	Trojan does not use an ice condenser.
7.0	Instrumentation and Controls			
	Reactor Trip System	Section 7.2	System functions are similar to D. C. Cook Units 1 and 2, Trojan	
	Engineered Safety Features Systems	Section 7.3	Systems functions are similar to D. C. Cook Units 1 and 2, Trojan	None.
	Systems Required For Safe Shutdown	Section 7.4	System functions are similar to D. C. Cook Units 1 and 2, Trojan	None.
	Safety Related Display Instrumentation	Section 7.5	Parametric display is similar to that of D. C. Cook Units 1 and 2, Trojan	Actual physical configuration may differ due to customer design philosophy.
	Other Safety Systems	Section 7.6	Operational Functions are similar to D. C. Cook Units 1 and 2, Trojan	None.
	Control Systems	Section 7.7	Operational functions are similar to D. C. Cook Units 1 and 2, Trojan	The Sequoyah Nuclear Plant has a 50 percent load rejection capability while that of the D. C. Cook Plant is 100 percent. The rod position indication for the Sequoyah Nuclear Plant and the D. C. Cook Plant is an analog system; Trojan's RPI is a digital system.

TABLE 1.3-1 (Continued)

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
9.0	Auxiliary Systems			
	Chemical and Volume Control System	Section 9.3.4	D. C. Cook Units 1 and 2, Trojan	The Sequoyah Nuclear Plant does not have deboration demineralizers.
11.0	Radioactive Waste Management			
	Source Terms	Section 11.1	D. C. Cook Units 1 and 2, Trojan	Differences are based upon plant operational influences.
	Liquid Waste Processing	Section 11.2	Performance characteristics similar to D. C. Cook Units 1 and 2, Trojan	The Sequoyah Nuclear Plant has a dissimilar segregated liquid drain system.
	Gaseous Waste Processing	Section 11.3	D. C. Cook Units 1 and 2 Trojan	None.
	Process Radiation Monitoring	Section 11.4	Functionally similar to D. C. Cook Units 1 and 2, Trojan	None.
15.0	Accident Analysis	Chapter 15	Similar to D. C. Cook Units 1 and 2, Trojan	The Accident Analysis sections have been updated. New sections have been added, e.g., single RCCA withdrawal, accidental depressurization of the RCS, computer code descriptions, etc.

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TABLE 1.3-2

DESIGN COMPARISON - SECONDARY CYCLE

Feature	Referenced FSAR Section	Sequoyah Nuclear Plant	Diablo Canyon	D. C. Cook	Zion
Turbine Generator					
Net Generator Output (kW)	10.1, 10.2	1,183,192	*1,026,000; **1,122,000	1,100,000	1,050,000
Turbine Cycle Heat Rate (Btu/kW-Hr)	10.1	9,871	*10,075; 10,033	*10,208; **10,232	***
Type/LSB Length	10.2	TC6F/44	TC6F/44	*TC6F/43; **TC6F/52	TC6F/44
Cylinders (No.)	10.2	1 H.P. - 3 L.P.	1 H.P. - 3 L.P.	1 H.P. - 3 L.P.	1 H.P. - 3 L.P.
Steam Conditions at Throttle Valve					
Flow (lb/hr)	10.2	14,254,200	*13,934,600; **14,239,300	14,120,000	13,989,300
Pressure (psia)	10.2	832	725	728	690
Temperature (°F)	10.2	522.7	507	507.5	501.5
Moisture Content (%)	10.1, 10.2	0.34	*.65; **0.53	NA	.25
Turbine Cycle Arrangement					
Steam Reheat Stages (No.)	10.1	2	2	1	1
Feedwater Heating Stages (No.)	10.1, 10.4.7, 10.4.9	7	6	6	6
Strings of Feedwater Heaters (No.)	10.1, 10.4.7, 10.4.9	3	3	3 Lowest Pressure, 2 All Others	3
Heaters in Condenser Neck (No.)		3		0	1
Heater Drain System (Type)	10.4.9	All Drains Pumped Forward	High Pressure Pumped Forward Low Pressure Cascaded	High Pressure Pumped Forward Low Pressure Cascaded	High Pressure Pumped Forward Low Pressure Cascaded
Condensate Pumps (No.)	10.1, 10.4.7	3	3	3	4
Condensate Booster Pumps (No.)	10.1, 10.4.7	3	3	3	4
Heater Drain Pumps (No.)	10.1, 10.4.9	3 H.P. - 3 L.P.	3	3	3
Main Feed Pumps (No. and Type)	10.1	2 - Turbine Driven	2 - Turbine Driven	2 - Turbine Driven	2 - Turbine Driven
Main Steam Bypass Capacity (%)	10.4.4	40%	40%	85%	40%
Final Feedwater Temperature		434.3	*432.[illegible], **432.9	*434.8; **430.5	NA
Condenser					
Type	10.1, 10.4.1	Single Pressure	Single Pressure	Single Pressure	Single Pressure
Number of Shells	10.1, 10.4.1	3	2	3	3
Design Back Pressure (In. Hg Abs)	10.1, 10.4.1	2	1.5	*1.71; **1.41	1.5
Total Condenser Duty (Btu/Hr)	10.1, 10.4.1	7.829×10^9	7.6×10^9 (Approx)	2.5×10^9 (Approx)	7.18×10^9 (Approx)

* Unit 1.
** Unit 2.
*** Commonwealth Edison will not release these heat rates.

A-56

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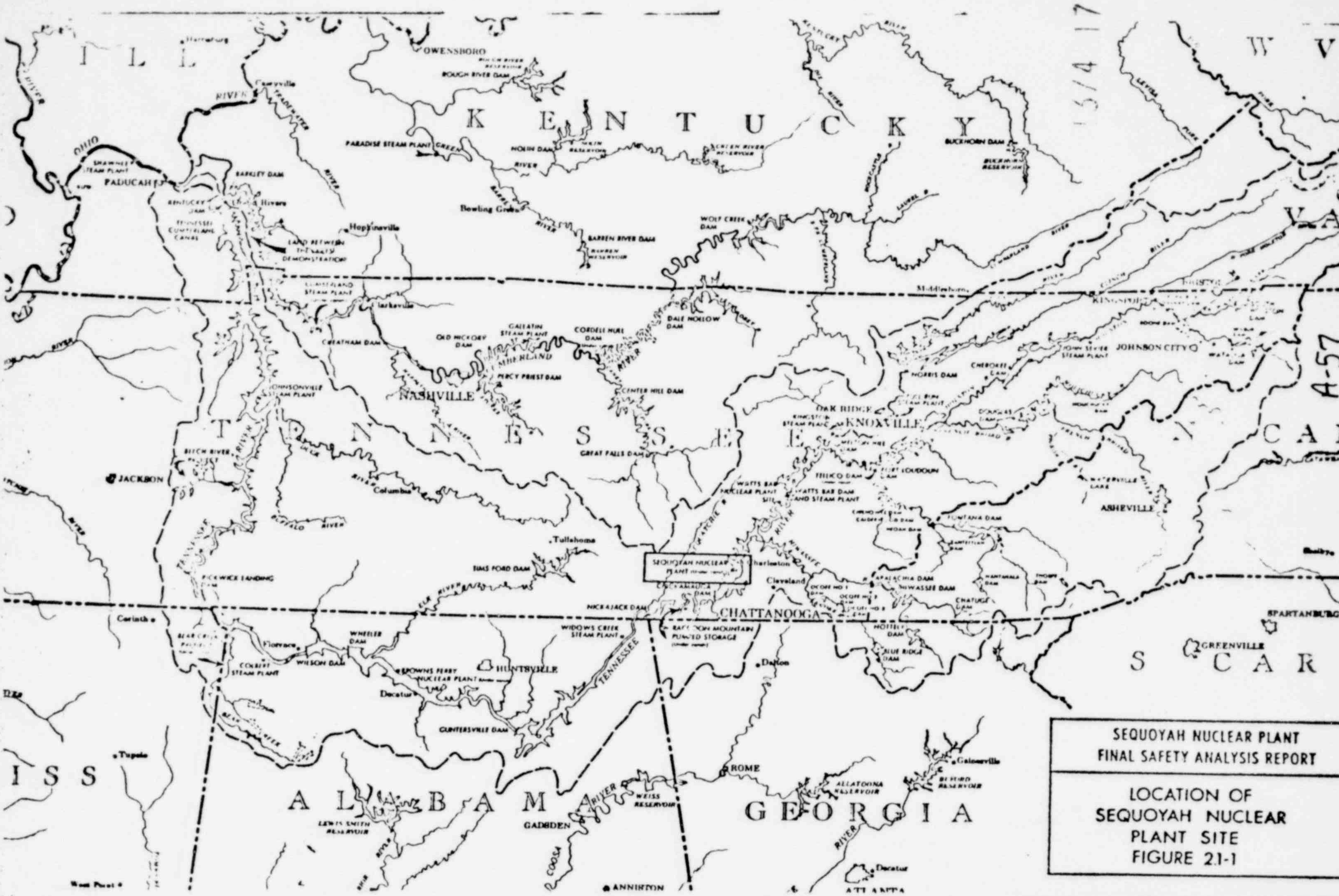
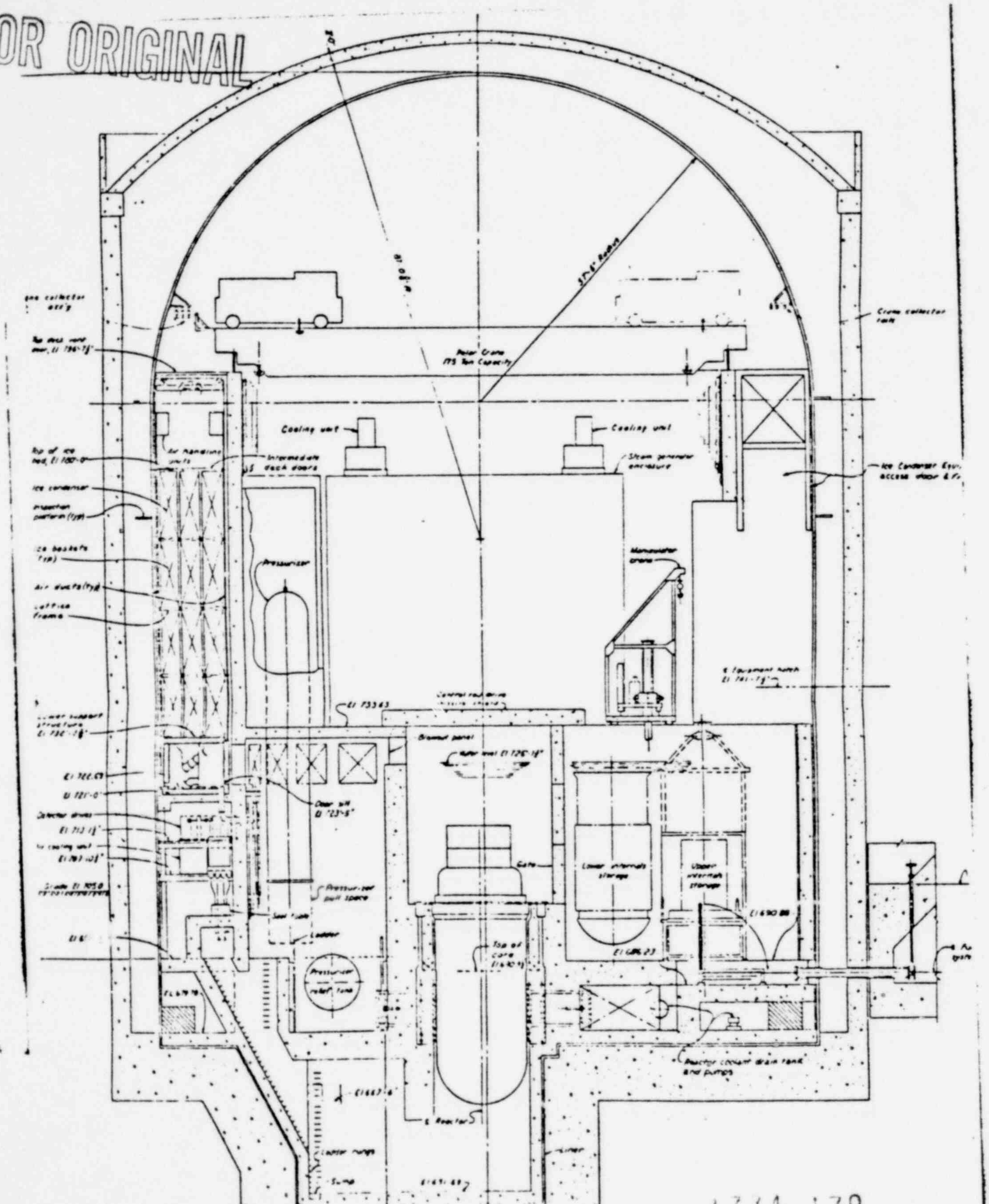


SEQUOYAH NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

LOCATION OF
SEQUOYAH NUCLEAR
PLANT SITE
FIGURE 2.1-1

POOR ORIGINAL

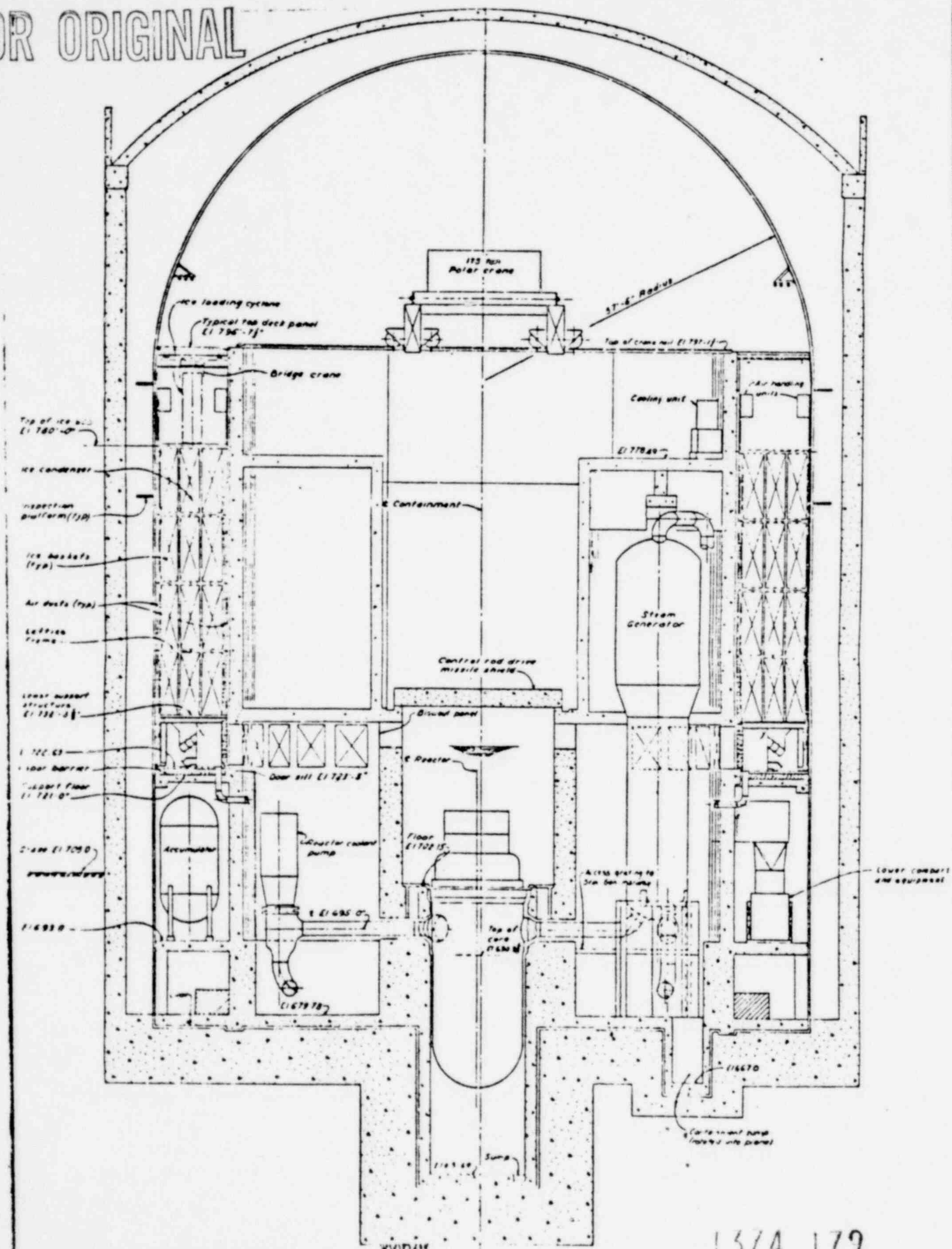


1374 178

SECTION D12-D12

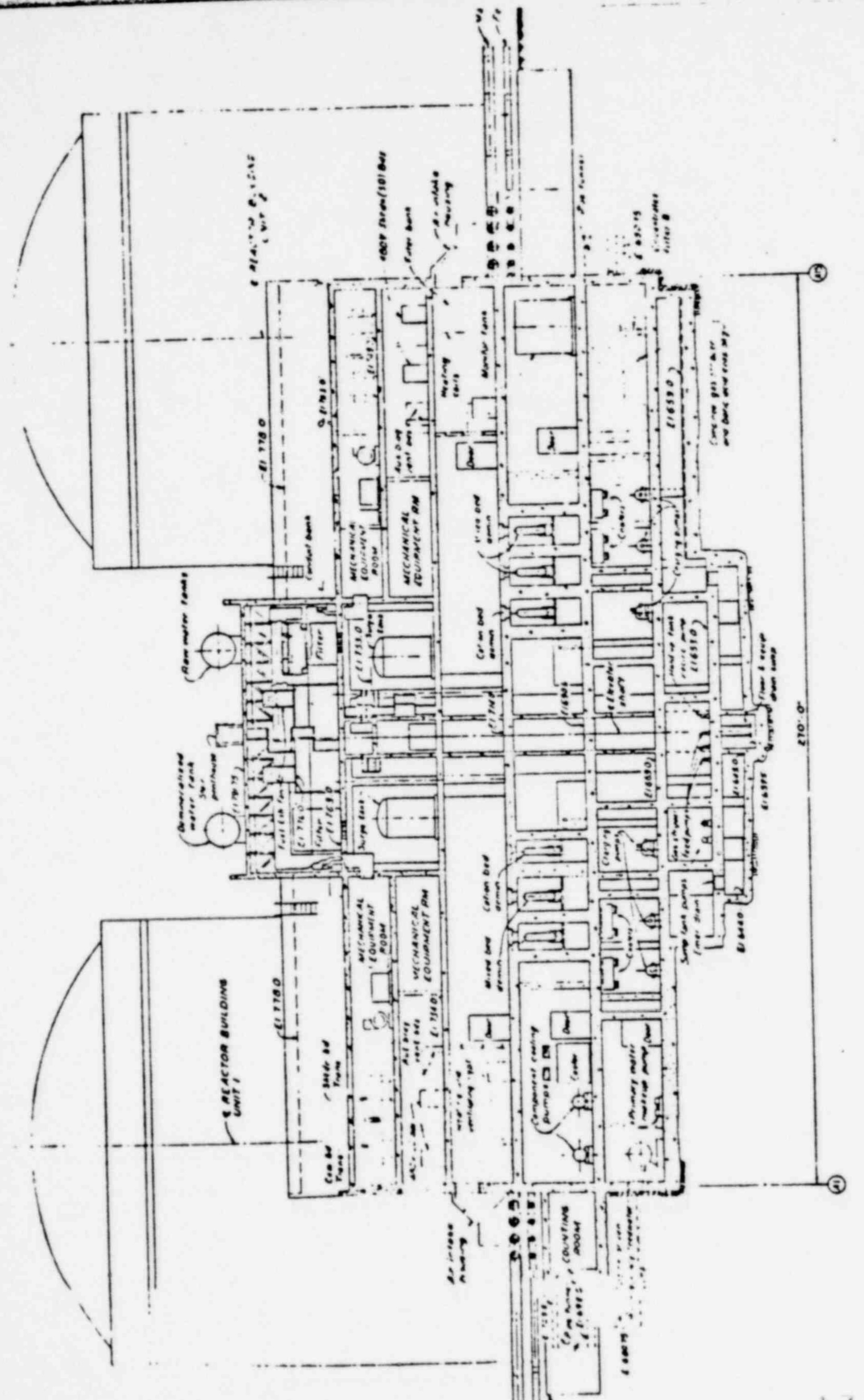
A-58

POOR ORIGINAL



SECTION E13-E13

1374 179



POOR ORIGINAL

1374 180

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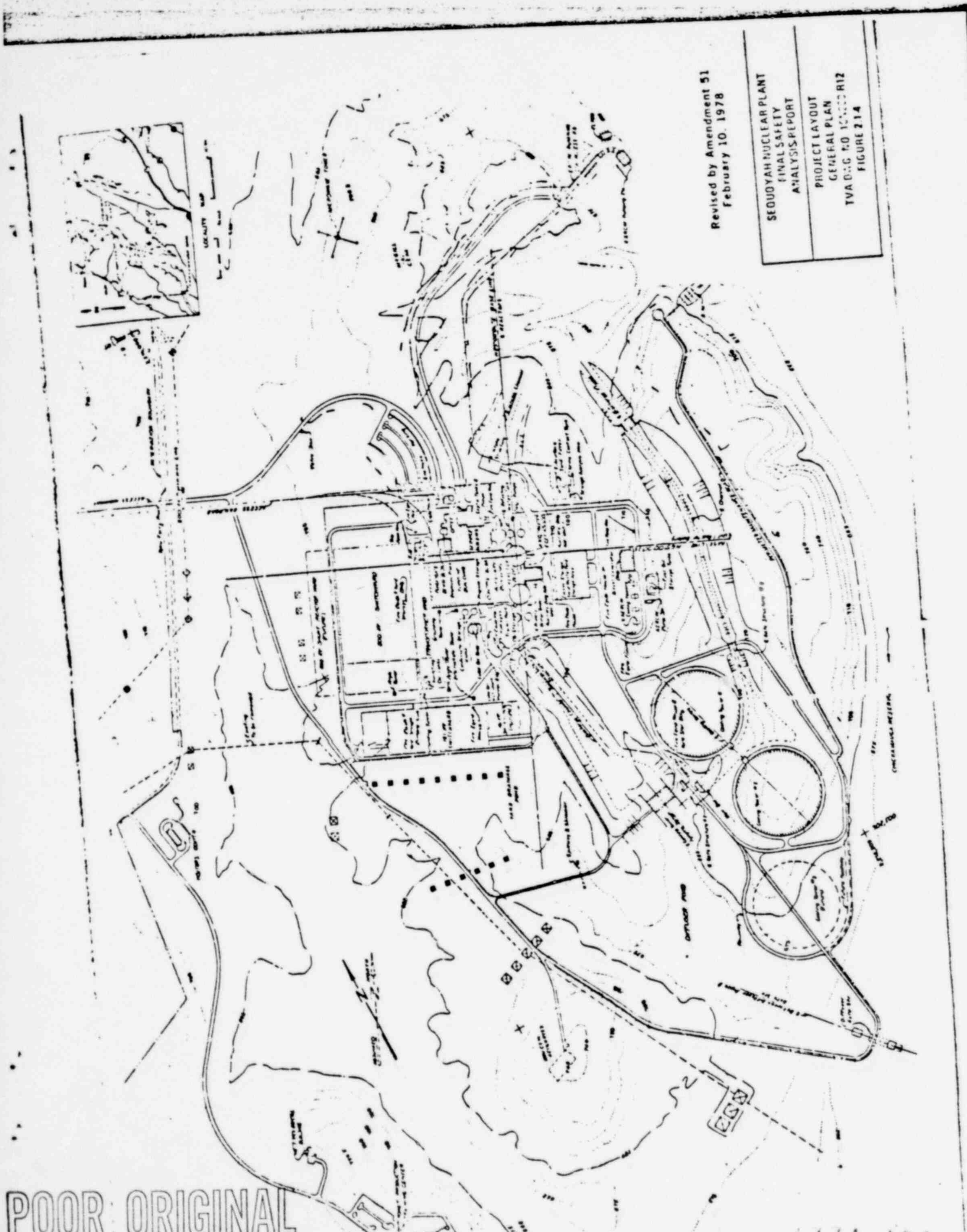
Revised by Amendment 51
February 10, 1978
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT
PROJECT LAYOUT
GENERAL PLAN
R12
FIGURE 2.14

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POOR ORIGINAL

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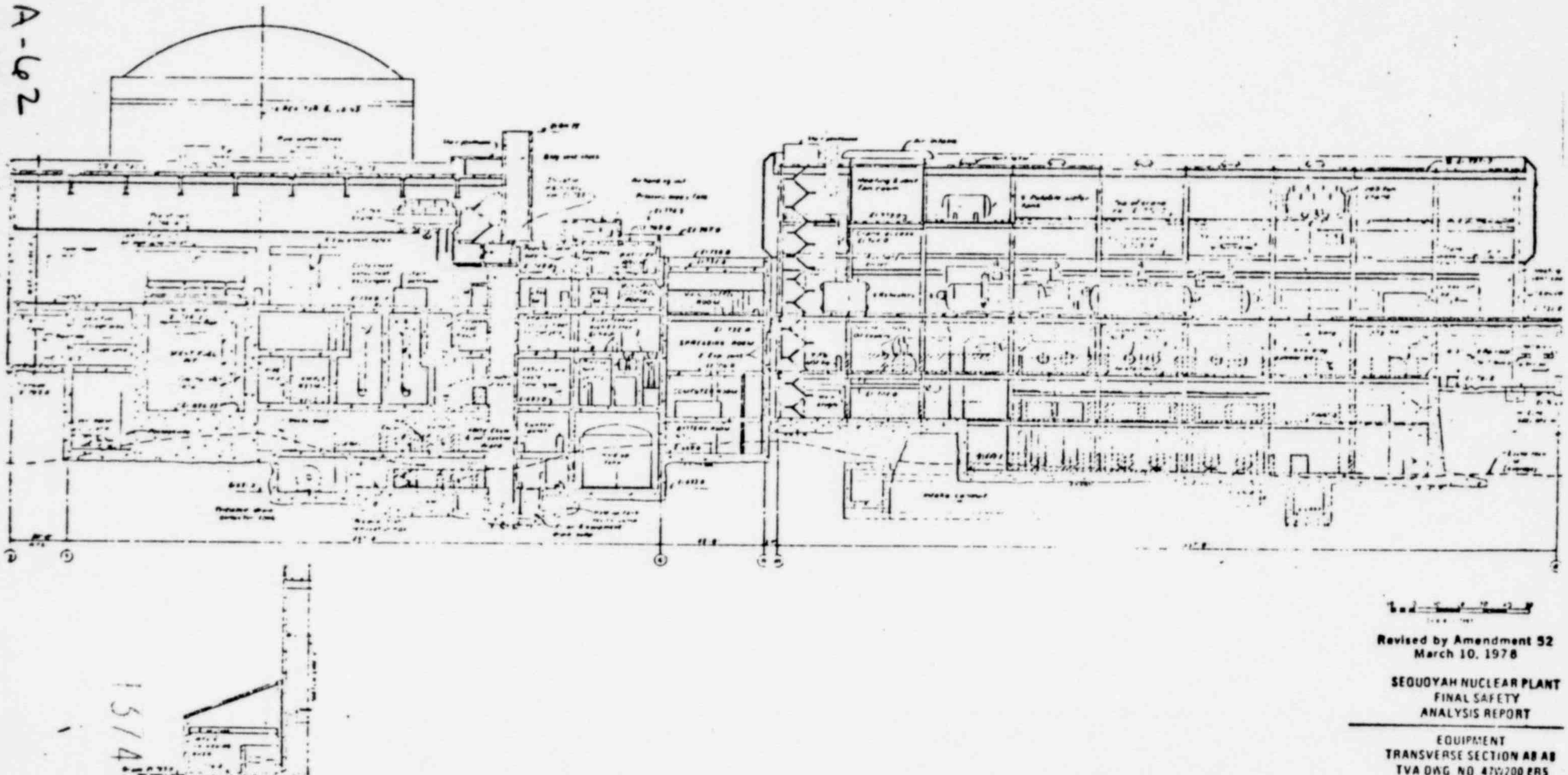


