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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
Revised: May 26, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
266TH ACRS MEETING
June 3-5, 1982
WASHINGTON, DC

Thursday, June 3, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. — ACRS Chairman's Report (Open)
- 1.1) Opening Statement
- 1.2) Items of interest regarding ACRS activities

2) 8:45 A.M. - 12:45 P.M. — Reactor Pressure Vessel Integrity (Open)
- 2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee on Metal Components (MB/EI)
- 2.2) 9:15 A.M.-12:45 P.M.: Meeting with NRC Staff and representatives of the nuclear industry

12:45 P.M. - 1:45 P.M. — LUNCH

3) 1:45 P.M. - 3:45 P.M. — Quantitative Safety Goals (Open)
- 3.1) Discuss proposed ACRS report to NRC regarding NUREG-0880, Safety Goals for Nuclear Power Plants: A Discussion Paper (DO/JMG/GRQ)

4) 3:45 P.M. - 5:45 P.M. — Reactor Safety Research (Open)
- 4.1) Discuss proposed ACRS report to NRC regarding the proposed NRC Safety Research Budget for FY 1984-85 and the long-range aspects of the "out-years" (1986-88) (CPS/et al/SD)

Portions of this session will be closed as necessary to discuss information the premature release of which would be likely to significantly frustrate the performance of the Committee's statutory function.

5) 5:45 P.M. - 6:15 P.M.	5.1) Report by C. P. Siess regarding 5/18/82 hearing on NRC Safety Research Program (CPS/SD)

Page Revised

Friday, June 4, 1982, Room 1046, 1717 H Street, NW, Washington, DC

6) 8:30 A.M. - 12:30 P.M.	Midland Plant Units 1 and 2 (Open) 6.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee (DO/DCF) 6.2) 9:00 A.M.-12:30 P.M.: Meeting with NRC Staff and Applicant Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter.
12:30 P.M. - 1:30 P.M.	LUNCH
7) 1:30 P.M. - 2:00 P.M.	Discuss Items for Meeting with NRC Commissioners (Open) 7.1) Disuss the following topics for the meeting with the NRC Commissioners . "Thermal Shock" of Reactor Pressure Vessels - The ACRS Acting Subcommittee Chairman (M. Bender) will provide a brief status report of activities related to the Committee's review and evaluation of the proposed NRC Staff plan of action to resolve this issue (see memo from NRC Chairman Palladino to Dr.P.G. Shewmon, ACRS Chairman, dated 3/25/82) . Quantitative Safety Goals - The ACRS Subcommittee Chairman (D. Okrent) will provide a status report regarding activities related to the ACRS review and development of comments regarding NUREG-0880, Safety Goals for Nuclear Power Plants, A Discussion Paper, dated 2/82 . Proposed ACRS Review of the Clinch River Breeder Reactor - The ACRS Subcommittee Chairman (M.W.Carbon) will make a brief presentation regarding the anticipated scope of and schedule for ACRS review of the CRBR

- Reactor Pressure Vessel Liquid Level/Inventory Instrumentation - The ACRS Subcommittee Chairman (William Kerr) will present a brief summary of the Committee's report dated 4/6/82 regarding this topic. Members of the Committee will be prepared to respond to questions regarding the Committee's comments and recommendations
- Proposed NRC Long-Range Research Program Plan - The ACRS Subcommittee Chairman (C.P. Siess) will present a brief summary of the Committee's report dated 4/5/82 regarding the Draft NRC Long-Range Research Plan for FY 1984-88 dtd. 3/15/82. Members of the Committee will be prepared to respond to questions the Commissioners may have regarding this matter.

8) 2:00 P.M. - 3:30 P.M. — Meeting with NRC Commissioners (Open)
  - 8.1) Meeting with NRC Commissioners to discuss items noted above

9) 3:30 P.M. - 4:45 P.M. — Future ACRS Activities
  - 9.1) Anticipated Subcommittee activities (PGS/MWL)
  - 9.2) Proposed ACRS activities (PGS/RFF)
  - 9.3) 3:45 P.M.-4:15 P.M.: Report by Dr. Moeller regarding consideration of seismic events in emergency planning (DWM/HA)
  - 9.4) 4:15 P.M.-4:45 P.M.: Report by Dr. Moeller regarding control room habitability in nuclear plants (DWM/HA)

10) 4:45 P.M. - 6:30 P.M. — Quantitative Safety Goals (Open)
  - 10.1) Discuss proposed ACRS report to NRC regarding NUREG-0880, Safety Goals for Nuclear Power Plants, A Discussion Paper (DO/JMG/GRQ)

Page Revised (5/24/82)

Saturday, June 5, 1982, Room 1046, 1717 H Street, NW, Washington, DC

11) 8:30 A.M. - 12:30 P.M.	Preparation of ACRS Reports (Open/Closed) 11.1) Discuss proposed ACRS reports to NRC regarding: 11.1-1) 8:30 A.M.-9:30 A.M.: Thermal Shock of Reactor Pressure Vessels (MB/EI) 11.1-2) 9:30 A.M. - 10:30 A.M.: Midland Plant Units 1 & 2 (DO/DCF) 11.1-3) 10:30 A.M.-12:30 P.M.: Quantitative Safety Goals (DO/JMG/GRQ) Portions of this session will be closed as necessary to discuss Proprietary Information and information that will be involved in an adjudicatory proceeding.
12:30 P.M. - 1:30 P.M.	LUNCH
12) 1:30 P.M. - 3:30 P.M.	Complete ACRS reports to NRC (Open) 12.1) Discuss proposed reply to Commissioner Gilinsky's inquiry regarding seismic methodology proposed by Dr. P. Jennings (DO/RS) 12.2) Complete discussion of reports noted above Portions of this session will be closed as necessary to discuss Proprietary Information and information that will be involved in an adjudicatory proceeding.

proposed rule will modify 10 CFR 50.34 (contents of applications: technical information) and contains the basic requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements".

**Metal Components*, June 7, 1982, Palo Alto, CA. The Subcommittee will be given a status report by nuclear reactor Steam Generator Owners Group on research results and any changes made in steam generator design/operation. Three Mile Island Unit 1 steam generator problems will also be discussed.

**Waste Management*, June 8, 1982, Washington, DC. The Subcommittee will review and comment on the Department of Energy's Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment; provide input for the Waste Management Chapter of the FY 1984 and FY 1985 Safety Research Program Review; review NRC Staff waste management activities; and discuss advances in waste management practices.

**Emergency Core Cooling Systems (ECCS)*, June 16 and 17, 1982, Idaho Falls, ID. The Subcommittee will discuss General Electric Company's request for a change in 10 CFR Part 100, Appendix K requirements, the NRC Staff's code audit capability, the LOFT ATWS test Nuclear Steam Supply System vendor code predictions and results, and the NRC work on operator accident guidelines and procedures.

**Reactor Radiological Effects*, June 23, 1982, Washington, DC. The Subcommittee will discuss NRC Staff proposed revision to 10 CFR 20 and the use of potassium iodide for thyroid blocking in the event of a radiation accident.

**Washington Public Power Supply System Unit 2 (WPPSS)*, June 23 and 24, 1982, Hanford, WA. The Subcommittee will continue the review of the application of Washington Public Power Supply System for an operating license for the WPPSS Nuclear Project Unit 2.

**Clinch River Breeder Reactor (CRBR) and Site Suitability*, June 24 and 25, 1982, Washington, DC. The Subcommittee will continue the site suitability review for the Clinch River Breeder Reactor.

**Perry Nuclear Power Plant Units 1 and 2*, June 28 and 29, 1982, Cleveland, OH. The Subcommittee will continue the review of the application of Cleveland Electric Illuminating Company for an operating license for the Perry Nuclear Power Plant Units 1 and 2.

**Systematic Evaluation Program*, June 30, 1982, Washington, DC. The Subcommittee will review the completion of the Systematic Evaluation Program review on Ginna.

**Grand Gulf Unit 1*, July 1, 1982, Washington, DC. The Subcommittee will continue the review of the Mississippi Power Company application for an operating license for Grand Gulf Unit 1.

**Extreme External Phenomena*, July 1, 1982, Washington, DC. The Subcommittee will review the Office of Nuclear Regulatory Research proposed FY 1984 and FY 1985 research funding and programs in this area for the Long-Range Research Plan.

**Reliability and Probabilistic Assessment*, July 1, 1982, Washington, DC. The Subcommittee will review the Office of Nuclear Regulatory Research proposed FY 1984 and FY 1985 research funding and programs for the Systems and Reliability Analysis (SARA) decision unit.

**Regulatory Activities*, July 6, 1982 (Tentative), Washington, DC. The Subcommittee will review proposed Regulatory Guides and Regulations.

**Safety Research Program*, July 7, 1982, Washington, DC. The Subcommittee will continue its review of the NRC Safety Research Program and budget for FY 1984 and FY 1985.

**Reactor Operations*, July 21 or 22, 1982, Washington, DC. The Subcommittee plans to discuss NRC's enforcement policy, the Inspection and Enforcement (IE) performance apprisal team inspection program and the IE regionalization program.

**Watts Bar*, Date to be determined (July), Washington, DC. The Subcommittee will continue the review of the application of Tennessee Valley Authority for an operating license for the Watts Bar Nuclear Power Plant Units 1 and 2.

**Safety Research Program*, August 11, 1982, Washington, DC. The Subcommittee will provide early input to the RES Staff for their preparation of the Long-Range Research Plan for FY 1985 through FY 1989.

**Transportation of Radioactive Materials*, Date and location to be determined. The Subcommittee will continue its review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.

**Metal Components*, Date to be determined, Washington, DC. The Subcommittee will continue the review of pressurized thermal shock.

ACRS Full Committee Meeting

June 3–5, 1982: Items are *tentatively* scheduled.

**A. Midland Nuclear Plant*—Operating license.

**B. Quantitative Safety Goals*—Proposed NRC policy regard Quantitative Safety Goals for Nuclear Power Plants (NUREG-0880).

**C. Reactor Safety Research*—Proposed NRC Safety Research budget for FY 1984 and FY 1985.

**D. Reactor Pressure Vessel Integrity*—Proposed NRC action plan to resolve concerns regarding repressurization of reactor pressure vessels following rapid cooldown transients.

**E. NRC Regulations*—Proposed NRC regulations regarding safety related matters including Application of TMI-2 Lessons Learned to Operating Reactors (10 CFR 50.34); Applicability of License Conditions and Technical Specifications in an Emergency (10 CFR 50.54/50.72); Accreditation of Testing Organizations (10 CFR 50.49(a)); and Evaluation of Alternate Decay Heat Removal Systems (Task Action Plan A-45).

**F. ACRS Subcommittee Activities*—Discuss the status of designated ACRS Subcommittee activities regarding safety related matters including consideration of seismic events in emergency planning; and proposed changes in seismic design methodology.

**G. Meeting with NRC Commissioners (Tentative)*—Discuss ACRS activities regarding quantitative safety goals for nuclear power plants, integrity of reactor pressure vessels, ACRS plans for review of the CRBR, the NRC Long-Range Research Program Plan, and instrumentation for detection of inadequate core cooling.

**H. Three Mile Island Nuclear Plant Unit No. 1*—Briefing regarding causes of and status of steam generator tube damage.

July 8–10, 1982: Agenda to be announced.

August 12–14, 1982: Agenda to be announced.

Dated: May 14, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-13661 Filed 5-18-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-313]

Arkansas Power & Light Co. (Arkansas Nuclear One, Unit 1); Exemption

I

The Arkansas Power and Light Company (the licensee) is the holder of Facility Operating License No. DPR-51, which authorizes operation of Arkansas Nuclear One, Unit No. 1. This license provides, among other things, that it is subject to all rules, regulations and

Issue Date:

October 8, 1982

MINUTES OF THE
266TH ACRS MEETING
JUNE 3-5, 1982
WASHINGTON, DC

CERTIFIED

The 266th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Chairman P. Shewmon at 8:30 a.m., Thursday, June 3, 1982.

[Note: For a list of attendees, see Appendix I. D. A. Ward and M. S. Plesset were not present for the meeting. D. W. Moeller was unable to attend on Thursday.]

The Chairman noted the existence of the published agenda for this meeting, and identified 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 also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC.

[Note: Copies of the transcript taken at this meeting are also available for purchase from the Alderson Reporting Co., Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. Chairman's Report (Open to Public)

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

Chairman Shewmon indicated that written statements had been received from Mrs. Mary Sinclair and Mrs. Barbara Stimeris related to the ACRS review of Midland 1 and 2. He noted that Mrs. Sinclair had requested time to make an oral statement to the Committee during the presentation on Midland. Chairman Shewmon also noted that Commissioner James K. Asselstine had assumed his duties of NRC Commissioner as of May 16, 1982 bringing the Commission to its full level of five commissioners. Also mentioned was the testimony given by C. P. Siess regarding the NRC Safety Research Program before the Subcommittee on Energy Research and Production of the U. S. House of Representatives Committee on Science and Technology. Chairman Shewmon mentioned a set of questions received by the Committee from the Staff of this Subcommittee on Energy Research and Production which will be discussed by the ACRS in Executive Session at the end of the day.

II. Operating License Review of Midland Plant Units 1 and 2 (Open to Public)

[Note: David C. Fischer was the Designated Federal Employee for this portion of the meeting.]

[W. Kerr did not participate in the review of the Midland Plant.]

A. Report of the ACRS Subcommittee

D. Okrent reviewed the history of the Midland license for the Committee He mentioned that the ACRS had done a particularly detailed review of Midland in 1969 and 1970 because the site was one that had a higher population density within three miles from the plant than did other proposed nuclear power stations. D. Okrent referred to a Committee letter dated November 18, 1976 which identified issues which should be considered in the OL review (see Appendix IV).

D. Okrent indicated that there are some special issues applicable to Midland. He called the Committee's attention to a history of quality control deficiencies at Midland during the construction period, noting some problems with cadwelds, bolts, and soil settling, as well as cracking at the foundation of the diesel generator building. He suggested that the Committee pay special attention to specific issues that dealt with the quality question. D. Okrent brought up a question concerning the seismic design rereview, a question of liquefaction problems with soils under many of the safety related structures and a dewatering scheme being proposed by the Applicant. Other topics mentioned for discussion were questions regarding whether a high point vent on the reactor vessel should be provided, whether provisions should be made for instrumentation to detect inadequate core cooling, whether less than favorable experience with high strength bolts required an explanation.

D. Okrent pointed out that there were no major issues regarding fire protection. He indicated that the Applicant is proposing an extensive program to evaluate systems interactions, similar to that being done at Indian Point. Although the integrated control system was modified somewhat to accommodate the process tertiary steam system he did not see a need for extensive Committee attention to these modifications. D. Okrent did point out that the Committee should decide whether to pursue the issue of turbine missiles as a specific or generic issue with regard to Midland.

D. Okrent identified several other potential issues which might be discussed as part of the Committee's Operating License Review (see Appendix IV):

- High copper content in the welds of the Midland 1 reactor vessel
- The status of the ongoing probabilistic risk assessment at Midland
- Commitment by the Applicant to install a third auxiliary feedwater pump in the nonseismic, Category-1 turbine building
- B&W emergency operating procedures
- Industrial security
- Steam generator overfill protection.

C. P. Siess summarized the ad hoc Subcommittee meeting on Midland Foundation Problems and Remedial Actions which was held on April 29, 1982. The problem at Midland is inadequately compacted fill that is partly granular and partly cohesive soils. He indicated that the consequence of this inadequate compaction was the differential settlement of certain safety related structures. This produced some cracking in the walls of some reinforced concrete structures. He indicated that the Subcommittee concluded, after presentations by the Staff and their consultants, that remedial measures being taken seemed appropriate to allay any particular concern about structural adequacy. He noted that the Subcommittee was satisfied with the dewatering system proposed by the Applicant to eliminate the hazard of liquefaction. However, the question of the seismic input to the liquefaction analysis was still open since the Subcommittee had not reviewed the seismic design spectrum during its meeting.

B. Statement by Mary Sinclair

Mary Sinclair, a citizen of Midland, Michigan, read a statement on the Midland Nuclear Plants (see Appendix V). M. Sinclair described the environment in the immediate vicinity of the Midland Plants, including the siting of an elementary school "immediately across the road from the Midland facility." She explained that her purpose was to present a public perception of the role of the Advisory Committee on Reactor Safeguards. The theme of her statement was that the public has lost confidence in the nuclear power plant licensing process.

C. Status of the NRC Staff Review

R. Hernan, NRR Project Manager for Midland, reviewed the SER open items individually, (see Appendix VI). A list of special review areas was presented as areas of particular concern to the NRC Staff. J. Ebersole questioned how the Staff was evaluating the soils settlement issue.

J. Kane, NRC Staff, indicated that the problem had been evaluated by measured building settlement and by making borings in the in-place fill material.

R. Hernan indicated that the Staff had looked closely at the unique process steam system at the Midland plant with regard to radiation monitoring in the case of a primary to secondary system leak. J. Ebersole pointed out that there is a vastly increased probability of secondary blowdown with such a system. B&W reactors are extremely sensitive to secondary system blowdown in view of the superheat design of the steam generators. He questioned whether the NRC Staff had looked into the combination of this increased probability of secondary blowdown in conjunction with a control system failure on feedwater overfilling the steam generator. This could result in an extremely rapid depressurization and thermal shock to the Midland 1 reactor vessel which does have a high copper content. He suggested that there is an unusual potential for very large thermal transients in this system. R. L. Tedesco, NRC Staff, pointed to the safety grade overfill protection system and the fact that only one steam generator would blow down should an accident occur. J. Ebersole expressed concern about the assumptions in the NRC analysis.

M. Bender expressed concern regarding the NRC's collective judgment as to the quality of the Midland plant. He questioned whether there was an integrated, comprehensive report on the problems of quality at Midland plant. R. L. Tedesco indicated that the Staff did not plan to produce an integrated report on this subject.

D. Okrent and R. Axtmann expressed concern about emergency preparedness and emergency planning at Midland. R. Axtmann inquired whether an emergency plan would be in place before startup. R. L. Tedesco indicated that a completed emergency plan might not be in place for low power operations, but that a tested plan must be available before the plant goes into full power operation.

R. Mattson, NRC Staff, indicated that steam generators should be protected against overfill from either the main or auxiliary feedwater systems. Equipment to provide this protection should be safety grade.

D. Okrent noted that this issue is particularly important on B&W plants because of their control sensitivity and he questioned the lack of urgency expressed by the NRC Staff at issuing a backfit requirement for operating plants. D. Okrent requested a written response within the next month regarding the NRC Staff position with respect to the issue of feedwater overfill protection. J. Ebersole requested that the NRC Staff include in its report an analysis of the consequences of continuing to pump cold main feedwater into the steam generator in the event of a main steam line failure. This procedure can lead to a severe secondary transient leading to the pressurized thermal shock problem in the reactor pressure vessel.

D. Quality Control Issues

W. Little, NRC Staff, Region III, presented a tabulation of NRC criteria for assessing contruction QA/QC at nuclear power plants (see Appendix VIII). As a result of the Staff's Systematic Assessment of Licensee Performance (SALP) review of Midland, the Staff had identified six areas which it plans to follow in more detail than currently required. M. Bender questioned how the Staff makes a final judgment regarding the overall plant adequacy. W. Little suggested that the Staff has to depend on its routine inspection program to assess the overall adequacy of plant construction.

J. Ebersole questioned how extensive the Staff effort would have to be in order to assure against a total failure of flow of service water. J. Kane, NRC Staff, indicated that the Staff has undertaken QA audit efforts sufficient to confirm loose fill, soft clays under pipes, the measurement of settlements and stresses on pipes. Based on an evaluation of the remedial measures taken by the Applicant, the Staff is convinced that the problems that have been identified are being adequately addressed.

D. Okrent requested an explanation of the six items that would require special quality assurance monitoring by the Staff. W. Little identified these as follows:

- Remedial actions related to soils problems
- Piping systems and supports
- Electrical power and supply distribution
- Instrumentation and control
- Design control and the control of design changes
- Reporting requirements and corrective action.

M. Bender suggested that the Staff prepare a comprehensive report identifying quality problems at the Midland site and containing an overall assessment of plant quality. R. Vollmer, NRC Staff, did not think the Staff would have any objection to preparing such a report. He indicated that, before this plant could be licensed, he expected Consumers Power to provide objective evidence that the plant had been designed and constructed in accordance with the application. M. Bender expressed concern that the construction problems found may suggest that greater care should have been taken during the construction phase. R. Vollmer thought that the audits the Staff is conducting regarding mechanical and structural details should provide sufficient assurance of the quality of construction.

E. Consumers Power Presentation Regarding Quality Assurance

D. W. Marguglio, Construction Quality Assurance Program Manager for Consumers Power Company (CPCo), described three major aspects of the quality assurance effort at the Midland site:

- NRC's increased Inspection Program
- External, independent audits and assessments by CPCo consultants (biennial audits)
- CPCo performed reinspections and rereviews.

The Committee discussed the apparent buildup in the quality assurance organization and its relationship to the fill material and electrical equipment qualification issues. P. G. Shewmon questioned whether the independent audits being conducted by Consumers Power have uncovered anything in the six areas that the NRC inspection teams have been concentrating their efforts. D. Marguglio indicated that a recent review found the timeliness of quality assurance corrective actions to be quite satisfactory.

J. Ebersole pointed out that numerous significant targets are in the direct path of potential turbine missiles. He questioned the position of the NRC Staff regarding the potential problem of both a turbine stop valve and control valve failure which could lead to turbine overspeed and disc failures. He mentioned attempts by the Applicant to put two trip systems on a single set of valves as a solution to the problem. R. Klecker, NRC Division of Engineering, explained the NRC's turbine missile guidelines as shown in Standard Review Plan III.5.1.3 (see Appendix X). He compared the Applicant's values for missile generation, strike and damage probabilities with NRC's Standard Review Plan numbers.

F. Seismic Review

J. Kimball, NRR Staff Seismologist, explained the Staff's position on the Midland Plant. Two alternatives were given to the Applicant after the Applicant's analysis at the construction permit stage had been reviewed and found to require reanalysis. The Applicant decided to use the site specific spectrum to replace the 0.12g modified Housner spectrum which was the original Midland design spectrum. The Staff and Applicant agree that the 84th percentile in the Midland site specific response spectra is a conservatve representation for the ground motion at the Midland site (see Appendix XI).

L. Reiter, Section Leader for Seismology in the NRR Division of Engineering, made some general comments regarding probabilistic estimates for the safe shutdown earthquake. He noted that reliance upon probabilistic estimates for very long return period earthquakes is not the way to alleviate concerns regarding earthquakes greater than the safe shutdown earthquake. In answer to a question by D. Okrent, L. Reiter indicated that one possible way to alleviate some of these concerns would be to make a study of events, the probability of which is high enough to be accurately estimated by available procedures, in order to develop a base from which to extrapolate less likely events. G. Knighton, NRC Staff, indicated that simply raising the g value for the plant site would not give confidence from the seismic point of view, as would a closer look at the capacity of the equipment or the design of the equipment to withstand more severe shaking. The Committee discussed the design of structures at nuclear plants in general with regard to their ability to withstand a seismic event.

R. Kennedy, President of Structural Mechanics Associates, consultant to Consumers Power, briefly summarized the criteria for the seismic margin review at the Midland plant (see Appendix XIII). R. Kennedy described the screening process to select structural elements, components, and distribution systems for seismic safety margin evaluation, and presented an example of analysis results for the borated water storage tank at Midland. There were no questions from the Committee.

T. R. Thiruvengadam, Consumers Power Co., reviewed the soils exploration program at Midland with regard to liquefaction potential and margins. He identified the diesel generating area, and the railroad bay area of the auxiliary building as the principal structures for which remedial measures against liquefaction were found necessary (see Appendix IX). He indicated that if these areas are dewatered and the ground water level is maintained at or below elevation 610, the structures would be safe against liquefaction for earthquakes with peak ground accelerations of 0.19g. He added that during normal operations of the dewatering system, the water level is maintained at elevation 595.

T. R. Thiruvengadam indicated that for an earthquake of magnitude 6 or 0.19g acceleration there is a factor of safety of 1.5 against the potential for liquefaction and for a 0.25g acceleration there is a factor of safety of 1.1 against the potential for liquefaction. In answer to a question by D. W. Moeller, T. R. Thiruvengadam indicated that the safety factor of 1 would imply the onset of liquefaction.