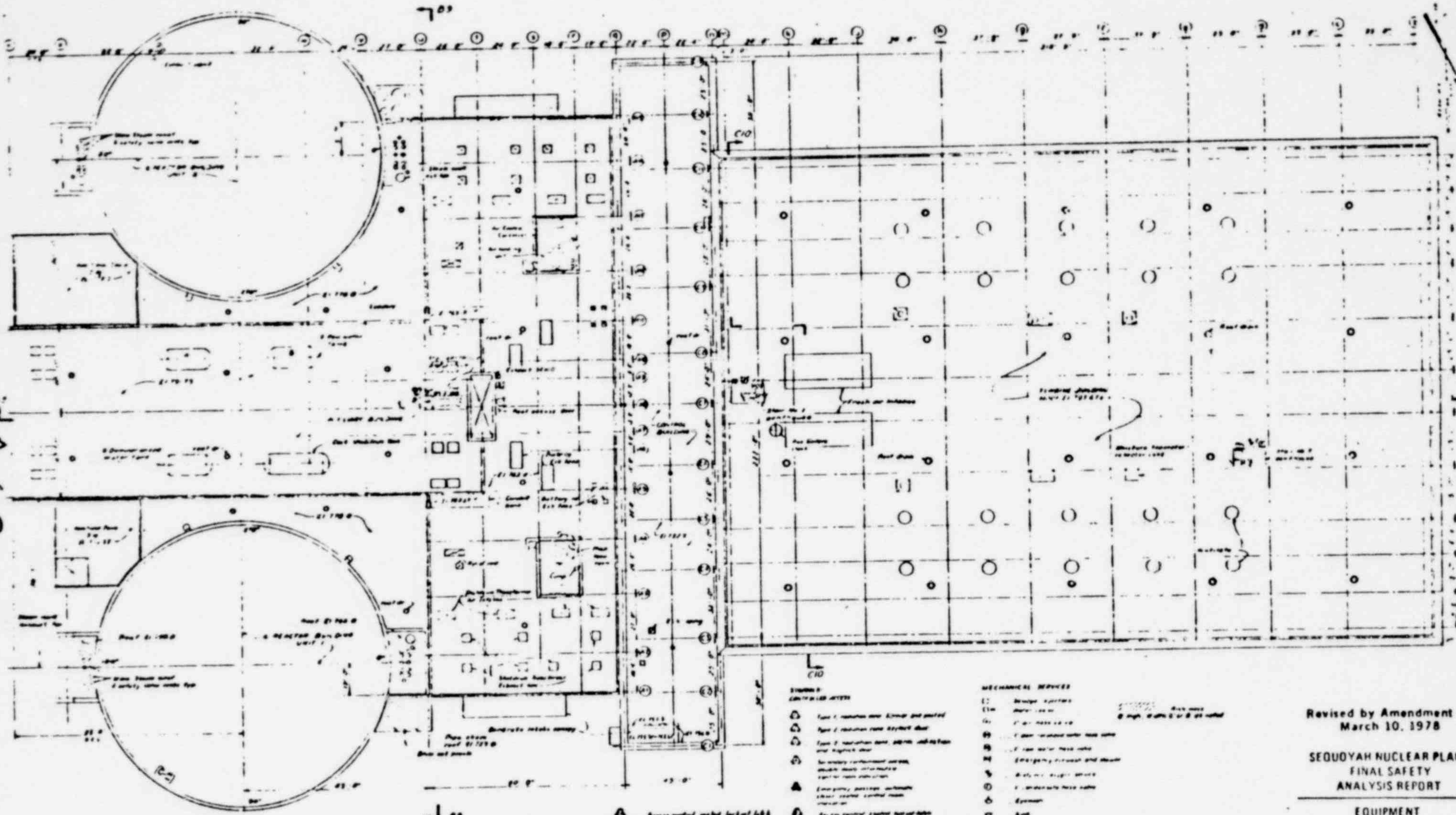
Revised by Amendment 52
March 10, 1978

SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

EQUIPMENT
TRANSVERSE SECTION AB AB
TVA DWG NO 47W200 ERS
FIGURE 1.2-8

1574 182

POOR ORIGINAL



Revised by Amendment 5
March 10, 1978

SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

EQUIPMENT
PLANS – ROOF
TVA DWG NO 47W200 1R7
FIGURE 1.2-1

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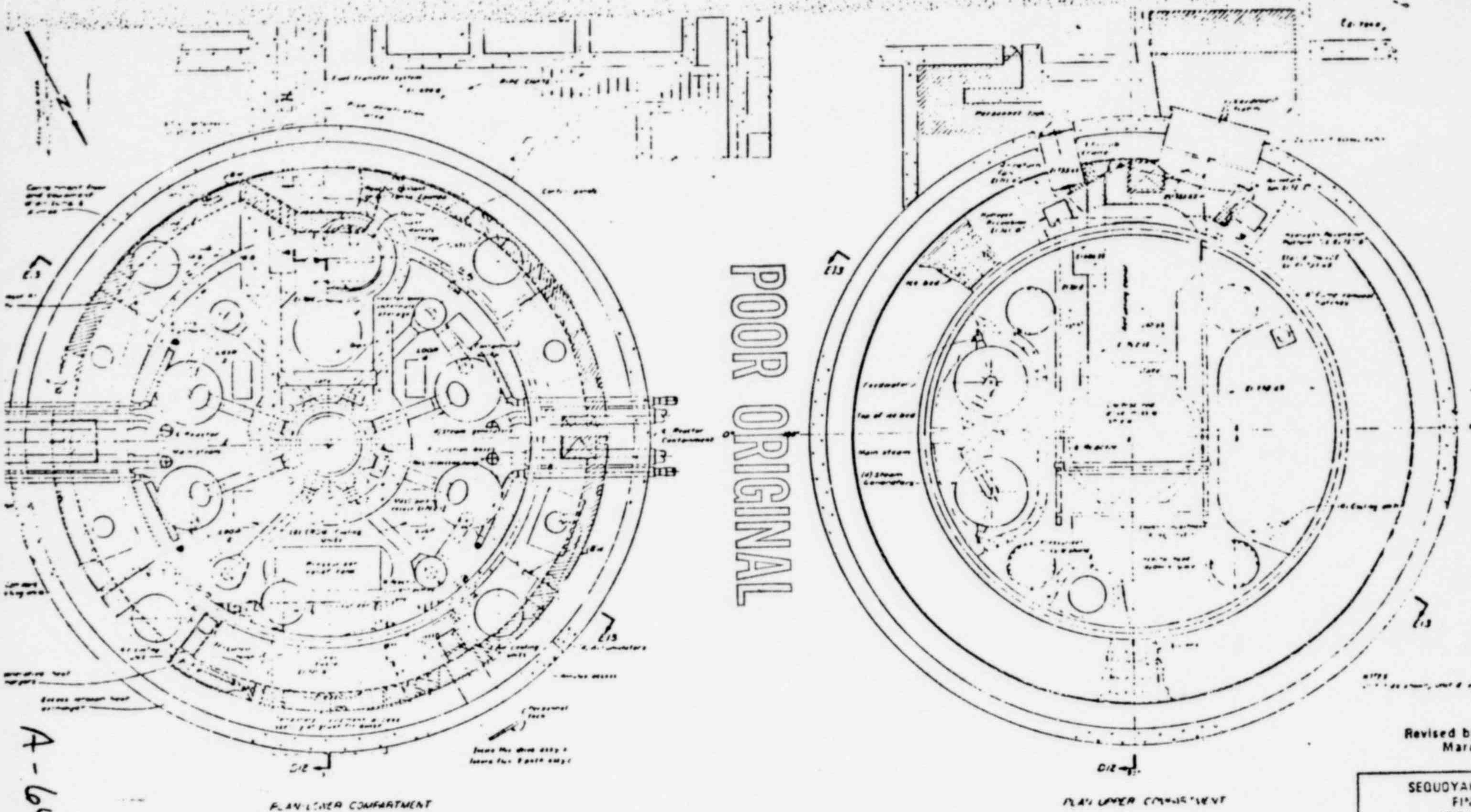

Revised by A
March

SEQUOYAH N
FINAL
ANALYS

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for these other facilities have been published and are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C.

1.4 Identification of Agents and Contractors

The Westinghouse Electric Corporation (Westinghouse) is supplying the nuclear steam supply system, including the first fuel loading, and the turbine-generators. TVA is responsible for the design of the remainder of the plant, and all other architect-engineer functions, and for the construction and operation of the plant.

Principal consultants utilized by TVA to perform selected design work and other specialized services include Western Geophysical Engineering, Inc. for soil foundation dynamic analyses, Engineering Data Systems, Inc. for seismic analysis of piping, Chicago Bridge and Iron Company for design and construction of the free-standing steel containments, and Pressay Corporation for certification of material for containment flexible seals.

1.5 Summary of Principal Review Matters

The evaluation performed by the staff included a review of the information submitted by the applicant, particularly with regard to the following matters:

We evaluated the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology, to establish that these characteristics had been determined adequately and had been given appropriate consideration in the final design of the plant, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facility, including the engineered safety features provided.

We evaluated the design, fabrication, construction, and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate rules, codes, and standards, and that any departure from these criteria, codes, and standards has been identified and justified.

We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. Conservative analyses were performed of these design basis accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

Appendix XII

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H Sebrey

We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operations, including the proposed organization, staffing and training program, the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the plant.

We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations, and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as reasonably achievable.

We will evaluate the financial position of the applicant to determine that the applicant is financially qualified to operate the Sequoyah Nuclear Plant, and will report on this matter in a supplement to this Safety Evaluation Report.

1.6 Outstanding Issues

We have identified outstanding issues in our review which have not been resolved with the applicant. We will complete our review of these items prior to issuance of an operating license, and will discuss the resolution of each of these items in a supplement to this report. These items are listed below and are discussed further in the sections of this report as indicated.

1. Bolted Connections in Component Supports (Section 3.9.2)

   The applicant has not yet furnished requested information on bolted connections in linear component supports in safety-related systems regarding support plate flexibility considerations in determining maximum bolt loads. We will report on our evaluation of this matter when the information is available.

2. Seismic Qualification of Instrumentation and Electrical Equipment (Sections 3.10, 7.2.2, 7.8.1)

   We have not yet completed our review of the Westinghouse-supplied Class 1E instrumentation and electrical equipment. For balance of plant equipment, confirmatory information is required on containment isolation valve motor operators. We will report further on this matter in a supplement to this report.

   BoP resolved, Small issues NSSS

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3. Fire Protection (Section 9.5)

We have not yet completed our review of the applicants fire protection program. We will complete this review prior to issuance of an operating license and will condition the operating license to assure implementation of all required modifications. We will report further in a supplement to this report.

4. Radiological Emergency Plan (Section 13.3)

The applicant has not yet provided responses to our request for additional information on this matter. All issues will be resolved prior to issuance of an operating license, and we will report further in a supplement to this report.

5. Acceptance Criteria for Plant Trip Test (Section 14.0) Resolved

The applicant has not yet provided information we requested on acceptance criteria for the turbine trip and generator load reject portions of the plant trip test from 100 percent power. We will report further in a supplement to this report.

1.7 Confirmatory Issues

As a result of our review, there are a number of matters for which we have completed our review and have determined positions which are acceptable to the staff and for which there appears to be no significant disagreement between the applicant and the staff. The applicant has been advised of our positions and we are awaiting confirmation of the applicant's commitment to comply with these positions and to provide appropriate information. These items will be reported in a supplement to the Safety Evaluation Report. These items, with appropriate references to subsections of this report, are stated below.

1. Single Failure in the Residual Heat Removal System (Section 5.3.2) Resolved

The applicant has not yet provided formal documentation of its agreement to provide a dedicated operator to monitor flow to the residual heat removal pumps during decay heat removal operations, pending installation of a flow alarm (See section 1.8 below).

2. Pressure-Temperature Limits for Heatup and Cooldown (Section 5.2.3) Resolved

The applicant has not yet provided confirmation of its statement that the proposed pressure-temperature limits for reactor vessel heatup and cooldown use an acceptable prediction for temperature shift.

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3. Inservice Inspection of Steam Generator Tubes (Section 5.2.6) Resolved

The applicant has not yet provided formal documentation of an inservice inspection program for the steam generator tubes. We will verify that an acceptable program is in place, and will report further on this matter in a supplement to this report.

4. Cold Shutdown Using Safety-Grade Equipment (Section 5.3.2)

The applicant has discussed with us the capability of the system to achieve cold shutdown using only safety-grade equipment and will provide appropriate confirmatory documentation. We will report further in a supplement to this report.

5. Design of Steam Generator and Pressurizer Supports (Sections 3.9.1, 6.2) Resolved

The applicant has not yet confirmed the assumption that, as in other plants, the pressure response to line breaks in the steam generator and pressurizer subcompartments has been utilized in evaluating the design of the equipment supports. We will report further in a supplement to this report.

6. Containment Response to Steam Line Break and Environmental Qualification of Westighouse Equipment (Sections 6.2.1, 7.2.2, 7.8.2)

Westinghouse has indicated that the containment temperature response to the small line break already analyzed will bound the response for the additional breaks we have requested be examined, but the applicant has not yet provided confirmatory information. Additional information is also forthcoming on environmental qualification of Westinghouse equipment. We will report further in a supplement to this report.

7. Upper Head Injection Preoperational Tests (Section 6.3.4)

The applicant has not yet submitted confirmatory documentation on tests already performed which reportedly demonstrated acceptable flow performance of the upper head injection system. We will report further in a supplement to this report.

8. Containment Sump (Section 6.3.4)

In fulfillment of the applicable requirements of Regulatory Guide 1.79, the applicant has performed scale model tests of the containment emergency sump performance and submitted reports which we have reviewed. The applicant has not yet responded formally to our requests for additional information to verify sump performance in the event of certain line breaks. We will report fully on these matters in a supplement to this report.

9. Bypassed Safety Injection Signal (Section 6.3.5) Resolved.

The applicant has indicated that sufficient time is available to respond effectively to postulated line breaks in the residual heat removal system when in the normal shutdown cooling mode when the safety injection signal is blocked, but has not yet provided information verifying actions required and time available. We will report further on this matter in a supplement to this report.

10. Loss-of-Coolant Accident Analysis (Sections 6.3.5, 15.3.2)

We have reviewed the loss-of-coolant accident analysis provided by the applicant and have requested information confirming that the most limiting case has been analyzed. We will report further in a supplement to this report.

11. Response Time Testing (Section 7.2.2)

The applicant has committed to measure channel response time including the sensors, but has not yet submitted the confirmatory information requested to assure acceptable implementation of this commitment.

12. Isolation Valve Interlocks and Position Indication (Section 7.3.2)

The applicant has not yet submitted documentation to confirm verbal information that position indication of two safety-related valves will be maintained when power is removed from the valves.

13. Post Accident Monitoring Separation Criteria (Section 7.5.2)

The applicant has not yet provided information varifying implementation of agreed criteria for separation and independence of post-accident monitoring channels.

14. Environmental Qualification of Balance-of-plant Equipment (Section 7.8.2)

The applicant has not yet provided confirmatory information on an environmental monitoring system or on the correction of errors in several tables in the Final Safety Analysis Report.

15. Diesel Generator and Remote Shutdown Testing (Section 14.0) Resolved

We require that the applicant perform tests in accordance with regulatory guides covering diesel generators and remote shutdown capability, or provide justification for exceptions to these guides. Confirmatory information has not yet been provided by the applicant. We will report further on this matter in a supplement to this report.

16. Boron Dilution (Section 15.2)

The applicant has not yet provided documentation confirming his procedures associated with alarm setpoints for the high flux alarm which provides protection against a boron dilution event during startup or shutdown.

17. Long Term Effects of Steam Line Break (Section 15.33)

The applicant has not yet provided information requested to verify operator actions related to long-term reactor vessel repressurization.

1.8 Staff Positions - Licensing Conditions

The staff has taken positions on certain issues requiring implementation and/or documentation after issuance of an operating license. The license will be conditioned as necessary to assure acceptable implementation of our positions. These items are listed below and are discussed further in the sections of this report as indicated.

1. Seismic Design of Structures and Components (Section 2.5) Resolved

The operating license will be conditioned to require evaluations showing margins available in structures and components to function during and after a design earthquake.

2. Inservice Testing After Commercial Operation (Section 3.9.1)

The operating license will be conditioned to assure implementation of an acceptable inservice testing program for pumps and valves after commercial operation.

3. Reactor Vessel Overpressurization (Section 5.2.2) Resolved

If equipment is not installed prior to initial fuel load to protect against startup and shutdown overpressurization transients, the operating license will be conditioned as necessary to require installation of such equipment at a later date. The applicant must provide acceptable justification for operation prior to installation of such equipment.

4. Loose Parts Monitor (Section 5.2.8) Resolved

We require installation of an acceptable loose parts monitoring system before initiation of startup testing after the initial fuel loading.

5. Flow Alarm in Residual Heat Removal System (Section 5.3.2)

The operating license will be conditioned to assure installation of a flow alarm to indicate loss of flow in the suction line to the residual heat removal pumps prior to startup following the first refueling outage.

6. Instrument Trip Setpoints (Section 7.2.7)

The operating license will be conditioned to assure receipt of requested information on the determination of instrument trip setpoints.

7. Effect of Power Transients on Safety Related Equipment (Section 7.3.2)

The operating license will be conditioned to require provision of an additional level of under- and over-voltage protection prior to startup following the first refueling outage.

1.9 Generic Issues

The Advisory Committee on Reactor Safeguards periodically issues a report listing various generic matters applicable to light water reactors. A discussion of these matters is provided in Appendix C to this report which includes references to sections of this report for more specific discussions concerning this facility.

The Nuclear Regulatory Commission staff continuously evaluates the safety requirements used in its review against new information as it becomes available. In some cases immediate action or interim measures are taken by the staff to assure safety. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing staff requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of our program for the resolution of these generic issues will be presented in a supplement to ths report.

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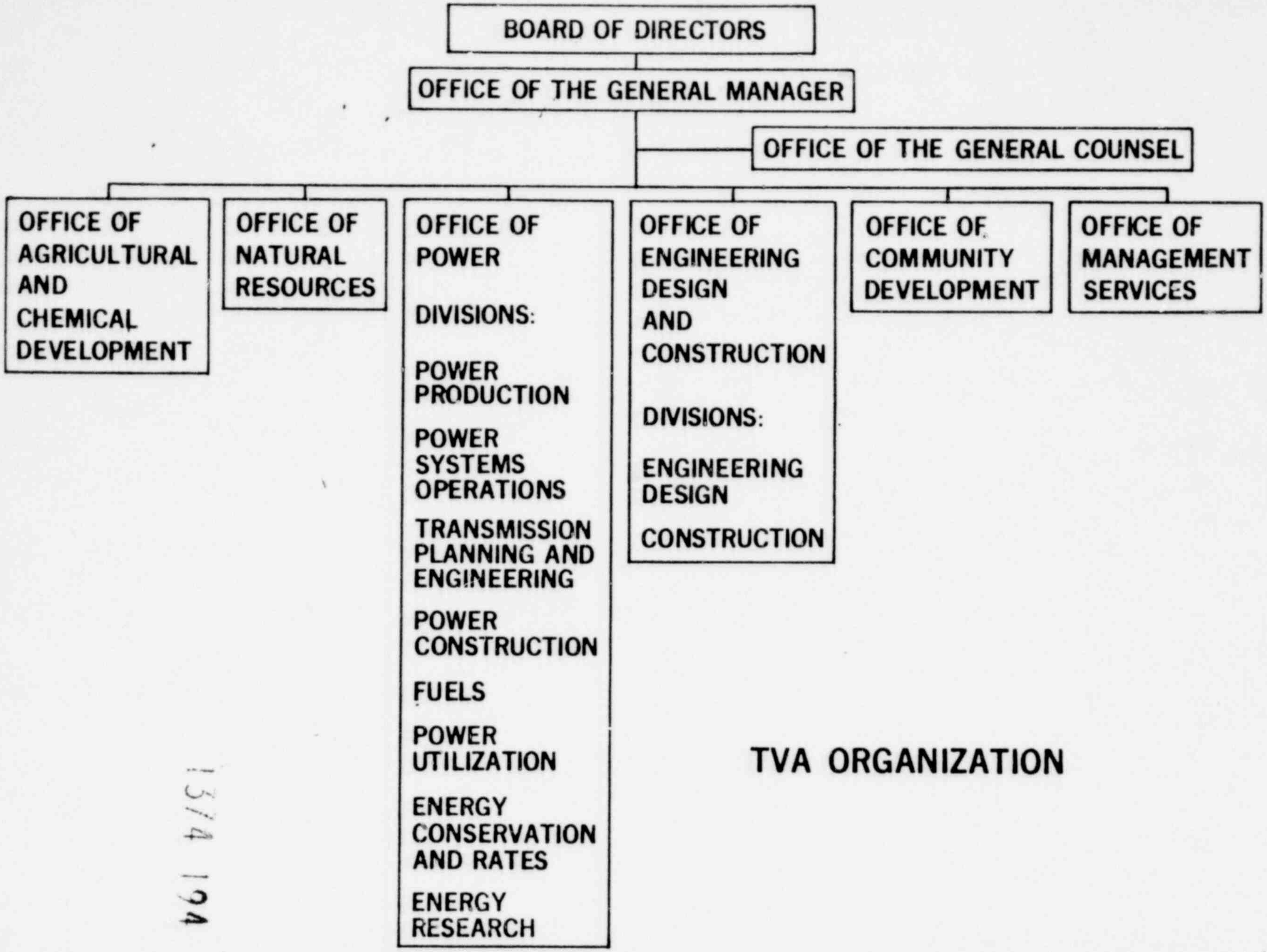
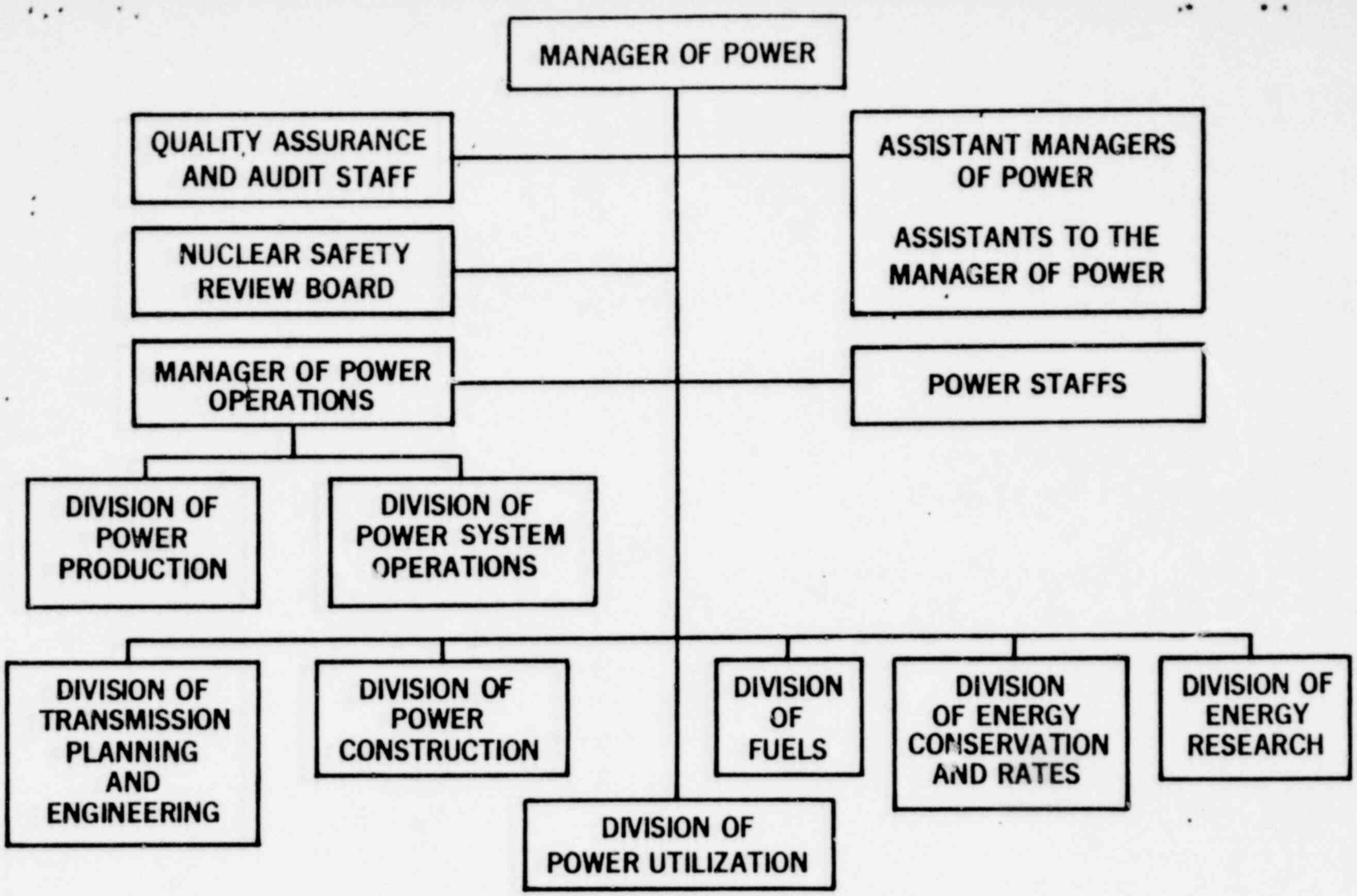
Appendix XIII



TVA ORGANIZATION

1374 194



OFFICE OF POWER ORGANIZATION

EMERGENCY PLAN INTERFACES

TENNESSEE COORDINATION

Tennessee Department of Public Health

Tennessee Office of Civil Defense and Emergency Planning

Tennessee Deaprtment of Public Welfare

Tennessee Department of Safety

Tennessee Department of Conservation

Tennessee Department of Agriculture

Tennessee National Guard

LOCAL COORDINATION

City and county Officials of Hamilton County

Sheriff's Department of Hamilton County

Civil Defense Director - Chattanooga - Hamilton County, Tennessee

Chattanooga Police

Rhea County Ambulance Service

Fire Departments - Chattanooga and Soddy-Daisy

Baroness Erlanger Hospital - Chattanooga

GENERAL SUPPORT COORDINATION

REAC/TS FAcility at Oak Ridge Hospital of the United Methodist Church

National Aeronautics and Space Administration - Huntsville, Alabama

U.S. Department of Energy - Oak Ridge, Tennessee

Alabama Department of Public Health 1374 196

Environmental Protection Agency, Region IV, Atlanta

Eastern Environmental Radiation Laboratory - Montgomery, Alabama

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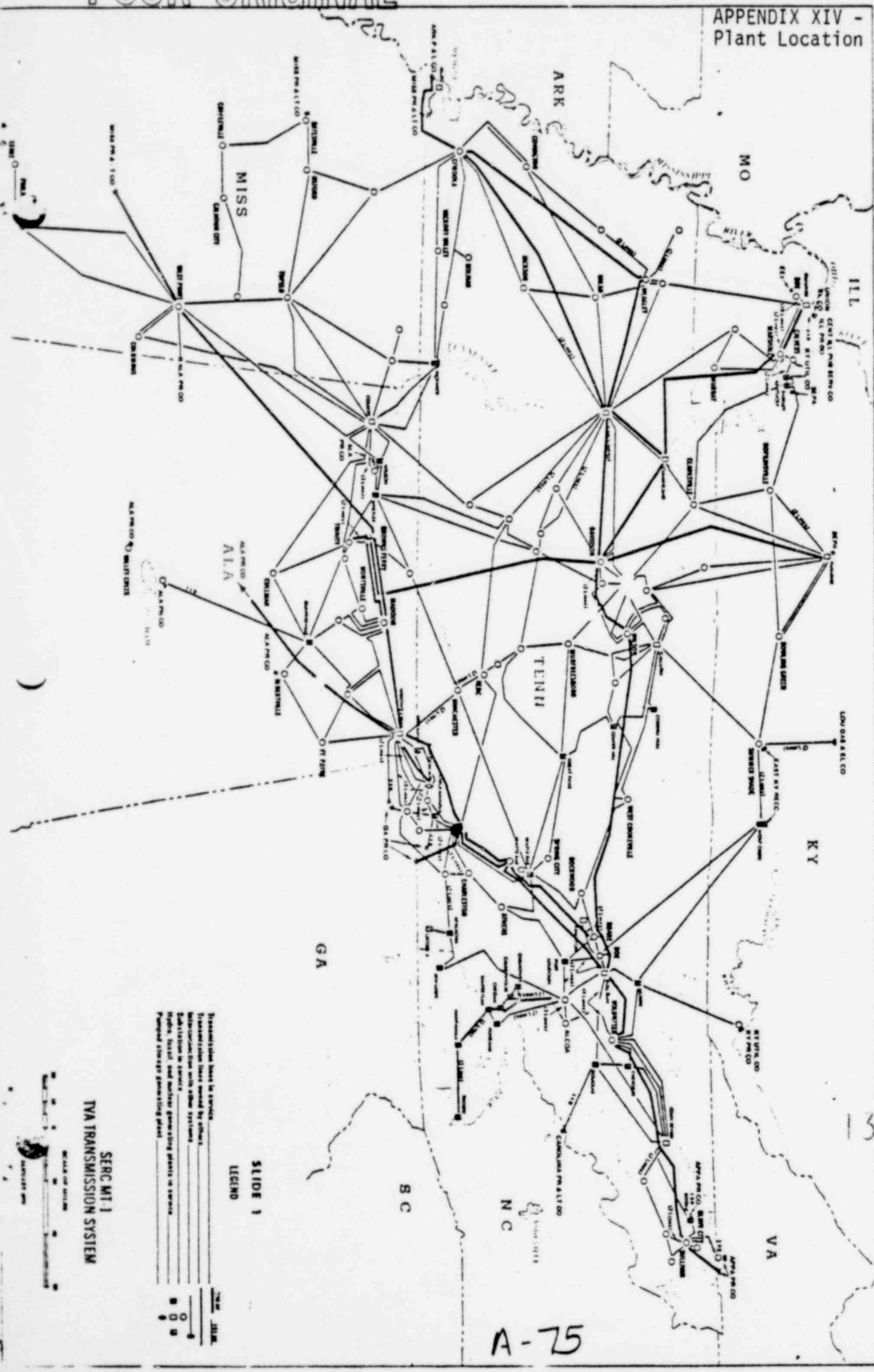
Appendix XIV

APPENDIX XIV - Sequoyah 1 and 2
Plant Location and Site



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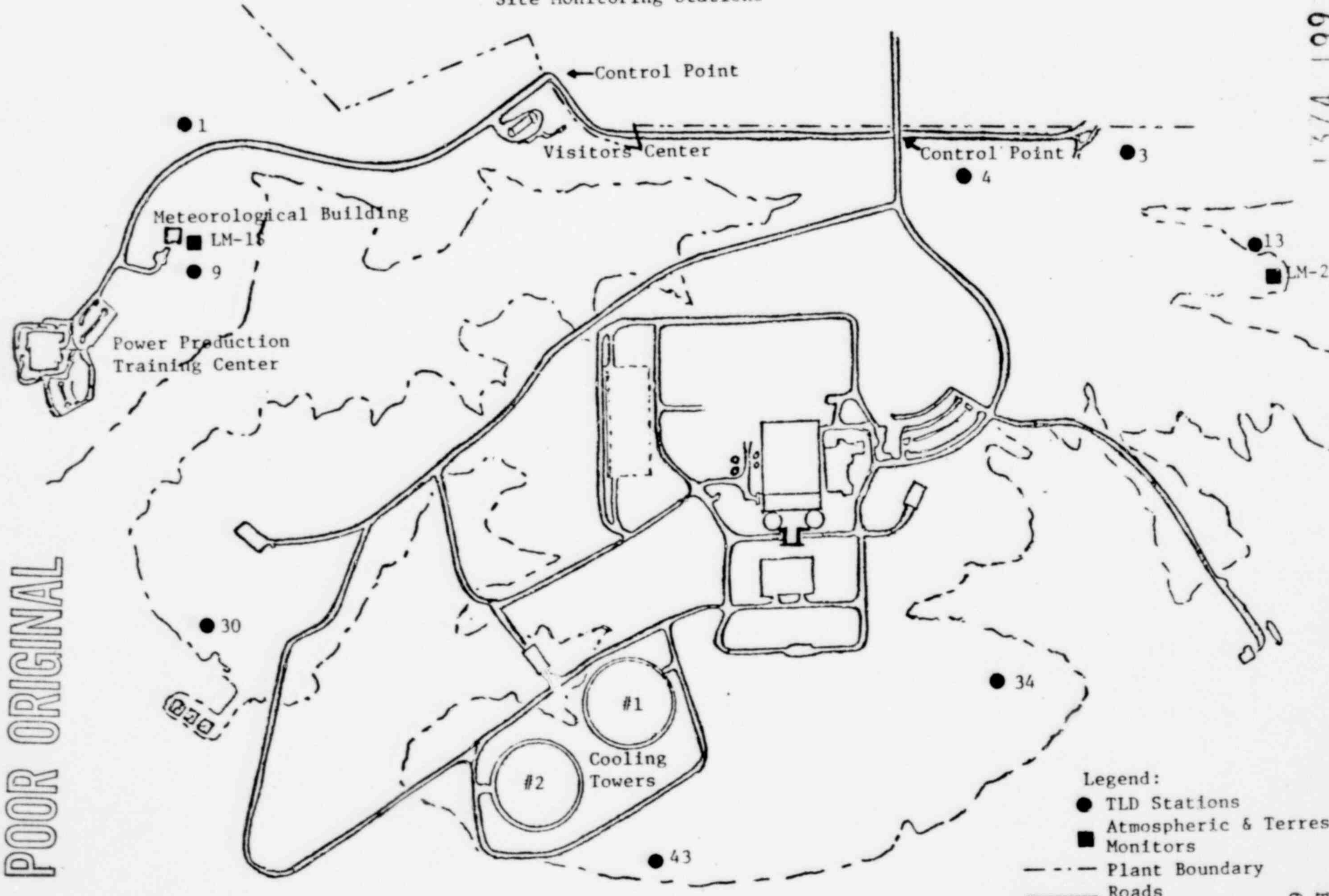
Sequoyah Nuclear Plant
Site Monitoring Stations



SLIDE 2

POOR ORIGINAL

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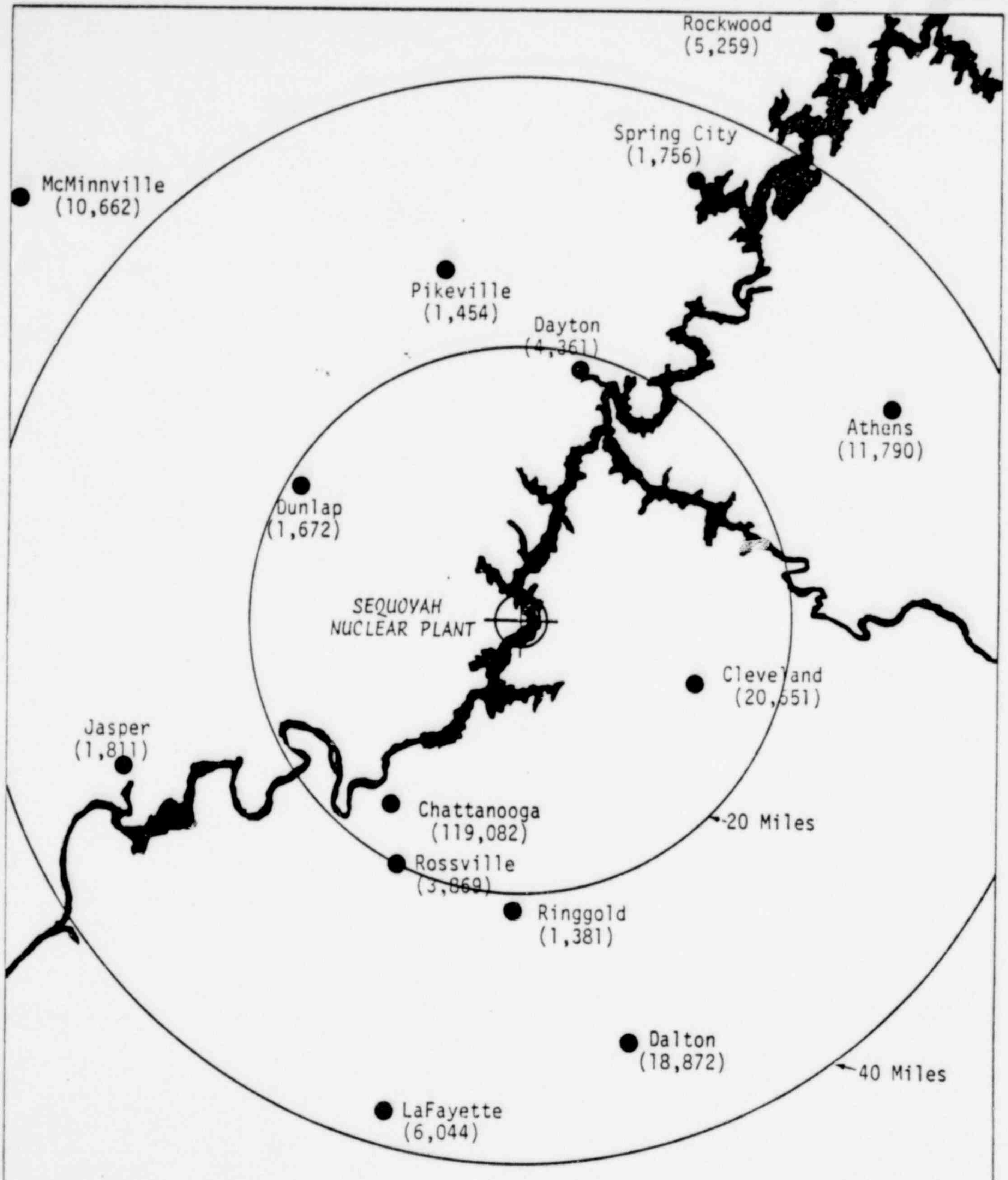
Exhibit C-1

A-76

Page 27
EEP-REP
6/27/77

SLIDE 3 IS A PHOTOGRAPH OF THE SITE

1374 200



POOR ORIGINAL

1374 201

Figure 1.2-8

MAJOR POPULATION AREAS
SEQUOYAH NUCLEAR PLANT

SLIDE 4

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SLIDE 5 IS A PHOTOGRAPH OF THE SITE

SLIDE 6 IS A CUT AWAY PICTURE OF THE PLANT

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Appendix XV

SLIDE 7-1

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

Nuclear Plant Units 1 and 2 - Comparison with McGuire

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
3.0	Containment	Section 3.8.2	McGuire	None
4.0	Reactor			
	Fuel	Section 4.2.1	McGuire	None
	Reactor Vessel Internals	Section 4.2.2	McGuire	Sequoyah Units 1 and 2 have thermal shields. McGuire has neutron pads.
	Reactivity Control	Section 4.2.3	McGuire	None
	Nuclear Design	Section 4.3	McGuire	None
	Thermal-Hydraulic Design	Section 4.4	McGuire	None
5.0	Reactor Coolant System	Sections 5.1, 5.2	McGuire	None
	Reactor Vessel*	Section 5.4	McGuire	None
	Reactor Coolant Pumps*	Section 5.5.1	McGuire	McGuire has higher flow due to impeller change.
	Steam Generators*	Section 5.5.2	McGuire	McGuire coolant volume is smaller.
	Piping*	Section 5.5.3	McGuire	None
	Residual Heat Removal System	Section 5.5.7	McGuire	None
	Pressurizer*	Section 5.5.10	McGuire	None

*All components designed and manufactured to Code edition in effect at date of purchase order.

APPENDIX XV - Sequoyah 1 & 2: Thermal & Hydraulic Design Parameters, Comparison with McGuire

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SLIDE 7-2

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
6.0	Engineered Safety Features			
	Emergency Core Cooling System	Section 6.3	McGuire	None
	Ice Condenser	Section 6.7	McGuire	None
7.0	Instrumentation and Controls			
	Reactor Trip System	Section 7.2	System functions are similar to McGuire.	None
	Engineered Safety Features Systems	Section 7.3	Systems functions are similar to McGuire.	None
	Systems Required for Safe Shutdown	Section 7.4	System functions are similar to McGuire.	None
	Safety Related Display Instrumentation	Section 7.5	Parametric display is similar to that of McGuire	Actual physical configuration may differ due to customer design philosophy.
	Other Safety Systems	Section 7.6	Operational functions are similar to McGuire.	None
	Control Systems	Section 7.7	Operational functions are similar to McGuire.	The Sequoyah Nuclear Plant has has a 50 percent electrical load rejection capability while McGuire has 100 percent.

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SLIDE 7-3

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
8.0	Electric Power			
	Offsite Power	8.2	McGuire	Sequoyah - 2 offsite sources 161 kV/6.9 kV
	Onsite Power	8.3	McGuire	Sequoyah - Tandem diesel generator arrangement

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SLIDE 7-4

DESIGN COMPARISON (EXCLUDING SECONDARY CYCLE)

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
9.0	Auxiliary Systems			
	Condensate Cleanup System	Section 9.3.4	McGuire	Sequoyah had condensate demineralizers backfitted.
11.0	Radioactive Waste Management			
	Source Terms	Section 11.1	McGuire	Differences are based upon plant operational influences.
	Liquid Waste Processing	Section 11.2	Performance characteristics similar to McGuire	None
	Gaseous Waste Processing	Section 11.3	Functionally similar to McGuire	None
15.0	Accident Analysis	Chapter 15	Similar to McGuire	Sequoyah has no untreated leakage paths to the environs.

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SLIDE 8

THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Sequoyah	McGuire
Reactor Core Heat Output, megawatts thermal	3411	3411
System Pressure, Nominal, pounds per square inch		
Minimum Departure from Nucleate Boiling Ratio at Nominal Conditions		
Typical Flow Channel	2.22	2.08
Thimble (Cold Wall) Flow Channel	>1.81	1.74
Thermal Flow Rate, pounds per hour	133.8×10^6	140.3×10^6
Effective Flow Rate for Heat Transfer, pounds per hour	122.8×10^6	134.0×10^6
Effective Core Flow Area, square feet	51.1	51.1 Average
Coolant Temperature		
Nominal Inlet, degrees Fahrenheit	545.7	558.1
Average Rise in Core, degrees Fahrenheit	67.8	62.7
Active Heat Transfer Surface Area, square feet	59,700	59,700
Active Heat Flux, BTU per hour-square foot	189,800	189,800
Maximum Heat Flux, for nominal operation, BTU per hour-square feet	474,500	440,300
Average Thermal Output, Kilowatts per foot	5.44	5.44
Maximum Thermal Output, for normal operation, Kilowatts per foot	12.20	12.60
Heat Flux Hot Channel Factor, F_Q	2.25	2.32
Peak Fuel Central Temperature at 100 percent Power, degrees Fahrenheit	3400	3250

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FUEL MECHANICAL DESIGN COMPARISON

Design Parameter	Sequoyah Units 1 & 2	McGuire Units 1 & 2	Typical Westinghouse Fuel
FUEL ASSEMBLY			
Rod Array	17 x 17	17 x 17	15 x 15
Number of Fueled Rods	264	264	204
Number of Spacer Grids	8	8	7
Number of Guide Thimbles	24	24	20
Inter-rod Pitch, inches	0.496	0.496	0.563
Average Thermal Output (4 Loop), Kilowatts per foot	5.4	5.4	7.0
FUEL PELLETS			
Density (theoretical), percent	95	95	94
Fuel Weight/Unit Length (per rod), pounds per foot	0.364	0.364	0.462
FUEL CLADDING			
Outside Radius, inches	0.187	0.187	0.211
Thickness, inches	0.0225	0.0225	0.0243
Radius/Thickness Ratio	8.31	8.31	8.68

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Appendix XVI

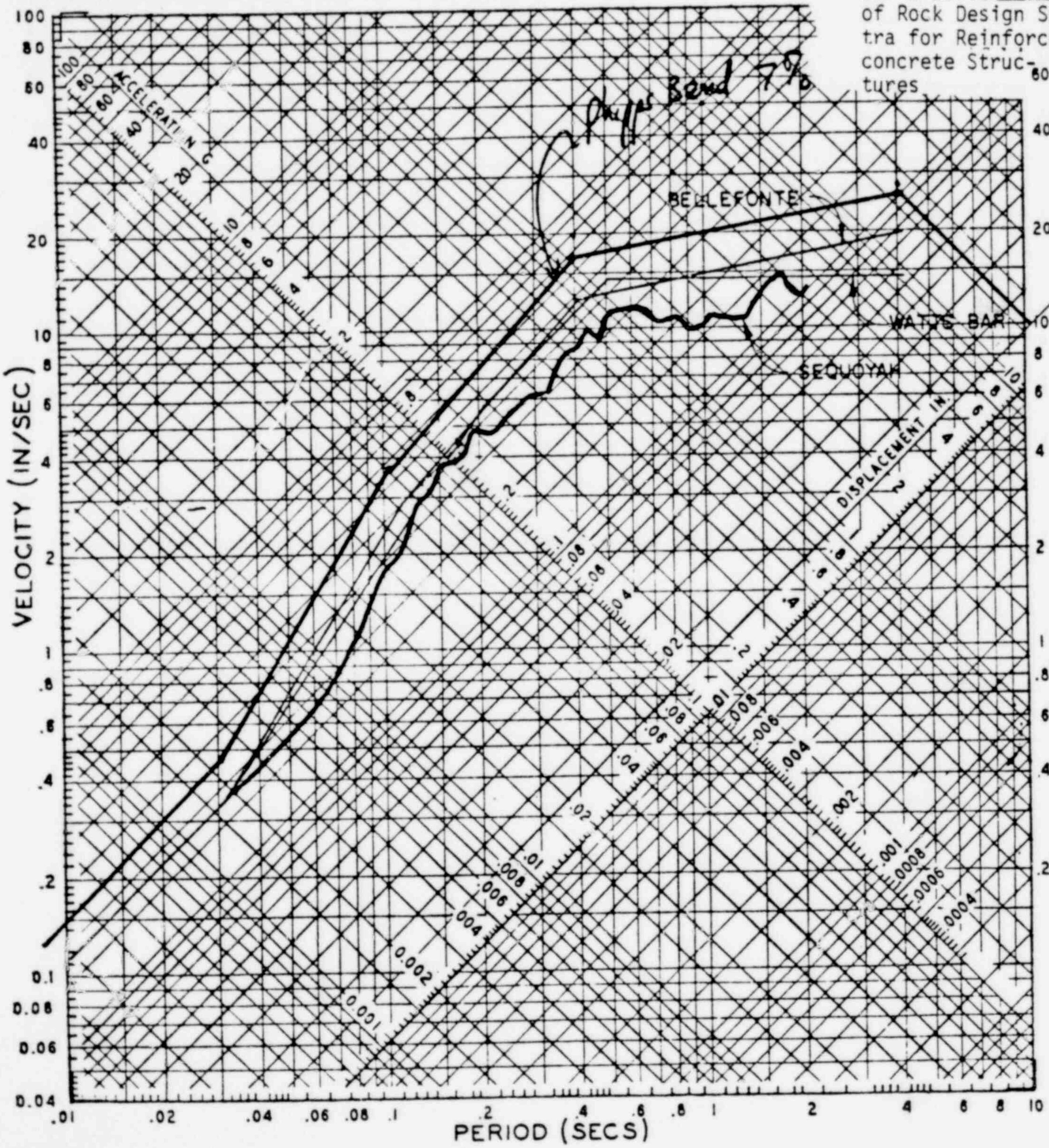
COMPARISON OF SEQUOYAH, WATTS BAR, AND BELLEFONTE NUCLEAR PLANTS TOP OF ROCK DESIGN SPECTRA FOR REINFORCED CONCRETE STRUCTURES

SEQUOYAH - 5% DAMPING

WATTS BAR - 5% DAMPING

BELLEFONTE - 7% DAMPING

APPENDIX XVI: Sequoyah 1 & 2: Comparison w/Watts Bar & Bellefonte Top of Rock Design Spectra for Reinforced concrete Structures



POOR ORIGINAL

A-86 FIGURE 3-8

FIGURE 2

1374 211

AIMS OF REVIEW

(1) MAKING A REALISTIC YET CONSERVATIVE ESTIMATE OF GROUND MOTION FROM THE CONTROLLING EARTHQUAKE.

(2) COMPARING THIS ESTIMATE WITH THE EXISTING SEISMIC DESIGN.

(3) DETERMINING THE SIGNIFICANCE OF ANY DIFFERENCE BETWEEN THE ABOVE.

1374 212

2

PARAMETERS FOR SITE SPECIFIC SPECTRA OF 1897 GILES COUNTY EARTHQUAKE

(1) BODY WAVE AND LOCAL MAGNITUDE RANGE
5.8 ± 0.5 (5.3-6.3)

(2) EPICENTRAL DISTANCE - LESS THAN 25 KILOMETERS

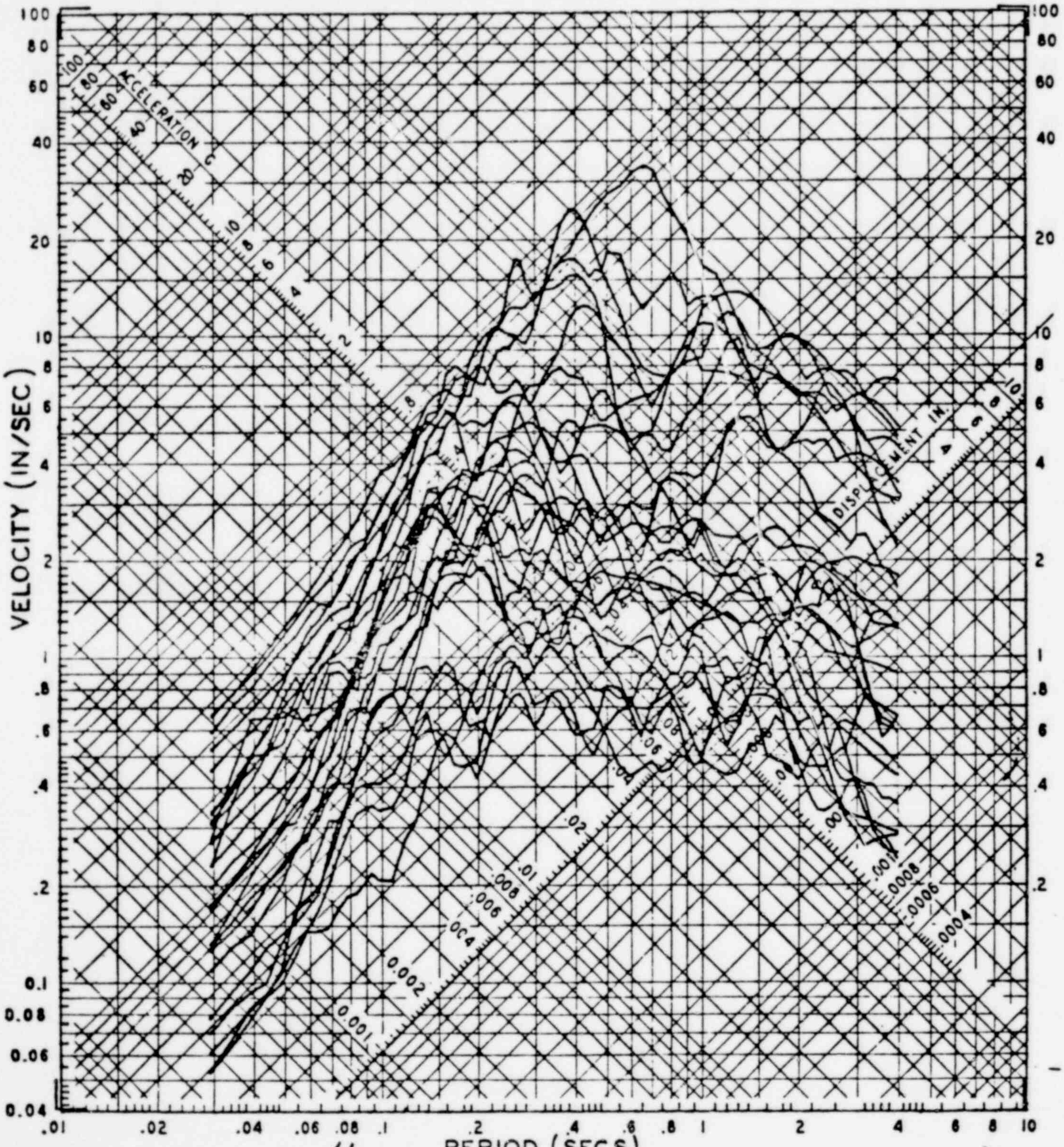
(3) SITE CONDITIONS - ROCK

1374 213

3

OVERPLOT OF RESPONSE SPECTRA FOR THIRTEEN US AND ITALY EARTHQUAKES - 7 % DAMPING

POOR ORIGINAL



4

FIGURE A-32 A-89

1374 214

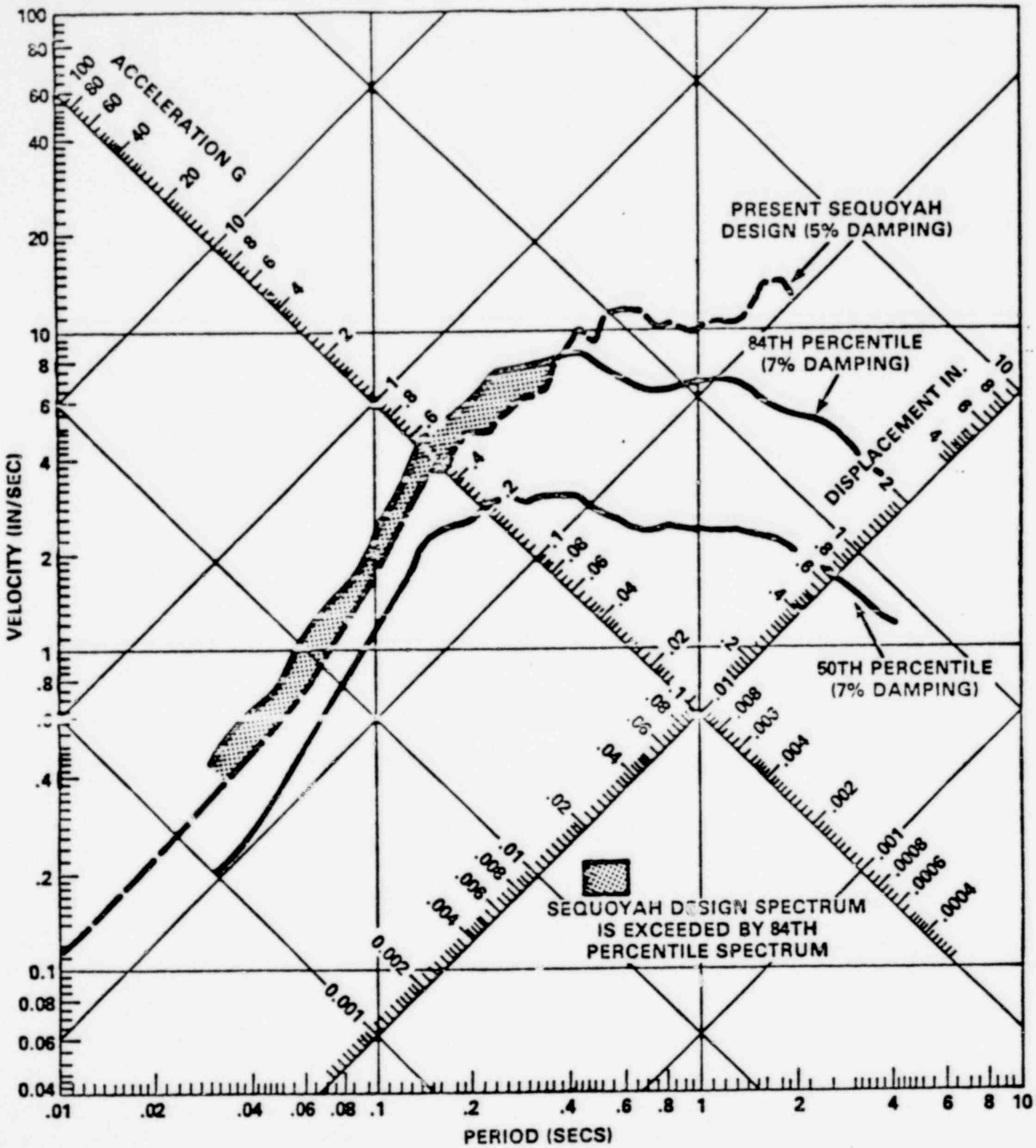


Figure 2-3

Comparison Of The Present Sequoyah Design Spectrum For Reinforced Concrete With Appropriately Damped 50th And 84th Percentile Site Specific Response Spectra.