D. Okrent summarized the various views of the ACRS consultants with respect to the seismic area. R. Holt, Western Geophysical Corp., consultant for Consumers Power, attempted to clarify and reconcile Midland numbers with the numbers estimated by Drs. Trifunac and Pomeroy, ACRS consultants.

G. Inadequate Core Cooling Instrumentation and Reactor Vessel Head Vent

R. Mattson, NRC Staff, indicated that a reactor vessel head vent will be required of the Applicant before licensing. The Staff and the Applicant have continued to discuss the core exit thermocouples as a means of detecting inadequate core cooling. R. Mattson indicated that these thermocouples would be upgraded and operational prior to fuel load. He added that the hot-leg monitoring system proposed by the Applicant is inadequate and has to be upgraded to include a vessel head tap.

L. Gibson, Section Head for Safety and Analysis, Consumers Power Co., presented Consumers Power's position with regard to venting their B&W designed reactor coolant system. He indicated that Consumers Power is in agreement with B&W, that the proper way to provide venting for the B&W design is through the use of vents at the top of the hot leg and the top of the pressurizer. L. Gibson expressed Consumers Power's belief that a level indicator in the reactor head does not provide additional margin for the operator to respond to an inadequate core cooling event. He indicated that Midland procedures call for trip of the reactor coolant pumps on a loss of subcooling margin in order to avoid void formation in the pumps. D. Okrent noted that there was definitely a philosophical difference between the Staff and the Applicant. R. Mattson indicated that, regardless of the Committee's report, the Staff was prepared to go to the Hearing Board with its current position.

R. Mattson mentioned the Semiscale/MOD-V which will model the B&W reactor system. The discussion involved recent TRAC calculations made by Los Alamos involving the ability to maintain single phase natural circulation cooling in the B&W design for certain small break LOCAs. He referred to a letter from the NRC Staff to H. Meyers of the Udall Committee which explains the Los Alamos calculations (see Appendix XIV) and suggested that the TMI-2 Hearing Board might require additional information regarding this matter. In response to an inquiry by D. Okrent, R. Mattson indicated that this matter was an open or outstanding issue which will be addressed in a supplement to the SER.

H. Slager, Consumers Power, provided the summary of experience at the Midland plant regarding bolting. During routine testing of reactor vessel anchor bolts, several failed in a ductile manner. They were found to be much softer than anticipated because of improper heat treatment. Because of this experience Consumers Power Company initiated a hardness testing program for all special purpose bolts. H. Slager noted that this is a QA problem which involved more than just record keeping. He indicated that the hardness test program was eventually extended to cover other bolts and similar problems were found with steam generator anchor bolts, reactor coolant snubber anchor bolts, and pipewhip restraint bolts (see Appendix IX). H. Slager indicated that Paladine Engineering Services was hired to perform an independent analysis of the Midland Reactor Vessel anchor bolts and found that the cracking mechanism was initial stress corrosion cracking followed by complete failure due to the low fracture toughness.

H. Slager explained that in order to avoid further stress corrosion cracking, Consumers decided to lower the prestress on the anchor bolts from 92 ksi to 6 ksi and add upper lateral supports to take up some of the potential seismic loads carried by the reactor vessel anchor bolts in the original design. H. Etherington suggested that it is not good engineering practice to let the design load exceed the prestress load on these bolts. The prestress load should be at least equal to the design load. T. R. Thiruvengadam indicated that that was the original intent, but the lost stiffness was now being taken up through the upper lateral supports.

Chairman Shewmon inquired whether the Staff had made any progress evaluating the use of this ASTM specification that has resulted in the placement of unsatisfactory material at two plants so far. C. D. Sellers, NRC Staff, indicated that the Staff does not have anything other than a technical assistance contract at Brookhaven that would address this matter. An NRC position addressing anchor bolt preload, material selection, hardness, inspection at receipt, and inspection in service would be formulated from the results of the Brookhaven contract.

J. J. Ray asked several questions pertaining to a.c./d.c. electrical system reliability. B. Harshe, Consumers Power, answered these questions as follows:

- Analysis for stability of the grid assumed a single failure such as a breaker that did not operate, line problems coincident with the fault such that there was stability long enough for backup relaying or backup switching to take place.

- With regard to d.c. supply, batteries are oversized and should last for approximately 4 1/2 hours under full load conditions.

- Load shedding analyses to verify extension of the 4 1/2 hour battery lifetime in the event of a blackout have not been done yet.

- All Consumers Power Nuclear Plants have top priority for restoration of power in the event of a blackout.

- Consumers Power System has blackstart capability through the use of the hydro facility at Leadington, diesel generators, and gas turbines.

C. Mark pointed out the unfavorable orientation of the plant turbines and questioned the NRC Staff's procedures for determining strike and damage probabilities. P. G. Shewmon asked the NRC Staff to explain their general approach with regard to the turbine missile strike probability P_2, and the damage probability P_3.

D. W. Moeller questioned whether the Applicant had considered the Bullock Creek Elementary School in its emergency planning. W. Bcckman, Consumers Power, indicated that there were actually two questions involved, the first involving the status of the Emergency Plan and the second with respect to the elementary school. He first indicated that the Midland County Emergency Plan has been reviewed by the State of Michigan. He pointed out that the school lies in Midland County and has an evacuation plan using buses. Information in answer to additional questions by D. W. Moeller concerning emergency planning are as follows:

- Saginaw and Bay County Emergency Plans will be submitted to FEMA for review.

- The Dow Chemical Co. and Consumers Power have reciprocal agreements regarding an accident at the nuclear power plant with plans for protecting the personnel and shutdown of certain facilities in the DOW Plant.

- Dow Chemical personnel participated in an emergency drill at the nuclear power plant, but Midland plant personnel do not yet participate in drills at the chemical plant.

D. Okrent expressed concern regarding the question of small break LOCAs and possible difficulties with natural circulation for B&W plants, and the Midland plant in particular. He explained that formation of a bubble at the top of the reactor coolant system hot leg would most certainly interfere with natural circulation if the reactor coolant pumps were tripped. He expressed displeasure with the fact that this item was not mentioned in the SER and that the Committee was given insufficient information to make a technical evaluation of it. He suggested that he would be more comfortable if the Committee did not go beyond a recommendation for 5% power operation prior to resolution of this issue.

In answer to a question by C. Mark, T. J. Sullivan, Consumers Power, indicated that the subject of control room habitability in the event of noxious gas release from Dow Chemical had been addressed. In answer to a question by J. Ebersole concerning the competency of the diesel generator building to handle a transformer failure and consequent fire, R. Burg, Bechtel Power Corp., indicated that they had looked at fire and also explosion with regard to the diesel generators and that the diesel generators can be controlled remotely from the main control panel for an indefinite period of time.

III. Reactor Vessel Integrity (Open to Public)

[Note: Elpidio Igne was the Designated Federal Employee for this portion of the meeting.]

A. Report of ACRS Subcommittee on Metal Components

M. Bender explained that the purpose of this meeting was to discuss pressure vessel integrity in response to a request by Chairman Palladino with regard to the short term program associated with the pressurized thermal shock issue and provide recommendations as appropriate to the Commission and the NRC Staff. He cited recommendations made to the Commission several months ago that suggested that the NRC Staff seek to familiarize themselves with the composition of the materials that are in the vessels in question, and that operating procedures to protect against thermal shock were in place in those plants. He mentioned that an active audit program was being conducted at the H. B. Robinson plant and some other plants where there is a comparable concern.

M. Bender introduced ACRS consultants M. Wechsler, Z. Zudans, I. Catton, G. Irwin, T. Theofanous, H. Kouts, and E. Abbott who were present at this meeting. He indicated that an ACRS working group had been formed and has addressed three separate issues.

- Thermal hydraulics questions that influence vessel temperature
- The materials question using the fracture mechanics approach
- Operational procedures.

M. Bender presented some of the problems that are posed with respect to this issue. He indicated that the presence of copper in the welds of the reactor vessel is the dominant problem that determines the amount of fracture toughness lost. It is important to judge the condition of the vessels. He added that a second problem involved the interpretation of fracture toughness determination based upon impact tests. Confusion in the interpretation of these hardness tests has influenced evaluation of how severe the thermal shock question really is. He mentioned another problem which involved understanding how control systems influence thermal transients and how the operator might respond if control systems do not work to control the thermal transient with existing control circuitry. It was mentioned that the question of whether the operator has a conflict in his operating decisions that prevents him from executing a timely, safe procedure is also important, especially regarding the adequacy of protection techniques.

M. Bender indicated that the NRC Staff has suggested that it would be worthwhile to reduce the fluence accumulation rate for vessels of concern. He suggested that the ACRS position would be to continue studying the problem, expecially with regard to proper control of operating conditions until the situation in better defined.

Chairman Shewmon suggested that the problem may be largely or completely avoided if the operator depressurizes the system to conditions near saturation. M. Bender mentioned an analysis by T. Theofanus of the way in which cooling rates could occur in the reactor vessel wall (see Appendix XV).

B. Presentation by the NRC Staff

H. Denton described the NRC approach to this problem as an action level or probability of pressurized thermal shock causing vessel failure in the range of 10^{-7} per vessel per year. F. Schroeder, NRC Staff, explained that the NRC approach involves trying to pick a limit on acceptable operation by analysing or constructing accident sequences and following the course of the events and probabilities that can be assigned to them. F. Schroeder discussed a tabulation of actual events and their characteristics (see Appendix XVI), including the final temperatures to which the water drops, and exponential time constant called beta to fit the actual transient with an exponential curve. F. Schroeder pointed to curves with values of temperature and pressure at which deterministic analyses predict cracks for a family of values of RT_{NDT}. He pointed out that such plots could actually define safe and nonsafe regions where a transient final temperature and specified beta can be plotted along with values of RT_{NDT} to identify the pressure necessary for crack initiation. From this procedure one could determine a pressure to stay below in order to avoid crack initiation. R. Klecker, NRC Staff, defined more specificially the NRC assumptions with regard to crack sizes including the length and depth of the crack and its location in the vessel. T. Schroeder pointed to cross plots which show the difference between the final temperature in the transient and the reference temperature in the vessel versus the probability of vessel failure for three different values of heat transfer coefficient. He pointed out that these curves are very steep and that a change in temperature of 20° increased the estimated probability of failure, given a particular event, by orders of magnitude.

F. Schroeder suggested as an NRC criterion that the limit on RT_{NDT} for operation should be 230° F for longitudinal welds. The Committee discussed the best estimate value of 230° F and the uncertainties in estimates of probabilities of vessel failure in severe transients. F. Schroeder discussed possible actions for plants that do not currently meet the criteria. Mentioned were operations improvements, instrumentation improvements, and pressure limiting control systems. He indicated that credit would be given if full volumetric, nondestructive examination of the vessel was performed. He added that the ultimate solution for plants that would exceed such a criteria would be an annealing of the vessel.

F. Schroeder pointed to NRC Staff goals involving flux reduction considerations and defense indepth features dealing with limits on vessel material properties, upgrading operational procedures, and improved instrumentation to allow the operator a better chance to stay out of trouble or possibly hardware improvements in the form of automatic pressure controls, warming ECCS water and required flux reduction rate at some RT_{NDT} threshold.

Chairman Shewmon suggested that the Staff's approach had substantial conservatisms and he questioned whether the Staff had taken account of certain mitigating factors

- Operator actions to ameliorate the accident

- Warm prestress phenomena applied to the pressure vessel

- Probability of a crack size distribution

- Probability of crack initiation leading to a core melt.

H. Denton indicated that the Staff was using a bounding approach in order to produce a regulatory decision in as expeditious a manner as possible and avoid getting mired in the technical details.

C. Presentations by Representatives from the Nuclear Industry

1. Westinghouse Owners Group

D. Speyer, Chairman of the Analysis Subcommittee of the Westinghouse Owners Group (WOG) mentioned a WOG report submitted to the NRC on May 28 entitled, Summary of Evaluations Related to Reactor Vessel Integrity. He defined the objectives of the Owners Group program

- Demonstrate no near-term safety issues in Westinghouse plants

- Reveal generically developed methodologies and tecnniques for addressing pressurized thermal shock

- Provide input to economic deliberations by utility members (see Appendix XVII).

D. Speyer indicated that analysis has shown that decay heat is very impo.tant to analysis of pressurized thermal shock or reactor vessel decay heat transients. Small amounts of decay heat have a beneficial effect and the absence of decay heat will result in a more severe cooldown. P. G. Shewmon and W. Kerr expressed concern that Westinghouse was not considering operator misactions in its analysis of operator actions during a cooldown transient. D. Speyer indicated that although the worst operator action is considered the infinite time for response to terminate the event, the models that Westinghouse used for analysis do include incorrect or improper actions or misactions by the operator. D. Speyer indicated that the Westinghouse best estimate calculation showed for full system pressure approximately 290° F. Below this temperature one could potentially expect crack initiation.

After the Westinghouse approach was described including assumptions, M. Bender asked the NRC Staff whether it had evaluated the Westinghouse approach and made a conclusion with regard to its reasonableness. R. Klecker, NRC Staff, explained that the Westinghouse and NRC Staff are similar except for the Staff's use of RT_{NDT} and betas for description of the temperature drop as opposed to the more severe infinite drop in temperature assumed by Westinghouse.

The Committee explored the Westinghouse approach with D. Speyer. J. Ebersole questioned whether there would be a substantial advantage if the reactor coolant pumps were tripped. D. Speyer indicated there would be a substantial benefit in the heat transfer coefficients but other aspects of the scenario would be much worse.

D. Speyer presented a table of frequency of transients which potentially initiate a crack by class of cooldown transients. It was pointed out from the table that excessive feedwater transients are very benign events with very low probabilities and small break LOCA events are relatively high probability events which tend not to be influenced by operator actions. In answer to a concern by M. Bender, D. Speyer indicated that for small break LOCA events, operator actions were not important since there was automatic actuation of safeguards equipment. Chairman Shewmon questioned why small break LOCA which is postulated to be a high probability event causes pressurized thermal shock. D. Speyer indicated that small break LOCAs result in stagnation in the effected loop.

M. Bender questioned why the excessive feedwater transient probability of thermalized shock was so low. J. Romancick, Westinghouse, indicated that the Westinghouse NSSS design incorporates a number of redundant backup features to isolate feedwater in the event of an excessive cooldown due to a feedwater transient. He indicated also that the inventory in the Westinghouse steam generator is very high, such that it takes a significant amount of water before a substantial amount of cooldown is felt by the system.

J. Ebersole pointed out that after a transient and subsequent safe shutdown, even though there is no actual significant damage, there may be severe monetary impact resulting from detailed fracture mechanics calculations to determine the potential for crack initiation and a decision as to whether inspection is necessary.

D. Speyer presented some recommendations for future research and development efforts in the area of pressurized thermal shock. He discussed fuel management techniques one of which could reduce fluence at the vessel wall. D. Speyer pointed out that in the area of human factors, operator performance could be enhanced through improved procedures and training and quantitative guidance. In answer to a question by Chairman Shewmon, D. Speyer indicated that the temperature limit methodology that Westinghouse is using is being factored into the status trees and function restoration guidelines that the operator will see. He also indicated that despite the appearance of two different approaches, the NRC Staff and Westinghouse are in close agreement on numbers.

D. Peck, Senior Consulting Engineer for Combustion Engineering, summarized the CE approach to pressurized thermal shock which was founded on two basic premises:

- No concern for newer vessels due to low copper materials
- No near-term concern on older vessels.

D. Peck indicated that CE is working on Emergency Procedure Guidelines as part of its post-TMI effort, although no guidelines were found that would cause a pressurized thermal shock event. He indicated that some improvements on the guidelines specifically aimed at pressurized thermal shock will be included in the next revision which will be submitted to the Staff in July, 1982. D. Peck indicated that CE had evaluated different kinds of transient scenarios using linear-elastic fracture mechanics analysis. He indicated that these analyses did not eliminate the preexistence of cracks but use acceptance criteria of crack arrest if there is crack initiation. D. Peck showed a summary of pressurized thermal shock evaluations (see Appendix XVIII). He explained that some transients scenario results include credit for warm prestress. He did point out that the main steam line break and anticipated operating occurrence results did not depend on warm prestress.

He did point out that the main steam line break and anticipated operating occurrence results did not depend on warm prestress. D. Peck also pointed out that the most challenging and most limiting transient for CE plants was the main steam line break.

The Committee discussed the assumptions used in the CE analysis, including reactor coolant pump trip, stagnation and the issue of depressurizations/repressurization. D. Peck made reference to vessel fluence reduction by fuel management and suggestions for future research and development on the subject of pressurized thermal shock (see Appendix XVIII).

J. Gasper, Manager of Reactor Fuel Technical Services at Omaha Power, presented a discussion of fluence reduction work at the Fort Calhoun Station and excess feedwater events at Rancho Seco and how they might relate to CE plants. He expressed confidence that the main steam line break is the bounding transient for Fort Calhoun, that CE reactors are based on the severity of this thermal transient, and the fact that no higher probability events of higher severity have been found. He expressed confidence that elastic fracture mechanics will probably always show that the vessel will not suffer a through wall crack. At Fort Calhoun he indicated that operator action can be shown to minimize the potential for repressurization. He indicated that training to preclude this event has been completed.

J. Gasper divided his specific comments on neutron flux reduction at the vessel wall into three categories: practical fuel management changes; potential fuel management changes; and reactor vessel wall shielding (see Appendix XIX).

J. Gasper indicated that if an excess feedwater transient were to occur at Fort Calhoun, there would not be dependence on operator action to terminate this type of event. He indicated that the large hot water inventory in the steam generator combined with control and safety systems would give the operator about 20 to 30 minutes to take action. W. Kerr suggested that control system failure might cause the excess feedwater transient. J. Herbst, CE, indicated that the reliability of the main feedwater isolation system is typically of the order of 10^{-3}. The Committee discussed the reliability of the main feedwater isolation system. M. Bender requested that J. Herbst furnish the Committee with information regarding the probability of the availability of the power supply for the main steam isolation valves. J. Ebersole questioned why there was a difference in the probability of an excess feedwater transient in the CE experience than in the Westinghouse work.

J. Gasper indicated that in the CE excess feedwater transient, the minimum temperature was around 440^{0} F which is a lot more probable. He added that if CE were to take the low temperature postulated by Westinghouse, it would yield an extremely low probability also.

J. Ebersole questioned whether there was a need for instrumentation to inform the operator that he has a mismatch in pressure and temperature or cold water in a highly pressured vessel. J. Gasper indicated that CE will provide such a signal with alarm functions pointing to minimum cooling or subcooling temperatures or the operators exceeding the 100^{0} F temperature.

B. J. Short, Project Manager for the B&W Owners Group Program, explained why B&W's approach to the pressurized thermal shock problem is correct (see Appendix XX). He pointed to plant specific analyses conducted which used crack arrest as an acceptance criteria and the same linear elastic fracture mechanics techniques used by Westinghouse and CE. He indicated that the results of these analyses showed that B&W reactor vessels are acceptable for their remaining lifetimes. M. Bender questioned whether B&W is taking credit for mixing in the downcomer. B. J. Short indicated that B&W considers mixing important and B&W is taking credit for it. B. Short discussed B&W efforts to reduce fluence or flux reduction and human action dependence to avoid a Rancho Seco type transient. J. J. Ray questioned whether changes had been made in the integrated control system (ICS) or the nonnuclear instrumentation systems voluntarily because B&W thought they were necessary after TMI-2. J. Taylor of B&W indicated that desensitization issues such as auxiliary feedwater flow control, auxiliary feedwater activation, and power supplies to the integrated control system were addressed but actual changes to the ICS were not made. He added that most of the changes were identified by the combination of B&W and the utilities with B&W plants. Work that was done by the Staff after the Crystal River event that suggested safety grade auxiliary feedwater and redundant power supplies has already been considered in some plants under construction.

L. Chano, Manager of the Division of Planning at GPU-Nuclear and Chairman of the B&W Owners Group Subcommittee on Materials, presented the GPU-Nuclear approach to pressurized thermal shock (see Appendix XXI). He indicated that a three dimensional mixing process was used and evaluated using the COMEX-1A computer code. He indicated that the analyses showed that there was indeed mixing in the downcomer in B&W vessels.

L. Chano indicated that GPU-Nuclear has an active material surveillance program which has generated accurate copper and phosphorous contents in the vessel wells. L. Chano pointed to a low leakage fuel management scheme being considered by GPU-Nuclear and several plant modifications done to avoid the Rancho Seco type of accident.

B. Hill, Licensing Engineer for Oconee, explained that Duke Power supports the vessel generic effort begun back in 1979. He indicated that a realization at Duke Power that a plant specific analysis was required for more realistic results, resulted in a Duke Power report issued January 2, 1981 using realistic material properties, fluence levels, and assessment of operating experience. He also explained that Duke Power supports the research and development effort concentrated in the areas of review of operating experience. This is so that nothing could potentially happen in the control room that might be a precursor to a transient. He also endorsed a material surveillance program and enhanced inservice examination of the vessel. B. Hill pointed to an 18 month fuel cycle that had been implemented on one unit and the transition taking place on one or two of the other units which involves a lowering of fluence levels. He also mentioned several improvements made at Oconee with regard to the discussion on the Ranch Seco type transient.

D. Comments by ACRS Consultants

T. Theofanous pointed out that the COMIX computer code suffers from a basic flaw because it uses a characterization of laminar diffusion which predicts complete mixing via a numerical diffusion technique. He indicated that the Staff's presentation of the design basis transient is based upon a number of calculations and results which have not been adequately detailed. He suggested that the ACRS should review the details and assumptions in these calculations. T. Theofanous also felt that the cooldown represented by the Staff seemed to be too fast.

F. Binford suggested that reactor operators need diagnostic assistance to cope with these transients, and procedures and training should be properly interfaced with the equipment the operator will have at his disposal. He also felt that the probabilistic approach being used in these analyses should be standardized so that the different vendor approaches could be more easily reconciled.

Z. Zudans suggested that the projection by Combustion Engineering would not be as optimistic had they not assumed that mixing was taking place prior to reaching the vessel downcomer. D. A. Peck of Combustion Engineering pointed out that Combustion Engineering systems generally have low head High Pressure Safety Injection (HPSI), such that when the system repressurizes the amount of HPSI water decreases. Therefore, he indicated, the addition of HPSI water is not a primary cooldown phenomenon. Z. Zudans suggested that this aspect should be explained in greater detail.

J. Ebersole questioned whether the Staff felt it was important to give the reactor operator the total perspective of the actions that would not been adequately detailed. He suggested that the ACRS should review the details and assumptions in these calculations. T. Theofanous also felt that the cooldown represented by the Staff seemed to be too fast.

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J. Ebersole questioned whether the Staff felt it was important to give the reactor operator the total perspective of the actions that would result should he act in one way or another. H. Denton agreed that examination of this point will be a priority item as the Staff visits the plants and examines their training program. H. Denton made some additional comments:

- There is reasonable agreement between metallurgists how to calculate fracture toughness parameters

- There is disagreement on the probabilities of various transients that actually cool the vessel and how to treat operator actions

- Once the transient lower temperature gets to the temperature of the vessel metal, temperature becomes an extremely important parameter.

M. Bender suggested that the NRC Staff had not given enough weight to treatment of the influence of material properties, and has dealt with them in an arbitrary and very conservative manner.

IV. Meeting with the NRC Commissioners (Open to Public)

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

A. Thermal Shock of Reactor Pressure Vessel

M. Bender explained that the Staff seems to be developing a regulatory framework based on probabilistic arguments which has some validity but also has the usual problems associated with probabilistic arguments. He added that the plan appears to require a considerable amount of investigative experiment work and dialogue with industry. M. Bender expressed concern that the Staff does not appear to have a full understanding of the physical problem with regard to pressurized thermal shock. Chairman Palladino requested a further explanation of that statement as to why he felt that the Staff did not understand the problem. M. Bender indicated that the Staff is interpreting virtually every aspect of the problem in a most conservative way and, while the materials probably have a lot of reserve capability, it is not being credited by the Staff approach.

P. G. Shewmon suggested that the Staff's assumptions, which are not very clear, lead to a bounding "guesstimate" which seems arbitrary and certainly conservative. C. P. Siess suggested that there would certainly be merit in having industry representatives and the Staff confer since industry has also gone through calculations to get bounding cooling curves, different curves that use different assumptions than the Staff. Commissioner Gilinsky suggested that protection in the reactor against a large break in the pressure vessel is one area in which there should be a healthy safety margin.