POOR ORIGINAL

5 A-90

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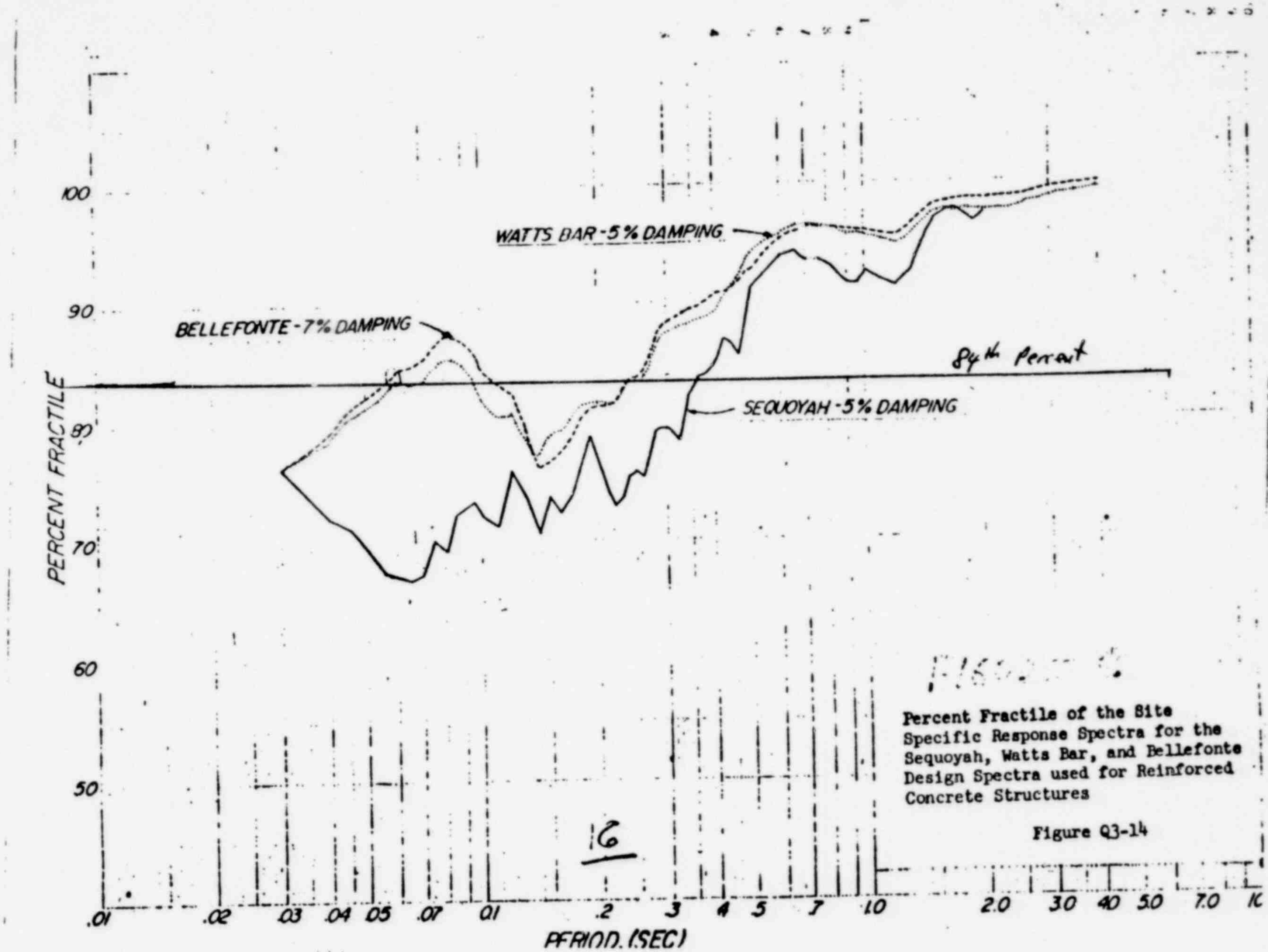
A-91

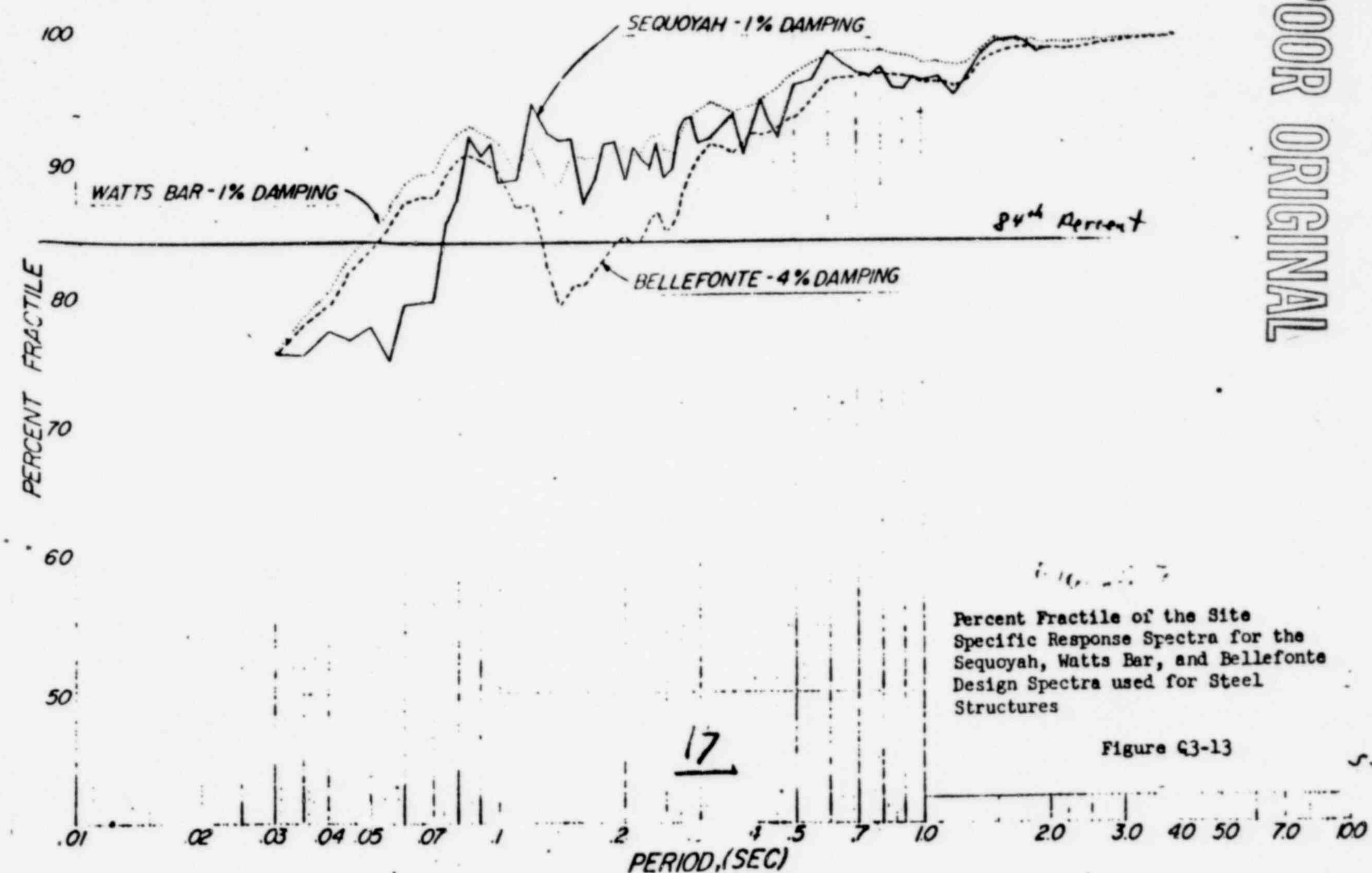
1374 216

46 5493



Percent Fractile of the Site Specific Response Spectra for the Sequoyah, Watts Bar, and Bellefonte Design Spectra used for Reinforced Concrete Structures

Figure Q3-14



Percent Fractile of the Site Specific Response Spectra for the Sequoyah, Watts Bar, and Bellefonte Design Spectra used for Steel Structures

Figure Q3-13

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17

POOR ORIGINAL

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INPUT PARAMETERS TO SEISMIC HAZARD COMPUTATIONS

(1) EARTHQUAKE ACTIVITY LEVELS FOR THE HOST TECTONIC PROVINCE AND THOSE SURROUNDING IT. THE ACTIVITY RATE FOR EACH PROVINCE WAS DETERMINED FROM THE SPECIFIC EARTHQUAKE HISTORY. THE B VALUES (RECURRENCE RATES) WERE ALL ASSUMED TO BE 0.57 (CHINNERY, 1979). THE UPPER INTENSITY CUTOFF WAS ASSUMED TO BE THE MAXIMUM HISTORICAL INTENSITY EXCEPT FOR THE HOST (AND CONTROLLING) PROVINCE WHERE THE MAXIMUM POSSIBLE INTENSITY WAS CONSERVATIVELY ASSUMED TO BE IX RATHER THAN VIII.

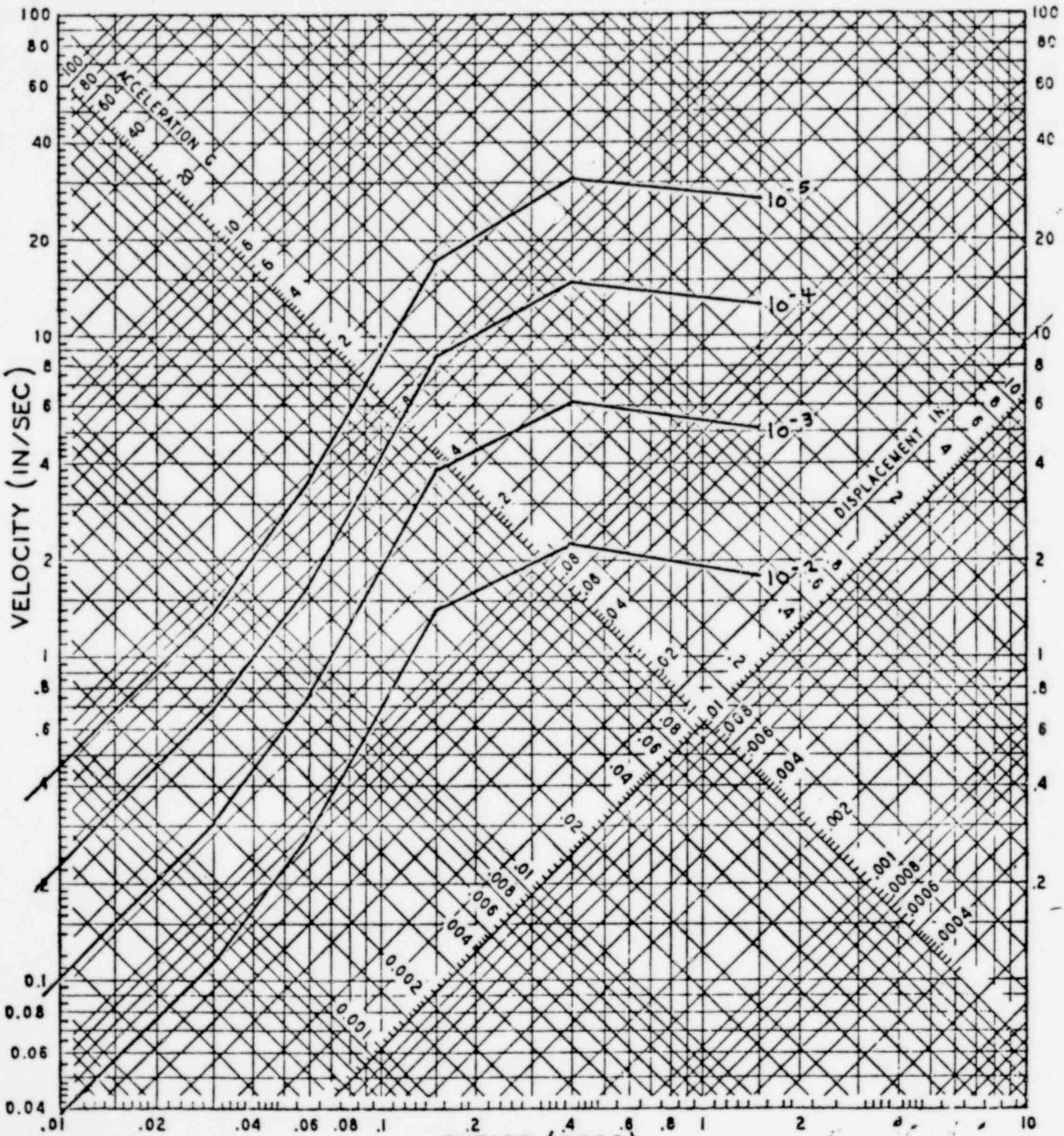
(2) THE INTENSITY FALL-OFF WITH DISTANCE WAS TAKEN TO BE THAT DETERMINED FROM THE 1886 CHARLESTON EARTHQUAKE (BOLLINGER, 1977).

(3) SITE INTENSITIES WERE CONVERTED TO PEAK ACCELERATION UTILIZING THE RELATIONSHIP DETERMINED BY MURPHY AND O'BRIEN (1978).

(4) PEAK ACCELERATIONS WERE CONVERTED TO SPECTRAL ACCELERATIONS AT SELECTED PERIODS UTILIZING SPECTRAL AMPLIFICATION FACTORS CALCULATED FROM THE 26 SITE-SPECIFIC SPECTRA NORMALIZED TO THE SAME PEAK ACCELERATION.

(5) THE DISPERSION ASSOCIATED WITH EACH OF THE LAST THREE RELATIONSHIPS WAS INCLUDED IN A TOTAL DISPERSION DEFINED BY A STANDARD DEVIATION FOR EACH PERIOD.

UNIFORM RISK RESPONSE SPECTRA WITH LIMITED DISPERSION ON UPPER LIMIT OF INTENSITY FOR SEQUOYAH, WATTS BAR, BELLEFONTE AND PHIPPS BEND PLANT SITES



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$I_{o_{MAX}} = IX$
$\beta = 1.312$

1374 219

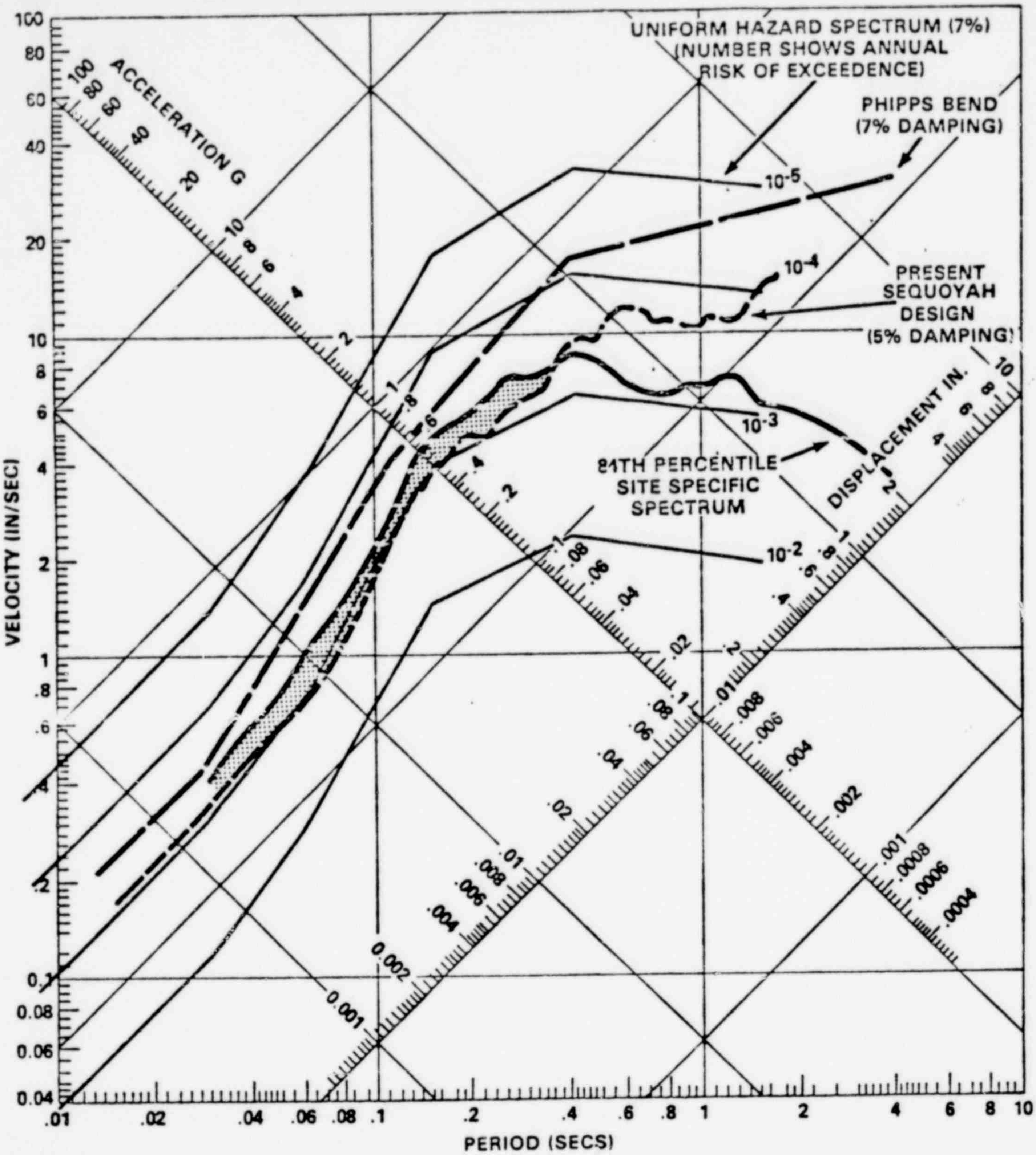

Figure 2-4

Comparison Of 7% Damped Uniform Hazard Response Spectra For The Sequoyah Site With The Present Sequoyah Design Spectrum For Reinforced Concrete, The 7% Damped 84th Percentile Site Specific Spectrum And The Phipps Bend Design Spectrum For Reinforced Concrete.

AVERAGE RISK OF EXCEEDANCE FOR SPECTRA AT PERIODS LESS THAN 0.5 SECONDS

SEQUOYAH DESIGN: 9.0×10^{-4} PER YEAR

SITE SPECIFIC EARTHQUAKE: 4.7×10^{-4} PER YEAR

PHIPPS BEND SSE: 2.3×10^{-4} PER YEAR

RELATIVE SEISMIC HAZARD

SEQUOYAH DESIGN VS SITE SPECIFIC EARTHQUAKE - 2x - (0.9-3.1)

SEQUOYAH DESIGN VS PHIPPS BEND SSE - 5x - (2.4-8.7)

1374 221

Sensitivity Study - 16th, 50th, and 84th Percentile Response Spectra for Original 13 Earthquakes, Original Plus 4 High Pairs, and Original Plus 4 Low Pairs 14

Lognormal Distribution - 7% Damping

Figure Q3-29 12 A-97

1374 222

CHARACTERIZATION OF SPECTRA IN TERMS OF INTENSITY
(UTILIZING TRIFUNAC AND BRADY, 1975 AND REG GUIDE 1.60)

SEQUOYAH DESIGN (REINFORCED CONCRETE)	INTENSITY VII
SITE SPECIFIC (84TH PERCENTILE)	INTENSITY VII-VIII
PHIPPS BEND	INTENSITY VIII

SOME REASONS FOR DIFFERENCES

1. LITTLE DATA AT INTENSITY VIII
2. 1897 GILES COUNTY MAY HAVE BEEN A WEAK VIII
3. DIFFERENCE IN SITE CONDITIONS

CONCLUSIONS

IT IS OUR CONCLUSION THAT THE DIFFERENCE IN ASSOCIATED SEISMIC HAZARD (RISK OF DESIGN SPECTRA BEING EXCEEDED BY EARTHQUAKE GROUND MOTION) BETWEEN THE PRESENT DESIGN AT SEQUOYAH AND THE APPROPRIATE SITE-SPECIFIC RESPONSE SPECTRUM IS NOT SUBSTANTIAL. THE REASONS FOR THIS ARE:

(1) FOR REINFORCED CONCRETE, THE PRESENT DESIGN AT SEQUOYAH REPRESENTS A MORE THAN MEDIAN DESCRIPTION OF THE CONTROLLING SITE-SPECIFIC GROUND MOTION.

(2) FOR REINFORCED CONCRETE, THE DIFFERENCES IN SEISMIC HAZARD ARE FACTORS OF 2 AND 3. THIS SEEMS VERY SMALL WHEN COMPARED TO THE ABSOLUTE SEISMIC HAZARD WHICH IS ON THE ORDER OF 10^{-3} TO 10^{-4}.

(3) IN OUR JUDGMENT, THERE ALREADY EXIST VARIATIONS IN SEISMIC HAZARD ASSOCIATED WITH DESIGN SPECTRA FOR OTHER PLANS IN THE EASTERN UNITED STATES THAT EXCEED FACTORS OF 2 OR 3.

(4) THE HAZARD ASSOCIATED WITH REINFORCED CONCRETE REPRESENTS A WORST CASE AND THE DIFFERENCE IN SEISMIC HAZARD WOULD BE EVEN LESS FOR OTHER MATERIALS.

1374 224

Appendix XVII

1374 225

TVA STRUCTURAL SEISMIC RE-EVALUATION

PURPOSE: DETERMINE THE MARGINS PRESENT IN THE CATEGORY I STRUCTURES FOR THE NEW REQUIREMENTS

GIVEN:

- SITE SPECIFIC RESPONSE SPECTRA-(84TH% RESPONSE SPECTRA)
- R.G. 1.61 DAMPING VALUES
- DYNAMIC & STRUCTURAL MODELS OF CATEGORY I STRUCTURES

OUTLINE OF RE-EVALUATED STRUCTURES

I. ROCK SUPPORTED STRUCTURES

SHIELD BLDG.

AUX. - CONTROL BLDG.

INTERNAL STRUCTURES

STEEL CONTAINMENT SHELL

II. SOIL SUPPORTED STRUCTURES

DIESEL - GENERATOR BLDG.

OTHER STRUCTURES

POOR ORIGINAL

1374 227

[redacted]

OUTLINE OF PROCEDURES USED IN RE-EVALUATION

- 84% RESPONSE SPECTRA vs. ACTUAL DESIGN SPECTRA
- DYNAMIC/STRUCTURAL ANALYSIS OF STRUCTURES USING:
    - 84TH % RESPONSE SPECTRA
    - RG 1.61 DAMPING VALUES
- DETERMINE AVAILABLE MARGINS AT CRITICAL LOCATIONS

POOR ORIGINAL

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A-102

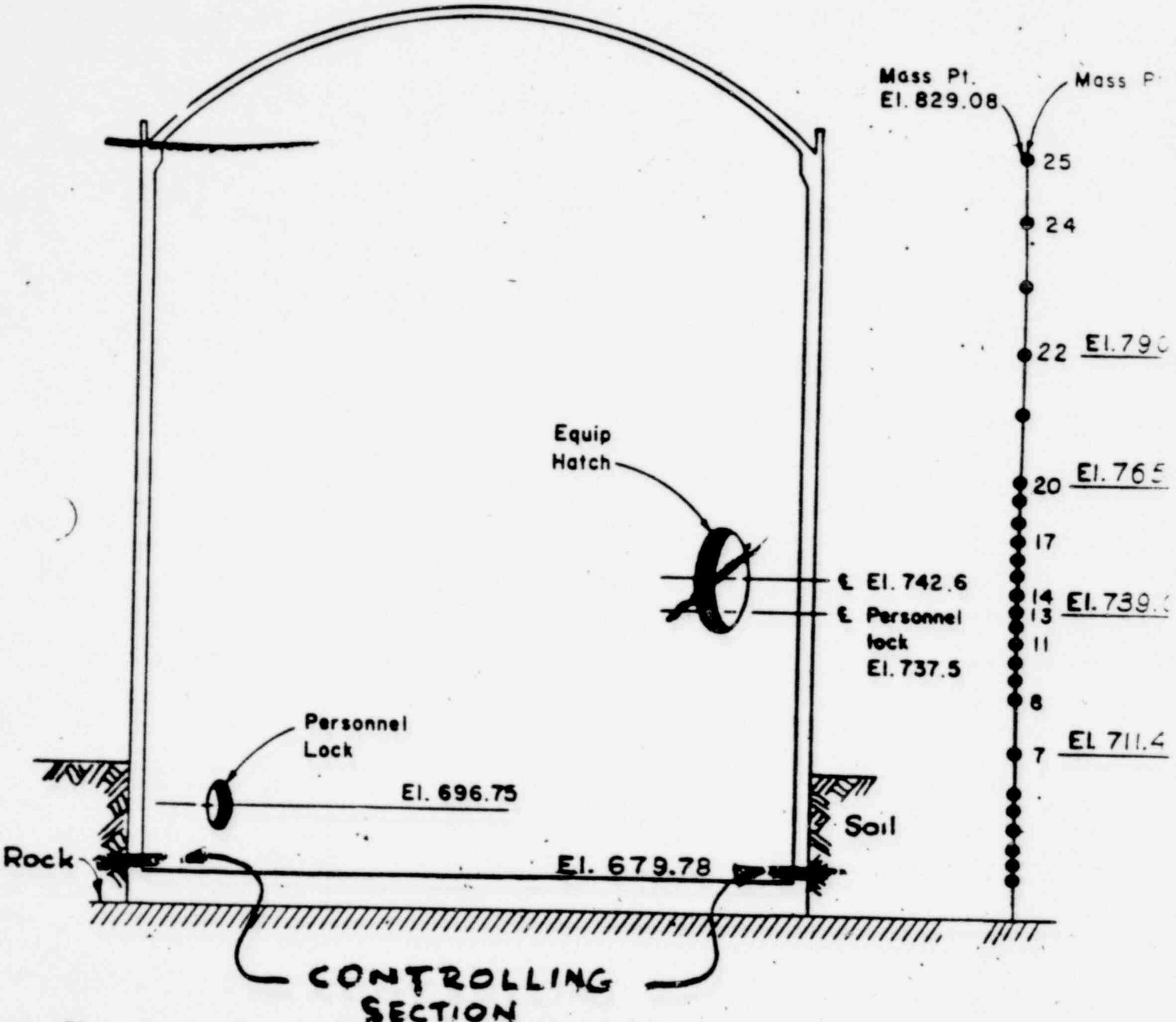


Figure 1 Section through Reactor Shield Building Looking West, Lumped Mass Model for Dynamic Analysis

1374 229

ROCK SUPPORTED STRUCTURES - FINDINGS

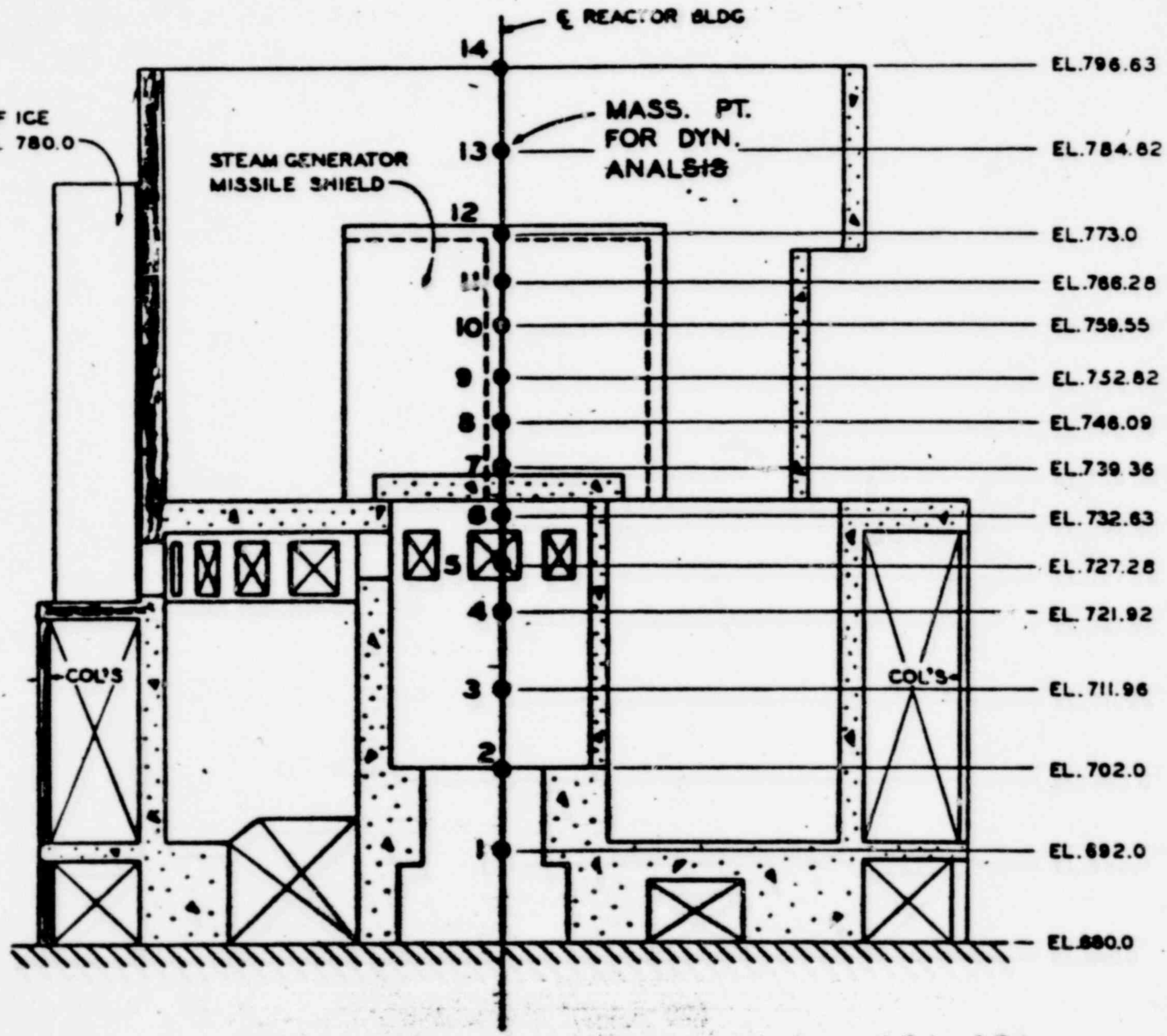
SHIELD BLDG.