P. G. Shewmon suggested that one of the critical questions with regard to this problem involves operator actions. Chairman Palladino questioned whether the Committee had any comments with regard to flow mixing problems involved. M. Bender indicated that the bounding computations were likely to signal which vessels are worrisome. He did point out that it should be decided whether it is justified to take some credit for mixing in the case of the high pressure injection system. It might be a legitimate conservatism that has not been credited. M. Bender did suggest that the Staff expand its dialogue with reactor operators concerning the importance of emergency procedures to a pressurized thermal shock transient.

H. Etherington pointed out that it is well understood that there is really no danger of a crack propagating in a cooldown unless the reactor is repressurized cold. He pointed out that the concensus of the ACRS subcommittee was that there may ultimately be a hazard but no real problem for the next few years. He noted, however, that the industry position that there is no problem for the full vessel lifetimes, is not subscribed to by the Staff. Chairman Palladino requested comments and thoughts from the Committee regarding a proposal by Congressman Markey for an in situ annealing demonstration program on an old, embrittled reactor vessel not currently in service.

B. Quantitative Safety Goals

D. Okrent previewed the ACRS' letter on Quantitative Safety Goals by indicating that the Committee will certainly emphasize that the Commission should not take any final action on a policy statement on safety goals until there is a proposed implementation plan that has been reviewed. He added that qualitative goals with quantitative guidance are useful but guidance to the public should be different from that given to the Staff and industry with the Staff having more design oriented goals. In addition, the probabilistic risk assessment which forms one of the bases for the quantitative safety goals should be treated with care because of the large uncertainties that exists in PRA, and the differences that are likely to appear from analyses by different groups for the same system or same plant.

Commission Ahearne questioned whether the ACRS would recommend any changes or suggestions for modifications in the quantitative goals. D. Okrent indicated that the Committee will comment in favor of some criterion on containment as well as on core melt or the prevention of core melt.

Chairman Palladino questioned whether the Committee would comment on the ALARA aspect. He summed up his position regarding ALARA as follows:

- Make efforts to bring operating plants down to an acceptable level of risk

- With regard to plants under design, go to the limit with cost beneficial modifications to meet ALARA.

D. Okrent agreed with the Chairman's concepts. Chairman Palladino and Commissioner Ahearne agreed upon the need for a proposed implementation plan for the safety goals and agreed that the Staff should move on development of such a plan.

C. Proposed ACRS Review of the Clinch River Breeder Reactor (CRBR)

M. W. Carbon pointed out that the review of the CRBR is quite different from a standard review of a light water reactor in that the ACRS is starting even with the Staff in the review and has had to work heavily with the applicant, the Department of Energy. He indicated that the ACRS is working in concert with the Staff but the Staff does not have positions on various technical matters. Chairman Palladino inquired as to particular key points on the ACRS review schedule with respect to the CRBR. M. W. Carbon indicated that the ACRS will review the site suitability at the July meeting and he added that the Commission may expect a construction permit letter in May 1983.

D. Reactor Pressure Vessel Liquid Level/Inventory Instrumentation

W. Kerr reviewed for the Commission the history of this issue starting shortly after the TMI-2 accident with recommendation for installation on operating reactors of an unambiguous water inventory instrumentation system. Palladino inquired whether the ACRS had evaluated the B&W proposal on inventory control. W. Kerr indicated that the B&W proposal was a concept that might be developed into a system but had not yet been developed into a system which could be reviewed.

In answer to a question by Chairman Palladino, W. Kerr and D. Okrent both indicated that the B&W proposal will require further study and additional explanatory details before a judgment can be made as to whether they are as far along with their concept as the other vendors.

In response to a question by Commissioner Ahearne, H. W. Lewis indicated that it was his personal view that the Commission has not yet decided what they want to measure and in what context they want to measure it. He suggested that a better water level indicator might have resulted from a more organized study of the real purpose of the instrumentation. He noted that the pumps on/pumps off issue has not been resolved. He did commend the Staff, however, for taking a more systematic and rational approach to the problem than it had done earlier. In answer to a question by Chairman Palladino, H. W. Lewis indicated that he no longer believed that these systems are counterproductive to safety. However, he suggested that the Commission put a greater premium on reactor operator training and judgment.

E. Proposed NRC Long-Range Research Program Plan

C. P. Siess explained that under existing procedures the ACRS is to review the Long-Range Plan at the draft stage along with user offices. He indicated that it was not clear what the result of the review was intended to be - should it be an input to the Research Staff or comments to the Commission. He explained that ACRS reports on the Research Program are collegial and a concensus of the full Committee, a process involving about ten ACRS subcommittees. It takes at least one full Committee review and close to three months to complete. He indicated that the ACRS plans to meet with the Staff in August to discuss the scope and format of the document. But, the ACRS would just as soon not be in the review process at the draft stage as far as review approval or input to the document. C. P. Siess indicated that the ACRS plans to continue to give the Staff input to its research program in the form of comments and recommendations whether indirectly through meetings or reports to the Commission as well as reports to the Congress. However, he added, the ACRS wishes to make the distinction between review of the Research Program which is almost a continuing effort and ACRS review of a particular NUREG on the Long-Range Research Plan. Chairman Palladino suggested that the ACRS write a letter regarding the basis for proposing that the ACRS not review the Long-Range Research Program Plan at the draft stage for consideration by the Commission.

F. Additional Discussion Issues

Commissioner Ahearne noted the ACRS comment in its letter concerning the draft of a proposed rule for environmental qualification of electrical equipment with respect to the fragmentation of the rule because seismic qualification was not treated. He requested further explanation on that comment. J. J. Ray indicated that it was in the nature of an alert because it left the utility or user with an incomplete picture of the environmental qualification issue, and especially as to whether qualification now for environmental purposes would require requalification later for seismic conditions with the potential for removal of expensive equipment.

In answer to a question by Commissioner Gilinsky as to what the Commission should do about this issue, J. J. Ray indicated that the Commission should proceed with the environmental qualification first and then with specifications for the development of the seismic requirements at a later date. The objective would be to have plants that have not finalized their environmental qualification have the benefit of the newly defined seismic requirements and perhaps do both qualification requirements in close enough timing so that they would not be required to replace equipment for seismic reasons that has been environmentally qualified.

M. Bender pointed out that the seismic qualification has to do with certain hardware that has to be physically oriented in the plant. The mounting arrangement has to be understood as well as just testing the hardware. He indicated that he had been concerned that some equipment would still be open to question with regard to seismic response after other environmental qualification had been done. Since it is anticipated that some seismic qualification is associated with the environmental qualifications, the lack of formal specifications for seismic qualification should not present as serious a matter as once expected.

Commissioner Ahearne questioned whether the ACRS had recommended in its report on SECY-82-111 that the Safety Parameter Display System (SPDS) be safety grade. W. Kerr explained that the comment in the letter was not specifically that the SPDS be safety grade, but that more thought be given to appropriate reliability requirements that should be used with the SPDS. Chairman Shewmon suggested that it might not be a good idea to mandate a requirement for safety grade in that it might hinder the actual development of the instrumentation by industry.

Commissioner Ahearne questioned the activities the ACRS had underway with regard to the area of high-level waste management. D. W. Moeller pointed to a June 8, 1982 subcommittee meeting review of the current status of DOE plans as well as NRC Staff efforts in this area.

H. W. Lewis presented additional views on the Quantitative Safety Goals. He referred to the question "How safe is safe enough", the use of risk aversion, ALARA, and the most exposed individual in the safety goal concept. He suggested that he would much rather see a completely arbitrary overall safety goal, rather than the methodology currently suggested. Chairman Palladino indicated that he had a problem with an arbitrary number because it would have to have some reference point in support of the goals proposed. He indicated that an advantage of this premise from which NRC is starting is that there is at least a reference point given by the probability values in the safety goal.

V. Consideration of Seismic Events in Emergency Planning

[Note: H. Alderman was the Designated Federal Employee for this portion of the meeting.]

D. W. Moeller reported the results from the Reactor Radiological Effects Subcommittee Meeting on May 14, 1982. He indicated that the primary result of small earthquakes on emergency planning would focus on the disruption of roads in the vicinity of a nuclear power plant. He indicated that in a discussion with the NRC Staff, the Staff indicated that backup communication systems and helicopters would be of value in the event of a small earthquake. A question came up as to whether similar preparations should be made with regard to a major earthquake. B. Grimes, NRC Staff, indicated that it would not be appropriate to take these measures with large earthquakes because of the massive nature of the disruption caused. D. W. Moeller indicated that B. Grimes had referred to a misinterpretation made by the Atomic Safety and Licensing Board which was under the impression that the Commission had referred to all earthquakes when discussing seismic events and emergency planning. B. Grimes comments were stated as follows:

- With regard to a small earthquake, the power plant would remain intact and emergency planning for a small earthquake would be beneficial.

- In the event of an intermediate earthquake, evacuation might not be possible and a suggestion might be made for the population to seek shelter.

- In the event of a major earthquake, little could be done offsite to help the indigenous population. Even if the plant survived, there would be no demand for electricity.

D. W. Moeller indicated that the NRC Staff is developing a position paper with regard to consideration of seismic events and emergency planning at nuclear power plants. He suggested that the Committee wait for issuance of the paper and review the draft at that time.

VI. Control Room Habitability in Nuclear Plants (Open to Public)

[H. Alderman was the Designated Federal Employee for this portion of the meeting.]

D. W. Moeller suggested that the ACRS full committee request that the NRC Staff conduct a two hour briefing at the July or August full committee meeting regarding control room habitability, with presentations made by the NRC Staff and possibly architect/engineering firms, consulting firms and the Institute for Nuclear Power Operation. He indicated that problems with regard to control room habitability have been pointed out through Licensee Event Reports (LERs). He indicated that outside consulting firms have been called in to look at control room habitability at certain plants and have pointed out many deficiencies which did not violate the plant's technical specifications. He pointed out that operators cannot inhabit the control room if dampers are set as often designed. Also noted was the fact that control room operators often lack confidence in control room ventilation systems as presently designed. Chairman Shewman recommended that the issue of control room habitability be put on the agenda for the August full committee meeting.

VII. ACRS Responses to Questions from the Subcommittee on Energy Research and Production (Open to Public)

The Committee briefly discussed a draft of responses to questions informally submitted to the ACRS subsequent to testimony given by C. P. Siess regarding the NRC Safety Research Program before the Subcommittee on Energy Research and Production of the U.S. House of Representatives on Science and Technology on May 18, 1982 (see Appendix XXII). The document was referred back for ACRS Staff revision for later consideration during the Meeting.

VIII. Executive Sessions (Open to Public)

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

A. ACRS Reports, Letters, and Memoranda

1. ACRS Interim Report on Midland Plant, Units 1 and 2

The Committee prepared a report to the Commissioners of its review of the Midland Plant Units 1 and 2 regarding the request for an operating license. The Committee concluded that, if due regard is given to comments in the body of the report, and subject to satisfactory completion of construction and staffing, operation at power levels up to 5 percent of full power is acceptable. ACRS recommendation regarding operation at full power has been deferred until the Committee has had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question of natural circulation in the presence of a small break LOCA.

2. ACRS Report on Pressurized Thermal Shock

The Committee prepared a report to the Commissioners of its review of the current status of the pressurized thermal shock problem.

The ACRS noted lack of sufficient information to evaluate the adequacy of an approach by the NRC Staff to develop a regulation based upon a combination of deterministic and probabilistic analyses.

3. ACRS Comments on Proposed Policy Statement on Safety Goals for Nuclear Power Plants (NUREG-0880, "A discussion Paper")

The Committee prepared a report to the Commissioners of its review of NUREG-0880, A Discussion Paper, recommending that final action on adoption of a policy statement on safety goals should be contingent upon proper evaluation and agreement on the implementation plan. The ACRS plans to provide further comments to the Commission after reviewing the Staff plan for implementation. M. Bender and H. W. Lewis appended additional comments. In the body of the ACRS report will also be found responses to the four questions raised by the Commission.

4. ACRS Review of the NRC Long-Range Research Plan

The Committee prepared a report to the Commissioners regarding termination of a formal ACRS report to the Commission on proposed Long-Range Research Plans. The ACRS expects to continue to receive the LRRP, both in draft and final form, and expects to utilize it in its review of and report on the NRC Safety Research Program and Budget for the Commission and the Congress.

5. Response to Commissioner Gilinsky Regarding Seismic Design Suggestions by Professor Paul Jennings

The Committee prepared a report to Commissioner Gilinsky recommending that the suggestions by Professor Paul Jennings on seismic design be considered within the context of a broad review of the NRC Staff's current seismic design practices including the NRC Staff's reassessment of Appendix A to 10 CFR 100. The ACRS suggested that Professor Jennings be invited to participate in this review.

6. ACRS Responses to Questions from the Subcommittee on Energy Research and Production

The Committee endorsed a response to questions received from the Staff of the Subcommittee on Energy Research and Production of the U.S. House of Representatives on Science and Technology with a one week grace period for comments by members.

B. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 267th ACRS Meeting, July 8-10, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

C. Nominations for New ACRS Member

The Committee discussed the qualifications of several prospective nominees to replace ACRS Member W. M. Mathis who is retiring and decided to invite the two leading candidates to the August full Committee Meeting to meet with ACRS Members.

D. Review of ICRP-26 and Proposed Changes to 10 CFR 20

D. W. Moeller has been asked to serve on the DOE Headquarters Ad Hoc Committee which will be reviewing ICRP-26/30 and the proposed changes to 10 CFR 20 in the context of practical operational problems envisioned by DOE. The Committee discussed the matter and decided that he should attend as an ACRS observer.

E. Participation in American Nuclear Society (ANS) Panels

W. Kerr has been invited to participate on a panel discussing the subject of degraded reactor cores at the ANS Annual Meeting being held in Los Angeles, June 6-10, 1982. D. Okrent indicated that he has also been invited to participate on a panel discussing quantitative safety goals at this same conference. The Committee offered no objection.

The 266th meeting of the Advisory Committee on Reactor Safeguards was adjourned on Saturday, June 5, 1982 at 12:25 p.m.

ACRS-1498

APPENDIXES
TO
MINUTES OF THE 266TH ACRS MEETING
JUNE 3-5, 1982

APPENDIX I

ATTENDEES
266TH ACRS MEETING
JUNE 3-5, 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Paul G. Shewmon, Chairman
Jeremiah J. Ray, Vice-Chairman
Robert C. Axtmann
Myer Bender
Max W. Carbon
*Jesse Ebersole
Harold Etherington
William Kerr
Harold W. Lewis
Carson Mark
William M. Mathis
Dade W. Moeller
David Okrent
Chester P. Siess

*Member Emeritus

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
William M. Baldewicz
Stuart K. Beal
Alden Bice
William M. Bock
Paul A. Boehnert
Don Bucci
Anthony J. Cappucci
Joseph Donoghue
Sam Duraiswamy
David C. Fischer
J. Michael Griesmeyer
Elpidio G. Igne
Kenneth D. Kirby
Morton W. Libarkin
John A. MacEvoy
Richard K. Major
Thomas G. McCreless
John C, McKinley
Thomas McKone
Austin Newsome
Gary R. Quittschreiber
Christopher Ryder
Richard P. Savio
Stanley Schofer
R. C. Tang

CONSULTANTS

E. Abbott
F. Binford
I. Catton
G. Irwin
H. Kouts
T. Theofanous
M. Wechsler
Z. Zudans

NRC STAFF ATTENDEES

266TH ACRS MEETING

Thursday, June 3, 1982

NUCLEAR REACTOR REGULATION

J. W. Clifford
A. Thandani
E. Goodwin
H. Denton
F. Schroeder
R. Woods
P. Cota
G. Knighton

NUCLEAR REGULATORY RESEARCH

N. Zuber

NUCLEAR MATERIAL SAFETY & SAFEGUARDS

R. Zimmerman

INVITED ATTENDEES

266TH ACRS MTG.

Thursday, June 3, 1982

Westinghouse Electric Corporation

J. A. Rumancik
R. J. Sero
J. D. McAdoo
T. A. Meyer

Combustion Engineering, Inc.

J. M. Westhoven
J. J. Herbst
D. A. Peck
P. J. Ayres

Duke Power Corporation

R. L. Gill

University of Maryland

G. Irwin

Carolina Power & Light Company

J. J. Sheppard

Babcock & Wilcox

F. Levandoski
J. H. Taylor
A. L. Lowe, Jr.
B. J. Short

Consumers Power Company

K. Drehubl
R.W. Huston
J. P. Kindinger
L. Gibson
J. Pastor
H. W. Sloyer
T. E. Hollowell

PUBLIC ATTENDEES

266TH ACRS MTG.

Thursday, June 3, 1982

K. J. Morris, OPPD
J. K. Gasper, OPPD
V. T. Chilson, Florida Power & Light Company
L. Erik Titland, Baltimore Gas & Electric
R. Leyse, NSAC
M. D. Patterson, Baltimore Gas & Electric
R. K. Mattu, NUS
M. S. Wechsler, Iowa State University

D. Speyer, Con Edison of New York
A. P. Rochino, GPU Nuclear
J. Bezila, Bechtel
B. Barnisin, GAI
P. A. Minson, ARC
D. J. Harvey, Interscience
L. Connor, Doc-Search Associates
A. C. Passwater, Union Electric
T. Kishbaugh, NUTECH
S. R. Phelps, EEI
R. G. Smith, SCP
R. S. Boyd, KMC
K. Gage, NIRS
J. Buckman, Inside NRC
P. Patniainen, IVO
W. Andrews, Southern Co. Services
P. Tremblay, NUS
J. O. Berga, EPRI
M. A. Barnisin, Gilbert Associates
B. Mahsen, IVO
P. Higgins, Atomic Industrial Forum
K. Gage, NIRS
C. Grochmal, Stone & Webster
K. C. Prasad, Bechtel

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NRC ATTENDEES

266TH ACRS MEETING

Friday, June 4, 1982

Division of Licensing

R. Hernan
E. G. Ademsam

Nuclear Reactor Regulation

E. Goodwin
R. L. Tedesco
R. Vollmer
G. Mazetis
L. Rubenstein, RRAB
E. Tomlinson, PSB
O. P. Chopra
W. Jensen
J. Kane, HGEB
J. Knight, CSE
W. Haass
R. S. Lee
L. E. Phillips
L. Reiter
R. Jackson
J. Kimball
A. Thadani
R. Frahm
J. Tsad, RRAB
R. Lobel, ASB
T. Chan, ASB
W. G. Kennedy, PTRB
C. D. Sellers

Executive Director for Operations

W. S. Little

Nuclear Regulatory Research

R. Zimmerman

Generic Information Branch

R. Woods

Division of Engineering

B. Elliot
R. Klecker

Division of Systems Integration

B. LeFave

APPLICANT ATTENDEES

266TH ACRS MEETING

Friday, June 4, 1982

CONSUMERS POWER COMPANY

B. Harshe
T. Thiruvengadam
D. Sommers
K. Drehobl
R. E. Berry
D. Karjala
R. L. Teuteberg
B. W. Marguglio
F. W. Buckman
J. F. Firlit
D. L. Sowers
G. B. Slade
J. J. Zabritski
W. Beckman
J. Pastor
J. Kindinger
G. Keeley
J. Carter
K. E. Drehobl
D. M. Budzik
R. W. Huston
N. J. Saari
W. R. Bird
J. P. Webb
D. A. Sommers
P. Jacobsen
T. Buczwinsch
J. Alderant
R. M. Hamm
I. J. Sullivan
D. T. Perry
M. Tepens
L. Gibson
R. B. Dweitt
P. A. Eblert
W. E. Garrity
W. J. Hall
H. W. Siager
R. Poland
A. L. Lowe, Jr.

Bechtel Power Corporation

M. Pratt
C. Patts, Jr.
M. Elgaby
H. M. Fontecilla
S. S. Afifi
W. C. P
R. Burg
T. Ballweg
K. C. Prasad
D. F. Lewis
M. A. Gerding

Babcock and Wilcox Company

B. Dunn
M. Liebmann
J. D. Carlton
B. Brooks
F. J. Levandoski
L. L. Joyner
A. L. Lowe, Jr.
J. H. Taylor

Quadrex

J. M. Nelson

Aptech

R. C. Cipolla

Weston Geophysical Corp.

R. J. Holt
G. Klimkicwicz

Pickard, Lowe & Garrick, Inc.

F. Hubbard

PUBLIC ATTENDEES

266TH ACRS MEETING

Friday, June 4, 1982

M. F. Conlan, Newhouse
J. L. Buckincin, Inside NRC
M. Coffman, WNEM TV
J. Duneem, WNEM-TV
J. S. McEwon, Jr., TSI
K. A. Perron, Fragema (France)
E. T. Murphy, Westinghouse
R. S. Boyd, KMC, Inc.
R. Leyse, EPRI
R. Kennedy, SBA
L. N. Rib, LNR Associates
M. Broad, NRDC
J. Maron, NUTECH
M. Sinclair, Citizen, Midland, MI
D. Crow, Government Accountability Project
D. Devlin, Midland Daily News
L. Chano, GPU Nuclear

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APPENDIX II

FUTURE AGENDA

JULY

Perry Nuclear Plant Unit 1--OL

Ginna Nuclear Plant--SEP review

Clinch River Breeder Reactor--Site Suitability Review

ACRS report to Commission on NRC Reactor Safety Research Program

ACRS comments regarding final version of Task Action Plan A-45, Evaluation of Alternate Decay Heat Removal Systems

Discuss/select panel of nominees to replace W. M. Mathis (DWM/MCG); discuss need to replace M. S. Plesset upon his retirement from ACRS membership

Subcommittee Reports

Subcommittee on Emergency Core Cooling System regarding proposed changes to 10 CFR 50, Appendix K for BWR's (MSP/PAB)

Subcommittee on Metal Components on the status of activities regarding steam generator tube integrity and proposed repairs/operation of TMI-1 (PGS/EI)

Subcommittee on Waste Management regarding DOE National Plan for Siting High Level Waste Repositories and Environmental Assessment (DWM/RCT/JCM)

Subcommittee on Reactor Radiological Effects regarding proposed changes in 10 CFR Part 20, and proposed use of potassium iodine for thyroid blocking (DWM/RCT)

AUGUST

Briefing by the NRC Staff regarding control room habitability in nuclear power plants

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

June 5, 1982

APPENDIX III
REVISED SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

MEMORANDUM FOR: ACRS Members
ACRS Technical Staff

FROM: G. Quittschreiber, Chief
Project Review Branch No. 2 Michele S. for

SUBJECT: REVISED SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

Please replace page 2 of your "Schedule of ACRS Subcommittee Meetings" with the attached sheet.

NOTE: (1) Grand Gulf Meeting on July 1st is cancelled.

(2) Extreme External Phenomena Meeting scheduled for July 1st is rescheduled to July 7th at 5:00 p.m.

(3) Reliability and Probabilistic Assessment Meeting scheduled for July 1st is rescheduled to July 7th at 4:00 p.m.

6/05/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JULY

Cancelled

~~1 (morning)~~ ~~(4 hours)~~	~~Grand Gulf 1 (Alderman) - Okrent, Bender, Ebersole, Siess. Purpose: To complete the operating license review.~~
7 5:00 pm ~~1 (afternoon)~~ (1 hour)	Extreme External Phenomena (Savio) - Okrent, ~~Bender~~, Siess, Mark. Purpose: To review the RES proposed FY 84-85 research funding and programs in this area for the Long-Range Research Plan.
7 4:00 pm ~~1 (afternoon)~~ (1 hour)	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) Okrent, Kerr, ~~Bender~~, Ebersole, Mark, Siess, Lewis. Purpose: To review the RES proposed FY 84-85 research funding and programs for the SARA Decision Unit.
~~1~~ 1	Electrical Systems/Qualification Program for Safety-Related Equipment (Savio/Cappucci) - Kerr, Ray, Ebersole, Mark, ~~Mathis, Ward, Bender~~. Purpose: To review the RES proposed FY 84-85 research funding and programs in this area for the Long-Range Research Plan.
6 (morning)	Human Factors (Fischer) - Ward, Lewis, Mathis, Ray, Moeller (tent.). Purpose: To review the NRC's proposed FY 84-85 programs and budget, and to develop specific comments on the Long-Range Research Plan.
6 ~~(tent.)~~ CANCELLED ~~(afternoon)~~	Regulatory Activities (Duraiswamy) - Siess, Bender, Carbon, Ward, Ray, Kerr. Purpose: To review proposed Regulatory Guides and Regulations.
7	Safety Research Program (Duraiswamy) - Siess, Okrent, Shewmon, Bender, Carbon, Mathis, Ward, Mark, Plesset, Kerr. Purpose: To continue discussion on the FY 84-85 NRC Safety Research Program Budget and prepare comments for use by the ACRS in its report to the Commission.
8-10	267th ACRS Meeting
14 (tent.)	CRBR Working Group on Structures and Materials (Cappucci/Quittschreiber) - Shewmon, Axtmann, Bender, Etherington, Siess. Purpose: To discuss crieria for elevated temperature design (N-47), including supports.
20 (tent.)	Reactor Radiological Effects (Tang/McKinley) - Moeller, Axtmann, Ebersole, Ray, Okrent (tent.). Purpose: (1) To review PWR Occupational Radiation Exposure histories and recent experiences in exposure reduction at several plants, (2) To discuss the status of requirement for "Radiation Protection Plan" (Rev. to Part 20), and (3) To discuss findings of Health Physics Appraisals.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JUNE

7 Metal Components (Palo Alto, CA, EPRI) (Igne) - Shewmon, Etherington, Ward, Mathis. Purpose: To be given a status report by owners group on research results and any changes made on steam generator design/ operation, and to be briefed on TMI-1 steam generator problems and possible fixes.

8 Waste Management (Tang/McKinley) - Moeller, Axtmann. Purpose: To review and comment DOE Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment; provide input for the Waste Management Chapter of the FY 84-85 Safety Research Program Review; review Staff waste management activities; and discuss advances in waste management practices.

9 Joint Reactor Radiological Effects and Site Evaluation (Tang/McKinley) - Moeller, Axtmann, Ebersole. Purpose: To review the RES proposed FY 84-85 research funding and programs in these areas for the Long-Range Resear n Plan.

16 & 17 ECCS (Idaho Falls,ID) (Boehnert) - Plesset, Ebersole, Mathis, Ward. Purpose: To discuss GE's request for change in Appendix K decay heat requirements, and an update and status of selected RES LOCA/ECCS Research Programs.

23 Reactor Radiological Effects (Tang/McKinley) - Moeller, Ebersole, Ray, Axtmann, Okrent (tent.). Purpose: (a.m.) To discuss NRC Staff proposed revision to 10 CFR 20. (p.m.) To discuss the use of KI for thyroid blocking in the event of a radiation accident.

24 & 25 Joint CRBR and Site Suitability (Boehnert/Alderman) - Carbon, Moeller, Ebersole, Mark, Okrent (tent.), Ray (tent.). Purpose: To discuss site suitability for CRBR.

28 & 29 Perry (Cleveland, OH) (Cappucci/Quittschreiber) - Ray, Axtmann. Purpose: To review the application for an OL and to conduct a site visit.

30 Systematic Evaluation Program (Major/Alderman) - Siess, Ebersole, Kerr, Lewis (part-time). Purpose: To review the completion of the Integrated Plant Safety Assessment Systematic Evaluation Program on Ginna.

6/05/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JULY

1 (morning) (4 hours)	Grand Gulf 1 (Alderman) - Okrent, Bender, Ebersole, Siess. Purpose: To complete the operating license review.
1 (afternoon) (1 hour)	Extreme External Phenomena (Savio) - Okrent, Bender, Siess, Mark. Purpose: To review the RES proposed FY 84-85 research funding and programs in this area for the Long-Range Research Plan.
1 (afternoon) (1 hour)	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) Okrent, Kerr, Bender, Ebersole, Mark, Siess, Lewis. Purpose: To review the RES proposed FY 84-85 research funding and programs for the SARA Decision Unit.
2	Electrical Systems/Qualification Program for Safety-Related Equipment (Savio/Cappucci) - Kerr, Ray, Ebersole, Mark, Mathis, Ward, Bender. Purpose: To review the RES proposed FY 84-85 research funding and programs in this area for the Long-Range Research Plan.
6 (morning)	Human Factors (Fischer) - Ward, Lewis, Mathis, Ray, Moeller (tent.). Purpose: To review the NRC's proposed FY 84-85 programs and budget, and to develop specific comments on the Long-Range Research Plan.
6 (tent.) CANCELLED (afternoon)	Regulatory Activities (Duraiswamy) - Siess, Bender, Carbon, Ward, Ray, Kerr. Purpose: To review proposed Regulatory Guides and Regulations.
7	Safety Research Program (Duraiswamy) - Siess, Okrent, Shewmon, Bender, Carbon, Mathis, Ward, Mark, Plesset, Kerr. Purpose: To continue discussion on the FY 84-85 NRC Safety Research Program Budget and prepare comments for use by the ACRS in its report to the Commission.
8-10	267th ACRS Meeting
14 (tent.)	CRBR Working Group on Structures and Materials (Cappucci/Quittschreiber) - Shewmon, Axtmann, Bender, Etherington, Siess. Purpose: To discuss crieria for elevated temperature design (N-47), including supports.
20 (tent.)	Reactor Radiological Effects (Tang/McKinley) - Moeller, Axtmann, Ebersole, Ray, Okrent (tent.). Purpose: (1) To review PWR Occupational Radiation Exposure histories and recent experiences in exposure reduction at several plants, (2) To discuss the status of requirement for "Radiation Protection Plan" (Rev. to Part 20), and (3) To discuss findings of Health Physics Appraisals.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JULY (CONT'D)

21 or 22	Reactor Operations (Major) - Mathis, Bender, Ebersole, Kerr, Moeller, Okrent, Ray, Ward. Purpose: (1) To discuss NRC's enforcement policy governing enforcement actions for violations of NRC regulations and license applications; (2) A discussion of regionalization effort within I&E; and (c) To discuss the current status of I&E's Performance Appraisal Team inspection program and Systematic Assessment of Licensee Performance program.

AUGUST

10 (tent.)	Watts Bar (Beal/Quittschreiber) - Ebersole, Bender, Ward. Purpose: To complete the review of the application for an OL.
10	Regulatory Activities (Duraiswamy) - Siess, Carbon, Ray, Kerr. Purpose: To review proposed Regulatory Guides and Regulations.
11	Safety Research Program (Duraiswamy) - Siess, Okrent, Plesset, Ward, Shewmon, Bender, Kerr, Moeller, Mark, Carbon (tent.). Purpose: To provide early input to the RES Staff for their preparation of the Long-Range Research Plan for FY 85-89.
12-14	268th ACRS Meeting
18	CRBR Working Group on Structures and Materials (Cappucci/Quittschreiber) - Shewmon, Axtmann, Bender, Etherington, Siess. Purpose: To discuss "leak before break" criteria for CRBR, overall leakages, and leak detection. Inservice inspection plan and the structural integrity of critical transition joints will also be discussed.

DATES TO BE DETERMINED

Date to Be Determined (late August)	WPPSS 2 (Hanford, WA) (Griesmeyer/Quittschreiber) - Plesset, Ebersole, Mark, Mathis, Ward (tent.). Purpose: To review application for an operating license.
Date to Be Determined	Transporation of Radioactive Materials (location to be determined) (Duraiswamy) - Siess, Bender, Mark. Purpose: To continue the review of the adequacy of the NRC package certification procedures.
Date to Be Determined	Metal Components (Igne) - Shewmon, Ward, Axtmann, Bender, Etherington, Mathis, Plesset. Purpose: To continue the review of pressurized thermal shock.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 7, 1982	Metal Components	(IGNE) Shewmon, Mathis Etherington, Ward Consultants: Dillon, Kassner, Berger

LOCATION: Palo Alto, CA (EPRI)

BACKGROUND:

Who proposed action: P. Shewmon

Purpose: Status report by owners group on research results and any changes on steam generator design/operation made. A briefing on TMI-1 steam generator problems and possible fixes will be presented by GPU.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Rev 1

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June 8, 1982	WASTE MANAGEMENT	(Tang, McKinley, Donoghue) Moeller, Axtmann
		Consultants: F. Parker, D. Orth, M. Steindler, S. Philbrick, G. Thompson, R. Foster

LOCATION: Room 1046, 1717 H St., NW, Washington, D.C.
8:30 am

BACKGROUND:

Who proposed action: D. W. Moeller

Purpose:

1. Review and comment on DOE Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment.
2. Provide input for Waste Management Chapter of FY 84-85 Safety Research Program Review.
3. Review Staff Waste Management activities.
4. Discuss advances in waste management practices.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. a) "National Plan for Siting High-Level Radioactive Waste Repositories and Environmental Assessment", (DOE/NWTS-4), February, 1982.
   b) Donoghue memo to Moeller on DOE Siting Plan, 4/29/82.
2. a) Siess memo to Safety Research Subcommittee on FY 84-85 Safety Research Program Review, 4/12/82.
   b) ACRS report to Palladino, "Comments on NRC Long-Range Research Plan, FY 1984-1988 (Draft NUREG-0784)", 5/5/82.
   c) "Comments on the NRC Safety Research Program Budget for FY 83" (NUREG-0795), 7/17/81, Ch. 7, pages 37-39.
   d) "Long-Range Research Plan, FY 1984-1988", (Draft NUREG-0784), 3/12/82, pages 8-1 to 8-19.
   e) ACRS Report to Congress on the NRC FY 1983 Safety Research Program (NUREG-0864), Ch. 7, pages 43-47, 2/12/82.
3. a) ACRS letter to Chairman Palladino on proposed rule for HLW disposal, 10 CFR 60, 9/16/81.
   b) Donoghue memo to Waste Management Subcommittee on HLW Management Bill, S-1662, 5/6/82.
4. a) Recent EPRI reports on radwaste processing techniques for nuclear power plants (EPRI NP-2334, EPRI NP-2335, EPRI NP-2338).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 9, 1982	JOINT REACTOR RADIOLOGICAL EFFECTS AND SITE EVALUATION	(TANG/MCKINLEY) Moeller, Axtmann, Ebersole, Cons.: Orth, Foster, Philbrick, Steindler (tent.), F. Parker

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To review NRC RES programs and budget for FY 84-85 in the following subelements: siting and environmental research, occupational protection, emergency preparedness, effluent control and chemical systems, fission products release and transport, and to provide input for the corresponding chapters of the FY 84-85 Safety Research Program Review Report. (Review will include NRC Staff's response to recommendations in NUREGs -0795 and -0864.)

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-0795, "Comments on the NRC Safety Research Program Budget for FY 83" dated July 1981.
2. NUREG-0864, "Review and Evaluation of the NRC SRP for FY 83", dated February 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 16-17, 1982	ECCS	(BOEHNERT) Plesset, Ebersole, Mathis, Ward Cons: Catton, Acosta, Dukler, Garlid, Schrock, Zudans

LOCATION: Idaho Falls, ID

BACKGROUND:

Who proposed action: M. Plesset/C. Siess

Purpose: To discuss (1) GE's request for change in Appendix K decay heat requirements; (2) Update and status of selected RES LOCA/ECCS Research Programs for Long Range Research Plan report by ACRS.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 23, 1982	Reactor Radiological Effects	(TANG/McKINLEY) Moeller, Ray, Axtmann, Okrent (tent), Ebersole Cons: Healy, Bair, Morgan Muller Inv. Speakers: Weiner (OSHA) Shleien (FDA) Fowinkle (St. of Tl Krimm (FEMA) Becker (ATA) Godwin (AL)

LOCATION: Washington, DC (Room 1046)

BACKGROUND:

Who proposed action: D. Moeller

Purpose: (Morning) To discuss NRC Staff proposed revision to 10 CFR 20.

(Afternoon) To discuss the use of Potassium Iodide for thyroid blocking in the event of a radiation accident.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. 10 CFR Part 20
2. Draft Revision to Part 20 (March 1982)
3. SECY-82-77 (Policy Issue on Potassium Iodide)
4. NRC Staff's testimony for Congressman Markey's hearing regarding Potassium Iodide (March 5, 1982)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 24-25, 1982	Combined CRBR and Site Suitability	(BOEHNERT/ALDERMAN) Carbon, Moeller, Ebersole, Mark, Okrent (tent.), Ray (tent.)
		Cons: F. Parker, R. Hosker, M. Trifunac, W. Lipinski, W. Kastenberg, R. Foster, Z. Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To discuss site suitability for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Site Suitability Report by the Office of Nuclear Reactor Regulation, USNRC in the matter of the Clinch River Breeder Reactor Plant, dated March 4, 1977 (to be revised in June or July).

NUREG-0833 "Environmental Impact Statement on the Siting of Nuclear Power Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 28 & 29	Perry	(CAPPUCCI/QUITTSCHREIBER) Ray, Axtmann

LOCATION: Cleveland, Ohio

BACKGROUND:

Who proposed action: NRR

Purpose: To review the OL application and to conduct a site visit.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NRR has committed to supply an SER by May 28, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 30, 1982	SYSTEMATIC EVALUATION PROGRAM	(MAJOR/ALDERMAN) Siess, Ebersole, Kerr, Lewis (part-time)
		Cons: Catton, Lipinski

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To review the completion of the Integrated Plant Safety Assessment Systematic Evaluation Program on Ginna.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant, NUREG-0821, May 1982 (distributed to members).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (morning - 4 hrs.)	GRAND GULF 1	(ALDERMAN) Okrent, Bender, Ebersole, Siess

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To complete the OL review. ACRS Interim Report dated Oct. 20, 1981 gave conditional approval for operation up to 5% of full power. Questions regarding the ability of the Mark III containment to withstand certain dynamic loads and regarding hydrogen control required answers before the ACRS would approve full power operation. The Committee was also concerned regarding the depth and experience of the operating and support staffs.

Two major issues to be resolved: Hydrogen Control
Seismic Qualification Review Team
(Equipment Qualification)

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SER issued 9/11/81.
2. SSER scheduled to be issued 5/15/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (p.m. - ~1 hr.)	EXTREME EXTERNAL PHENOMENA	(SAVIO) Okrent, Bender, Siess, Mark

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the RES proposed FY 1984 - 1985 research funding and programs in this area and to develop comments on the Long-Range Research Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposed funding and program plans in this area as are available and the Long-Range Research Plan.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (p.m. --~1 hr.)	RELIABILITY AND PROBABILISTIC ASSESSMENT	(GRIESMEYER/QUITTSCHREIBER) Okrent, Kerr, Bender, Ebersole, Mark, Siess, Lewis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To review the RES proposed FY 1984-FY 1985 research funding and programs for the SARA decision unit.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 2, 1982	ELECTRICAL SYSTEMS/QUALIFICATION PROGRAMS FOR SAFETY-RELATED EQUIPMENT	(SAVIO/CAPPUCCI) Kerr, Ray, Ebersole, Mark, Mathis, Ward, Bender

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the RES proposed FY 1984-1985 research funding and programs in this area and to develop comments on the Long-Range Research Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposed funding and program plans in this area as are available and the Long Range Research Plan.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
July 6, 1982 (morning)	HUMAN FACTORS	(FISCHER) Ward, Lewis, Mathis, Moeller (tent.), Ray
		Cons: Arnold, Buck, Debons, Keyserling, Pearson, Salvendy, Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: The review the NRC's proposed FY 84-85 programs and budget, and to develop specific comments on the Long-Range Research Plan as they relate to Human Factors. RES will present its FY 84-85 Human Factors research budget proposal to the ACRS. The Subcommittee will then have discussions with NRC/RES and user offices.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

RES proposed budget as submitted to the EDO.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
~~JULY 6, 1982~~ ~~(tentative)~~ (afternoon) CANCELLED	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Bender, Carbon, Ward, Ray, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review proposed Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 7, 1982	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Okrent, Shewmon, Bender, Carbon, Mathis, Ward, Mark, Plesset, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Routine Process

Purpose: To continue discussion on the FY 1984 and FY 1985 NRC Safety Research Program and Budget and to prepare comments for use by the ACRS in its report to the Commission on the FY 1984 and FY 1985 NRC Safety Research Program Budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 14, 1982 (TENTATIVE)	CRBR WORKING GROUP ON STRUCTURES AND MATERIALS	(CAPPUCCI/QUITTSCHREIBER) Shewmon, Axtmann, Bender Etherington, Siess Cons. Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Shewmon

Purpose: To discuss criteria for elevated temperature design (N-47), including supports.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be supplied later.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 20, 1982 (tent.)	REACTOR RADIOLOGICAL EFFECTS	(TANG/MCKINLEY) Moeller, Axtmann, Ebersole, Ray, Okrent (tent.)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: (1) To review PWR Occupational Radiation Exposure histories and recent experiences in exposure reduction at several plants.
(2) To discuss the status of requirement for "Radiation Protection Plan" (Rev. to Part 20).
(3) To discuss findings of Health Physics Appraisals.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees," dated March 1981.

2. NUREG-0855, "Health Physics Appraisal Programs," dated March 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 21 or 22, 1982	REACTOR OPERATIONS	(MAJOR) Mathis, Bender, Ebersole, Kerr, Moeller, Okrent, Ray, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: W. Mathis

Purpose: (a) To discuss NRC's enforcement policy governing enforcement actions for violations of NRC regulations and license applications. Including a discussion of the types of enforcement actions available to the NRC and the circumstances under which they will be used.

(b) A discussion of the regionalization effort within NRC's Office of Inspection and Enforcement, which is beginning to place the responsibility for technical reviews with the regional field offices. Current progress, future aims, relationship between regional offices, headquarters could be topics for discussions.

(c) Current status of IE's Performance Appraisal Team (PAT) inspection program and the Systematic Assessment of Licensee Performance (SALP) program. How IE perceives these programs' interface with the INPO evaluation programs would also be of interest.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Nuclear Regulatory Commission, 10 CFR 2, "General Statement of Policy and Procedure for Enforcement Actions." Revised general statement of policy. Effective Date: March 9, 1982

2. Memorandum for: W. Kerr From: M. Libarkin, Subject: NRC Enforcement Policy, dated May 12, 1982.

3. Nuclear Regulatory Commission, SECY-82-150, Subject: "The Performance Appraisal Team (PAT) Inspection Program," dated April 8, 1982.

4. Memorandum for: Mr. Ward and Mr. Bender, From: Dr. Kenneth D. Kirby, Subject: The IE Performance Appraisal Team Inspection Program, dated May 7, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 10, 1982 (TENTATIVE)	WATTS BAR	(BEAL/QUITTSCHREIBER) Ebersole, Bender, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To complete the review of the application for an OL.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER due June 26, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 10, 1982	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Carbon, Ray, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review proposed Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 11, 1982	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Okrent, Plesset, Ward, Shewmon, Bender, Kerr, Moeller, Mark, Carbon(tent.)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: RES Staff

Purpose: To provide early input to the RES Staff for the preparation of the Long-Range Research Plan for FY 1985-1989.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 18, 1982 (TENTATIVE)	CRBR WORKING GROUP ON STRUCTURES AND MATERIALS	(CAPPUCCI/QUITTSCHREIBER) Shewmon, Axtmann, Bender, Etherington, Siess Cons. Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: P. Shewmon

Purpose: To discuss "leak before break" criteria for CRBR, overall leakages, and leak detection. Inservice inspection plan and the structural integrity of critical transition joints will also be discussed.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

W-ARD-0185
Memo, Cappucci to Shewmon, "Proposed Review Plan for CRBR Working Group on Structures and Materials," dated 4/22/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Late August	WPPSS-2	(GRIESMEYER/QUITTSCHREIBER) Plesset, Ebersole, Mark, Mathis, Ward (tent)

LOCATION: Hanford, WA

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER, April 12, 1982 without seismic evaluation.
SSER with seismic evaluation due June 4, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To Be Determined	TRANSPORTATION OF RADIOACTIVE MATERIALS	(DURAISWAMY) Siess, Bender, Mark Cons: Langhaar, Shappert, Zudans

LOCATION: To be determined

BACKGROUND:

Who proposed action:

Purpose: To continue review of the adequacy of the NRC package certification procedures.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To be determined	Metal Components	(IGNE) Shewmon, Ward, Axtmann, Bender, Etherington, Mathis, Plesset Consultants: Kouts, Theofanous, Catton, Zudans, Irwin, Abbott, Binford, Fitzsimmons

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: P. G. Shewmon

Purpose: To continue the review regarding pressurized thermal shock.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

November 18, 1976

APPENDIX IV
EXCERPTS FROM MIDLAND PROJECT STATUS REPORT

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: SUPPLEMENTAL REPORT ON MIDLAND PLANT UNITS 1 AND 2

Dear Mr. Rowden:

In response to a request from Chairman D. M. Head of the Midland Atomic Safety and Licensing Board, the Advisory Committee on Reactor Safeguards has reviewed the record pertaining to the Midland Plant Units 1 and 2 as reported in its letter of June 18, 1970. The items listed below are those items referred to in its paragraph on "other problems related to large water reactors" which had been previously "identified by the Regulatory Staff and the ACRS," and which the Committee considered applicable to the Midland Plant. Following each item, the Committee has included an amplifying statement based on ACRS reports on other similar commercial nuclear reactor power plants which had been reviewed during the months prior to the Committee's review of the Midland Plant. Copies of the referenced ACRS reports are attached.

1. Separation of protection and control instrumentation - The Applicant proposed using signals from protection instruments for control purposes. The Committee believed that control and protection instrumentation should be separated to the fullest extent practicable, and recommended that the Applicant explore further the possibility of making safety instrumentation more nearly independent of control functions. (Three Mile Island, 1/17/68).

2. Vibration and loose parts monitoring - The Committee recommended that the Applicant study possible means of in-service monitoring for vibration or the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system, and implement such means as found practical and appropriate. (Palisades, 1/27/70).

3. Potential for axial xenon oscillations - The Applicant was continuing studies on the possible use of part-length rods for stabilizing potential xenon oscillations. Solid poison shims were to be added to the fuel elements if necessary to make the moderator temperature coefficient more negative at the beginning of core life. (Three Mile Island, 1/17/68).

ATTACHMENT F

4. The behavior of core-barrel check valves in normal operation - The Applicant had proposed core-barrel check valves between the hot leg and the cold leg to insure proper operation of the ECCS under all circumstances. Analytical studies had indicated that vibrations would not unseat these valves during normal operation. The Committee desired that this point be verified experimentally. (Three Mile Island, 1/17/68).

5. The potential consequences of fuel handling accidents - The Committee believed that further study was required with regard to potential releases of radioactivity in the unlikely event of gross damage to an irradiated subassembly during fuel handling and the possible need for a charcoal filtration system in the fuel handling building. The Committee recommended that this matter be resolved in a manner satisfactory to the Regulatory Staff. (Hutchinson Island, 3/12/70).

6. The effects of blowdown forces on core internals - The Committee recommended that the Regulatory Staff review the effects of blowdown forces on core internals and the development of appropriate load combinations and deformation limits. (Three Mile Island, 1/17/68).

7. Assurance that LOCA-related fuel rod failures will not interfere with ECCS function - The Committee desired to emphasize the importance of work to assure that fuel-rod failures in loss-of-coolant accidents will not affect significantly the ability of the ECCS to prevent clad melting. (Three Mile Island, 1/17/68).

8. The effect on pressure vessel integrity of ECCS induced thermal shock - The Committee recommended that the Regulatory Staff review analyses of possible effects, upon pressure-vessel integrity, arising from thermal shock induced by ECCS operation. (Oconee, 7/11/67).

9. Environmental qualification of vital equipment in containment - The Committee recommended that attention be given to the long-term ability of vital components, such as electrical equipment and cables, to withstand the environment of the containment in the unlikely event of a loss-of-coolant accident. (Palisades, 1/27/70).

10. Instrumentation to follow the course of an accident - This item related to the development of systems to control the buildup of hydrogen in the containment, and of instrumentation to monitor the course of events in the unlikely event of a loss-of-coolant accident. (Hutchinson Island, 3/12/70).

11. Improved quality assurance and in-service inspection of primary system - The Committee continued to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommended that the Applicant implement those improvements in quality practical with current technology. (Oconee, 7/11/67).

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachments:
1. Request from Chairman D. M. Head, AS&LB, dated 10/14/76
2. Report on Midland Plant Units 1 & 2, dated 6/18/70
3. Report on Hutchinson Island Unit No. 1, dated 3/12/70
4. Report on Palisades Plant, dated 1/27/70
5. Report on Three Mile Island Nuclear Station Unit 1, dated 1/17/68
6. Report on Oconee Nuclear Station, Units 1, 2, and 3, dated 7/11/67

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 14, 1976

RECEIVED

OCT 15 AM 9 12

Dr. Dade W. Moeller
Chairman, Advisory Committee
on Reactor Safeguards
1016 - H Street
Washington, D.C. 20555

RE: CONSUMERS POWER COMPANY (MIDLAND PLANT, UNITS 1 & 2), DOCKET NOS. 50-329/330

Dear Dr. Moeller:

The U.S. Court of Appeals for the District of Columbia Circuit in Aeschliman v. NRC, Appeal Nos. 73-1776 and 73-1867 (July 21, 1976), ruled that your Committee's report on the Midland facility should be returned to the ACRS for clarification, in particular for further elaboration on the reference to "other problems".