EARTHQUAKE LOAD	INCREASE
ELEV. vs ACCELERATION	26% (TOP)
MOMENT	22% (BASE)
SHEAR LOAD	17% (BASE)
VERTICAL LOAD	60% (BASE)
VERTICAL ACCELERATION	54% (TOP)

RESULTS

- BASE OF THE STRUCTURE CONTROLS DESIGN
- 0. 3% OVERSTRESS IN REINFORCING STEEL (AISC)
- 5.0% OVERSTRESS IN CONCRETE (ACI-318)

1374 230



1374 231

Figure 2 Reactor Building, Interior Concrete Structure Sectional Elevation Looking West, Lumped Mass Model for Dynamic Analysis

ROCK SUPPORTED STRUCTURES - FINDINGS

INTERNAL STRUCTURES

EARTHQUAKE LOAD			INCREASE
ELEV.	VS.	ACCELERATION	59% (TOP)
		MOMENT	33% (BASE)
		SHEAR	25% (BASE)
		VERTICAL LOAD	112% (BASE)
		VERTICAL ACCELERATION	99% (TOP)

RESULTS

- NO OVERSTRESS OF REINFORCING STEEL OR CONCRETE
- MARGINS FOR CRITICAL LOCATIONS

CRANE WALL	15% (TENSION STEEL)
ICE CONDENSER FLOOR	57% (TENSION STEEL)
	55% (CONCRETE)
ICE CONDENSER COLUMNS	56% (BUCKLING)

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1374 233

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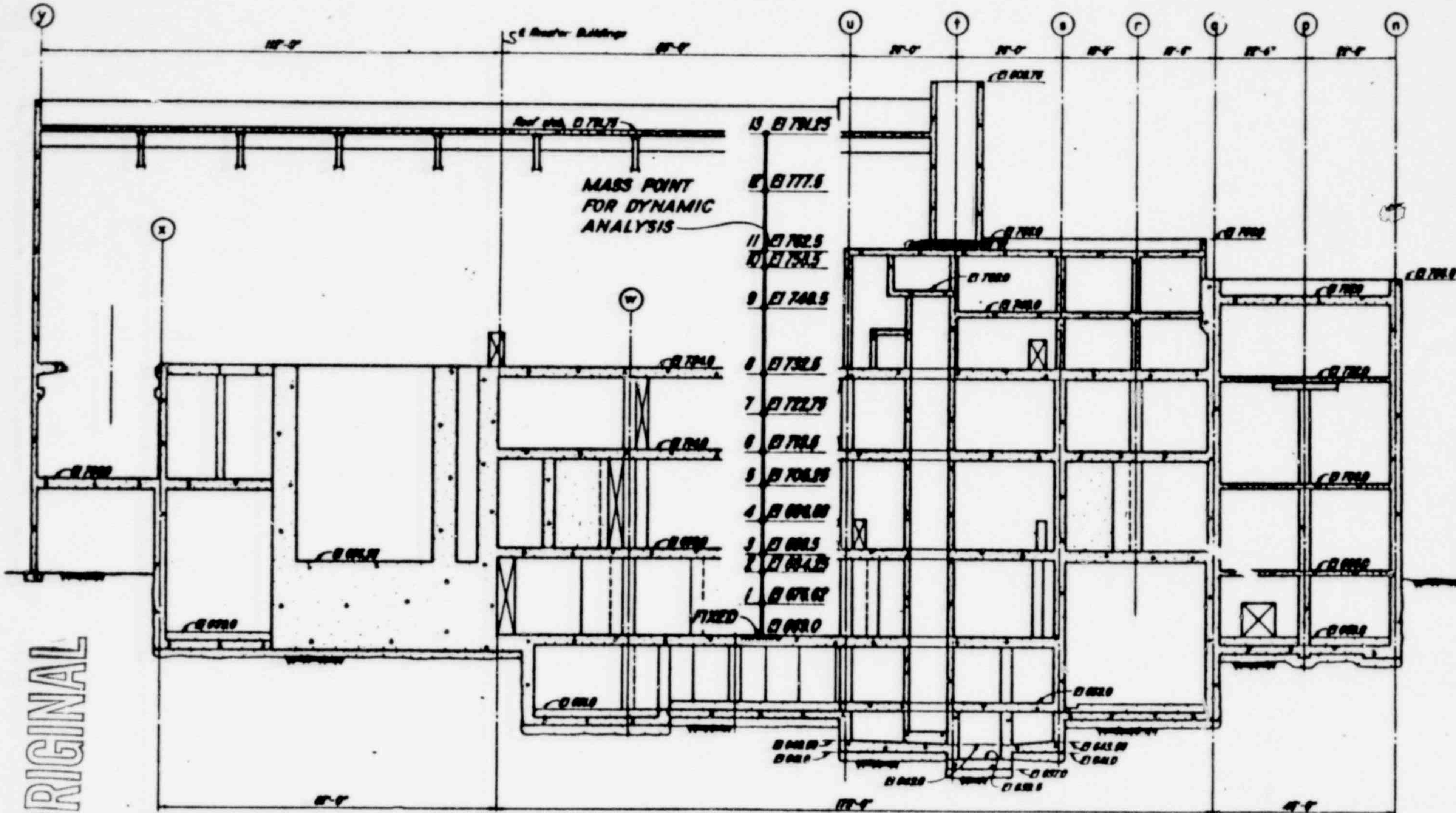


Figure 3 Sectional Elevation of Auxiliary Control Building Lumped Mass Model for Dynamic Analysis

ROCK SUPPORTED STRUCTURES - FINDINGS

AUX/CONTROL BLDG.

EARTHQUAKE LOAD			INCREASE
ELEV.	VS.	ACCELERATION	60% (TOP)
		SHEAR	30% (BASE)
		BENDING MOMENT	33% (BASE)
		VERTICAL ACCELERATION	153% (TOP)
		VERTICAL LOAD	57% (BASE)

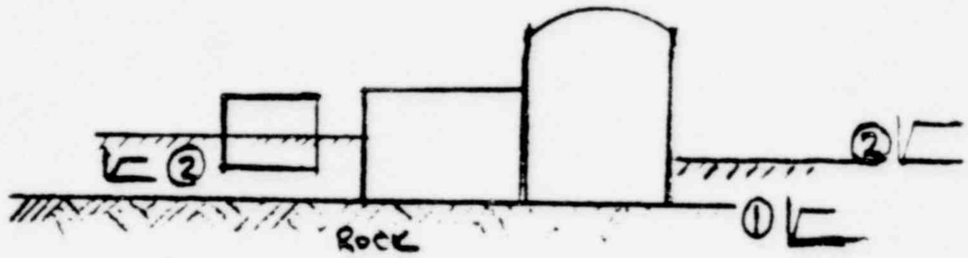
RESULTS

- NO OVERSTRESS OF REINFORCING STEEL OR CONCRETE
- MARGINS FOR EXHAUST STACK - 21% (TOTAL SECTION SHEAR)

1374 254

SOIL SUPPORTED STRUCTURES

SEISMIC DESIGN OF STRUCTURES



- SITE DESIGN RESPONSE SPECTRA ①
- CALCULATED FREE FIELD SURFACE RESPONSE SPECTRA ② OBTAINED BY VARYING SOIL PROPERTIES AND SOIL DEPTH FROM SURFACE TO BEDROCK. THIS REPSONSE SPECTRA WAS APPLIED AT THE FOUNDATION OF THE SOIL SUPPORTED STRUCTURES.
- THE NEW 84TH % RESPONSE SPECTRA IS FOR ALL FREQUENCIES OF INTEREST LESS THAN THE OLD DESIGN RESPONSE SPECTRA {② }
- BY INSPECTION THE SOIL SUPPORTED STRUCTURES MEET THE NEW DESIGN REQUIREMENTS.

1374 235

SEISMIC QUALIFICATION OF EQUIPMENT & COMPONENTS

- NEW FLOOR RESPONSE SPECTRA WILL BE COMPUTED BASED ON ONE OF THE FOUR ORIGINAL DESIGN EARTHQUAKES.
- THE SELECTED EARTHQUAKE'S AMPLITUDE WILL BE INCREASED BY FACTOR SUCH THAT ITS RESPONSE SPECTRA WILL ENVELOP THE NEW 84TH% DESIGN RESPONSE SPECTRA.

1374 236

CONCLUSION:

THE CATEGORY I STRUCTURES ARE ACCEPTABLE FOR THE GSB'S 84TH % RESPONSE SPECTRA & R. G. 1.61 REQUIREMENTS.

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Appendix XVIII

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APPENDIX XVIII: Title: Sequoyah 1 and 2 Seismic Design Margins Review of Piping & Mechanical Equipment

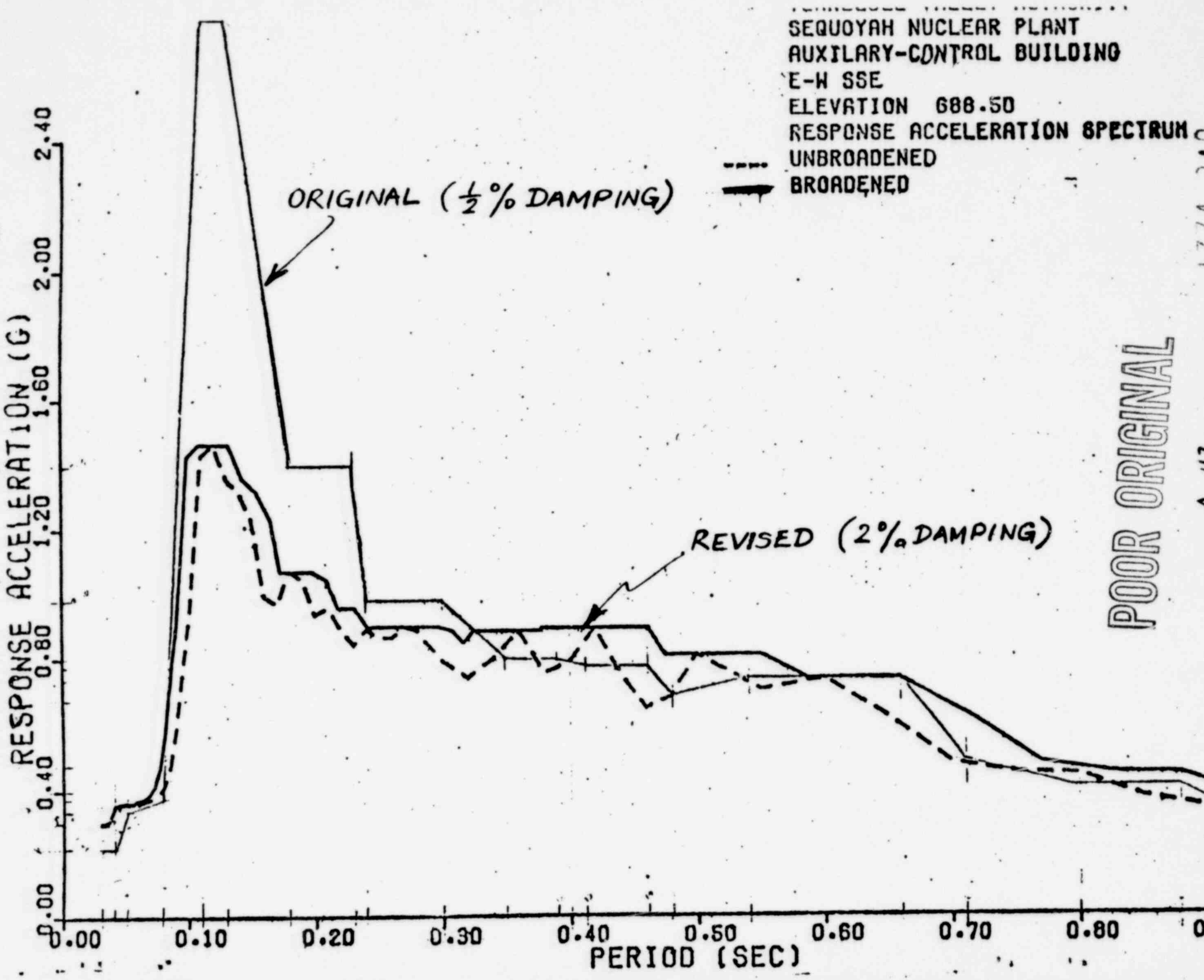
SEQUOYAH UNITS 1 AND 2

SEISMIC DESIGN MARGIN REVIEW

PIPING AND MECHANICAL EQUIPMENT

- AUXILIARY FEEDWATER AND THE ESSENTIAL RAW COOLING WATER PIPING SYSTEMS WERE SELECTED FOR REANALYSIS ON THE BASIS OF THEIR SIGNIFICANCE IN ACHIEVING A SAFE SHUTDOWN.
- REGIONS OF HIGH SSE PLUS DEAD WEIGHT PLUS PRESSURE STRESSES WERE IDENTIFIED IN THESE SYSTEMS BASED ON THE ORIGINAL FLOOR RESPONSE SPECTRA AND 1/2% DAMPING
- PIPING SYSTEMS WERE REANALYZED USING THE 84 PERCENTILE EARTHQUAKE RESPONSE SPECTRA AND REG. GUIDE 1.61 DAMPING (2%)
- SEISMIC MARGINS WERE QUANTIFIED IN REGIONS OF HIGH STRESS
- PIPING SUPPORTS WERE EVALUATED ON THE BASIS OF REVISED DESIGN LOADS
- SELECTED MECHANICAL AND ELECTRICAL EQUIPMENT IN SAFE SHUTDOWN SYSTEMS WAS EVALUATED AGAINST THE REVISED FLOOR RESPONSE SPECTRA

1374 259



POOR ORIGINAL

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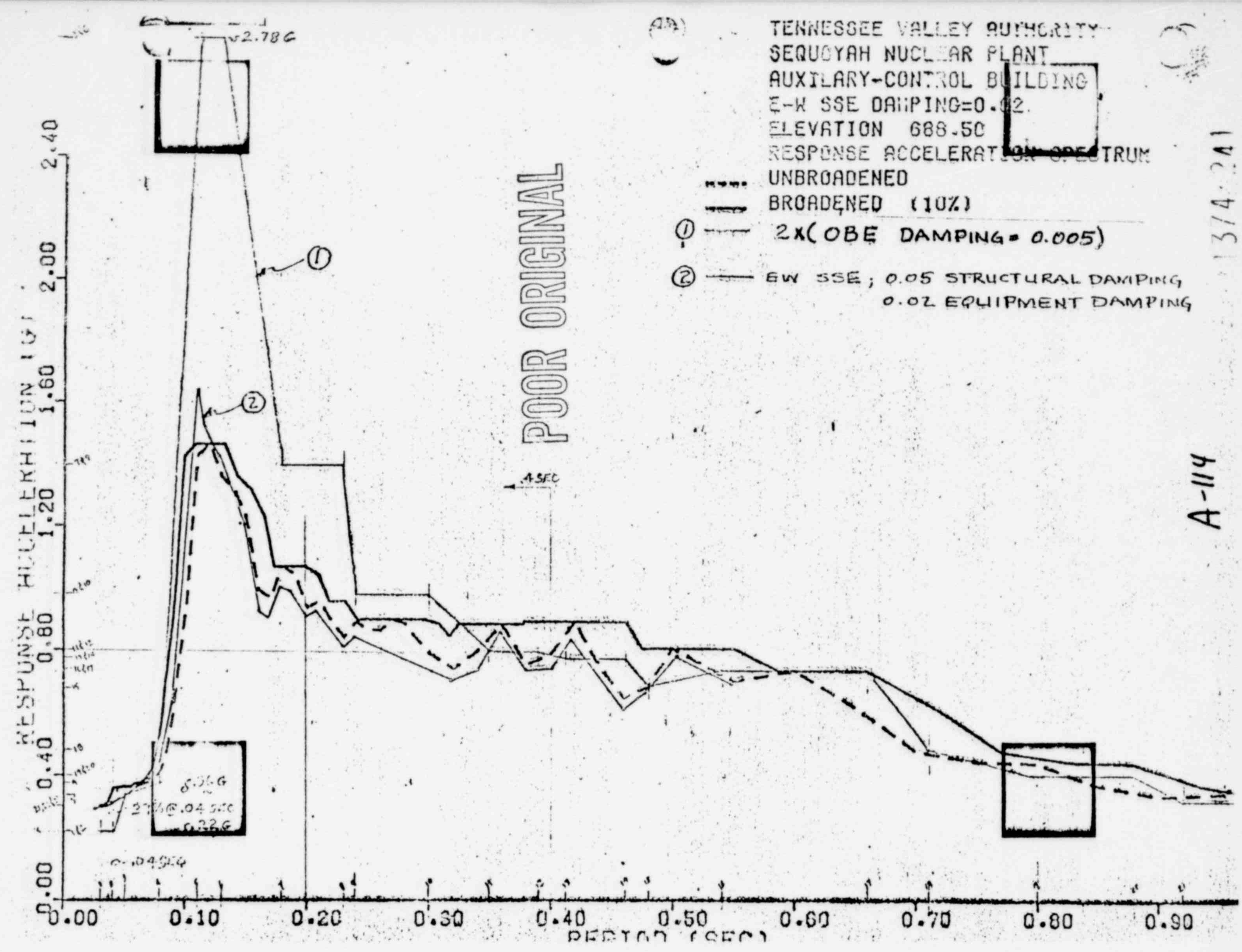
1374 240



POOR ORIGINAL

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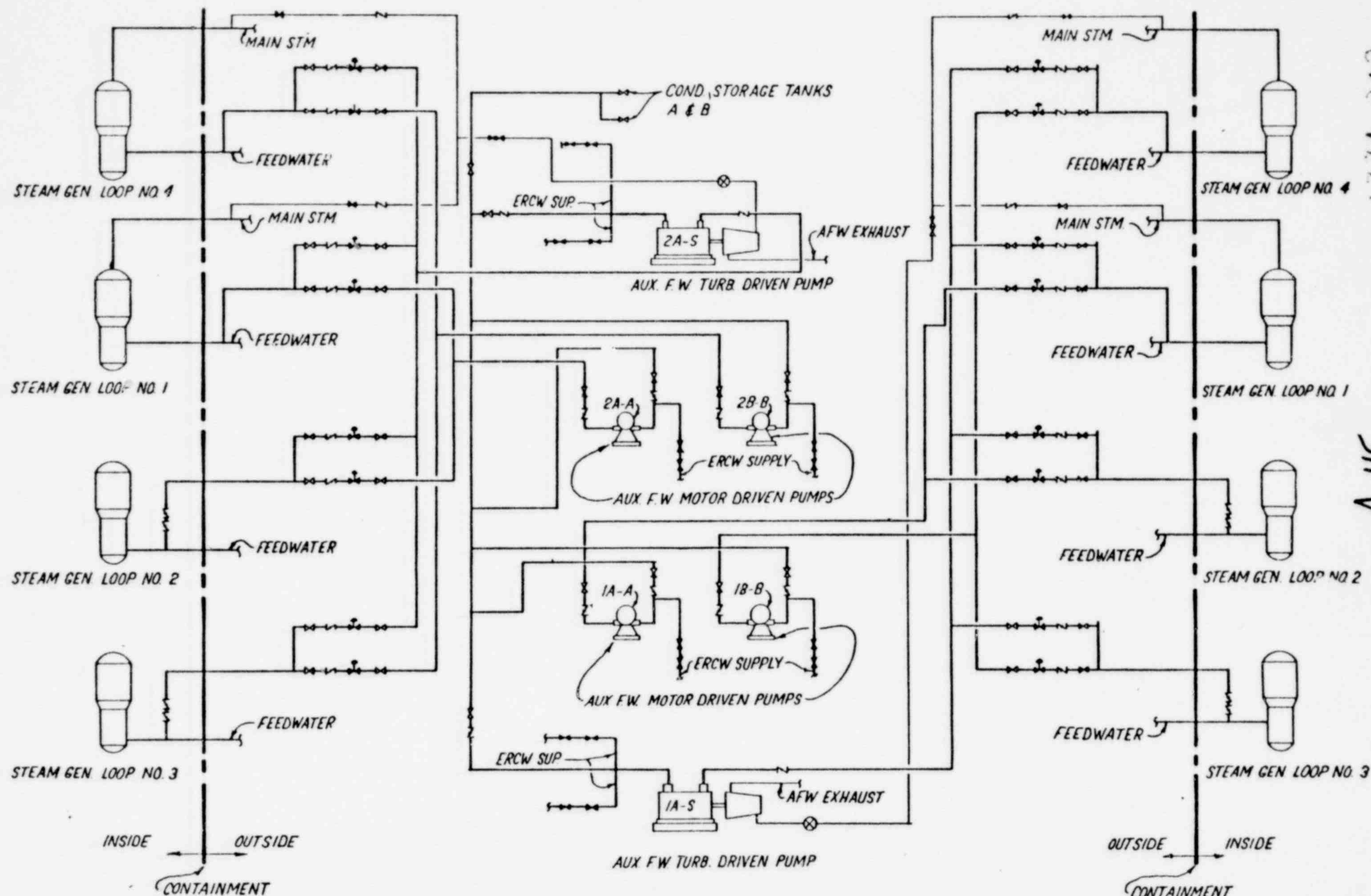
1374 241



AUXILIARY FEEDWATER

POOR ORIGINAL

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SYSTEM PROBLEM NUMBER	SEISMIC	WEIGHT STRESS (S_{DL})	PRESS. STRESS (S_{LP})	DESIGN SEISMIC / 1/2% DAMPING				84 PERCENTILE / RG 1.61 DAMPING			
				SSE STRESS	TOTAL STRESS	ALLOW. STRESS	SSE STRESS / ALLOW. STRESS	SSE STRESS	TOTAL STRESS	ALLOW. STRESS	SSE STRESS / ALLOW. STRESS
AFW N2-3-1A,-2A	AB 690'	370	280	24,495	25,145	36,000	.680	24,425	25,075	36000	.678
AFW N2-3-3A	AB 714'	857	2830	21,097	24,784	36000	.586	7287	10974	36000	.202
AFW N2-3-4A	AB 714'	311	3044.	20258	23,613	36000	.563	6715	10070.	36000	.187
ERCW N2-67-2A	AB 734'	668	828	27,798	29,294	36000	.772	15270	16,766	36000	.424
ERCW N2-67-3A	AB 714'	1621	531	23,434	25,589	36000	.651	15089	17241	36,000	.419

POOR ORIGINAL

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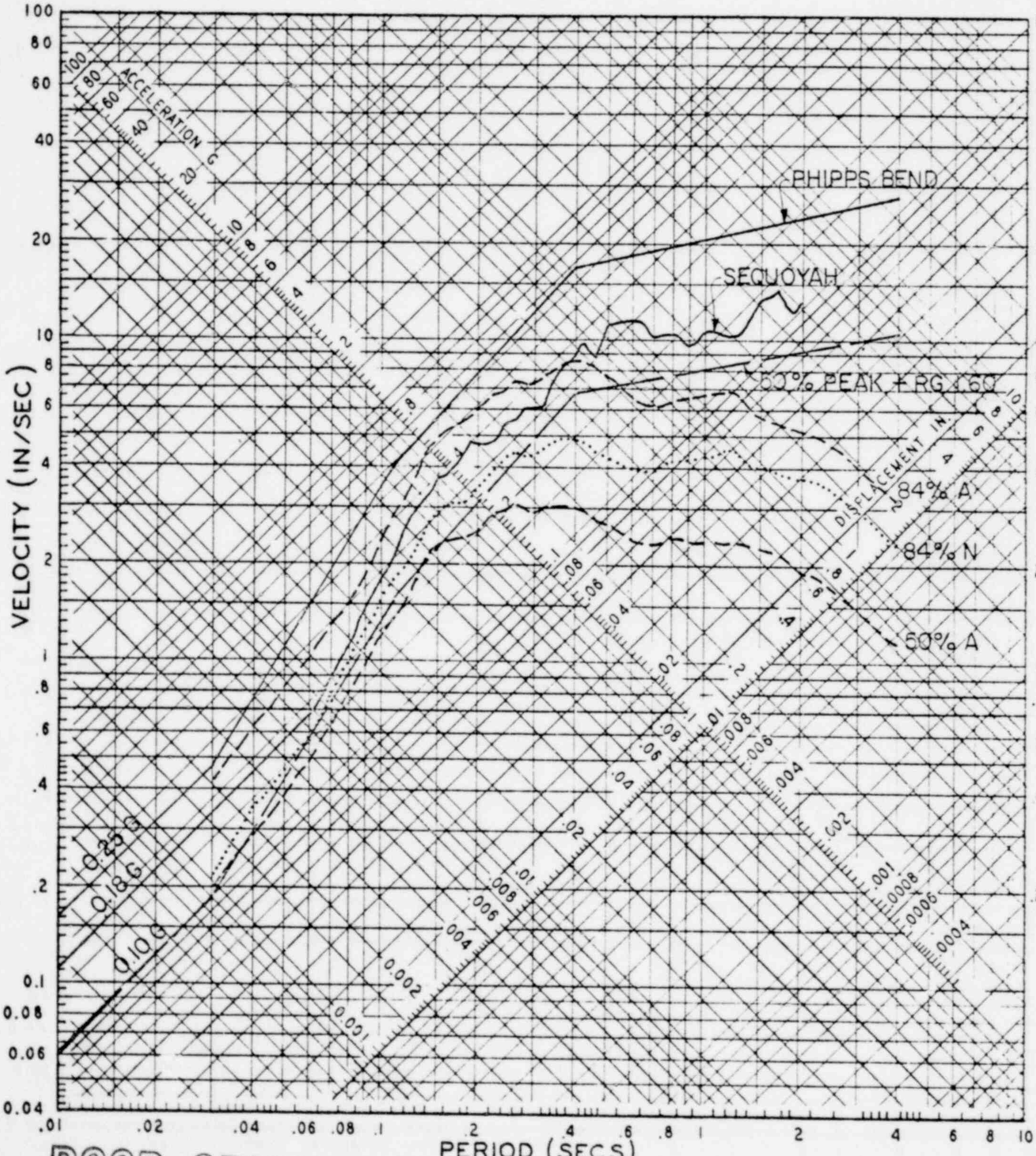
QUALIFICATION SUMMARY OF SELECTED MECHANICAL

COMPONENTS AND EQUIPMENT

COMPONENTS	VENDOR	NATURAL FREQ.	QUALIFICATION METHOD
AUX. FEEDWATER PUMPS MOTOR, TURBINE	INGERSOLL-RAND	RIGID/FLEX	COMB. TEST & ANALYSIS
CONTROL VALVES	MASONEILAN INT.	RIGID/FLEX	COMB. TEST & ANALYSIS
COMPONENT COOLING WATER PUMP	DELAVAL	RIGID	ANALYSIS
COMPONENT COOLING WATER HEAT EXCHANGER	INDUS. PROC.	RIGID	ANALYSIS
MAIN ATMOS. RELIEF VALVES	WESTING.	RIGID/FLEX	TEST AND ANALYSIS
MAIN STEAM CHECK VALVES	ATWOOD & MORRILL	RIGID	ANALYSIS
MAIN STEAM SAFETY VALVES	CROSBY	RIGID	ANALYSIS
EMERG. DIESEL GEN.	BRUCE GM	RIGID	TEST AND ANALYSIS
ERCW PUMPS/MOTOR	JOHNSTON PUMP CO.	RIGID/FLEX	ANALYSIS
AUX. AIR COMPRESSOR	INGERSOLL-RAND	RIGID/FLEX	TEST
AUX. CONTROL AIR RECEIVERS	INGERSOLL-RAND	RIGID	ANALYSIS
HVAC DUCTS	WYLE (QUALIF. TESTS)	FLEX	TEST

COMPARISON OF SEQUOYAH AND PHIPPS BEND DESIGN SPECTRA FOR REINFORCED CONCRETE STRUCTURES WITH VARIOUS SITE SPECIFIC SPECTRA