This Atomic Safety and Licensing Board has been reconvened by the Commission to conduct the reopened proceedings required by the above-identified Court decision. This reopened hearing includes the issue of clarification of the ACRS report. As required by the Court, we are hereby returning the ACRS report of June 18, 1970, with its supplement of September 23, 1970, to you for clarification. Would you advise us of what action your Committee is taking or plans to take with regard to Midland in response to the Court order. We would also appreciate an estimate of the time that will be required for the clarification called for by the Court.

A prompt reply would be helpful to the Board in assessing scheduling requirements for the reopened proceeding.

Very truly yours,

Daniel M. Head, Chairman
Atomic Safety and Licensing Board

Enclosure: ACRS report

cc w/o encl: Harold L. Reis, Esquire
Myron M. Cherry, Esquire
Jane A. Axelrad, Esquire
James N. O'Connor, Esquire

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 16, 1977

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: ADDITIONAL REQUEST FOR INFORMATION FROM THE MIDLAND AS&LB

Dear Mr. Rowden:

The Committee has received an additional request from the Atomic Safety and Licensing Board in the Midland case for further elaboration and "treatment" of matters mentioned in the Committee's Supplemental Report to you of November 18, 1976 and attachments thereto. That report, you may recall, was written in response to a previous request which followed directly from the decision in Aeschliman vs. NRC. A copy of the most recent AS&LB request, dated January 28, 1977, is attached.

Although the Committee is willing to provide reasonable and necessary clarification of its recommendations and opinions, we believe that the Board in this case has misinterpreted the Aeschliman decision and has embarked on a course which, if pursued, could involve the Committee in an unnecessary and potentially unending series of requests for clarification and elaboration of its reports, in connection with not only the Midland proceeding, but other proceedings as well. The Board's "three areas of comment" are addressed below:

I.

The Board notes two specific paragraphs of interest to the Midland proceeding in a set of ACRS meeting minutes (106th ACRS meeting held February 6-8, 1969) during which the Midland project was discussed, and the Board requests "further comment under the rules set forth in the Aeschliman case" regarding these two paragraphs "as well as any other 'matters of concern' (including any matters mentioned in furnished or unfurnished minutes)" and requests that these matters be treated fully by the Committee in accordance with the following excerpt from Aeschliman vs. NRC:

> "At a minimum, the ACRS report should have provided a short explanation, understandable to a layman, of the additional matters of concern to the Committee, and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail."

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ATTACHMENT G

In the opinion of the Committee, the Board has incorrectly concluded that all topics discussed during an ACRS review, and recorded in the meeting minutes, are "matters of concern" to the Committee in the context of the Aeschliman decision. "Items of concern" to the ACRS at the completion of its review are identified in the Committee's report and have been explained in the Committee's Supplemental Report of November 18, 1976, in language "understandable to the layman" as required by the Aeschliman decision. Many other items of interest are documented and discussed during the course of an ACRS review and are not identified as matters of concern in the ACRS report. Some of these items are considered satisfactory or are adequately resolved by amendment of the application or other means during the review process. Some represent points of general information, some represent matters that the Committee explores on a generic basis.

It should be noted that the Aeschliman decision did not address the content of ACRS meeting minutes or other information available to or considered by the Committee but was limited (see Attachment 2) to those matters identified in ACRS reports as items of concern. To require that the ACRS address in its report every item discussed or considered during the course of a review is impractical and unnecessary.

For example, the suitability of the Midland Plant for the proposed Midland site was discussed at length during six Subcommittee meetings held on January 22 and February 4, 1969, and March 24, April 24, June 10, and September 14, 1970, and at five full Committee meetings held on February 6, 1969, and April 9, May 8, June 11-13, and September 17-19, 1970; appropriate safety features were included in the design for this reactor at this site. The minutes of these meetings have been in the public domain since 1974.

II.

This section of the Board's request deals with the substance of the Committee's Supplemental Report of November 18, 1976 and requests that the Committee further clarify one of its recommendations, specifically, that the Committee specify the "danger" that is of concern if instrumentation and control are not separated; further describe the type of separation required (e.g., physical or other); and specify a standard for conformance.

The Board further notes that this illustration is only an example of an area where a problem may exist and further elaboration of other matters may also be required.

The Committee appreciates the Board's desire and interest in understanding the issues identified by the Committee but does not agree with the method being used to develop this understanding. The Committee's Supplemental Report dated November 18, 1976 did provide a brief description of the items considered to have been problems by the Committee and specific cross references to other applicable cases, as required by the Court in Aeschliman vs. NRC.

The desire for additional clarification by the Board with respect to specific questions of this nature is best served by:

* Examination of the record related to the Midland review and the review of other cases specifically cross-referenced by the Committee.

* Discussion with the NRC Staff who participate in the Committee's review process, are thoroughly familiar with the problems and issues involved, and are participants in the hearings.

The example chosen by the Board is itself a case in point. The matter of separation of control and protection instrumentation relates to reducing the probability of failure due to a common cause and is dealt with generically by Section 7.3 of the NRC's Standard Review Plan, which provides guidance to Staff reviewers; the Committee provided a specific reference, in its November 18, 1976 Supplemental Report, to the Three Mile Island Nuclear Station, Unit 1, in response to the Court's order to provide a "cross-reference to the previous reports in which those problems and the measures proposed to solve them were developed in more detail." The July 11, 1973 Safety Evaluation of the then Directorate of Licensing in the matter of Three Mile Island, Unit 1, deals directly with this ACRS concern in Section 7.5, "Separation of Control and Protection Systems" and the Committee's August 14, 1973 report on operation of Three Mile Island, Unit 1, indicates that this matter was no longer of concern for the Three Mile Island case. In the Midland case, the Committee will review the adequacy of the final design as it exists at the time it reviews the Midland Plant for an operating license.

In general, we believe that examination of the implementation of the Committee's advice and of any resulting changes in the application are best left to the NRC Staff which plays a direct role in the hearing, and that any evidence relating to such matters should be sought from them. Indeed, the Court in Aeschliman itself notes, "This is not to say that an ACRS report must contain detailed factual findings of the kind necessary to aid judicial review. Under Commission rules, when ACRS conclusions are controverted, a factual record is compiled anew before the Licensing Board."

The NRC Staff (previously, the AEC Regulatory Staff) has routinely addressed itself to the comments and recommendations in ACRS reports for many years as part of the NRC hearing process. A typical example is to be found in Supplement No. 1 to the Directorate of Licensing's Safety Evaluation for Three Mile Island, Unit 1, dated October 15, 1973. Chapter 4 of that document is addressed entirely to the issues raised in the ACRS report of August 14, 1973.

III.

This section of the Midland Board's most recent request points to perceived "ambiguities" resulting from an examination of several ACRS reports provided as references in the Committee's Supplemental Report of November 18, 1976. The Board notes that those references contain "ambiguities" similar to the ones cited by the Court in Aeschliman and points, by way of example, to the Committee's reference to "other problems" in it's Hutchinson Island report of March 12, 1970. The Board asks that any of the "other problems" which apply to Midland be identified and described as the Court directed.

The Committee's Supplemental Report of November 18, 1976 was provided as ordered by the Court to identify those "other problems" which had been considered applicable to the Midland Plant at the time of the CP review and which were noted generically in the ACRS report of June 18, 1970. Any items not so identified in the Committee's November 18, 1976 report were not considered applicable to Midland during the CP review.

The Committee will be in a position to update this list and address the current status of specific items when it has completed its review for an Operating License for the Midland Plant. This review has not yet been scheduled.

In summary, the Committee believes that the response already provided in its Supplemental Report of November 18, 1976, fully meets the requirements of the Aeschliman Court since:

(1) The Court requested elaboration only of those items referred to in the Committee's original report as "other problems" and no others.

(2) The Committee's Supplemental Report of November 18, 1976, did provide a "short explanation understandable to a layman of the additional matters of concern to the Committee and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail" as specifically directed by the Aeschliman decision.

(3) The Committee's Supplemental Report of November 18, 1976, fully identified all additional matters of concern to the Committee during its CP review of the Midland Project.

The ACRS does not feel that any further clarification of its reports on Midland is necessary.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachments:
1. F. J. Coufal, Chairman, AS&LB letter to M. Bender, ACRS, dated January 28, 1977.
2. Excerpt from the decision in Aeschliman vs. NRC.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Attachment 1

January 28, 1977

Myer Bender, Chairman
Advisory Committee on Reactor
Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

RE: MIDLAND PLANT UNITS 1 AND 2

Dear Mr. Bender:

The Board has reviewed the reports in evidence in this case by the Advisory Committee on Reactor Safeguards (ACRS) (Staff Exhibits 1, 2 and 3) and has decided to return those responses to the ACRS for further elaboration. These responses were originally submitted as a result of the decision in Aeschliman vs. NRC ___ F.2d ___, (DC Cir. 1976), slip opinion at 21. The Board has received two responses, both dated November 18, 1976, one including a copy of some minutes of an ACRS meeting discussing Midland and the other having no such enclosure. We have three areas of comment.

I.

The minutes mentioned contain references which we believe require further comment under the rules set forth in the Aeschliman case. Two of these are:1/

> "c. Exclusion area and low population zone - the exclusion area extends 1100 meters from the proposed plant and includes a portion of the Dow plant, including 53 Dow employees; the low population zone extends to three miles and includes all of the Dow plant and part of the City of Midland. The site received a-34 index rating when compared to the hypothetical reference site (considering the maximum population in the Dow complex).

1/ Others may exist. We presently focus on these because of their relationship to current suspension hearings.

"g. Other aspects - ... the Committee mentioned but did not explore in any depth: the suitability of B&W reactors for marginal sites, protection required against reactor vessel splits, cavity flooding systems, and the use of process steam in products to be consumed by people."

Neither the ACRS letter dated June 18, 1970, nor the one dated November 18, 1976, furnished to meet the requirements of Aeschliman, mention these matters. We believe that the court, in the words that are set out in footnote 2 below requires that these matters, as well as any other "matters of concern" (including any matters mentioned in furnished or unfurnished minutes) be treated fully by the Committee.

The significance of the rating system referred to in item (c) and the hypothetical reference site is not apparent nor are there explanatory references cited. Furthermore, the Board does not understand what the ACRS means by "the suitability of the B&W reactors for marginal sites" in item "g."

II.

We are concerned with the adequacy of some responses in the November 18, 1976, letter to meet the Aeschliman test. To illustrate we set out the first of the eleven topics in the letter:

"1. Separation of protection and control instrumentation - The Applicant proposed using signals from protection instruments for control purposes. The Committee believed that control and protection instrumentation should be separated to the fullest extent practicable, and recommended that the Applicant explore further the possibility of making safety instrumentation more nearly independent of control functions. (Three Mile Island, 1/17/68).

2/ "At a minimum, the ACRS report should have provided a short explanation understandable to the laymen of the additional matters of concern to the Committee and a cross-reference to previous reports in which those problems and the measures proposed to solve them were developed in more detail."

It is unclear to the Board what this paragraph means. The danger is not specified and it is unclear as to whether the "separation" mentioned refers to a physical separation of components or to the necessity for separate energy sources for signals and controls or to some other separation. No standard is set for the Applicant's (now Licensee's) conformance. The referenced documentation (Three Mile Island, January 17, 1968) says no more. There is in that document a list of references (some marked ACRS Office Copies Only) which may clarify the matter. But no direction is given as to which of these references is relevant to the particular subject.

This illustration is exemplary only and whether the same infirmity exists in other items is a problem we have not had the opportunity to address. We furnish this now so that the Committee is made aware of our concern and so that further elaboration is not delayed.

III.

The letter of the ACRS to Chairman Rowden, November 18, 1976, referred to other ACRS letters. Those letters contain items which have ambiguities similar to those disapproved in Aeschliman. For example, the March 12, 1970 letter on Hutchinson Island stated:

> "Other problems related to large water reactors have been identified by the Regulatory Staff, and the ACRS and cited in previous ACRS Reports" (p. 3).

Those items, we feel, need to be identified if they apply to Midland and if they do, to be described as the Court directed. See footnote 2 hereof.

* * *

We write this under what we perceive to be out duty under the direction given in the Aeschliman case3/ without waiting to fully identify all of the possible areas

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3/ A "sua sponte" request for elaboration.

of concern relative to the November 18, 1976, letter. We do so because we are in the midst of suspension hearings and will need a resolution of this matter as soon as it may reasonably be furnished.

Respectfully submitted,

Frederic J. Coufal, Chairman
Atomic Safety and Licensing
Board

(2) The Court concluded in the Aeschliman decision that:

> The ACRS report in this case must be evaluated in light of the congressional purposes. While the reference to "other problems" identified in previous ACRS reports may have been adequate to give the Commission the benefit of ACRS members' technical expertise, it fell short of performing the other equally important task which Congress gave ACRS: informing the *public* of the hazards. At a minimum, the ACRS report should have provided a short explanation, understandable to a layman, of the additional matters of concern to the committee, and a cross-reference to the previous reports in which those problems, and the measures proposed to solve them, were developed in more detail. Otherwise, a concerned citizen would be unable to determine, as Congress intended, what other difficulties might be lurking in the proposed reactor design. Since the ACRS report on its face did not comply with the requirements of the statute, we believe the Licensing Board should have returned it *sua sponte* to ACRS for further elaboration of the cryptic reference to "other problems." [11]
>
> Turning to the propriety of discovery directed to individual ACRS members and ACRS documents, we conclude it was not error to deny these requests. ACRS' unique role as an independent "part of the administrative procedures in chapter 16 of the act," *supra*, is sufficiently analogous to that of an administrative decisionmaker to bring into play the rule that the "mental processes" of such a "collaborative instrumentalit[y] of justice" are not ordinarily subject to probing. *United States* v. *Morgan*, 313 U.S. 409, 422 (1941). This rule is particularly apropos in light of ACRS's collegial composition such that no individual may speak for the group as a whole. Where an ACRS report on its face omits material

[11] This is not to say that an ACRS report must contain detailed factual findings of the kind necessary to aid judicial review. Under Commission rules, when ACRS conclusions are controverted, a factual record is compiled anew before the Licensing Board. *See* 10 C.F.R., pt. 2, App. A, V(f)(1) (1976).

information, the appropriate course is not discovery but to return it for supplementation. *Cf. Dunlop* v. *Bachowski*, 421 U.S. 560, 574-75 & n. 11 (1975). We merely hold here that neither the Atomic Energy Act nor general principles of administrative law required the Commission to grant Saginaw's discovery requests.[38]

On remand, the ACRS report should be returned to the ACRS for clarification of the ambiguities noted above.

[38] The case as presented calls upon the court to make no decision whether the Federal Advisory Committee Act, 5 U.S.C. App. I § 10(b) (Supp. III, 1973), entitles a party upon proper request to have access to data which were before the ACRS.

APPENDIX V
STATEMENT FROM MARY SINCLAIR ON THE
MIDLAND NUCLEAR PLANTS, JUNE 4, 1982

Mary Sinclair
June 4, 1982

<u>Statement Before the Advisory Committee on Reactor Safeguards, Washington, D.C. on the Midland Nuclear Plants</u>

My purpose in being here today is to bring some insights to you on the role of the Advisory Committee on Reactor Safeguards (ACRS) as the public perceives it, and to compare it to what has actually been accomplished for public safety by this Committee as seen by those few of us who have a long and extensive experience with the nuclear power licensing process.

The public has lost confidence in the nuclear power plant licensing process. Since this ACRS review is a part of that process, the reasons for this loss of confidence should be of importance to you.

Chairman Palladino has appointed a Task Force to study the nuclear licensing process. In my view, this forum for review of the Midland nuclear plants can provide much substance for studying the reasons for loss of public confidence and what is deficient in your review methods.

This loss of confidence in nuclear plant licensing has come about in spite of long and detailed licensing staff reviews and a supposedly objective overview by this prestigious ACRS. It has occurred primarily because very serious problems have come to light at many plants following operating license approval after these extensive reviews. These problems have been the source of much pain, anxiety and high cost to the public. The ACRS must accept some responsibility.

Now, I was personally very impressed with the tough questions that the ACRS subcommittee asked the NRC staff and the applicant in Midland last week. I was also impressed with the range and quality of expertise that this Committee has.

Therefore, it is important to look at other factors to identify reasons why the ACRS reviews are losing their value as far as the public is concerned.

I believe I should establish my credentials for what I am to say here today.

As a science writer and editor, I was aware of the role of the ACRS from its beginnings. I have followed the patterns of its work much more closely since I first became involved in the Midland nuclear plant licensing. The misleading propaganda with which the Midland nuclear plants, as well as all nuclear plants, were being promoted in the late '60's, as well as the incredible record of poor quality control that was being disclosed during the Palisades operating license hearings at that time where Consumers Power Co. was also the applicant, brought me into the Midland nuclear plant licensing procedure. I have been involved in nuclear plant licensing since that time--over 10 years.

In 1970, I also asked for a special graduate program at the University of Michigan which would allow me to study in any department of the University, if the department head approved, that would increase my knowledge and access to libraries and other resources in order to follow the Midland n-plant licensing with more expertise and to follow nuclear power development in general. The Midland project was touted as the world's largest nuclear-industrial project, tieing together as it does a huge chemical complex, The Dow Chemical Co. and two large nuclear plants. It was being built by Consumers Power Co. and Bechtel, the same utility and architect engineer who had done so poorly at the Palisades nuclear plant. I considered it a worthy commitment of my time, energy and study to follow its construction since these plants are two miles from my home. My Masters' thesis at the University of Michigan was on the environmental and social impact of nuclear power. It is from this background that I am making the following observations.

I have a little good news. The ACRS is good at identifying many current safety problems and potential hazards in nuclear power. I have found the Committee's data very useful on many occasions and have applied what I was able to during the Midland nuclear plant licensing. But, the Committee itself has not followed up on their own recommendations. I mentioned a number of useful suggestions that the ACRS Committee had made in 1969 that applied to the Midland nuclear plants and other installations in my statement before the Midland subcommittee on May 20. I also proved that 10 years or more later, little or nothing had been done about many of them at nuclear installations.

Where citizen intervenors have taken up your ideas, however, there have been positive results. For example, in 1969, the ACRS recommended that some method should be devised to control the hydrogen that would form over the reactor core under certain accident conditions. By March, 1979, this had not been followed up at many plants including Three Mile Island-2, and there it had frightening consequences. With strong citizen intervenor participation in the Midland licensing procedures, however, it was followed up and there were hydrogen recombiners in these plants already installed in March, 1979. This demonstrates that third party input has an important safety value.

Those of us who have a comprehensive knowledge of what has been happening at the construction site at Midland have followed details of what is revealed in the hearing transcripts and documents. We have listened to the summaries, especially as they have to do with QA and QC matters, that the NRC staff and the applicant have provided for the review of the subcommittee on Midland. We assure you that these summaries grossly misrepresent and whitewash the extent of the problems of these nuclear plants.

Barbara Stamiris, a citizen intervenor, who has done extensive work on the soil settlement issue, stated in her oral statement at the Midland review that she was "shocked" at the glossing over of serious problems and the omissions in these summaries. As she promised, she has written a detailed and documented review for the ACRS of the extent of QA and QC problems which were not mentioned. Her statement indicates the extent to which both the NRC staff and the applicant have covered-up the seriousness of these issues.

NRC inspectors themselves have made some significant comments which were not relayed to you.

Last August, Joseph Kane who is chief geotechnical engineer for the NRC staff stated that if safety were the major consideration that removal and replacement of the diesel generator building would have been the better option even with the amount of stress that building showed before preloading began, But since costs and construction schedules are so important at Midland, that option couldn't be exercised, Joseph Kane said. (p 4209-4210) In other words, the NRC geotechnical specialist had assessed the serious problems with that building years ago and came to the same conclusion that our expert witness, Dr. Charles Anderson, made in his final statement to the ACRS subcommittee on May 21 of this year. But the ACRS subcommittee was not advised of this.

Another Region III inspector, Eugene Gallagher, said under oath, "You're talking about a plant 70% complete that is crippled." (p. 2466) He also said the problems at Midland were unprecedented at any other site. (p. 2463) In another instance, Gallagher said he had a hard time giving quality control assurance as far as Consumers Power Co. was concerned because the utility and Bechtel had obviously had a hard time moving soil from one place to another and doing it right, and in operating the plant they would be handling highly sophisticated equipment some of which had never been in use before. (p. 2441)

Many other significant items have not been discussed. For example, underground safety piping is overstressed in at least 12 locations due to uneven settlement. This piping--the cooling lifelines of the plant--will require a permanent stress monitoring system. Unusual corrosion of stainless steel piping was identified in 1979, but the combined effect of this corrosion with settlement stress has not been addressed. (Feb. 18, '82 Testimony)

This plant will operate in an area with more chemical pollutants than is usual because of the proximity of The Dow Chemical Co. It is known that the combination of radiation and chemicals result in multiple factor interaction that can be significant due to synergistic effects. The corrosion potential of this interaction has not been addressed. Corrosion is a serious problem for nuclear plants even in better environments.

Quality control problems have beset the Midland nuclear plants construction for years. The ASLB Appeals Board found the QA so bad in the earliest construction of the plant, that they elicited a promise of reform from Cons Power Co on this issue in 1973 as a condition of affirming the license. In spite of this, the same QA-QC problems occurred. When our attorney brought this matter to the Appeal Board's attention, the Board finally angrily wrote, "What we have here is a pattern of repeated, flagrant and significant QA violations of a non-routine character--coupled with an unredeemed promise of reformation." (Letter to L. Manning Muntzing, Nov. 26, 1973)

This is about as adverse a characterization of management attitude as could be made. ~~But~~ Add to this the fact that both Bechtel and Consumers Power Co. knew that the soil was poorly compacted sitewide in 1977 but went ahead and built safety-related and other buildings on it anyway, and you have an attitude that approaches criminal negligence.

The letter to Muntzing led to the first show cause hearing on quality control in the country. But, this did not improve matters as Ms. Stamiris has pointed out in her statement to this Committee.

Michigan's other reactors have had serious problems that should have been identified before ACRS approval was given.

At the Palisades nuclear plant, the rad waste holding tank never was able to operate. Did the applicant's or NRC staff review disclose this to the ACRS in their summary review of that plant? The company continued to operate the plant and not only didn't notify the NRC, but deliberately concealed the fact. Did the staff or the company disclose the extent of the QA problems at Palisades? These have resulted in one of the poorest operating records in the nation and have been a costly burden to ratepayers and stockholders alike.

The ACRS is responsible for oversight in proposed changes at operating facilities (p. 387, Nuclear Safety, Vol. 20, No. 4, July, 1979) However, the attempt to experiment with a full loading of plutonium fuel at the Big Rock nuclear plant was approved by the Atomic Energy Commission simply with an amendment to the original license with no public notice and no ACRS review. No environmental impact statement had ever been made for Big Rock. As one of the early plants, it did not have, and still does not have, the basic safety systems that are required at other plants. Yet, Consumers boasted in 1972 that they were going to begin the plutonium economy for the nation at Big Rock. (Nucleonics Week, Dec. 2, 1972)

Only an extraordinary and dedicated effort by a citizens' group, the West Michigan Environmental Action Council, halted full plutonium loading at Big Rock. Partial loading of plutonium, however, has gone on at Big Rock for years. Citizens who are fighting fuel compaction at Big Rock were allowed by their Board to review the issue of the possibility of criticality ~~of~~ in compacting spent fuel storage for this reason. Without expert witnesses or funds, their ability to explore this issue is very limited. Some of the expertise of this ACRS Committee should be there to help them. Their hearings come up this month. Here is a major risk being forced on people since we have so much excess electrical capacity in Michigan (over 35% by Consumers Power Co. testimony) that we have no need for the Big Rock facility to operate at all. But, the ACRS remains distant from the real problems citizens face even though citizens are paying them to provide safety.

On the national scene, ACRS, of course, approved TMI-2 with all of its many problems which brought it to within an hour of a total meltdown in March, 1979. ACRS also approved Brown's Ferry where a fire disclosed the weakness of design for electrical equipment that controls safety systems. The same problem exists at the Donald C. Cook plants in Michigan. The Union of Concerned Scientists has recommended that those plants be shut down for this and other deficiencies. But no reviews and recommendations are made on these types of issues. The public must wait for accidents to happen to get any action.