1374 245



POOR ORIGINAL

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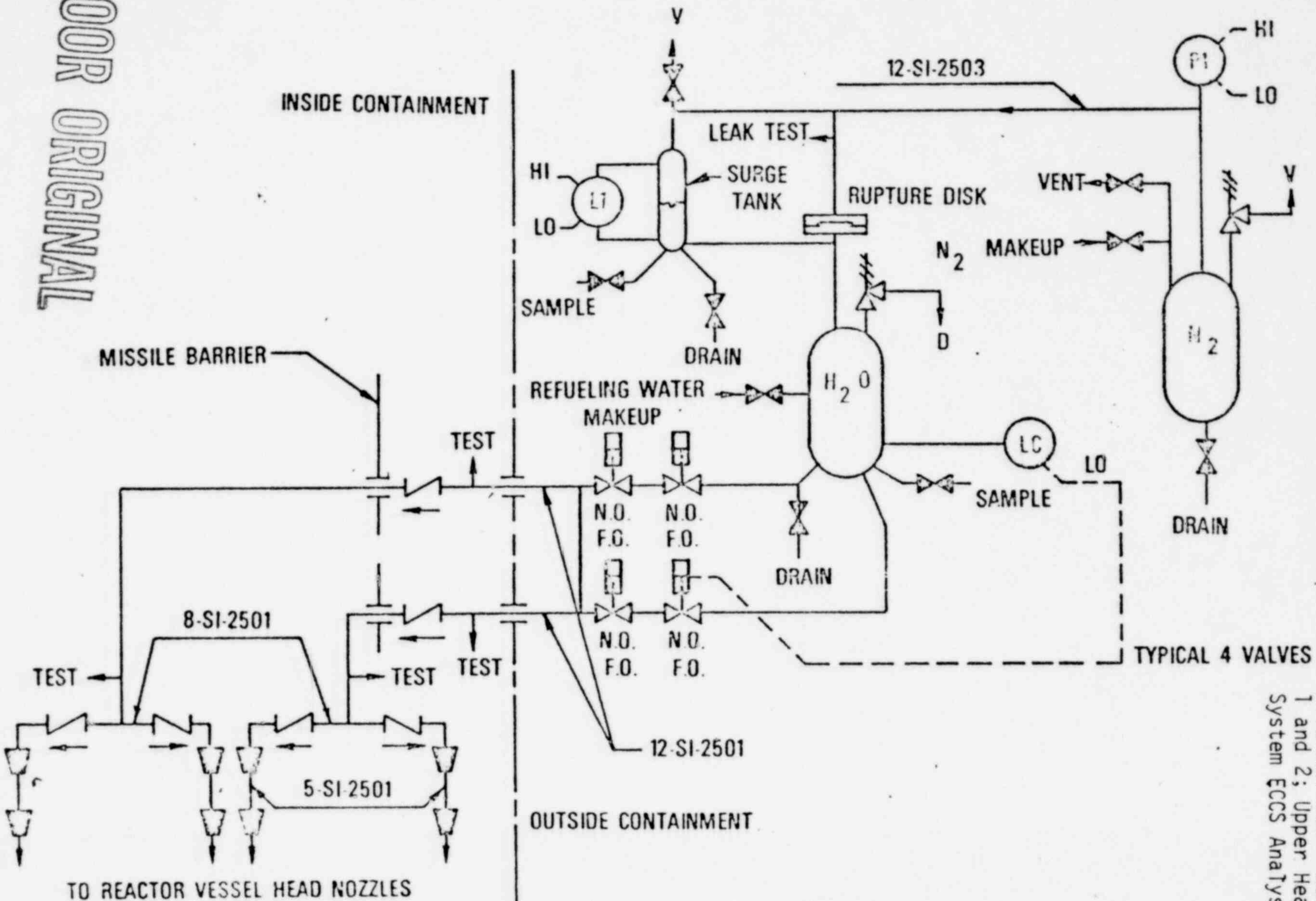
Appendix XIX

UPPER HEAD INJECTION SYSTEM FLOW DIAGRAM



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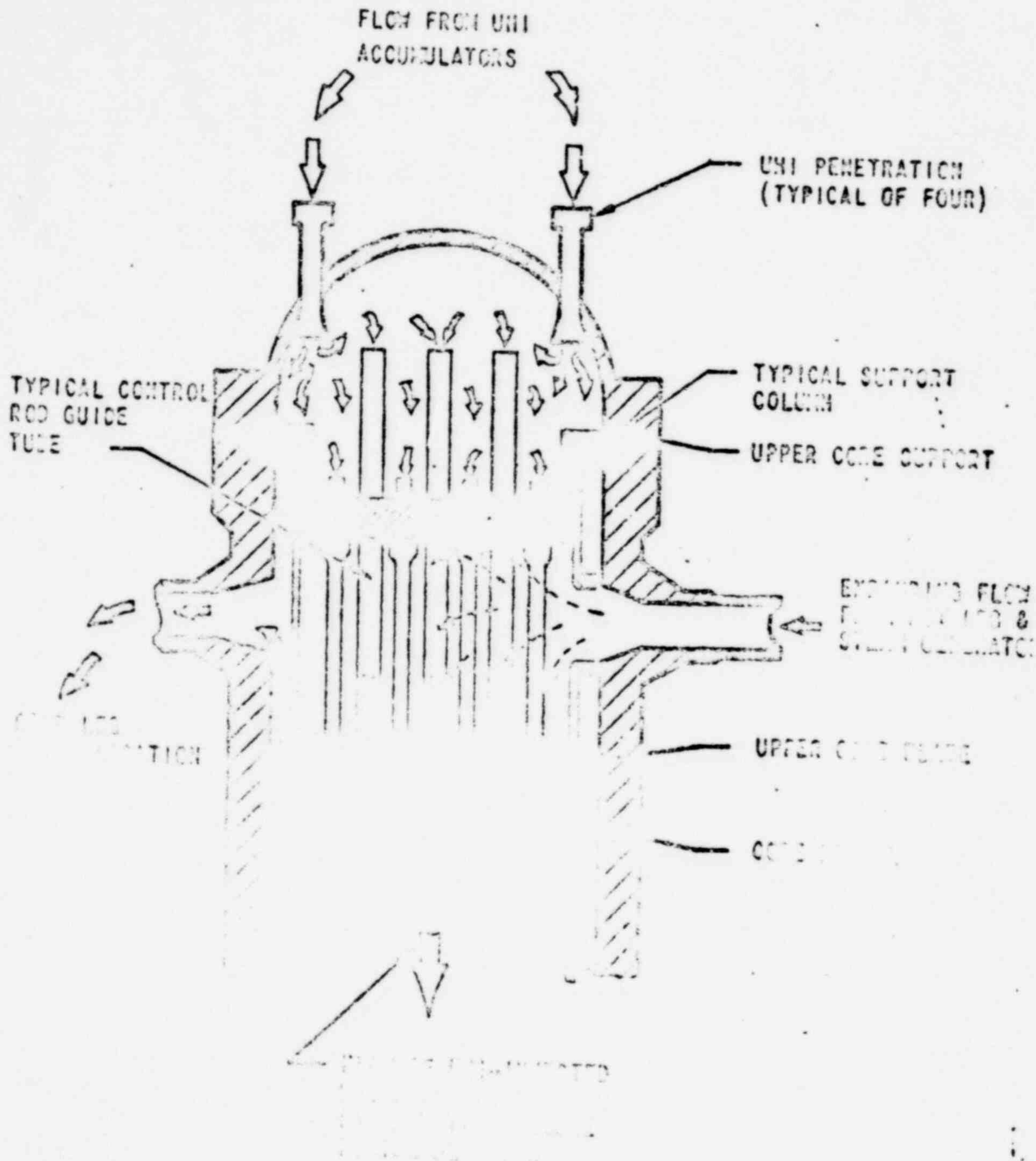
APPENDIX XIX - Sequoyah Units 1 and 2; Upper Head Injection System ECCS Analysis

ANALYTICAL MODEL

- CONFORMS WITH APPENDIX K REQUIREMENTS
- SAFETY EVALUATION REPORT ISSUED APRIL 1978
- SEQUOYAH RESULTS MEET ACCEPTANCE CRITERIA OF 10 CFR 50.46

1374 248


FLOW FROM UNI
ACCUMULATORS
UNI PENETRATION
(TYPICAL OF FOUR)
TYPICAL CONTROL
ROD GUIDE
TUBE
TYPICAL SUPPORT
COLUMN
UPPER CORE SUPPORT
UPPER

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SUMMARY OF RESULTS

TIME SEQUENCE OF EVENTS

ACTION	C_D = 0.6 DECLG PERFECT MIXING (SEC)	C_D = 0.6 DECLG IMPERFECT MIXING (SEC)
SI SIGNAL	4.8	4.8
UPPER HEAD ACCUMULATOR INJECTION	2.62	1.82
COLD LEG ACCUMULATOR INJECTION	19.4	19.9
UPPER HEAD ACCUMULATOR DELIVERY COMPLETED	26.3	23.1
PUMPED INJECTION	29.8	29.8
END OF BYPASS	58.0	48.0
END OF BLOWDOWN AND BOTTOM OF CORE RECOVERY	128.0	71.8
COLD LEG ACCUMULATOR EMPTY	128.9	120.2

POOR ORIGINAL

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COMPLIANCE WITH APPENDIX K 10CFR50.46

RESULT	C_D = 0.6 DECLG PERFECT MIXING	C_D = 0.6 DECLG IMPERFECT MIXING
PEAK CLAD TEMP. (^{o}F)	2111.	2190.
PEAK CLAD TEMP. LOCATION (FT)	7.5	7.5
LOCAL ZR/H2O REACTION (MAX. %)	4.07	7.63
LOCATION OF MAX. LOCAL ZR/H2O (FT)	7.5	7.5
TOTAL ZR/H2O REACTION (%)	<0.3	<0.3
HOT ROD BURST TIME (SEC)	72.8	65.2
HOT ROD BURST LOCATION (FT)	6.0	7.0

LICENSED CORE POWER (MWT), 102% OF	3411
PEAKING LINEAR POWER (KW/FT), 102% OF	12.25
PEAKING FACTOR (AT LICENSE RATING)	2.25

SUMMARY

- ANALYSIS PERFORMED WITH APPROVED MODEL RESULTS IN PCT < 2200°F

- SEQUOYAH ECCS MEETS THE REQUIREMENTS OF THE ACCEPTANCE CRITERIA PRESENTED IN 10CFR50.46

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Appendix XX

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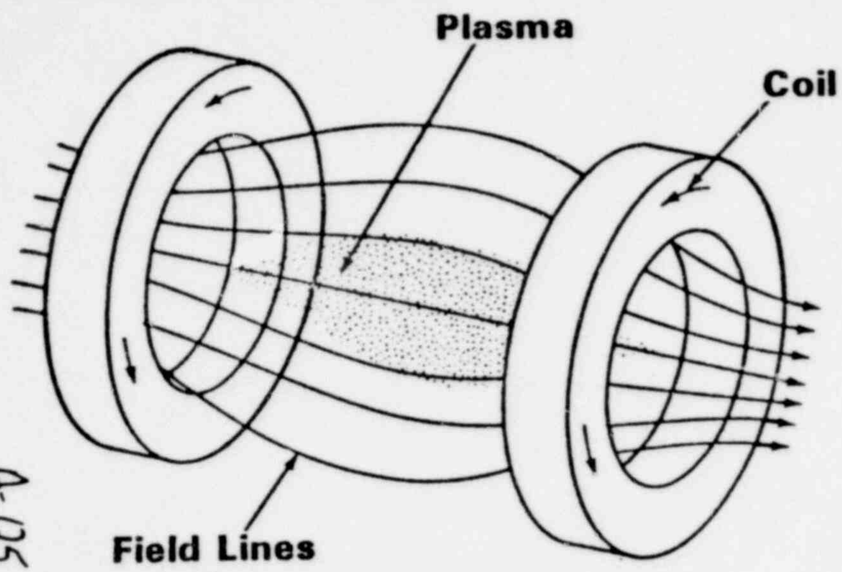
MAGNETIC FIELD CONFIGURATIONS

IV. Fundmentals of Fusion



Open System - Simple Magnetic Mirror

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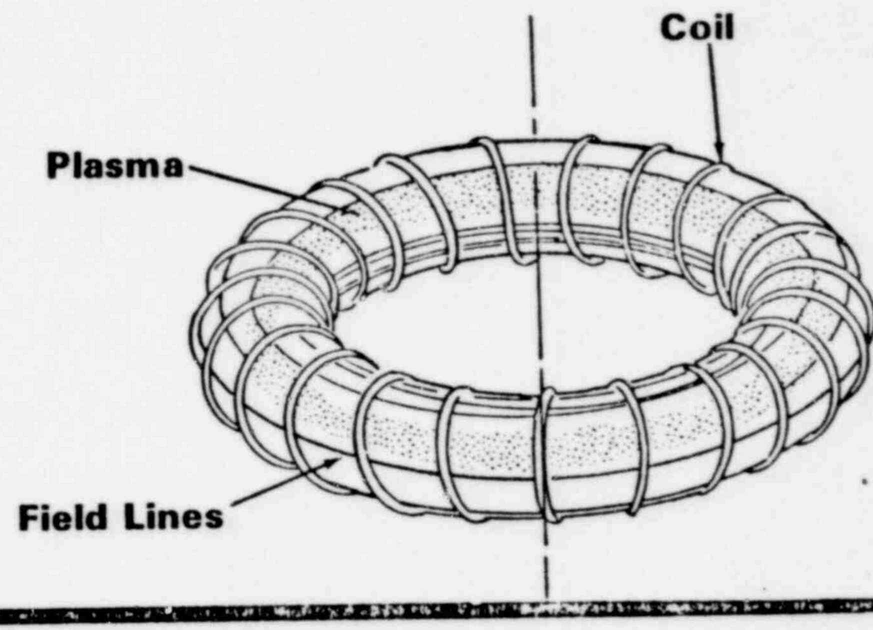
Closed System - Simple Torus



POOR ORIGINAL

APPENDIX XX: DOE's Magnetic Fusion Program

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/ugraph Frame

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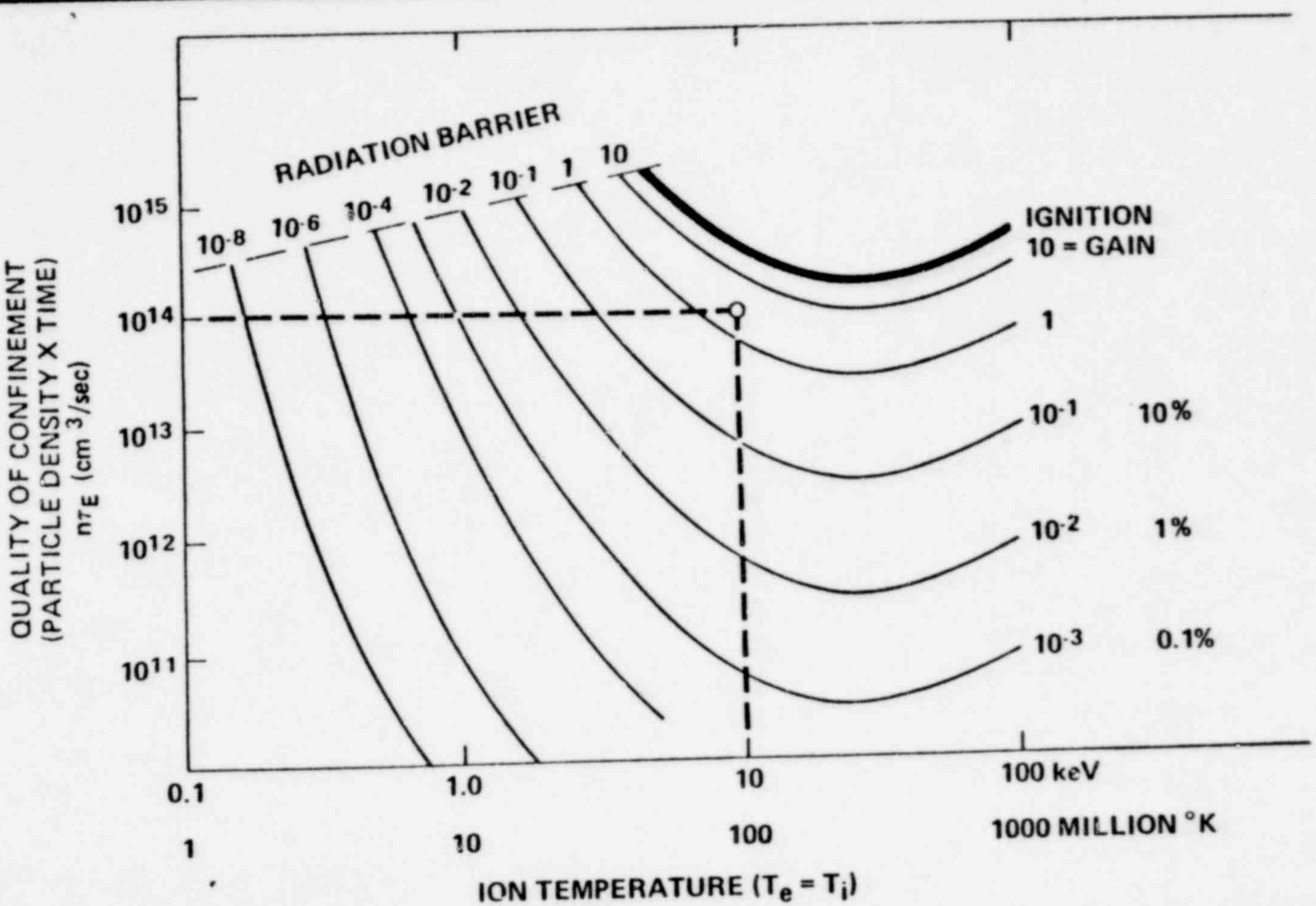
POOR ORIGINAL

GOAL FOR POWER REACTORS

IV. Fundamentals of Fusion



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534-B

NEW DEVICES COMPLETED

- DOUBLET III — FEBRUARY 1978
- ALCATOR C — APRIL 1978
- ISX-B — JULY 1978
- PDX — NOVEMBER 1978
- TANDEM MIRROR — OCTOBER 1978
- EBT-S — SEPTEMBER 1978

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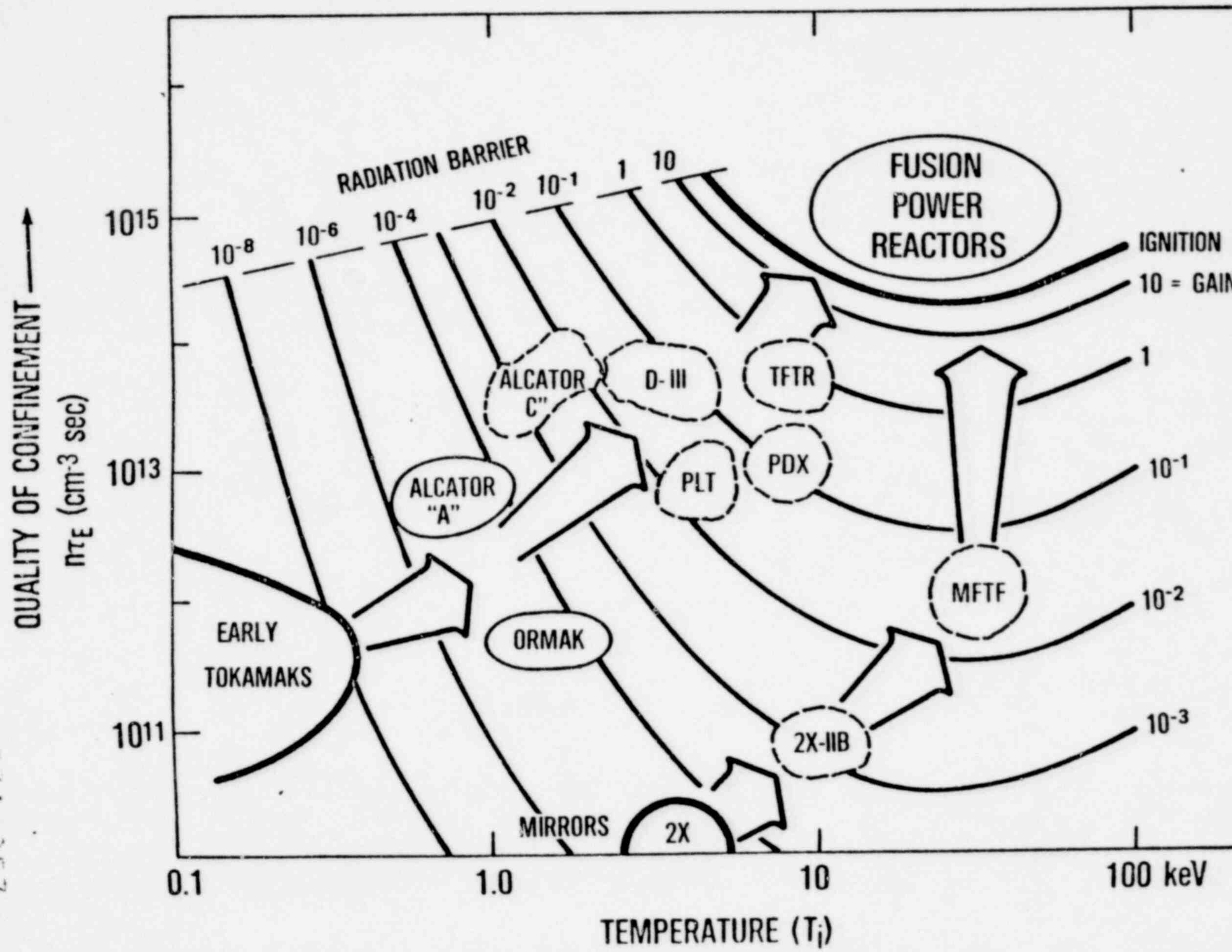
TECHNICAL PROGRESS AND OUTLOOK IN MAGNETIC FUSION



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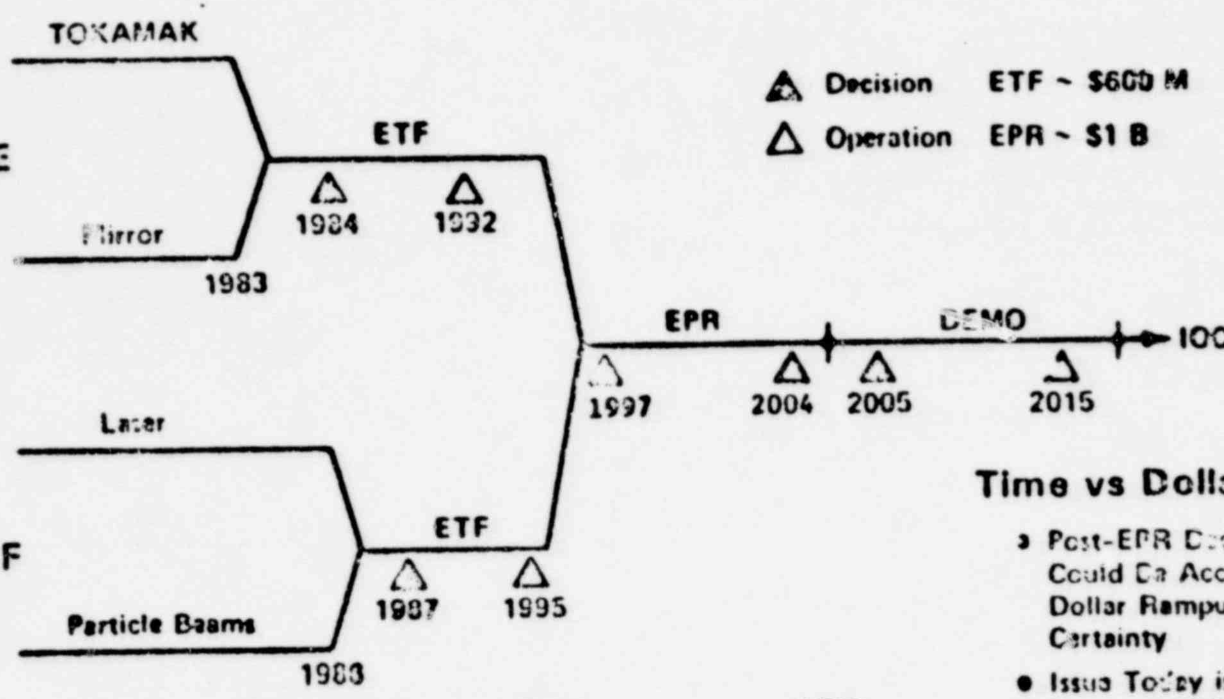
OPERATING CHARACTERISTICS OF FUSION DEVICES

	CURRENT EXPERIMENTS	TFTR	ETF	COMMERCIAL TOKAMAK
POWER (MWT)	0	20	500	2000
POWER PULSE LENGTH (SEC.)	0.1–0.3	0.5	30	200
OPERATING DUTY CYCLE	0.1%	0.2%	50%	90%
DEVICE AVAILABILITY	20–30%	50%	50%	75%
ION TEMPERATURE (KEV)	1–10	5–10	10	13
$N\tau$ ($10^{20} M^{-3}$ SEC.)	≤ 1	≤ 1	1–2	1–2

Fusion Development

(Reference Budget Case)



Key Milestones	ETF Decision	ETF Operation	EPR Operation
Cumulative Costs To Key Milestones	$4 B	$11 B	$18 B

Time vs Dollar Tradeoff

- Post-EPR Decision Program Could Be Accelerated By Dollar Rampup with Greater Certainty
- Issue Today is Program Through ETF Period Up To EPR Decision Point

78-8556-M/12-20

TFTR SAFETY ISSUES

OFFICE OF FUSION ENERGY, DOE

I. MAGNETIC FUSION PROGRAM OVERVIEW

J. E. BAUBLITZ, CHIEF
REACTOR SYSTEMS & APPLICATIONS BRANCH

II. TFTR W. A. MARTON
PROJECT TECHNOLOGY BRANCH

A. GENERAL DESCRIPTION

B. SAFETY ISSUES

C. IMPORTANT SYSTEMS

D. SAFETY ANALYSIS REPORT

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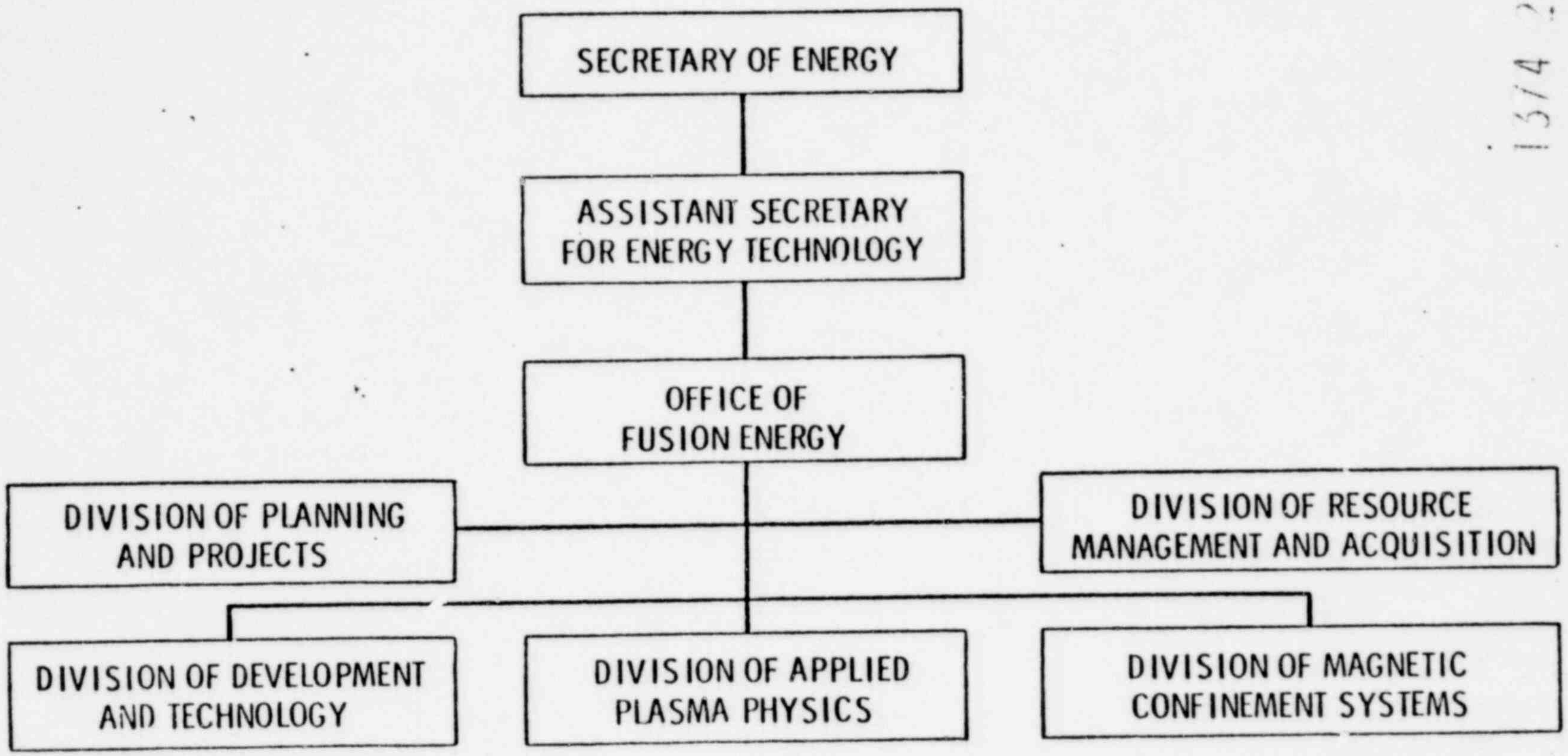

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Appendix XXI

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OBJECTIVES OF TFTR

- TO STUDY THE PHYSICS OF DEUTERIUM-TRITIUM PLASMAS
  - ACHIEVE PLASMA PARAMETERS NEAR THOSE REQUIRED FOR POWER REACTOR, T = 5-10 KEV, DENSITY x CONFINEMENT $>10^{13}$, POWER DENSITY=1 W/$\underline{CM}^3$
  - ACHIEVE SCIENTIFIC BREAKEVEN, FUSION ENERGY OUT = PLASMA HEATING ENERGY IN
  - BEGIN TO INVESTIGATE PHYSICS BEHAVIOR OF SELF-HEATED (BY ALPHAS) PLASMA
- TO BEGIN DEVELOPING FUSION ENGINEERING EXPERTISE WITH TRITIUM

POOR ORIGINAL

APPENDIX XXI: Tokamak Fusion Test Reactor (TFTR) Program

A-133

1314 263

TFTR PROJECT STATUS

- TEC $239M
- R&D $100M
- BEGIN OPERATION — 3/82
- DESIGN COMPLETE
- 75% EQUIPMENT ON ORDER
- CONSTRUCTION 35% COMPLETE
- PARTICIPANTS
  - PRINCETON
  - EBASCO/GRUMMAN DESIGN
  - GIFFELS A/E
  - TERMINAL CONSTRUCTION CO.

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PRINCETON PLASMA PHYSICS LABORATORY

- NEW YORK CITY, PHILADELPHIA – EACH 50 MILES AWAY
- POPULATION — 0-10 MILES IS 200,000 (FARMS/TOWNS)
  0-50 MILES IS 16,000,000
- LABORATORY — 900 PEOPLE
- CAMPUS/TOWN — 3 MILES WEST
- TFTR SITE BOUNDARY — 125 METERS

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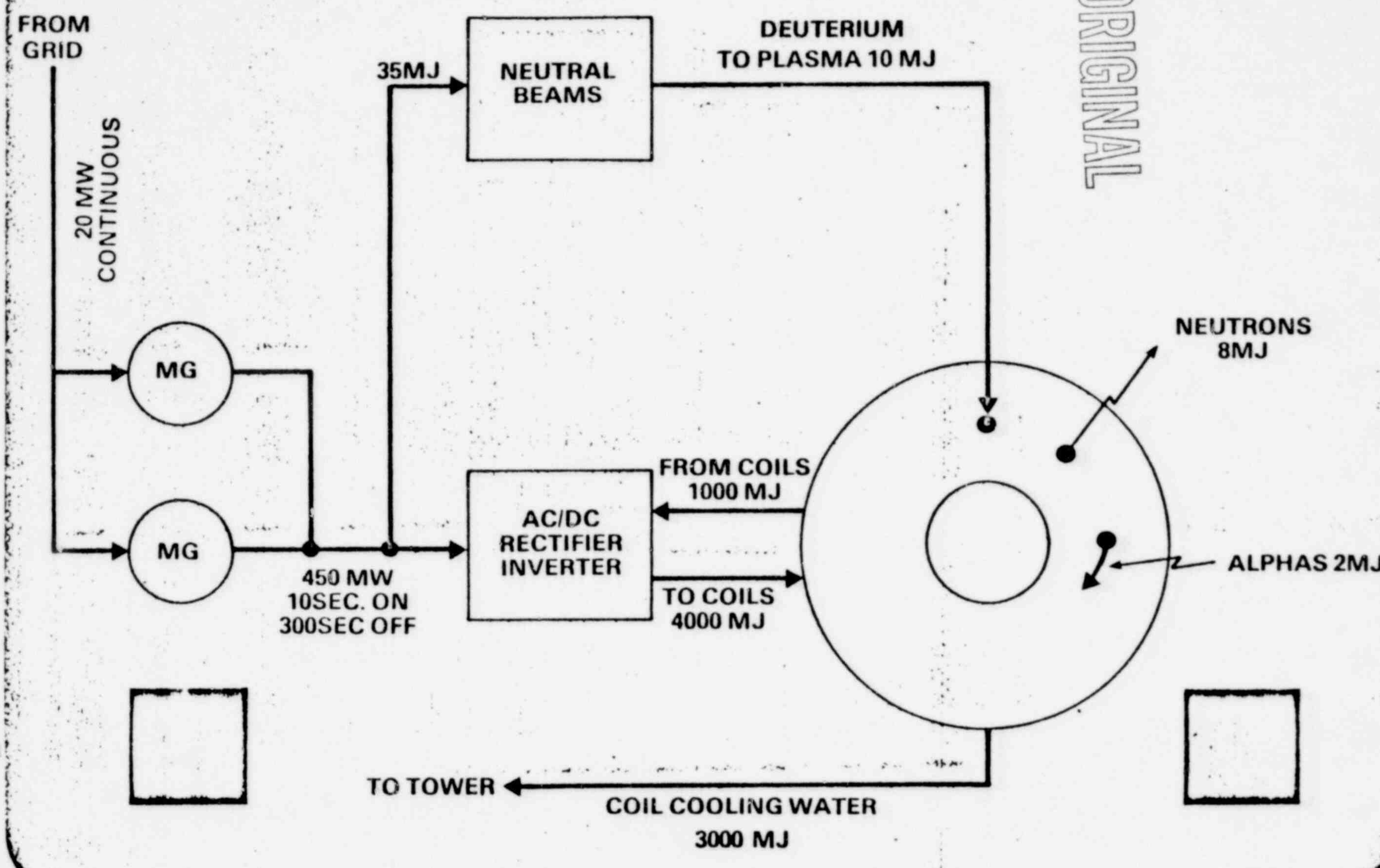
TFTR COMPLEX

- 20-MW POWER IN
- MG BUILDING — 2 MG'S, TOTAL 660 MW, 9,000 MJ WITH 4,500 MJ DELIVERABLE, 400 RPM, 10 SEC PULSE, 5 MINUTE RECHARGE
- ENERGY CONVERSION BUILDING — 600 MW DC RECTIFIER/INVERTER, 75 KA TO COILS
- NEUTRAL BEAM BUILDING — 120 KV, 20 MW, 0.5 SEC
- TEST CELL — 45M x 35M x 15M HIGH, WALLS 4 FT., ROOF 5 FT.
- NEUTRAL BEAM TEST CELL
- MOCKUP AREA
- BASEMENT — WATER SYSTEMS, REINFORCED TRITIUM VAULT

A-136

1374 266

TFTR ENERGY FLOW



POOR ORIGINAL

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1374 267

TFTR RADIATION DOSE CRITERIA

(SITE BOUNDARY)

EVENT	DOE CRITERIA OBJECTIVE/LIMIT
MOST INTENSE NATURAL PHENOMENA & VERY LOW PROBABILITY ACCIDENTS (REM/OCCURRENCE)	5/25
MOST PROBABLE NATURAL PHENOMENA & LOW PROBABILITY ACCIDENTS (REM/OCCURRENCE)	1/25
OPERATIONAL OCCURRENCES (MREM/YEAR)	100/500
NORMAL OPERATIONS (MREM/YEAR)	10/500

BASIS: DOE MANUAL CHAPTER 0524 STANDARDS FOR RADIATION PROTECTION

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TFTR RADIATION DOSE CRITERIA

(ON SITE)

POOR ORIGINAL

EVENT	DOE CRITERIA OBJECTIVE/LIMIT
ACCIDENTS (REM/OCCURRENCE)	25/25
NORMAL OPERATION AND OPERATIONAL OCCURRENCES (MREM/YEAR)	1000/5000 CONTROLLED AREA 100/500 UNCONTROLLED AREA

BASIS: DOE MANUAL CHAPTER 0524 STANDARDS FOR RADIATION PROTECTION

A-139

1314 269

TFTR SAFETY CONCERNS

(SITE BOUNDARY)

- **MAGNETIC FIELDS — 3% OF BACKGROUND**
- **RADIATION**
  - **NEUTRONS — DIRECT, ACTIVATION OF AIR, WATER**
  - **GAMMA**
  - **TRITIUM**

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1574 270

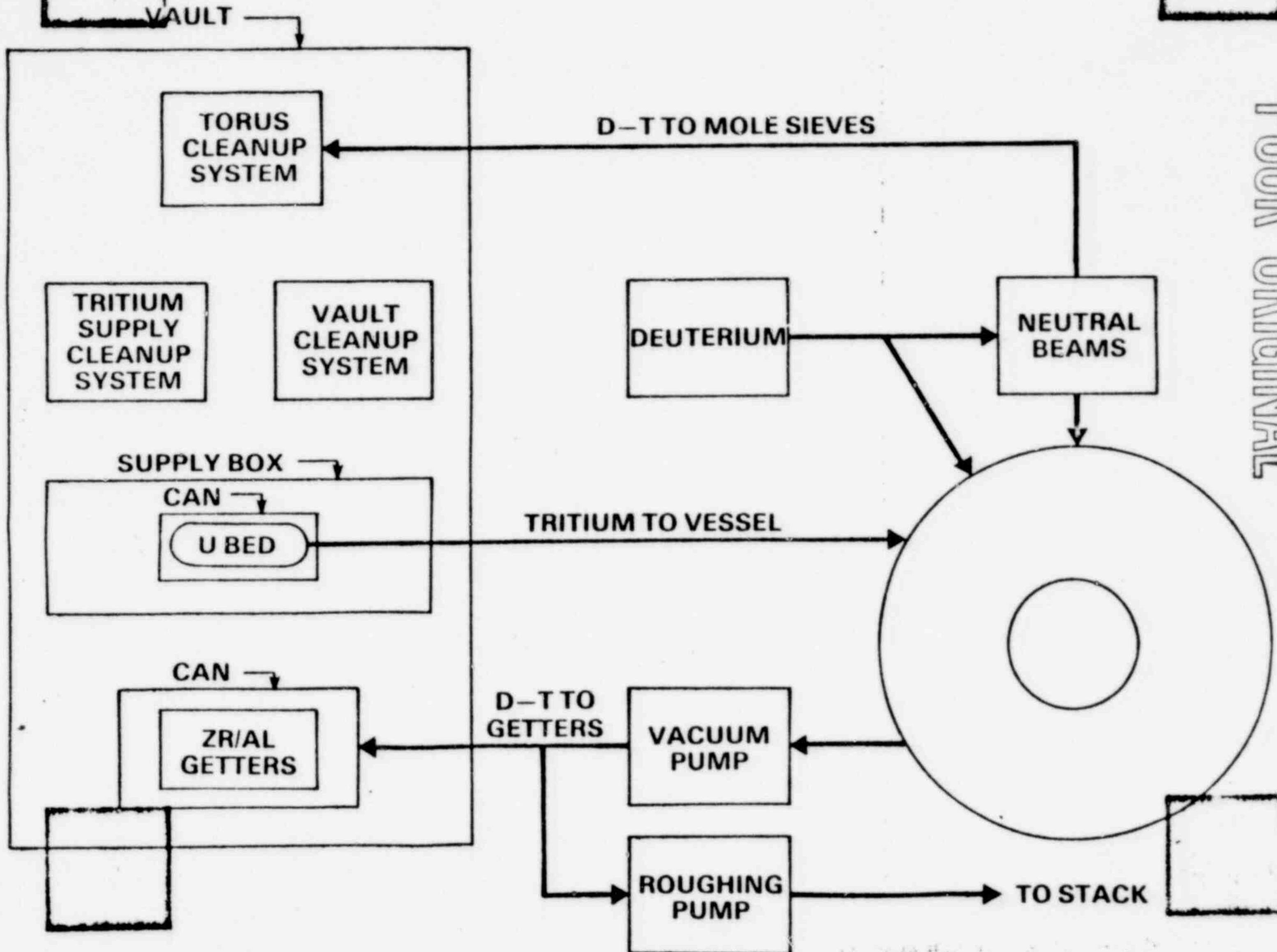
TFTR SAFETY CONCERNS ON SITE

- **ELECTRICAL**
  - HIGH VOLTAGE UP TO 120 KV
  - HIGH CURRENT UP TO 2.5 MA
  - MAGNETIC FIELDS UP TO 100,000 GAUSS
- **MECHANICAL**

  INTERACTING CURRENTS AND FIELDS
  - COIL CENTERING FORCE — 6,000,000 LBS.
  - COIL OVERTURN MOMENT — 10,000,000 FT. LBS.
  - COIL BURSTING — 5,000 PSI
  - ROTATIONAL — MG AND TURBOPUMPS
- **FIRE AND EXPLOSION**
  - INSULATING OIL
  - HYDROGEN
  - CRYOGENS
- **RADIATION**
  - NEUTRONS — DIRECT, ACTIVATION OF AIR, WATER, COMPONENTS
  - GAMMA
  - TRITIUM
- **INDUSTRIAL**
- **NATURAL PHENOMENA**

A-141

TFTR GAS FLOW



A-142

BASIC DIFFERENCES PWR AND TFTR

POOR ORIGINAL

1. STORED ENERGY (MJ)

PWR		TFTR	
30,000	CORE THERMAL	9,000	MG ROTATIONAL
400,000	PRIMARY COOLANT THERMAL	1,300	COILS MAGNETIC
400,000	STEAM GENERATOR THERMAL	3,000	COILS THERMAL
830,000	TOTAL	25	PLASMA MAGNETIC
		10	PLASMA THERMAL
		9,000	TOTAL

2. STORED CURIES

PWR	TFTR
10^9	5×10^4

3. STORED CHEMICAL (MJ)

PWR		TFTR	
10,000	ZR/H_2O (5%)	50	H_2/O_2
10,000	H_2/O_2		

4. DECAY HEAT

PWR	TFTR
7% TO 0	NONE

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FIRE SAFETY CRITERIA

- PPPL SAFETY MANUAL
- MINIMIZE COMBUSTIBLES
- PROVIDE SEPARATION BETWEEN COMBUSTIBLES AND IGNITION SOURCE
- ALARMS AND AUTOMATIC SPRINKLERS/HALON
- RESPONSE BY TRAINED PERSONNEL

A-144

1374 274

TFTR ELECTRICAL SAFETY CRITERIA

- PPPL SAFETY MANUAL
- FAILURE OF 2 INDEPENDENT INTERLOCKS
- NATIONAL ELECTRIC CODE FOR SEPARATION BETWEEN POWER, SIGNAL, CONTROL
- SAFETY INTERLOCK SYSTEM SEPARATED FROM ALL OTHERS

A-145

TFTR CONTROL SYSTEM PHILOSOPHY

- ALL CONTROL FROM CONTROL ROOM BY COMPUTERS
- ALL EQUIPMENT SELF PROTECTING BY HARDWIRE INDEPENDENT OF COMPUTERS

A-146

1314 276

TFTR TRITIUM HANDLING PHILOSOPHY

- LIMIT ON-SITE INVENTORY TO 50,000 CI
- STORE IN SOLIDS (URANIUM OR ZR/AL) IN VAULT DESIGNED FOR MOST INTENSE CONDITIONS
- ADMINISTRATIVELY CONTROL INVENTORY IN TORUS, NEUTRAL BEAMS TO SMALL QUANTITY SINCE RISKS ARE HIGHER IN TEST CELL (ELECTRICAL, MECHANICAL, MOST PROBABLE DESIGN)
- PROVIDE SEPERATE CLEANUP CAPABILITIES (SUPPLY, VAULT, TORUS, TEST CELL)
- OFF-SITE PROCESSING OF TRITIUM

POOR ORIGINAL

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1374 277

FLOODS

- PROBABLE MAXIMUM PRECIPITATION, 48 HOURS
- PROBABLE MAXIMUM FLOOD (10^{-4} PER YEAR)
  - FLOODS TO 91 FT. MSL FROM BEE BROOK
  - SITE AT 95–100 FT. MSL

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1374 278

TFTR TORNADO CRITERIA

TYPE	CRITERIA
MOST INTENSE ($P = 10^{-7}$)	245 MPH ETC.
MOST PROBABLE ($P = 10^{-6}$)	110 MPH ETC.

BASIS: REVIEW HISTORY LAST 25 YEARS, RADIUS 50 MILES

FIND PROBABILITY OF ANY OCCURRENCE AT SITE

DEVELOP WINDSPEED PROBABLITY DISTRIBUTION USING FUJITA SCALE

FOR GIVEN P FIND PROBABILITY OF SPECIFIC TORNADO AND IT'S WINDSPEED

A-149

1574 279

TFTR EARTHQUAKE CRITERIA

TYPE	CRITERIA
MOST INTENSE	MM VII .13g HORIZONTAL .085g VERTICAL
MOST PROBABLE	MM VI .07g HORIZONTAL .043g VERTICAL

BASIS: 1. REVIEW HISTORY LAST 300 YEARS, RADIUS 200 MILES. USE TRIFUNIC BRADY ACCELERATIONS FOR GEOLOGIC PROVINCE.

2. INCREASE INTENSITY BY 1 AT FAULTS, ATTENUATE TO TFTR SITE.

3. MOST PROBABLE IS LARGEST IN NEW JERSEY.

POOR ORIGINAL

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1374 280

QUALITY ASSURANCE PLAN

- APPLIES TO ALL PHASES (DESIGN TO OPERATION)
- LEVELS APPROPRIATE TO SHUTDOWN AND SAFETY SIGNIFICANCE

  CRITICAL — LONG SHUTDOWN, OFF-SITE DOSE >1 REM

  MAJOR — THREE WEEK SHUTDOWN, OFF-SITE DOSE >0.1 REM

  MINOR — NO SAFETY IMPLICATION

BASIS: DOE MANUAL CHAPTER 0820 QUALITY ASSURANCE PLUS NASA INPUT

A-151

OPERATIONS PHILOSOPHY

- FORMAL PROCEDURES FOR OPERATION AND MAINTENANCE
- PPPL SAFETY REVIEW COMMITTEE
- OPERATIONAL SAFETY REQUIREMENTS

  FACILITY — DOE CONTROLLED

  SUBSYSTEMS — PPPL CONTROLLED

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1374 282

TEST CELL BUILDING

- DESIGNED FOR MOST PROBABLE CONDITIONS:
  NOT A CONTAINMENT BUILDING
- ZONED VENTILATION, ALL SUBATMOSPHERIC,
  TRITIUM VAULT AND TEST CELL LOWEST
- TRITIUM SEAL TO OTHER SPACES
- CONTINUOUS VENTING UP STACK
- RECIRCULATION DURING ACCIDENT
- STEAM SPARGE/CONDENSE TRITIUM CLEANUP
  SYSTEM WITH LOW VENT RATE

A-153

13/4 283

TRITIUM SUPPLY SYSTEM

- DESIGNED FOR MOST INTENSE CONDITIONS
- LOCATED IN HARDENED VAULT
- IN ARGON GLOVE BOX
- TWO DOUBLE-WALLED URANIUM BED GENERATORS WITH ONE SPARE WITH COOLING (ARGON) AND HEATING (ELECTRICAL)
- URANIUM GOOD GETTER, RELEASES TRITIUM AT REASONABLE TEMPERATURE 400°C, BUT PYROPHORIC
- DOUBLE-WALLED PIPES, VACUUM IN MIDDLE ANNULUS

POOR ORIGINAL

A-154

1374 284

TRITIUM CLEANUP SYSTEMS

- DESIGNED FOR MOST INTENSE CONDITIONS
- LOCATED IN HARDENED VAULT
- SUPPLY CLEANUP — 10 CFM
- TORUS CLEANUP — 50 CFM
- VAULT CLEANUP — 1000 CFM
- ALL BASED ON CATALYTIC RECOMBINING OF TRITIUM TO WATER FORM FOLLOWED BY ABSORPTION ON MOLECULAR SIEVE BEDS — 99.9% CLEANUP
- REDUNDANCY AND CROSS CONNECTING FEATURES
- INITIATION AND CONTROL AUTOMATIC AND INDEPENDENT OF COMPUTER. ALSO MANUAL CONTROL OUTSIDE TRITIUM SEAL
- SEPARATE TEST CELL CLEANUP — 46,000 CFM BASED ON STEAM SPARGING FOLLOWED BY CONDENSING ON AIR CONDITIONING COILS — 90% CLEANUP OF HTO

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1574 285

POWER SYSTEMS

- MOTOR GENERATORS — BELOW GRADE, CONVENTIONAL WATER TURBINE CONSTRUCTION, LUBE OIL COOLING CRITICAL
- RECTIFIER/INVERTERS — CONVENTIONAL DESIGN BUT LARGE, 600 MW
  - SEND ENERGY BACK TO MG'S
  - SHORT OUT COILS THROUGH RECTIFIERS AND CURRENT DECAYS
  - CROWBAR COILS AND CURRENT DECAYS

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1374 286

STANDBY POWER

- **BATTERIES**
  - MAINTAIN BREAKERS ENERGIZED ONE MINUTE UNTIL START DIESEL
  - PROVIDE COOLING WATER AND LUBE OIL TO MOTOR GENERATORS FOR FOUR HOURS UNTIL STOPPED
  - SAFETY COMPUTER
  - TRITIUM MONITORING
- **DIESEL (ONE)**
  - MAINTAIN SYSTEMS (VACUUM, CRYOGENIC) FOR QUICK START
  - RUN CLEANUP SYSTEMS IF ACCIDENT HAS OCCURRED
  - IF WILL NOT START, BUTTON UP FACILITY

POOR ORIGINAL

A-157

1374 297

COMPUTER

- 13 INTERCONNECTED MINI COMPUTERS
- CONTROLS ALL OPERATIONS (8,000 CONTROL POINTS)
- COLLECTS DATA (4,500 CHANNELS)
- PROCESSES DATA
- SAFETY COMPUTER MONITORS KEY SAFETY PARAMETERS, INITIATES ALARMS, INITIATES SAFETY ACTIONS (ALL BACKED UP BY SEPARATE HARDWIRE CHANNELS)

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1374 288

WASTE SYSTEMS

- COOLING WATER PURIFICATION LOOPS — RESINS, FILTERS DISPOSED AS SOLIDS. TRITIUM CONTAINING SOLIDS ALSO

- WASTE TANKS (THREE 1,500 GAL.) DUMP WATER TO BEE BROOK/DEVILS BROOK/ MILLSTONE RIVER AT 10% OF MPC.

- CHEMICAL WASTES (COOLING TOWER BLOWDOWN) PER STATE LAWS

A-159

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PSAR CONTENTS

- SITE DESCRIPTION
  - GEOGRAPHY AND DEMOGRAPHY
  - METEOROLOGY
  - HYDROLOGY
  - GEOLOGY AND SEISMOLOGY
- FACILITIES AND SYSTEMS
- WASTE MANAGEMENT
- HEALTH/SAFETY PROGRAM
- MANAGEMENT CONTROLS FOR OPERATION
- RESEARCH AND DEVELOPMENT
- QUALITY ASSURANCE
- ACCIDENT ANALYSIS

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1374 290

SYSTEMS DISCUSSED IN THE TFTR PSAR

TOKAMAK
- VACUUM VESSEL
- FIELD COILS
- STRUCTURE
- SHELDING

GAS DELIVERY
- TRITIUM
- NON TRITIUM

NEUTRAL BEAM

VACUUM PUMPING
- TORUS
- NEUTRAL BEAMS

CRYOGENIC SUPPLY
- NITROGEN
- HELIUM

REMOTE MAINTENANCE

DIAGNOSTICS

AUXILIARIES
- HVAC
- TRITIUM CLEANUP
- FIRE PROTECTION
- COOLING WATER
- COMMUNICATIONS
- LIGHTING

I & C
- CICADA
- SAFETY & PROTECTION
- MONITORS

ELECTRICAL
- ELECTRICAL POWER
- SUPPLY & DISTRIBUTION
- PULSED ENERGY CONVERSION
- EMERGENCY POWER
- INTERLOCKS

WASTE MANAGEMENT
- LIQUID
- GASEOUS
- SOLID

FOR EACH OF THE ABOVE SYSTEMS THE FOLLOWING ITEMS WERE DISCUSSED:

(A) FUNCTIONAL REQUIREMENTS FOR NORMAL AND OFF-NORMAL OPERATING CONDITIONS
(B) DESIGN BASES TO MEET FUNCTIONAL REQUIREMENTS
(C) SYSTEM DESCRIPTION
(D) SAFETY EVALUATION OF DESIGN
(E) PRE-OPERATIONAL TESTS AND INSPECTIONS
(F) REQUIREMENTS AND PROVISIONS FOR SURVEILLANCE AND PREVENTATIVE MAINTENANCE

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REVIEW OF PSAR

- DRAFT PSAR PREPARED BY EBASCO/GRUMMAN
- REVIEWED AND ISSUED BY PRINCETON — 8/77
- REVIEWED BY
  - DOE CHICAGO OPERATIONS OFFICE SAFETY DIVISION
  - DOE OFFICE OF FUSION ENERGY
  - DOE DIVISION OF OPERATIONAL SAFETY
  - W. STRATTON WITH LASL AND UNIVERSITY OF WISCONSIN
- FORMAL DOE COMMENTS TO PRINCETON — 10/77
- DOE/PRINCETON MEETING TO DISCUSS COMMENTS — 11/77
- PRINCETON FORMAL RESOLUTION OF MOST COMMENTS — 12/77
- DOE AUTHORIZATION TO PROCEED WITH CONSTRUCTION — 1/78
- DOE AGREEMENT TO PRINT PSAR AND RESOLVE REMAINING COMMENTS IN FSAR — 2/78

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BOTTOM LINE

(ON SITE)

- DURING OPERATION NO ACCESS WITHIN TRITIUM SEAL BOUNDARY. LIMITED (NOT NORMAL) ACCESS TO OTHER BUILDINGS ON SITE. CONTROL ROOM IN LABORATORY/OFFICE BUILDING
- AFTER 100 PULSES, ONE HOUR WAIT TO GAIN ACCESS TO TEST CELL
- REMOTE MAINTENANCE INSIDE IGLOO
- IF TRITIUM RELEASE, NO ACCESS UNTIL CLEANED UP, EXCEPT FOR EMERGENCY WITH TRITIUM SUIT

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BOTTOM LINE

(SITE BOUNDARY DOSE)

- **NORMAL 1,000 PULSE OPERATION**

• NEUTRONS/GAMMA	5.0		
• TRITIUM — 100 CI HTO	1.0		
• ACTIVATED AIR	.4		
TOTAL	6.4	MREM/YR	(0-50 MILES 50 MANREM)

- **OPERATIONAL OCCURRENCE (LARGE LEAK IN TORUS)**

  TRITIUM — 4 CI HTO — .04 MREM (2 MANREM)

- **LOW PROBABILITY ACCIDENT (BREAK IN NEUTRAL BEAM LINE WITH FIRE)**

  TRITIUM — 180 CI HTO — 1.8 MREM (90 MANREM)

- **VERY LOW PROBABILITY ACCIDENT (MASSIVE DESTRUCTION OF TEST CELL WITH FIRE)**

  TRITIUM — 6,200 CI HTO — 2,730 MREM (3,000 MANREM)

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TFTR UPGRADE

- MORE HEATING
- LONGER PULSE, 1.5 SECONDS
- HIGHER FUSION YIELD, Q = 2
- BETTER CAPABILITY TO STUDY PHYSICS OF ALPHA PARTICLES
- SOME POSSIBILITY FOR "LOCAL IGNITION" (ALPHA PARTICLE HEATING OF PLASMA = LOSS OF HEATING DUE TO CONDUCTION AND RADIATION)
- ALL DESIGN AND SAFETY CRITERIA REMAIN UNCHANGED

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Appendix XXII

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POOR ORIGINAL

PRELIMINARY SEQUENCE OF EVENTS (TMI-2, 3/28/79 INCIDENT)

APPENDIX XXII: Preliminary Sequence of Events at TMI-2

The following sequence of events for the TMI-2 incident of 3/28/79 has been formulated by B&W engineers using available plant data. This chronology has been constructed from numerous sources and has not been totally confirmed. It may not be precise in either event occurrence or sequence.