The ACRS has approved through their routine review process Diablo Canyon, Zimmer, Davis Besse, Rancho Seco, Ginna, Crystal River and TMI-1 and 2 and many other nuclear plants which have subsequently had serious problems or have had serious flaws found in construction or design.

These problems are disturbing and painful for the public immediately affected. They are costly for everyone. Is it any wonder that public confidence in nuclear power plant licensing is so low?

I am convinced that ACRS itself is competent. But, all of these facts and examples point to the fact that the method of review which ACRS has accepted in the past is faulty. In fact, from a legal, technical and scientific point of view, I contend that the data base on which ACRS has been depending writing ~~for~~ its final letters on nuclear plants to the chairman of the NRC is and always has been untenable and unsound.

This same kind of review had been planned as the routine for Midland. But we decided at Midland to change the routine and create a third party objective view through citizen disclosure of important facts from the record that have not been revealed. I believe that Barbara Stamiris and I have demonstrated that without some objective probing of the summaries provided by the NRC staff and the applicant, the ACRS has a very limited and inaccurate data base for its review. (Both the Kemeny Commission and Rogovin Reports following the TMI accident have recommended third party review through funding citizen intervention). We have also provided an objective review of a critical unresolved problem, -the soil settlement issue and resulting question of the structural integrity of at least two safety-related buildings, -with Dr. Charles Anderson as our consultant.

It has been obvious from all that transpired in Midland ~~last week~~ on May 19-21 that Consumers Power Co. is bitterly opposed to any objective review of that plant's construction.

But, at least this time you have some additional facts to consider beyond those of the staff and the applicant.

However, the fact remains that the ACRS meets and, on such a speedy basis, hearing only the nuclear promoters with no independent review, makes judgments of great importance to the public that affects their health, safety and economy. This process raises questions about the credibility of ACRS judgment. These reviews are also a costly item for citizens as part of nuclear regulation.

The fact that every letter without exception at the operating license stage always and inevitably states after 6 or 8 paragraphs of recommendations, that the facility can be operated without undue risk to public safety also raises questions about the credibility of your deliberations.

In the past, of course, for the general public, this type of letter from such an august body has provided an illusion of an assurance of safety. But the many serious problems now make people realize that this illusion of safety is worse than no review at all.

Some nuclear facilities we now know are so bad they should never have been allowed to operate. TMI-2 is one, Palisades, with its poor quality control and negligence in management is another. Diablo Canyon and Zimmer should not be allowed to operate. With its history of QA violations and incredibly poor management decisions and attitudes, Midland should not be allowed to operate either.

My conclusion is this, that to the extent that the ACRS review letter for the operating license is primarily based on the limited and carefully controlled information from the NRC staff and applicant with no objective third party review, the ACRS becomes just another part of the promotional package for the nuclear industry. It raises the question, in these difficult times, with the excess reserve electrical capacity that we have in Michigan and in this country whether the public can afford promotion of a technology that cannot make it in the free market system.

In the decision-making process with nuclear technology, human survival is the issue. There is no margin for error, failure - or the chance to do it better next time. The consequences are irreversible, In this sense, these decisions are as profound as mankind has ever had to make. You are the professionals upon whom we must depend to seek and establish the proper course for this technology for humanity. We cannot allow to do any less than just that.

Mary Sinclair

Mary Sinclair, M.S
5711 Summerset Dr.
Midland, MI 48640

E.Gallagher

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in the same kind of a way that we were talking about Mr. Selby.

A. I think you're comparing apples and oranges here. You're talking, one, about a failure to identify an item of non-compliance or withdrawing an item of non-compliance, and on the other hand, you're talking about a $27 million-plus fiasco.

There are no comparisons. You're talking about a plant that's 70 percent complete, that is crippled. You're not talking about an insignificant error in an inspection report.

Q. Actually, Mr. Gallagher, what we're talking about, I think, is a reasoned business judgment, a managerial judgment by the chief executive officer of a large utility, who has had years of experience in that, versus the judgment of a reactor inspector civil engineer.

I appreciate the fact that you note there are differences, because in fact there are. And it seems to me that simply stating that the chief executive officer ought to be held accountable doesn't take into account or consideration how large companies are run. And I wonder if you are really familiar with that?

MR. PATON: I object, Mr. Chairman. Mr. Gallagher was asked if he had any ideas which would be helpful to the Board, and because he made a suggestion, he's now

E. Gallagher

sites, to a much different degree, however.

Q But there have been, in fact, problems on other nuclear sites with something as simple as soils, haven't there?

A To a much lesser extent. The degree of the problem is what's important here. The extent of what has occurred at the Midland facility is unprecedented at any other facility.

Q The point remains, however, that other people have had some problems with something as simple as soils, or haven't they?

A Yes, of course.

Q In fact, a recent bulletin has been issued covering not only Midland but other plants as well, is that right?

A I wrote the bulletin.

Q So the answer is that, yes, a recent bulletin has been issued with regard to soils for not only this plant, but others?

A Excuse me. It was a circular; Inspection and Enforcement Circular.

Q To someone like me, they're the same. I'm sorry.

A It has a different regulatory posture.

Q So your answer is, yes, in fact there has been

Gallagher

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from my own shortcomings of what is technically feasible; but nevertheless, I certainly have some uneasiness about the suitability that the fixes can be accomplished successfully.

CHAIRMAN BECHHOEFER: Let me interrupt for one thing. If your recommendation concerning Mr. Selby, together with the recommendation on qualifications of QC inspectors--those were both put into effect, would you have less reservations, or is there anything else you can suggest to us as well?

THE WITNESS: What I'm simply saying is that the complexity of the remedy itself is somewhat difficult to come to grips with.

I do have one other, I guess, provision that might provide some better reasonable confidence that this task can be accomplished, and that is that the NRC as well provide a full-time geotechnical representative to observe, to witness, to inspect, to take independent measurements throughout the remedial fixes, and in doing so, provide the NRC with continuous confidence information, starting with the dewatering system installation, the monitoring of structures, preloading of the borated water storage tank, valve pits, underpinning the auxiliary building and field water valve pits, and piping systems embedded in the fill.

In other words, have the NRC have independent and continuous observation of the soils settlement remedies

E. Gallagher Testimony

post-December '79 testimony?

A To be honest with you, I have some real difficulties with that phrase "reasonable assurance". I'd like to ask the Board what you understand "reasonable assurance" to mean, because, quite frankly, I'm not certain.

Q Well, may I ask-- I don't know whether this will help. I will try.

In what areas do you have reservations about this statement?

MR. PATON: Judge Decker, could I interject? I think he said he had reservations as to the meaning of the word.

MR. DECKER: I understand that.

6 A I would agree that there is certainly the necessary tools and systems in effec[illegible]vide some acceptable level of confidence that the task can [illegible]plished. The reservation that I have is that having been so close to this problem for two and a half to three years, and knowing that simply the Company could not take soil material from one point of the site and place it in a sufficient manner to support the structures on another place on the site, and then recognize that we have extremely complex sophisticated and, in some cases, unprecedented remedial actions at a nuclear power plant, I have to have some reservation as to whether or not it can be successfully accomplished, and that may be just

be doing to the structure, and we would like some criteria by which you are going to evaluate the acceptability of these problems, the settlement and the cracking."

CHAIRMAN BECHHOEFER: And your response more or less anticipated what I viewed as a followup question. If these kinds of problems should have been anticipated, did Consumers or Bechtel, as the case may be, take sufficient -- assuming that those problems should have been anticipated, are there ways where the surcharging could have been planned by which these problems could be either minimized or alleviated?

MR. ZAMARIN: Chairman Bechhoefer, I hate to object to a Board question, but I do not think that he has testified that these problems were not anticipated. I think what he said in his last answer was there was a different approach taken; that was, by measuring cracks and things of that nature. And your question just now is based upon the assumption that these were not anticipated, and I do not think he said that.

CHAIRMAN BECHHOEFER: My question was a hypothetical: If these problems were ones which should have been anticipated, are there ways of minimizing or alleviating these problems? My next question after that would be: Did the company take these into account? My question was a hypothetical leading to --

MR. ZAMARIN: The problem I had with it, I think it implied that they had not considered them.

(Board conferring.)

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a better solution. If you are considering the other facets -- that is, the cost, the impact on schedule, and these are facets that engineers must address -- then it may not be the superior option. And I am saying "may not," because the decision to go with the preloading or the surcharging has an inherent assumption that ultimately it can be proven that the problems with overstressing and the pipes and the cracking of the structures are not so severe that it cannot be demonstrated that it is acceptable. So there is a risk in going with preloading.

We have not reached the bottom line with regards to accepting the effectiveness of the surcharge. There are studies now being conducted. They have not yet been submitted to the NRC to which we must address ourselves.

CHAIRMAN BECHHOEFER: Let me ask the followup question so we do not get too confused.

Back in '78, should problems such as -- well, problems of additional distortion of the installed conduits and pipes and additional cracking of the building, should those problems have been anticipated?

WITNESS KANE: Yes. And I think the biggest difference between the Applicant and the NRC over the preloading program has been Consumers saying, "We want to go ahead and do this and make measurements and demonstrate its adequacy," and on the other side we have the NRC saying, "We have trouble accepting this observational method when in fact we know what surcharging can

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NRC Geotechnical Specialist - Joseph Kane

concerning whether some of these things should have been anticipated.

BY MS. STAMIRIS:

Q Do you believe some of these problems should have been anticipated?

MR. ZAMARIN: I will object to the question. I do not know what the "problems" are.

CHAIRMAN BECHHOEFER: Are you referring to the two Mr. Kane mentioned?

MS. STAMIRIS: Yes. The additional distortion of piping and the additional cracking.

CHAIRMAN BECHHOEFER: Right.

WITNESS KANE: It seems to me there are two questions. One had to do with removal and replacement as being superior. And now the soils is the second question. Which one am I being asked to address?

BY MS. STAMIRIS:

Q Why don't you state one and answer it?

MR. ZAMARIN: Can we have a question of the witness?

CHAIRMAN BECHHOEFER: Mr. Kane can answer the question about if, with 20-20 hindsight, would removal and replacement have been a better option in 1978?

WITNESS KANE: The answer depends on the facts that must be addressed. When you are considering it from the standpoint of safety alone, it is my opinion that removal and replacement is

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JUNE 4, 1982

MIDLAND ACRS MEETING - INTRODUCTION BY NRC STAFF

OL REVIEW HISTORY TO DATE

11/18/77	OL APPLICATION DOCKETED
EARLY 1979	PREPARATION OF SER IN PROGRESS
3/28/79	TMI-2 ACCIDENT -- MIDLAND REVIEW SUSPENDED
EARLY 1981	OL REVIEW FULLY RESUMED
2/5/82	DRAFT ENVIRONMENTAL STATEMENT ISSUED
MARCH-APRIL 1982	FINAL SER OPEN ITEM RESOLUTION MEETINGS HELD WITH APPLICANT
5/6/82	SER ISSUED
5/20-21,6/2/82	ACRS SUBCOMMITTEE MEETINGS

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APPENDIX VI
NRC STAFF REVIEW STATUS

MIDLAND SER OPEN ITEM STATUS

SER OPEN ITEM NO.	DESCRIPTION OF ITEM	NEXT ACTION	APPROX. DATE
1.	NEARBY EXPLOSIVE HAZARDS	STAFF/APPL. MTG.	JULY 82
2.	TURBINE MISSILES	STAFF COMPLETE REVIEW	JUNE 82
3.	TORNADO MISSILE PROTECTION	STAFF/APPL. MTG.	JUNE 82
4.	ANALYSIS OF RCS & CORE COMPONENTS	APPL. SUBMIT ANALYSIS	MARCH 83
5.	SOILS SETTLEMENT ISSUE	APPLICANT SUBMITTAL	ONGOING
6.	SEISMIC & ENVIRONMENTAL QUALIFICATION OF EQUIPMENT	STAFF CONDUCT SITE AUDITS	ENV. JUNE 82 SEISMIC 09/82
7.	NATURAL CIRCULATION COOLDOWN ANALYSES	APPL. SUBMIT ANALYSIS	AUGUST 82
8.	HPI LINE MAKEUP NOZZLE CRACKING	APPLICANT SUBMITTAL	JULY 82
9.	REACTOR VESSEL HEAD VENT	APPLICANT SUBMITTAL	JUNE 82
10.	SECONDARY SYSTEM CONTAINMENT VALVE TESTING	STAFF COMPLETE REVIEW	JUNE 82
.	LEAK TESTING OF DHR AND RBCWS CONTAINMENT VALVES	STAFF COMPLETE REVIEW	JUNE 82
12.	APPENDIX R (FIRE PROTECTION)	STAFF COMPLETE EVALU.	JUNE 82
13.	AFW RING HEADER DISTORTION	APPLICANT SUBMITTAL	AUGUST 82
14.	EMERGENCY PREPAREDNESS PLAN	STAFF COMPLETE EVALU.	JUNE 82
15.	CONTROL ROOM DESIGN REVIEW	APPL. SUBMIT REPORT	DECEMBER 82
16.	SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS	APPL. SUBMIT RELIABILITY ANALYSIS	JUNE 82

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SPECIAL REVIEW AREAS

- QUALITY ASSURANCE
- REMEDIAL ACTIONS RELATING TO SOILS SETTLEMENT
- PROCESS STEAM SYSTEM
- B&W NSSS SENSITIVITY
- REACTOR VESSEL ANCHOR BOLTS
- PROXIMITY TO DOW FACILITIES
- SEISMIC RE-EVALUATION

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IMPROVEMENTS TO MIDLAND PLANT DESIGN

- PORV'S AND PORV BLOCK VALVES
- AFW SYSTEM
- FEEDWATER OVERFILL PROTECTION
- ANTICIPATORY REACTOR TRIP SYSTEM
- PRESSURIZER: LEVEL INDICATION AND HEATERS
- INADEQUATE CORE COOLING INSTRUMENTATION
- POST-ACCIDENT MONITORING INSTRUMENTATION
- HOT LEG VENTS
- FOGG LOGIC ADDED TO ESFAS
- OPERATIONAL IMPROVEMENTS

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APPENDIX VII
AGENDA FOR MIDLAND REVIEW

INCOMPLETE TENTATIVE SCHEDULE
MIDLAND PLANT UNITS 1 & 2
266TH ACRS MEETING
JUNE 4, 1982
(8:30 A.M. - 12:30 P.M.)

APPROXIMATE TIME		SPEAKER
	I. INTRODUCTION	
8:30 a.m.	A. Subcommittee Chairman's Report	D. Okrent
8:50 a.m.	B. Comments by Members of the Public	M. Sinclair
9:00 a.m.	C. Status of the NRC Staff Review 1. SER Open Items 2. Licensing conditions	R. Hernan
9:10 a.m.	D. Discussion [ascertainment of Committee preferences (from the topics listed in Item VI) for topics to be discussed]	
	II. QUALITY CONTROL ISSUES (Does experience in Midland indicate the need for a broader review of the Quality Control?)	
9:30 a.m.	A. Summary of NRC Experience and NRC Staff Position	W. Little
9:50 a.m.	B. Applicant Response	B. Marguglio
10:00 a.m.	C. Discussion	
10:15 a.m.	********* BREAK *********	
	III. SEISMIC	
10:25 a.m.	A. Staff Summary of Proposed Requirements and Probabilistic Estimates of Increased Sizes	L Reiter/J. Kimball
10:35 a.m.	B. Margin Study Results (including Liquefaction)	T R. Thiruvengadam
10:50 a.m.	C. Discussion	
	IV. INADEQUATE CORE COOLING INSTRUMENTATION AND HEAD VENT	
11:05 a.m.	A. Staff Position	R. Mattson
11:10 a.m.	B. Applicant Response	L. Gibson
11:15 a.m.	C. Discussion	

APPROXIMATE TIME		SPEAKER
	V. BOLTING EXPERIENCE	H. Slager
11:20 a.m.	A. Summary by Applicant	
11:30 a.m.	B. Discussion	
11:35 a.m.	VI. OTHER POSSIBLE TOPICS FOR DISCUSSION	
	A. Fire Protection	
	B. Systems Interaction	
	C. Integrated Control System	
	D. Turbine Missiles	
	E. High Copper Vessel	
	F. Process Steam	
	G. ATWS	
	H. AC/DC Systems Reliability	
	I. Probabilistic Risk Assessment	
	J. Auxiliary Feedwater System Reliability	
	K. Organization, Management, and Training	
	L. Emergency Operating Procedures	
	M. Industrial Security	
	N. Items from Previous ACRS Letters	
	O. SG Overfill Protection	
	P. Natural Circulation in the Event of a SBLOCA	
	Q. Other	

W. Little, NRC Reg III
266th ACRS MTC
June 4, 1982

APPENDIX VIII
NRC STAFF POSITION ON QA/QC

FIGURE 1

CRITERIA FOR ASSESSING CONSTRUCTION QA/QC

MANAGEMENT ATTITUDE

REGULATORY REQUIREMENTS AND INDUSTRY STANDARDS

ORGANIZATION FOR QUALITY

STAFFING FOR QUALITY

TRAINING FOR QUALITY

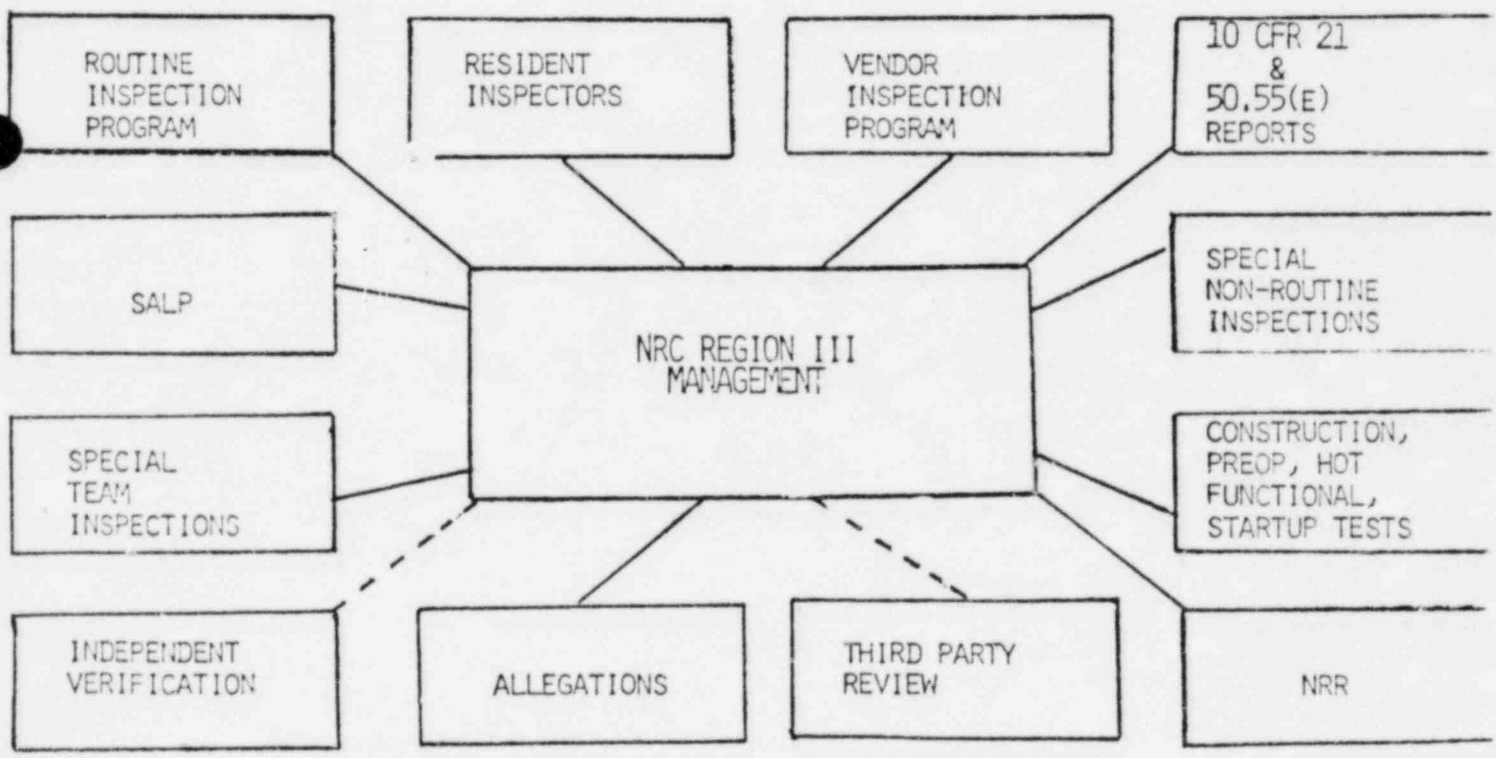
TIMELY PROBLEM IDENTIFICATION

EFFECTIVE PROBLEM RESOLUTION
- APPROPRIATE IMMEDIATE ACTION
- IDENTIFICATION AND CORRECTION OF ROOT CAUSE
- PREVENT REPETITIVE PROBLEMS

MOTIVATION FOR QUALITY

FIGURE 2

SOURCES OF INFORMATION - MIDLAND PERFORMANCE



———————— SOURCES CURRENTLY IN USE

------------ SOURCES WHICH MAY BE USED

APPENDIX IX
PRESENTATION BY CONSUMERS POWER

INCOMPLETE TENTATIVE SCHEDULE
MIDLAND PLANT UNITS 1 & 2
266TH ACRS MEETING
JUNE 4, 1982
(8:30 A.M. - 12:30 P.M.)

APPROXIMATE TIME		SPEAKER
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8:50 a.m.	B. Comments by Members of the Public	M. Sinclair
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11:10 a.m.	B. Applicant Response	L. Gibson
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11:30 a.m.	B. Discussion	
11:35 a.m.	VI. OTHER POSSIBLE TOPICS FOR DISCUSSION	
	A. Fire Protection	
	B. Systems Interaction	
	C. Integrated Control System	
	D. Turbine Missiles	
	E. High Copper Vessel	
	F. Process Steam	
	G. ATWS	
	H. AC/DC Systems Reliability	
	I. Probabilistic Risk Assessment	
	J. Auxiliary Feedwater System Reliability	
	K. Organization, Management, and Training	
	L. Emergency Operating Procedures	
	M. Industrial Security	
	N. Items from Previous ACRS Letters	
	O. SG Overfill Protection	
	P. Natural Circulation in the Event of a SBLOCA	
	Q. Other	

MIDLAND PROJECT QUALITY ASSURANCE

PRESENTATION TO ACRS

JUNE 4, 1982

CONSUMERS POWER COMPANY

B W MARGUGLIO

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NRC INSPECTIONS:

- INCREASED INSPECTION PROGRAM
- TEAM INSPECTION--MAY, 1981
- RESIDENT INSPECTOR SINCE 1978

EXTERNAL AUDITS AND ASSESSMENTS:

- BIENNIAL QUALITY ASSURANCE AUDIT--1976--
  NUCLEAR AUDIT & TESTING COMPANY

- BIENNIAL QUALITY ASSURANCE AUDITS--1978 & 1980--
  MANAGEMENT ANALYSIS COMPANY

- SPECIAL QUALITY ASSESSMENT--1981--
  MANAGEMENT ANALYSIS COMPANY

- RE-REVIEW OF SUPPLIER RADIOGRAPHIC FILM--
  HARTFORD STEAM BOILER

- PRESERVICE INSPECTION AUDITS

- FUTURE BIENNIAL QUALITY ASSURANCE AUDITS

- FUTURE INPO ASSESSMENTS

REINSPECTIONS AND RE-REVIEWS:

- RE-REVIEW OF ENVIRONMENTAL AND SEISMIC QUALIFICATION TESTS
- RE-REVIEW OF FSAR
- RE-REVIEW OF DOCUMENTS WHICH UNOFFICIALLY COULD HAVE ALTERED DESIGN REQUIREMENTS
- SPECIAL RE-REVIEW FOR DESIGN SPECIFICITY AND TOLERANCING
- SEISMIC MARGIN ANALYSIS
- CONTROL ROOM DESIGN RE-REVIEW FOR HUMAN FACTORS
- SPECIAL DESIGN RE-REVIEW BY BECHTEL CORPORATE
- RE-REVIEW OF BECHTEL PROCUREMENT PACKAGES
- RE-REVIEW OF SUPPLIER QUALITY RECORDS (NSSS & AE)
- RE-ASSESSMENT OF LOW ALLOY, QUENCHED & TEMPERED BOLTS--
7/8" & ABOVE

REINSPECTIONS AND RE-REVIEWS (CONT'D):

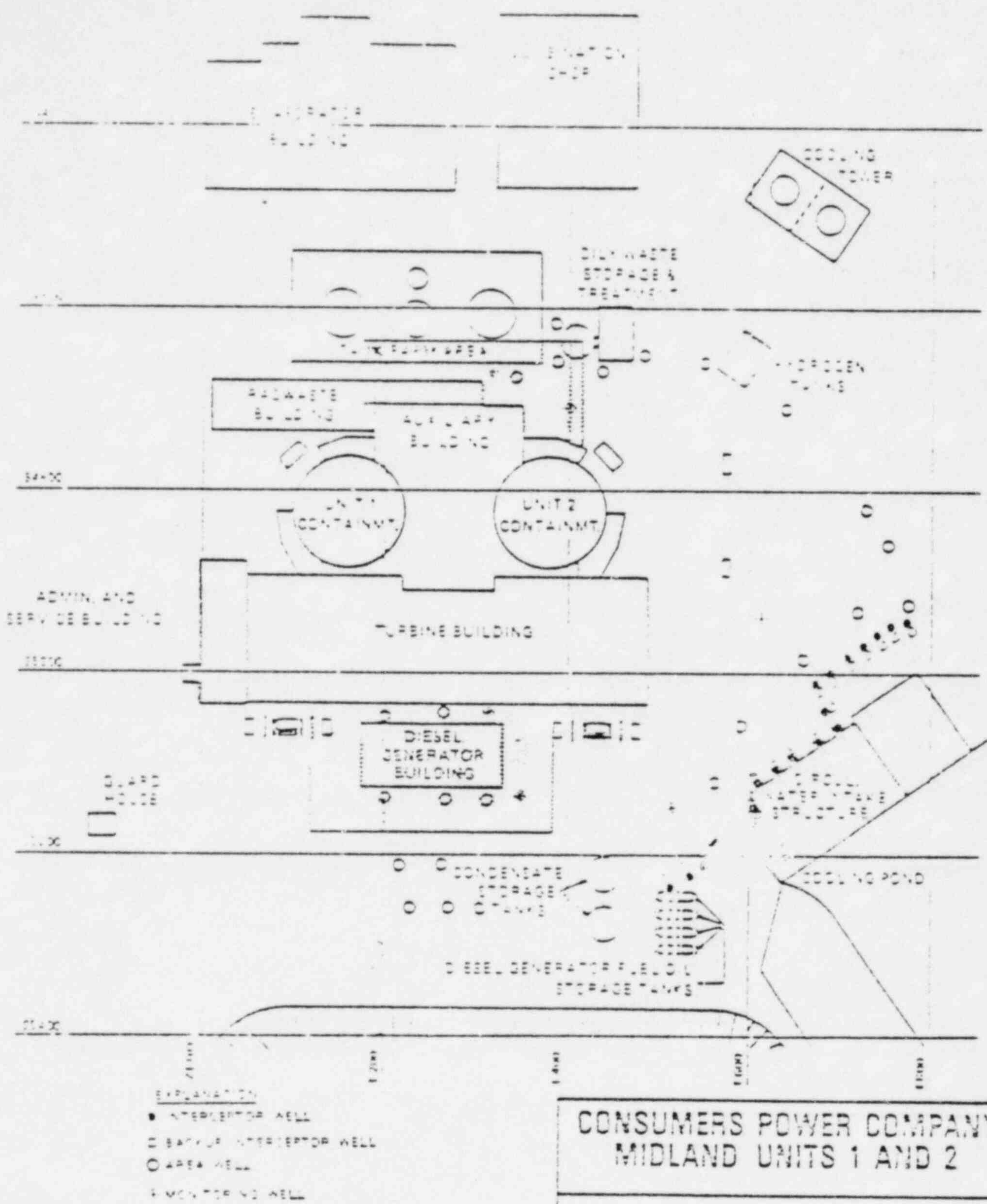
- SPECIAL OVERINSPECTION OF SUPPLIED ELECTRICAL EQUIPMENT FOR WORKMANSHIP
- 100% RE-REVIEW OF B&W FIELD-ORIGINATED RADIOGRAPHS
- SAMPLING RE-REVIEW OF BECHTEL FIELD-ORIGINATED RADIOGRAPHS
- RE-REVIEW OF CONSTRUCTION PROCEDURES
- RE-REVIEW OF QUALITY CONTROL INSTRUCTIONS
- MIDLAND PROJECT QUALITY ASSURANCE DEPARTMENT REINSPECTIONS
- SPECIAL CABLE INSTALLATION REINSPECTION
- OVERVIEW OF BECHTEL SUPPLIER AUDITS

CONCLUSIONS:

- RECOGNIZED PAST PROBLEMS
- STRENGTHENED QA PROGRAM
- GAINED EXTENSIVE KNOWLEDGE OF PLANT
- IMPROVED QUALITY OF CURRENT WORK
- EXPECT TO MEET ALL REQUIREMENTS

SEISMIC ISSUES

A-81



CONSUMERS POWER COMPANY
MIDLAND UNITS 1 AND 2

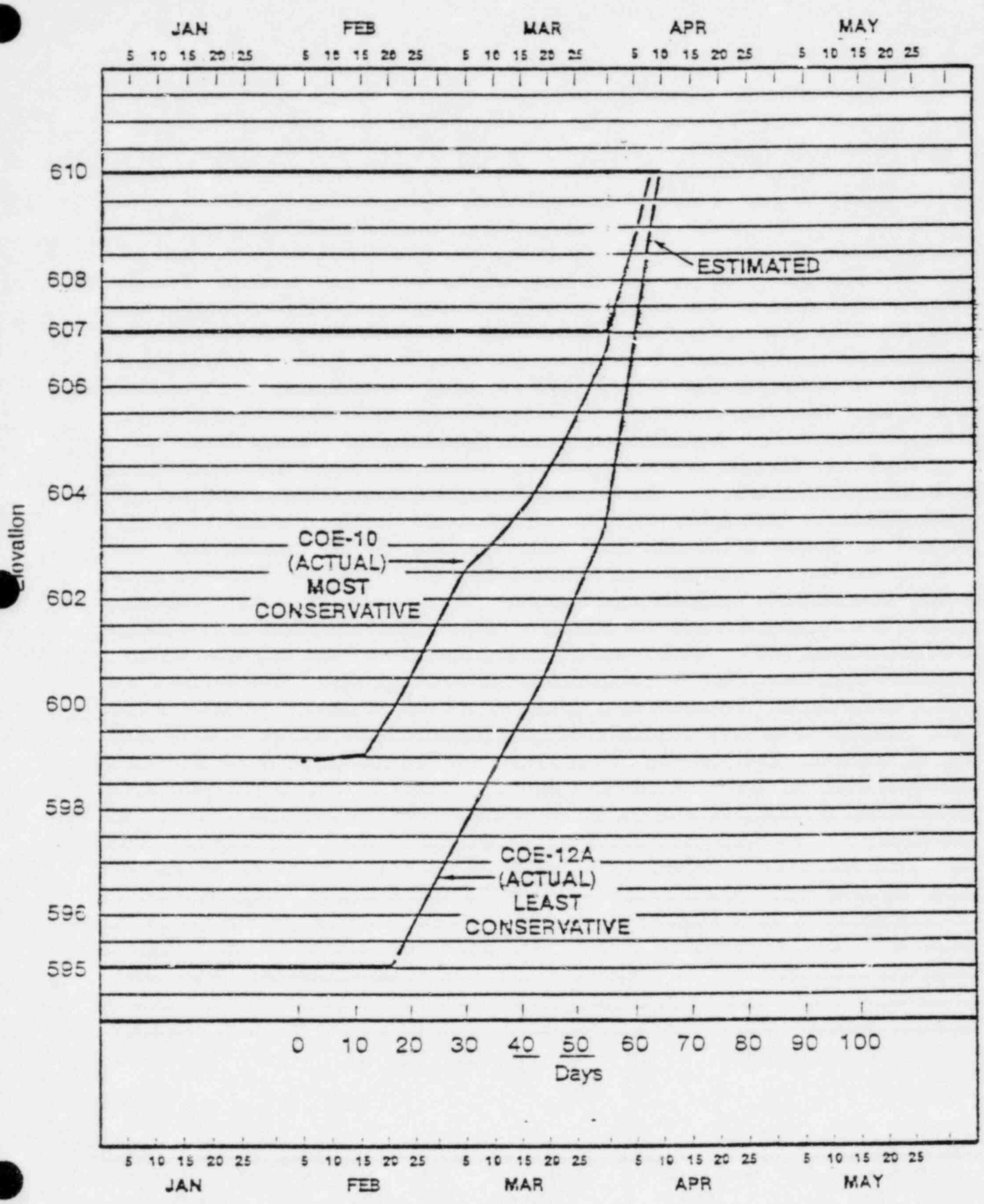
PLAN OF PERMANENT
DEWATERING SYSTEM

FIGURE 1

A-82

MIDLAND UNITS 1 AND 2
ACRS 4 27 82

G-2604-37



MIDLAND UNITS 1 AND 2
ACRS 4/7/82

EVALUATION OF LIQUEFACTION POTENTIAL
UNDER DIESEL GENERATOR BUILDING

SEED'S STANDARD PENETRATION TEST METHOD

FOR EARTHQUAKE MAGNITUDE M = 6
AND GROUND WATER LEVEL AT ELEV. 610.00

ACCELERATION	FACTOR OF SAFETY
0.19 G	1.5
0.25 G	1.1

BOLTING AND OTHER HIGH STRENGTH MATERIAL

MIDLAND LOW-ALLOY QUENCHED AND TEMPERED BOLTING

- UNIT 1 REACTOR VESSEL ANCHOR BOLTS
- PIPE WHIP RESTRAINT BOLTS
- STEAM GENERATOR ANCHOR BOLTS
- REACTOR COOLANT PUMP SNUBBER ANCHOR BOLTS
- LOW-ALLOY QUENCHED AND TEMPERED BOLT SURVEY

A-86

UNIT 1 REACTOR VESSEL ANCHOR BOLTS

- THREE BOLTS FAILED WITHIN 8 MONTHS OF PRELOADING
- CRACKING MECHANISM - STRESS CORROSION CRACKING FOLLOWED BY FRACTURE DUE TO LOW TOUGHNESS
- PRELOAD - APPROXIMATELY 92 KSI
- HARDNESS - AS HIGH AS 48 HRC
- RESOLUTION
  - Lower Prestress - 6 KSI Max
  - Upper Lateral Supports
  - Limit Accident Loads for 70% of Proof Load

A-87

REACTOR VESSEL ANCHOR BOLT PRELOAD ALLOWABLE STRESS

PROPERTY RATIO METHOD

- Assume Stud Has Corrosion Crack of Unknown Size But Crack Is Subcritical in Relation to Stress and Fracture Toughness (if crack exceeded critical size, stud would have broken)
- Long-Term Loading

  $K_{ISCC} = \sigma_{NEW} \bullet C \sqrt{a}$ (1)

  $K_{IC} = \sigma_{OLD} \bullet C \sqrt{a}$ (2)

  Divide Equation (1) by Equation (2) and Solve for σ_{NEW}

  $$\sigma_{NEW} = \frac{K_{ISCC}}{K_{IC}} \bullet \sigma_{OLD} \quad (3)$$

- Assuming a Conservative Value for K_{ISCC}/K_{IC} and 86 KSI for the Old Preload, the New Preload Which Will Not Result in Crack Extension Is 12 KSI
- A Safety Factor of 2 Is Applied to 12 KSI Resulting in a 6 KSI Allowable

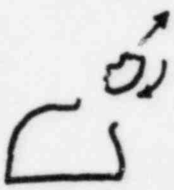
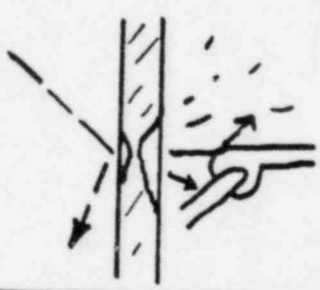
G-2525-15

A-88

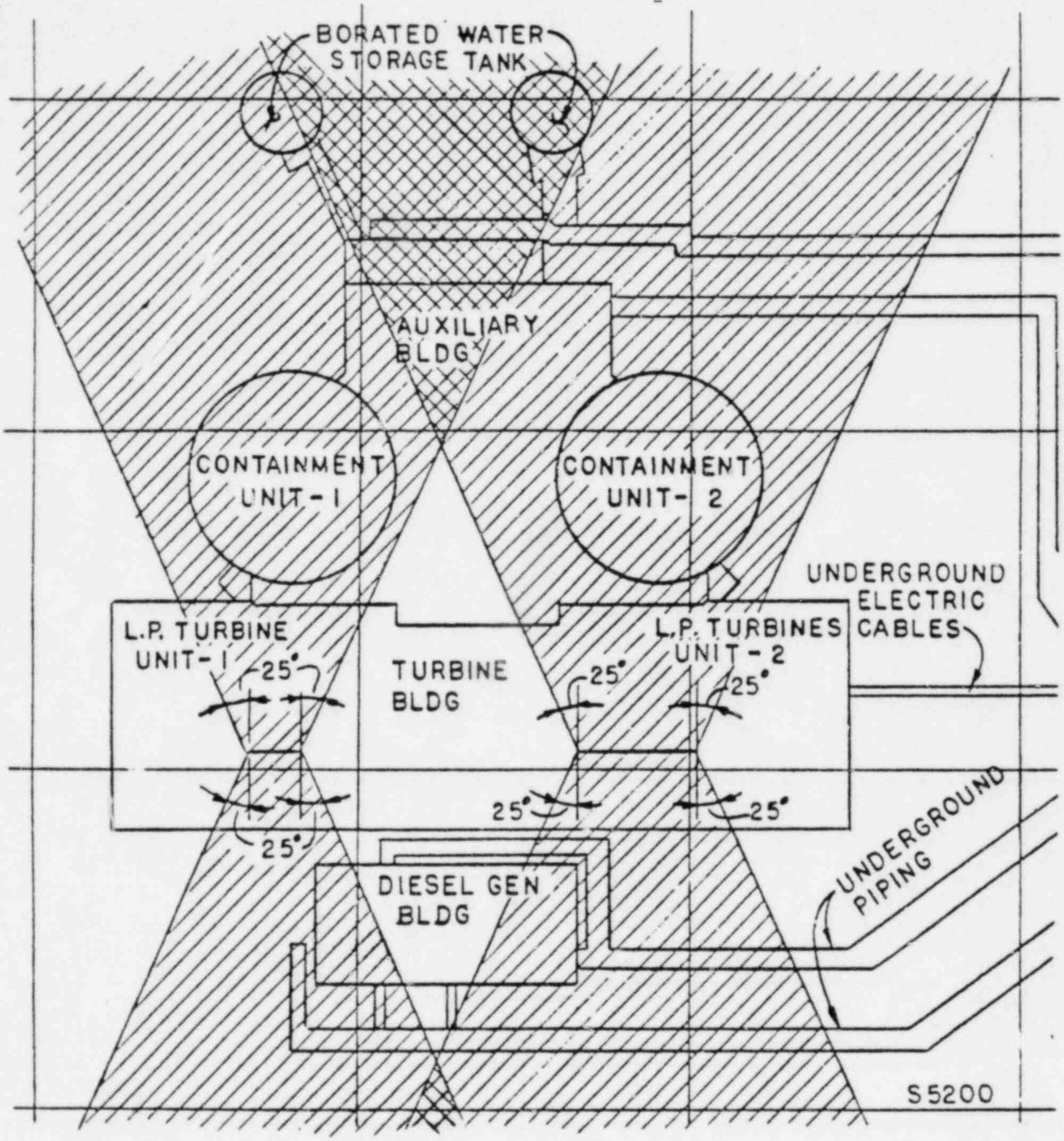
Bolts-9-A

TURBINE MISSILES

REFERENCES: SRP 3.5.1.3, 10.2, 10.2.3
RG 1.115

$P_4 = P_1 P_2 P_3$ PROBABILITY, YR^{-1}	EXISTING PROCEDURE	PROPOSED PROCEDURE
MISSILE GENERATION, P_1 	DESIGN 6×10^{-5} OVERSPD 4×10^{-5} ASSUMED $= 10^{-4}$ - - - - - - - - TIME DEPENDENCE - - - - - - DISK CRACKING EXPERIENCE - - - - - - CONTROL SYST. TEST INTERVAL	KEEP $< 10^{-4} - 10^{-5}$ DEPENDING ON ORIENTATION BY SUITABLE: · MATERIALS . DESIGN . OPERATION AND . ISI VENDOR CLAIMS P_1 $10^{-6} - 10^{-10}$
STRIKE, P_2 	$< 10^{-3}$ CALCULATED FOR EACH FACILITY - - - - - - - - UNCERTANTIES: · MISSILE CHAR. · TECHNIQUE · TREATMENT OF BARRIERS AND OBSTACLES	KEEP $< \approx 10^{-3}$ BASED ON: · PREVIOUS REVIEW EXPERIECE · FACILITY TOUR · JUDGEMENT
DAMAGE, P_3 	ASSUMED $= 1$ - - - - - - - - UNCERTANTY: DEFINITION OF DAMAGE	COMBINE WITH P_2
UNACCEPTABLE, P_4	$< 10^{-7}$	$< 10^{-7}$ (NO CHANGE)

CONSUMERS POWER COMPANY
MIDLAND PLANT UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT



APPLICANT	P_1	$P_2 P_3$	P_4
DES. SPD	8.7×10^{-9}		UNIT 1, 1.4×10^{-9}
OVER SPD	5×10^{-9}		UNIT 2, 1.5×10^{-9}
	1.4×10^{-8}	$\times 10^{-1}$	1.5×10^{-9}
NRC SRP	10^{-4}	10^{-1}	10^{-5}
MUST OBTAIN	---	---	10^{-7}

MIDLAND ACRS

CP STAGE - MMI=VI (PGA = 0.12G)
(1970) MODIFIED HOUSNER SPECTRUM

OL STAGE ISSUES

MICHIGAN BASIN

SITE SPECIFIC SPECTRA

SEISMIC HAZARD ANALYSIS

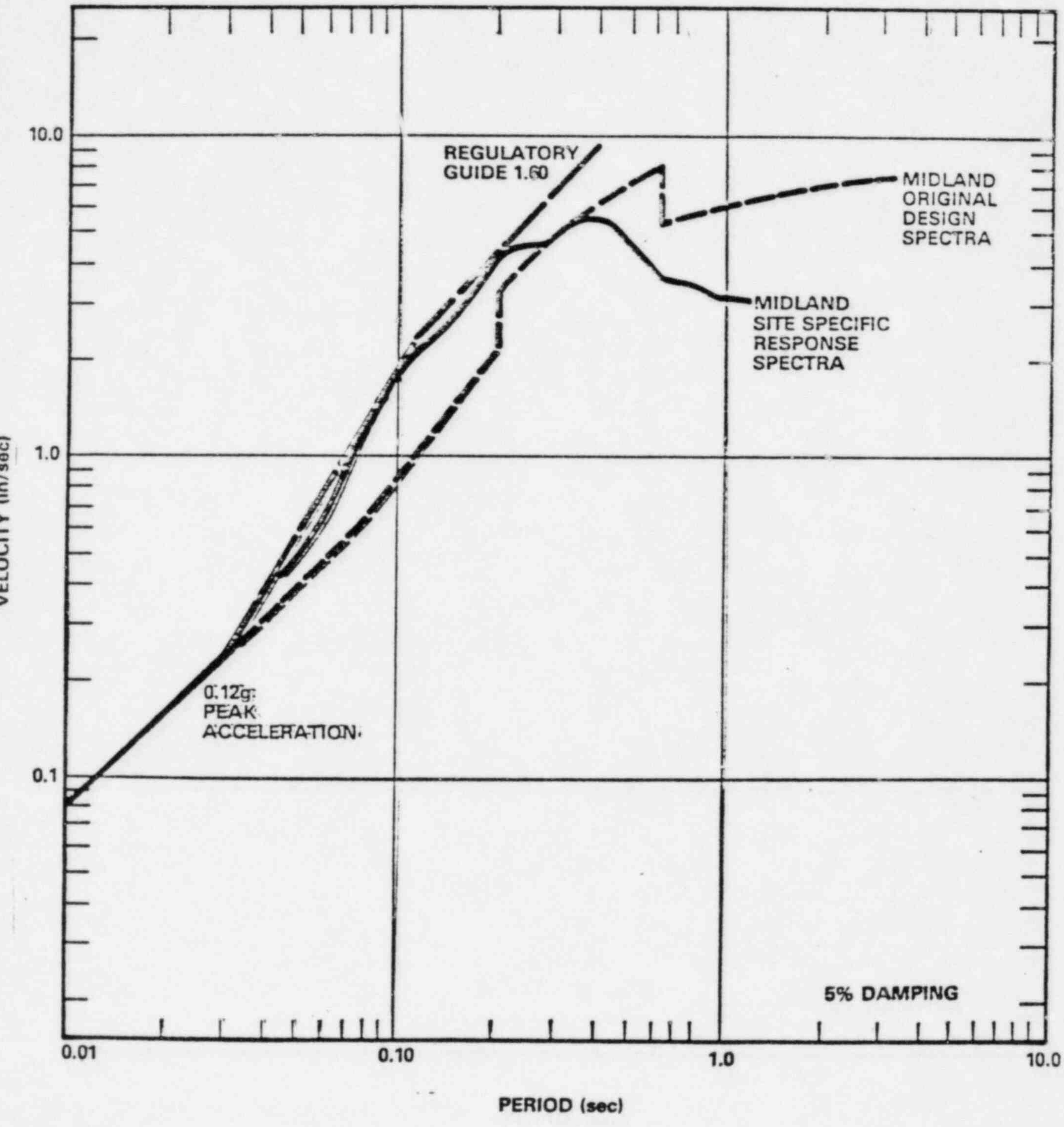


Figure 2.5.2-1 Comparison of Midland Original Design Spectra, Site Specific Spectra, and Regulatory Guide 1.60 Spectra

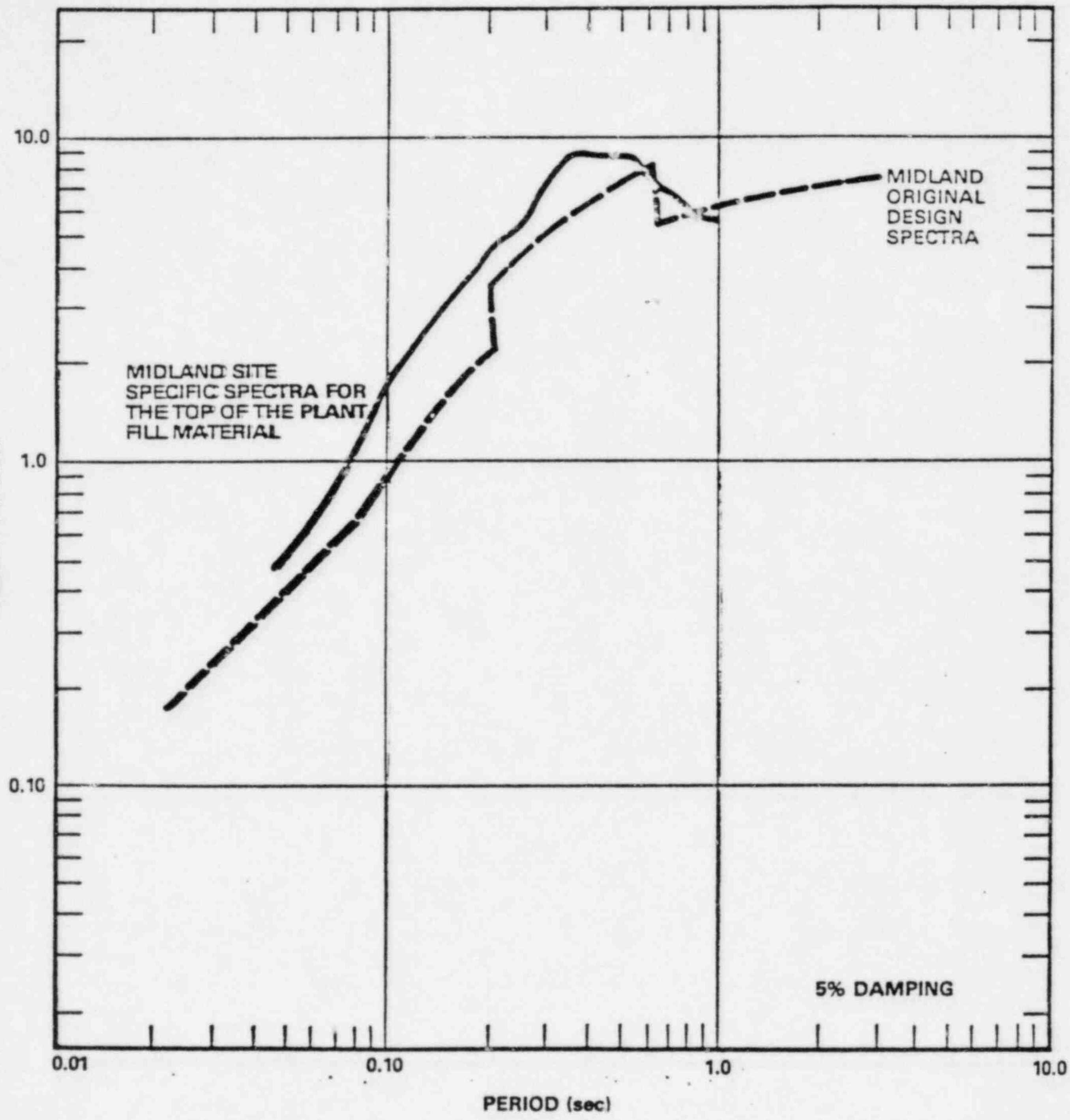


Figure 2.5.2-2 Comparison of Midland Original Design Spectra and Site Specific Spectra for top of Fill Material