Time, Minutes	Event
Prior to turbine trip	The initiating events could have come from numerous postulated causes. For purposes of this sequence, they are relatively unimportant. The prime effect is that it led to a loss of main feedwater (MFW) booster pumps.
0	Main feedwater pumps are tripped. Almost simultaneously, the turbine trip occurs.
0.10	Pressurizer pressure increases to the EROV setpoint of 2270 psig.
0.15	Secondary side pressure peaks at 1070 psig and is limited by steam relief valves.
0.20 Trip	RC pressure trip setpoint reached (2355 psig at hot leg tap) and system pressure peaks at about this value.
	Indications from pump discharge pressure are that auxiliary feedwater pumps (one turbine driven, two electric) are running at this point; however, no level change occurs in steam generators.
0.25	Pressurizer level peaks at 255 inches (indicated) and starts to decrease with system contraction.
0.30	Quench tank pressure is increasing.
0.50	Pressurizer level is at a minimum of 158 inches and starts to increase. Hot leg temperature is at a minimum of 577°F and starts to increase slowly.
1.0	OTSG level indication on the startup range is 10 inches. OTSG pressure holds at about 1025 psig.
2.0	OTSG pressure starts a steady decrease. HPI flow is initiated by ESFAS on low RC pressure (HPI setpoint = 1600 psig).
3.0	The quench tank's increasing pressure levels off at 120 psig. Relief valve setpoint is 150 psig.
4.75	The hot and cold leg temperatures start increasing at a more rapid rate. Analytical simulation indicates that this occurs when the HPI is turned off. Site information notes that operator terminates HPI fully at 5.1 minutes.

4374 297

Time, Minutes	
5.0	Pressurizer level indicates a slowing and then continues to increase as the hot leg temperature is increasing.
6.0	Pressurizer level indicates a full pressurizer and the quench tank pressure increases beyond the relief valve setpoint of 150 psig.
	RC pressure reaches a minimum of 1350 psig with a hot leg temperature of 584°F. This indicates hot leg is in saturation condition.
8.0	Auxiliary feedwater flow is initiated to both OTSG's. This is indicated by immediate OTSG repressurization to ~1025 psig and OTSG level change.
9.0	RC pressure peaks out at 1900 psig and starts to decrease. Hot leg temperature peaks out at 597°F.
11.0	Pressurizer level indication is restored. It stabilizes out at 375 inches at 15 minutes.
16.0	Quench tank pressure drops suddenly, indicating the rupture disk has blown (setpoint = 200 ± 25 psig).
18.0	The decreasing RC pressure stabilizes at 1115 psig.
22.0	The RCS temperature stabilizes at a hot leg of 553°F and a cold leg of 548°F. The temperature decrease from start of auxiliary feedwater to this stabilization represents a 200°F/hr cooldown. Reactor building pressure is 1.6 psig and increasing. Two feet level is restored in both OTSG's.
50.0	The startup level indication shows OTSG B level increasing and OTSG A level decreasing. Pressure increases in both OT[illegible]
60.0	During the 22-60 minute period, the system parameters have stabilized in the saturation condition of a pressure of ~1[illegible] psig, temperature of ~550°F. RC flow indication is decreas[illegible] from 69 (initial) to 50 x 10^6 lb/hr. The reactor building pressure is 2.2 psig and increasing.
73.0	Two RC pumps are tripped (in Loop B). Reactor coolant flow rate decreases in Loop B.
78.0	OTSG B pressure drops from 950 psig to 140 psig in 18 minut[illegible]
80.0	T_{hot} follows T_{sat}. ΔT across the core equals about 5°F.
100.0	Both remaining RC pumps are tripped.
114.0-120.0	T_{hot} and T_{cold} diverge rapidly. T_{hot} > 620°F in less than 15 minutes.
132.0	Site information notes that EMOV relief line was isolated initially. RC pressure starts decreasing more rapidly.

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POOR ORIGINAL

Time, Minutes	Event
139.0	RCS has depressurized to 670 psig and RCS hot leg temperature is at maximum scale of 620°F. At 620°F, system would have superheating at upper elevations as long as pressure was below saturation pressure of 1772 psig.
	RCS shows rapid re-pressurization.
159.0	OTSG B level ramped up from 5% to 65% in 48 minutes.
180.0	OTSG B main steam isolation valves and turbine bypass valves are closed. RCS pressure peaks at 2120 psig.
180.0-204.0	Regulation by EMOV block valve reduces RCS pressure.
204.0	HPI comes on (1600 psig signal).
216.0m	HPI pump 1c to Loop A turned off. RC pressure decreases stepwise. RB pressure increases stepwise.
290.0m (4.83 hr)	RB pressure hits 4 psig. Building fan cooler comes on.
318.0m (5.3 hr)	RCS pressure increases rapidly from 1250 to 2120 psig in 35 minutes. The EMOV block valve is closed, one HPI (1A) is on.
354.0 (5.9 hr)	OTSG A level is ramped up from 50% to 95% on operating range in 1 hour and to 100% in 1.5 hour. OTSG A pressure starts to decrease toward zero.
450.0 (7.5 hr)	The EMOV block valve is opened. RCS pressure starts to decrease (2050 psig to 480 psig in 1 hr, 45 min).
519.0 (8.65 hr)	RC system pressure reaches 600 psig, core flood tank setpoint.
588.0 (9.8 hr)	RB pressure spike to 28 psig occurs.
630.0 (10.5 hr)	T_{hot} Loop A reappears on scale, decreases to 525°F in 1/2 hr.
678.0 (11.3 hr)	T_{cold} Loop A increases in about 5 minutes from 190°F to 400°F.
750.0 (12.5 hr)	HPI flow increased to 400 gpm. T_{hot} in Loop A decreases.
810.0 (13.5 hr)	T_{cold} Loop A decreases.
948.0 (15.8 hr)	Pump 1A is started.

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Hrs., Minutes	Event
Thereafter	Condenser vacuum re-established. SG A begins steaming to condenser. RCS cooled to approximately [illegible]°F, 1000 psi. Letdown line cooled to permit flow and relief valve being reseated (estimated 14-16 gpm flow). Some fuel incore thermocouples reading about [illegible]°F. RB pressure below 1 psi. High radiation in reactor containment and auxiliary building.

POOR ORIGINAL

1374 300

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Appendix XXIII

APPENDIX XXIII: Areas for Improvement following TMI-2 Accident

LIST A

I. INSTRUMENTATION

Core liquid level

Containment Isolation Signal (waste water storage/transfer)

II. BASIC DESIGN

Further study of anomalous transients

IV. ADMINISTRATION

Instrumentation to follow the course of an accident

V. EMERGENCY PLANNING

Emergency equipment (aux. building vent)

contingency plans and arrangements for further degradation of affected unit

1374 302

LIST B

I. INSTRUMENTATION

Relief valve position indication

II. BASIC DESIGN

Cause of initial feedwater loss

High point and reactor vessel vents (remotely operated)

III. RESEARCH

Assess stress on instruments & electrical insulation

Decontamination and recovery

IV. ADMINISTRATION

Special procedures and operator training

Onsite inspectors and instructions

V. COMMUNICATIONS

Vendor-licensee emergency contact - dedicated phone lines

Notification of state and Federal officials

VI. EMERGENCY PLANNING

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Cleanup equipment

LIST C

I. INSTRUMENTATION

Qualification of auxiliary building instrumentation and apparatus

II. BASIC DESIGN

RHR System and auxiliary building

Loop seal on pressurizer

Further study of anomalous transients

Qualification of equipment to tolerate accident environment (both protection and process control)

Containment spray additives

Use of normal process equipment during emergency conditions

III. RESEARCH

Path of core melt

Path of fission product release

Reliability of existing engr'd safeguards

Site hydrological criteria

Computer assistance to operators (long-term online prediction)

Improved containment/core catcher

IV. ADMINISTRATION

Startup check list/operational

Reserve support to site

Special procedures and operator training

(Continued)

LIST C - continued

VI. EMERGENCY PLANNING

Evacuation

KI pills

Contingency arrangements in general

1374 305

LIST D

III. RESEARCH

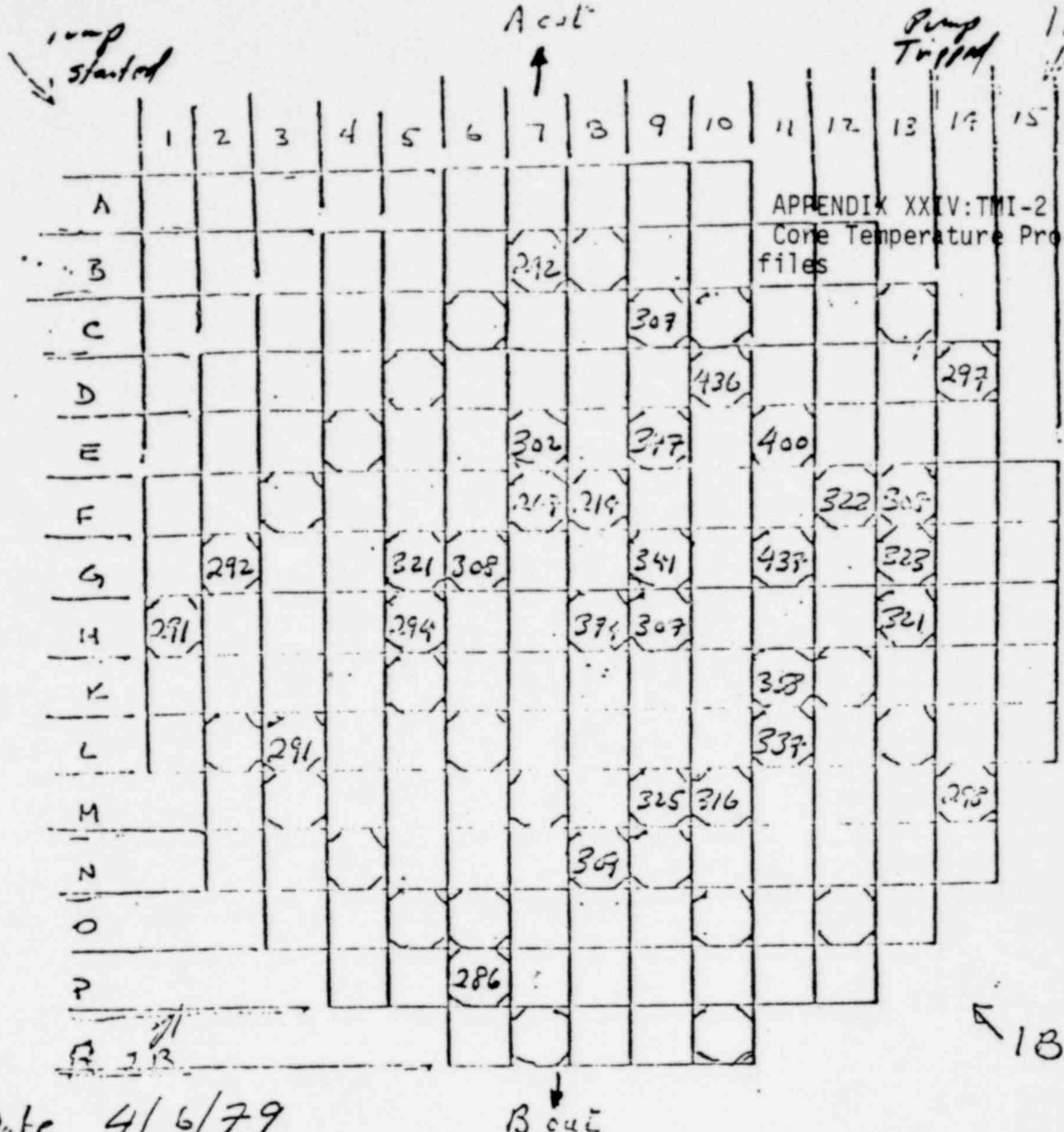
Reliability of existing engr'd safeguards

Computer assistance to operators (long term online prediction)

1374 307

Appendix XXIV

APPENDIX XXIV: TMI-2 Core Temperature Profiles



Date 4/6/79

Time 1000

RCS Pressure 1125 psig

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Before RCS pump trip

POOR ORIGINAL

Inlet 285 F

Outlet 282 F

Pressure 560 F

P_{cont} -1.2 psig

Pressurizer 242"

T_{cont} 83

LOCATION OF FUEL ASSEMBLIES CONTAINING BURNABLE POISON RODS

THREE MILE ISLAND NUCLEAR STATION UNIT

FIGURE 4

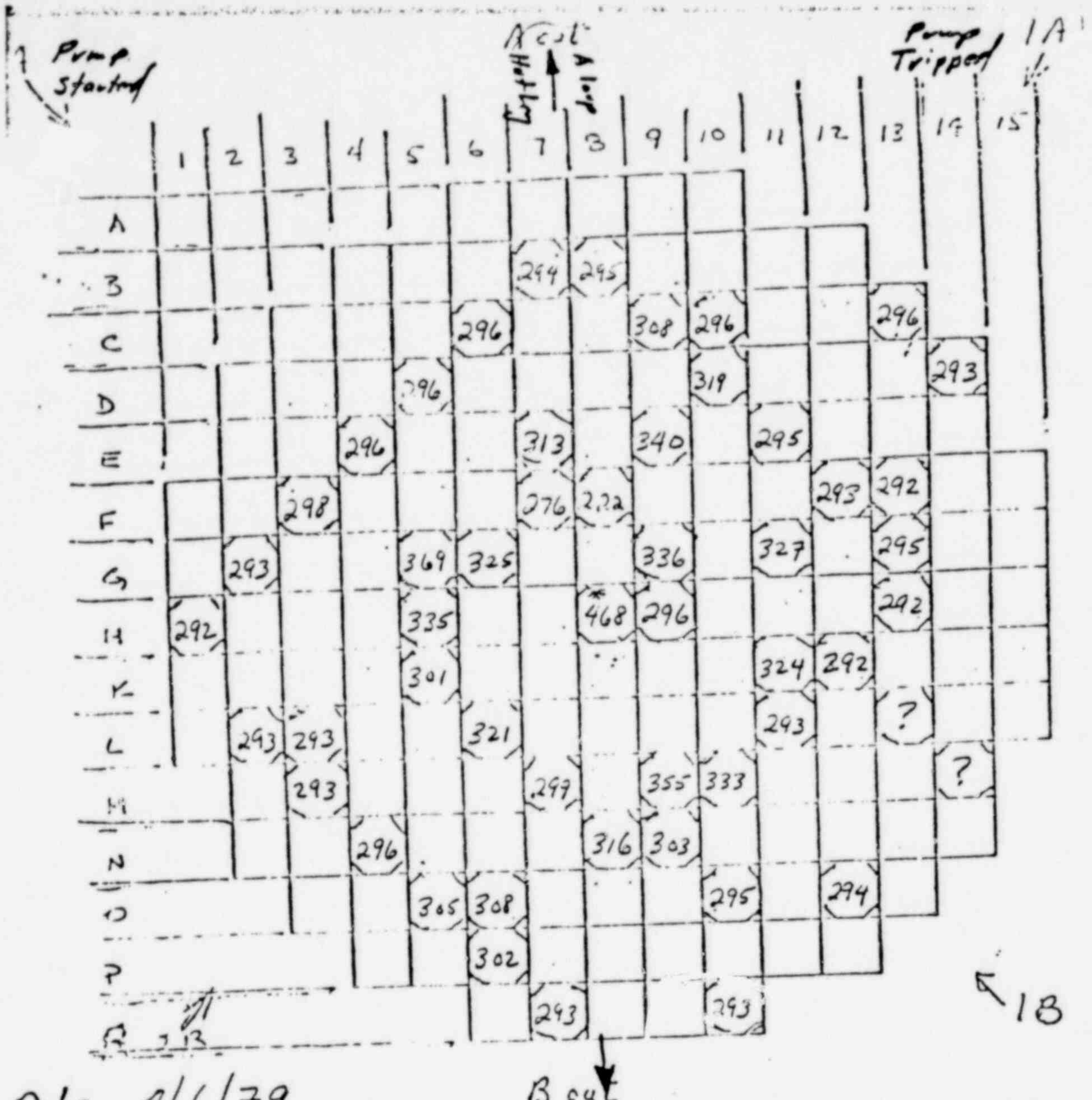
Press SGA 16 L_A 355"

Press SGB 14 L_B 303" Hydrogen Conc Inline: ______

Letdown Flow ______ Recombiner ______

CONTAINMENT:

Pressure -1.2 psig Temp. 83F Level of Water ______



Date 4/6/79
Time 1447

* during transient this TC reading varied widely it has steadied out but may be suspect.

? no reading

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...ess. ...ressure ______

... inlet ______

... inlet ______

...ressure ______

Pressurizer ______

...ress SGA ______

...ress SGB ______

...e flown Flow ______

After RCS pump trip

LOCATION OF FUEL ASSEMBLIES CONTAIN BURNABLE POISON RODS

THREE MILE ISLAND NUCLEAR STATION UN

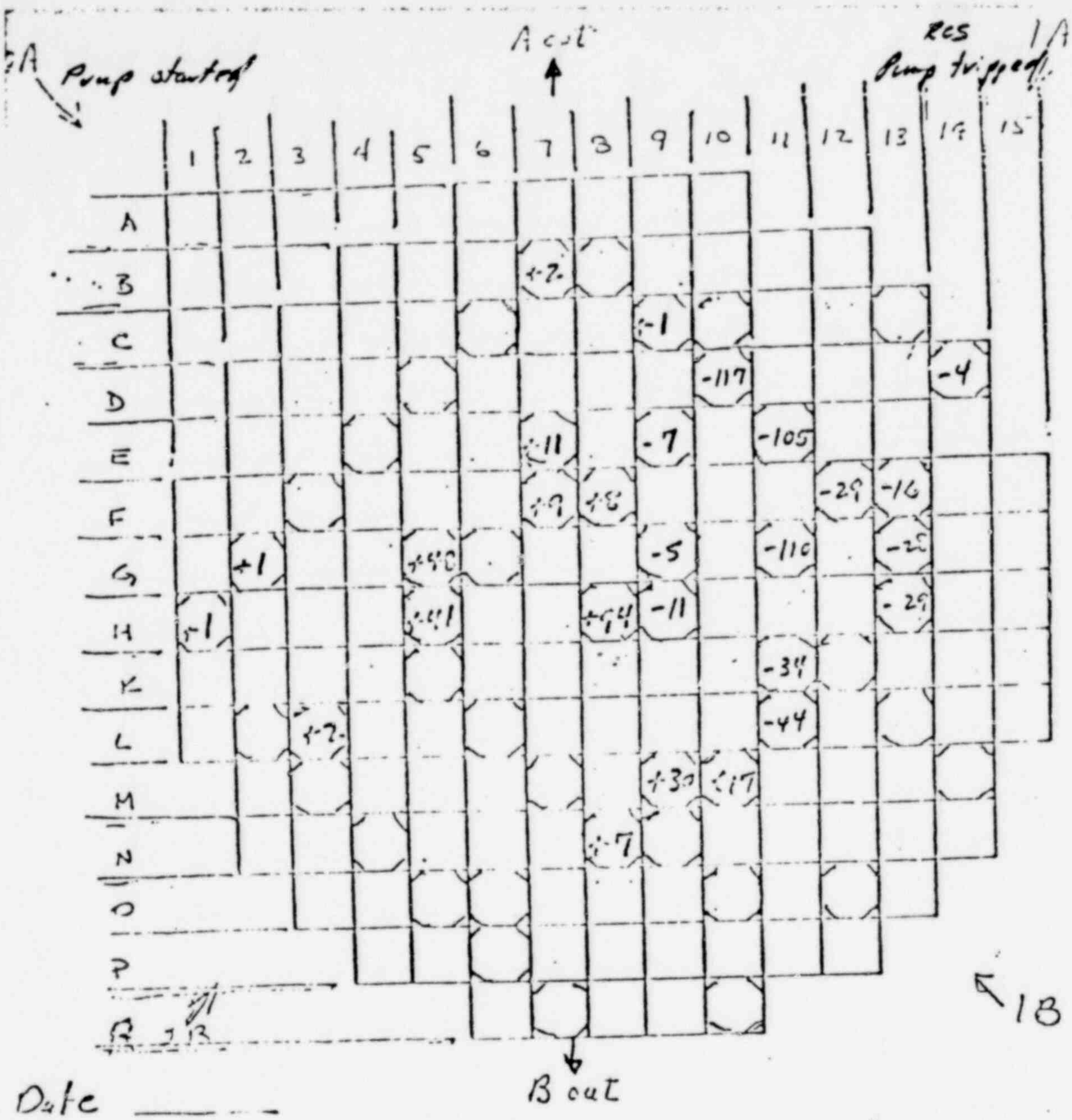
FIGURE 4

Hydrogen Conc Inline: ______

Recombiner ______

CONTAINMENT:

Pressure ______ Temp. ______ Level of Water ______



Date ______

Time ______

RCS. Pressure ______

ΔT (Readings From 1000 → 1147)

4/6/79

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Changes due to pump trip

Inlet ______

Inlet ______

Pressure ______

Pressurizer ______

Press SGA ______

Press SGB ______

Letdown Flow ______

LOCATION OF FUEL ASSEMBLIES CONTAIN BURNABLE POISON RODS

THREE MILE ISLAND NUCLEAR STATION UNI

FIGURE 4

Hydrogen Conc Inline: ______

Recombiner ______

CONTAINMENT:

Pressure ______ Temp. ______ Level of Water ______

Appendix XXV

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

April 10, 1979

APPENDIX XXV: Regulatory Guides

Mr. Lee V. Gossick
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS ACTION ON PROPOSED REVISIONS OF REGULATORY GUIDES

Dear Mr. Gossick:

During its 228th meeting, April 5-7, 1979, the ACRS concurred in the regulatory position of Regulatory Guide 1.140, Revision 1, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Sincerely,

Max W. Carbon
Chairman

cc: H. Denton, NRR
R. Minogue, OSD
G. Arlotto, OSD
S. J. Chilk, SECY

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bcc: ACRS Members
J. Jacobs
H. Voress

Appendix XXVI

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APPENDIX XXVI: Schedule for ACRS Report to Commissioners on RSR Budget

March 8, 1979

ALTERNATE C - Proposed RSR Review Process

	NRC Budget Process (for FY-80 Budget)	ACRS Report Preparation (1978 Report
Dec 30, 1978	ACRS 1978 RSR report to Congress	
April 1, 1979	Report by RES re implementation of ACRS recommendations	
May/June		ACRS review proposed implementation of ACRS recommendations by NRC in 1980 budget
July 1979	RES submits proposed FY-81 budget to Commissioners	ACRS submits report to NRC re proposed implemenation of ACRS recommendations
July/Aug 1979	Commissioners review budget proposals	ACRS conduct overall review of proposed NRC RSR Program
Sept 1979	NRC submits budget to OMB for review	
Oct/Nov/Dec 1979	OMB Reviews NRC budget RES will keep ACRS informed of proposed changes	
Dec 1979		ACRS Annual Report to Congress on NRC RSR program
Jan 1980	Administrations proposed budget to Congress	
May 1980	Authorization of FY-81 programs by Congress	
"Sept" 1980	Appropriations approved by Congress for FY-81 budget	

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Appendix XXVII

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

April 7, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON RECENT ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Hendrie

During its 228th meeting, April 5-7, 1979, the Advisory Committee on Reactor Safeguards reviewed the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2. During this review, the Committee had the benefit of discussions with the NRC Staff.

Our study of the accident at Three Mile Island has shown that it is very difficult for a PWR plant operator to understand and properly control the course of an accident involving a small break in the reactor coolant system accompanied by other abnormal conditions.

The Committee recommends that further analyses be made, as soon as possible, of transients and accidents in PWRs that involve initially, or at some time during their course, a small break in the primary system. The computer codes used for these analyses should be capable of predicting the conditions observed during the accident at Three Mile Island, including thermal-hydraulic effects and clad and fuel temperatures. The range of break sizes considered should include the smallest that could be deemed significant, and should consider a range of break locations.

The Committee believes that the analyses recommended above will demonstrate, as has the accident at Three Mile Island, that additional information regarding the status of the system will be needed in order for the plant operator to follow the course of an accident and thus be able to respond in an appropriate manner. As a minimum, and in the interim, it would be prudent to consider expeditiously the provision

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of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. Early consideration should be given also to providing remotely controlled means for venting high points in the reactor system, as practical.

The foregoing recommendations apply to all pressurized water reactors.

The recommendations in IE Bulletin 79-05A, dated April 5, 1979, are believed to be generally suitable for Babcock and Wilcox facilities, on an interim basis. However, the Committee believes that the actions listed in Item 4b. under the heading, "Actions To Be Taken by Licensees," may prove to be unduly prescriptive in view of the uncertainties in predicting the course of anomalous transients or accidents involving small breaks in the primary system.

With regard to Three Mile Island Unit 2, the Committee believes that decisions should be made expeditiously with regard to contingency measures which may be prudent concerning containment and reactor cooldown as a backup to the currently planned cooldown procedure.

The Committee is continuing its review of these and other concerns arising from this accident and will provide further advice as it is developed.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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Appendix XXVIII

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

April 23, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Hendrie:

Reference: Memo to Commission from James L. Kelley, Deputy General Counsel, "ACRS Consultants as Witnesses in Hearings: Provision of Counsel" dtd. 3/12/79

I have been provided with the referenced memo noted above regarding legal support for ACRS consultants who are subpoenaed as witnesses at NRC hearings and would like to offer the following comments.

Contrary to Mr. Kelley's view, the ACRS believes strongly that legal support should be provided for any subpoenaed consultants at both depositions and hearings. These consultants are appearing as a result of their work for the ACRS, and we must back them up by providing such support. Further, in addition to the aspect of fulfilling our obligations, it seems likely that we will lose some of these valued people if we don't provide this support. Ms. Nordlinger commented that Drs. Trifunac and Luco "evidenced great apprehension," expressed anxiety," found the experience of being deposed "grueling and unpleasant," and felt that they were "almost badgered" to give simplified answers. I believe we are surely asking too much to expect that a technical consultant should, in effect, act as his own lawyer based on an information sheet and long distance phone calls as recommended by Mr. Kelley. This is especially so when all other parties including NRC Staff consultants will be represented by legal counsel.

In this connection, the cost of providing such support should be secondary, within reasonable limits. This cost is unlikely to represent a heavy burden on NRC resources, however, since the consultants will presumably only be called in "exceptional circumstances."

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We also consider it important that a lawyer be present to ensure that the consultants are not probed regarding "the reasoning process of the collegial ACRS report." Reliance on the judicial record, as proposed by the General Counsel, is undesirable since a specific hearing board or the parties involved in a deposition may not be familiar with this judicial history or may not consider the history applicable to the particular hearing at hand.

In summary, we request that ACRS consultants subpoenaed to appear at NRC depositions and hearings be provided legal counsel when they are presenting testimony based on work done for the ACRS.

Sincerely,

Max W Carbon

Max W. Carbon
Chairman

cc:
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne
L. Bickwit, OGC
S. Chilk, Secretary to Comm.

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Appendix XXIX

APPENDIX XXIX

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, D. Thompson, Executive Officer for Operations Support, IE to D. B. Vassallo, Assistant Director for LWRs, NRR, Information for Board Notification - Davis-Besse Units 2 & 3 and Midland Units 1 & 2, dated March 1, 1979, and Attachments.

2. Letter, Dr. Schnurer, Dept. of Interior, FRG to J. D. Lafleur, Deputy Director, Office of International and State Programs, Information Exchange Discussions, dated February 19, 1979, and enclosures.

3. Report, C. Michelson, Decay Heat Removal During a Very Small Break LOCA for a B&W 205-Fuel-Assembly PWR, January 1978.

4. Regulatory Guide 1.140 (Rev. 1), Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.

5. I&E Bulletin 79-05, Nuclear Incident at Three Mile Island - Supplement, dated April 5, 1979.

6. I&E Bulletin 79-05A, Nuclear Incident at Three Mile Island - Supplement, dated April 5, 1979.

7. PNO-79-67K, Preliminary Notification of Event or Unusual Occurrence, Nuclear Incident at Three Mile Island, dated April 5, 1979.

8. Memorandum, J. H. Bickel, ACRS Fellow to ACRS, Subject: Three Mile Island Unit 2 Incident and a Quick Comparison with the WASH-1400 Evaluation, dated April 4, 1979.

9. Highlights, Three Mile Island Subcommittee Meeting Minutes, Washington, DC, April 4, 1979.

10. Preliminary Notification of Event on Unusual Occurrence, PNO-79-67 A through J.

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APPENDIX XXIX

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

11. Summary of the March 12, 1979 Meeting of the Subcommittee on Sequoyah Nuclear Power Plant, Units 1 and 2.

12. Executive Summary and Conclusion of the Zion Systems Interaction Study performed by Fluor Pioneer, Inc.

13. Paper, R. E. Alexander, NRC Staff, Regulatory Strategy for Reducing Occupational radiation Risks in the Nuclear Fuel Cycle.

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