90°
88°
86°
84°
82°
80°
78°
76°
Lake Superior
WISCONSIN
Lake Michigan
BIG ROCK POINT
Lake Huron
MIDLAND
MICHIGAN
PALISADES
Lake Ontario
Lake Erie
NE
200 MILES
300 MILES
350 MILES
ILLINOIS
INDIANA
OHIO
PENNSYLVANIA
MARYLAND
WEST VIRGINIA
KENTUCKY
VIRGINIA
TENNESSEE
NORTH CAROLINA

SITE SPECIFIC SPECTRUM

A) 22 SETS OF ACCELEROGRAMS

M_L = 4.9 TO 5.5 (5.3)

EPIC. = 7 TO 33 (17.6)

B) SENSITIVITY TESTS

DISTANCE RESTRICTION

PARKFIELD RECORDS

25 SETS OF ACCELEROGRAMS

M_L 4.9 TO 5.6 (5.4)

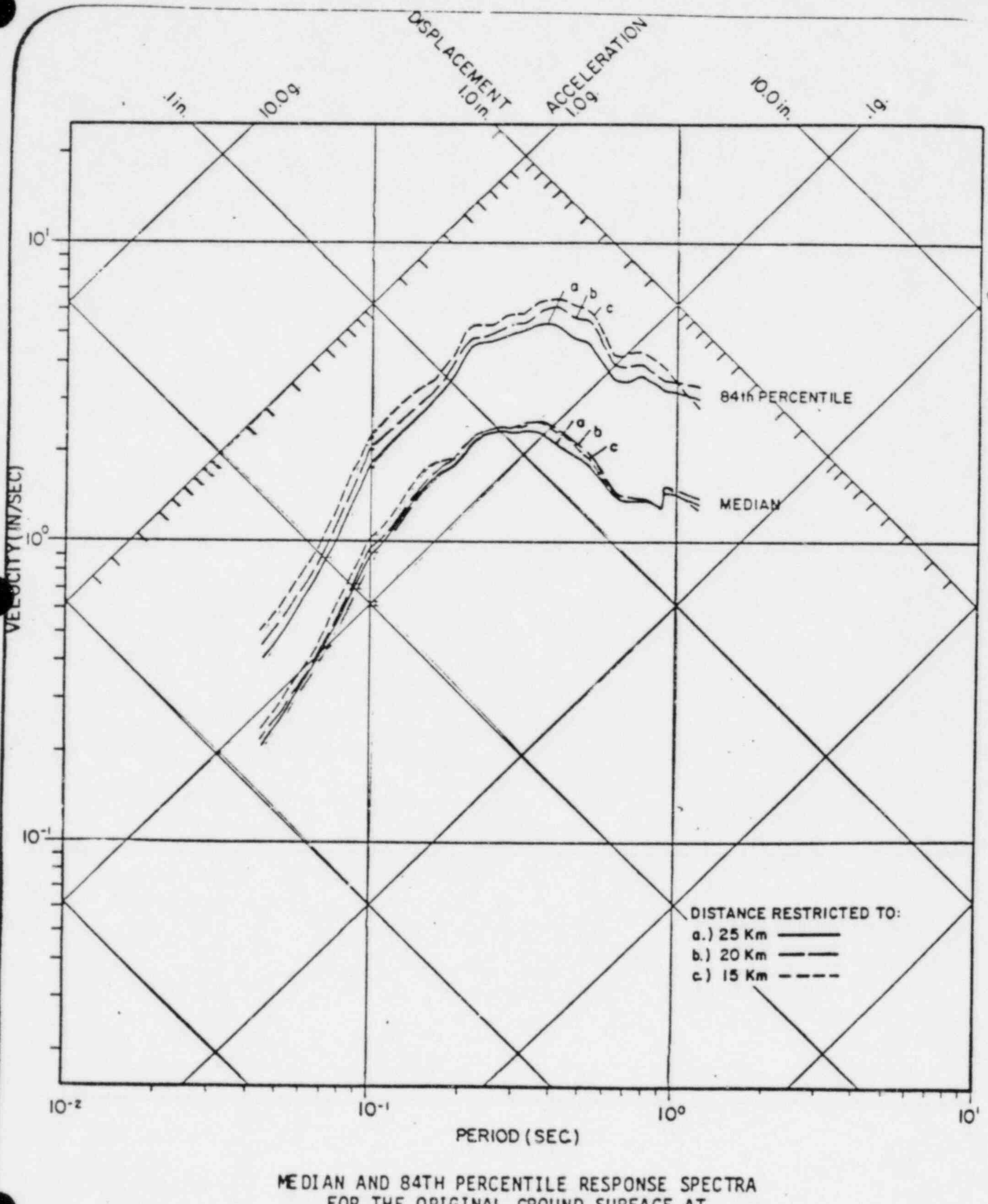
EPIC. = 7 TO 33 (17.1)

SSP WITH PARKFIELD EXCEEDS SSP WITHOUT PARKFIELD

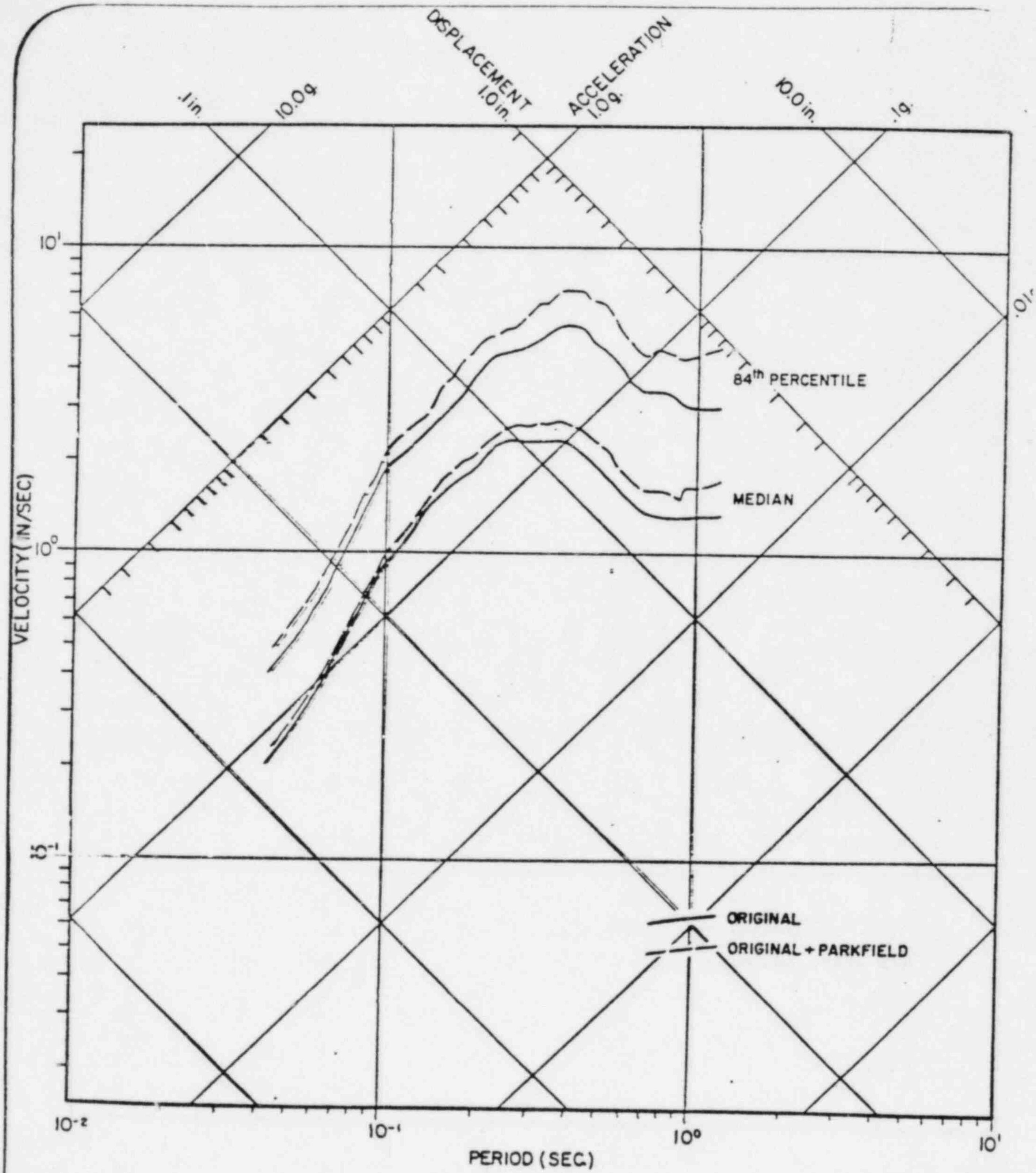
BY 1.1 TO 1.39 AT FREQUENCIES 2 TO 20 HZ.

STAFF POSITION M_{BLG} = 5.3 SPECTRUM SHOULD

INCLUDE PARKFIELD RECORDS



MEDIAN AND 84TH PERCENTILE RESPONSE SPECTRA
FOR THE ORIGINAL GROUND SURFACE AT
MIDLAND NUCLEAR POWER PLANT
(5% OF CRITICAL DAMPING)



MEDIAN AND 84TH PERCENTILE RESPONSE SPECTRA
FOR THE ORIGINAL GROUND SURFACE AT
MIDLAND NUCLEAR POWER PLANT
(5% OF CRITICAL DAMPING)

SEISMIC HAZARD ANALYSIS:

A) ATTEMPT TO QUANTIFY LOW SEISMICITY NEAR MIDLAND

USED IN A RELATIVE SENSE (COMPARED MIDLAND TO 5 OTHER CUS SITES)

B) 3 SOURCE ZONE MODELS USED

C) MIDLAND 0.50 TO 0.70 INTENSITY UNITS LOWER THAN 5 OTHER SITES

NUTTLI AND HERRMANN (1978) $I_o = 2m_B - 3.5$

MIDLAND 0.25 TO 0.35 m_B UNITS LOWER THAN 5 OTHER SITES

$M_{BLG} = 5.3 - 0.25$ TO $0.35 = 5.0$

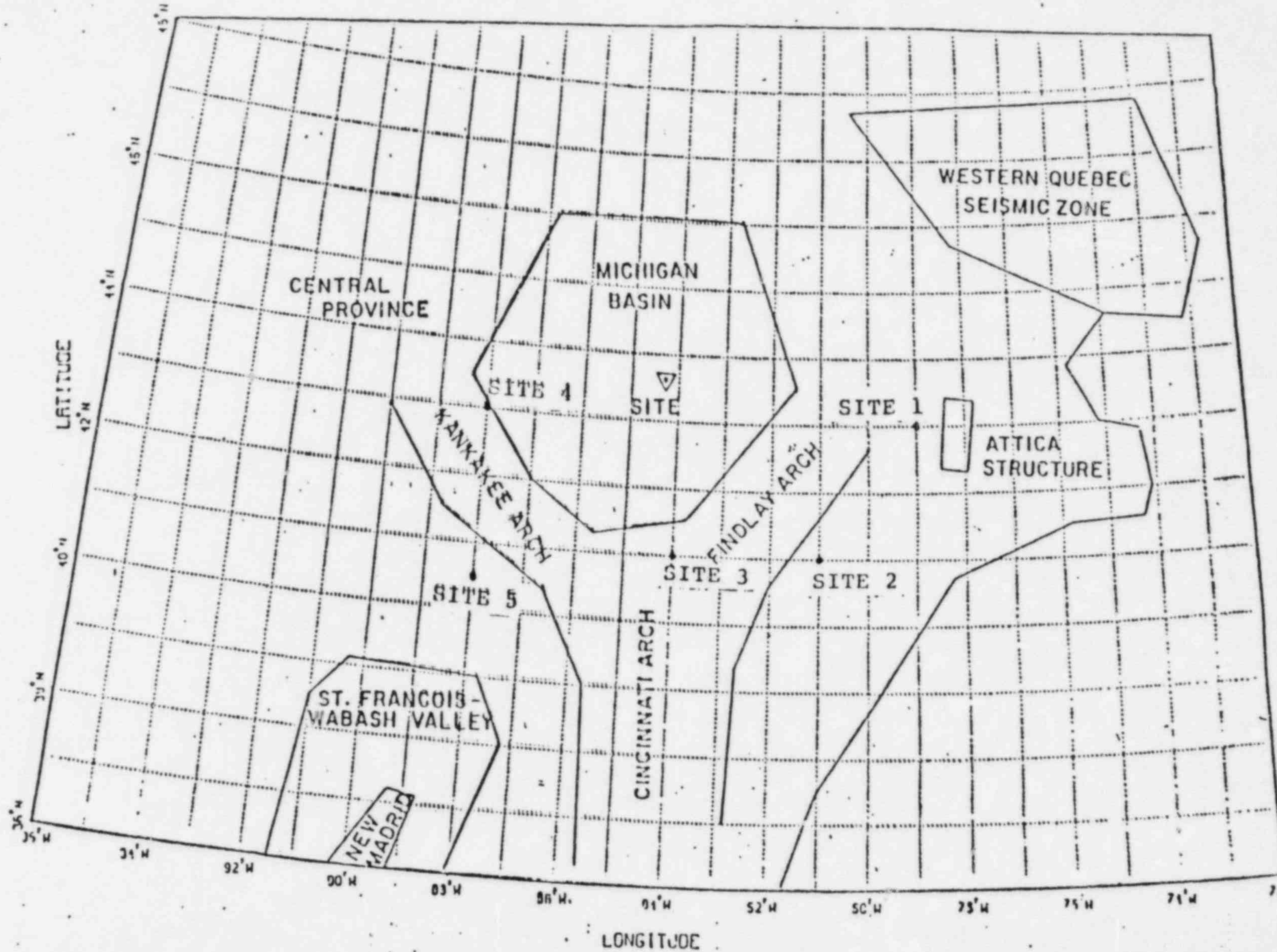
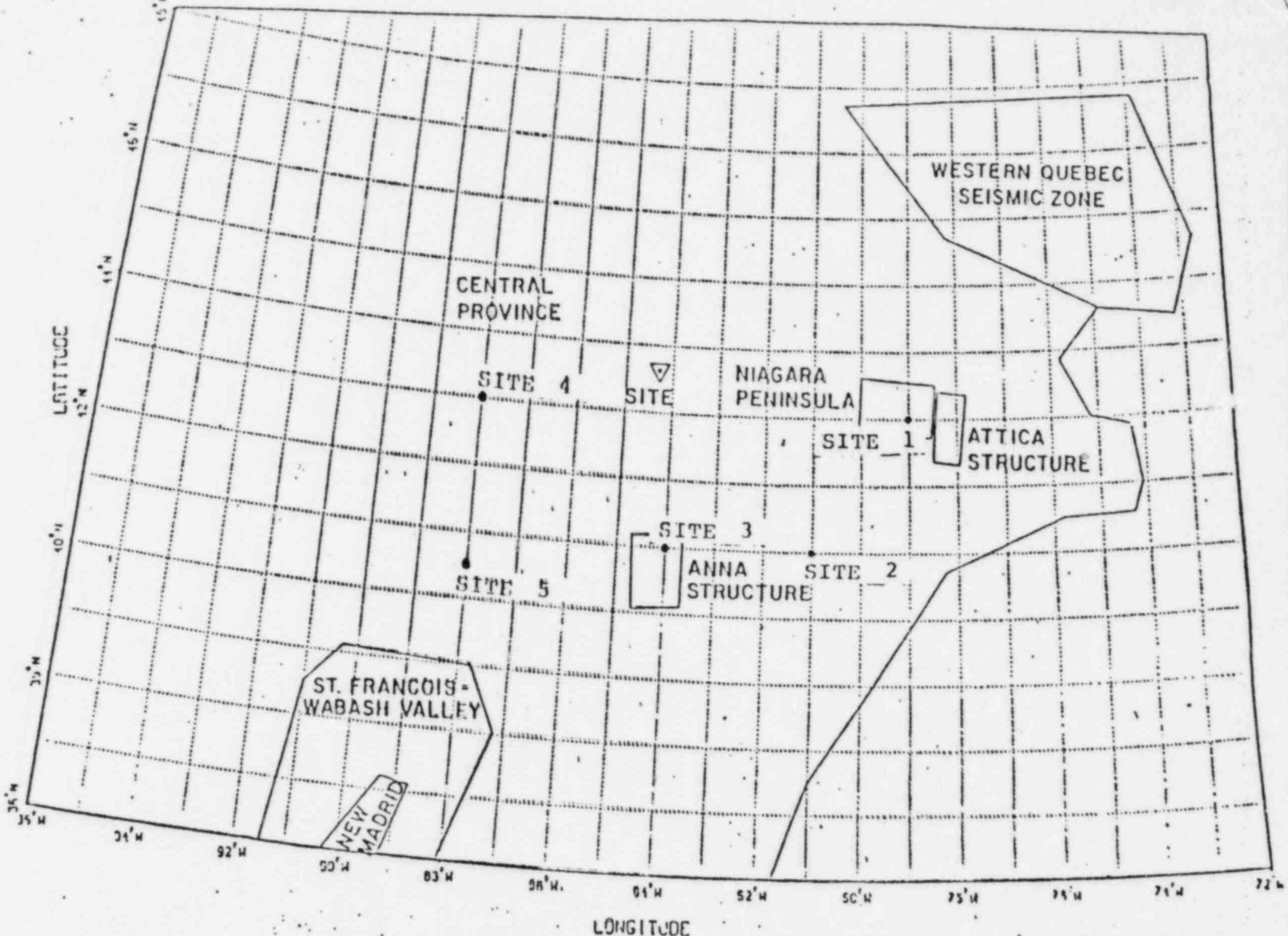


Figure 3

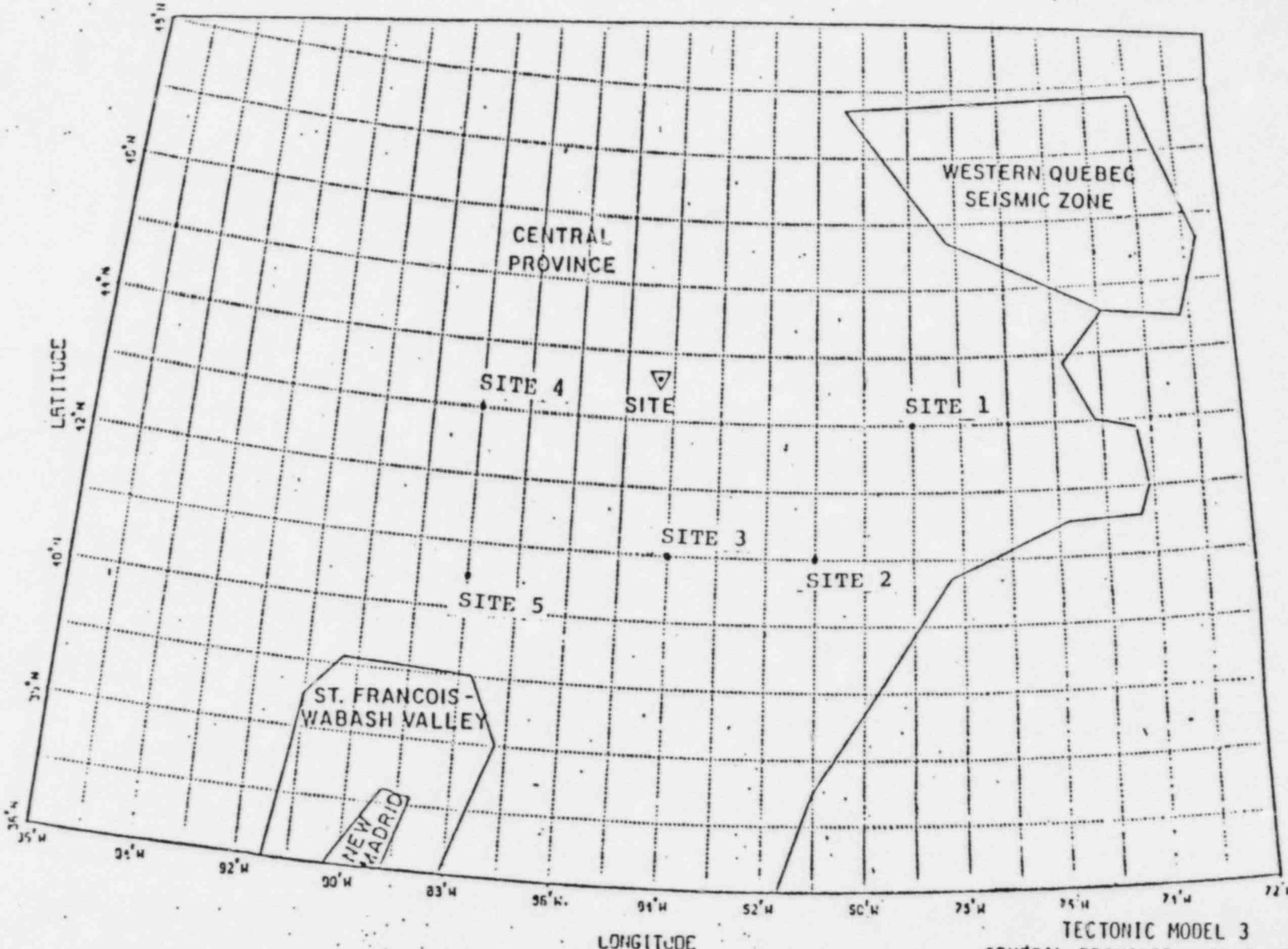
TECTONIC MODEL
MICHIGAN BASIN-CINC
ARCH STRUCTURE

A-100



TECTONIC MODEL 2
CENTRAL PROVINCE - ANNA, OHIO
AND ATTICA, N.Y. TECTONIC STRUCTU

Figure 4



TECTONIC MODEL 3
CENTRAL PROVINCE - ANNA, OHI
AND ATTICA, N.Y. NOT CONSTRAIN
TO STRUCTURES

Figure 5

D) TARGET MAGNITUDE OF M_{BLG} = 5.0 IS CONSERVATIVE FOR MIDLAND

IS SSP WITHOUT PARKFIELD RECORDS CONSERVATIVE FOR AN M_{BLG} = 5.0 TARGET MAGNITUDE?

STAFF USED 4 GROUND MOTION ATTENUATION RELATIONSHIPS

MIDLAND 0.50 TO 0.70 MMI UNITS LOWER
MIDLAND 0.25 TO 0.35 M_{BLG} UNITS LOWER

M_{BLG} = 5.0 WOULD BE .68 TO .71 TIMES THAT OF
M_{BLG} = 5.3 (5.3 WOULD BE 1.40 TO 1.46 TIMES THAT OF A 5.0)

ABOVE FACTORS SIMILAR TO RELATIVE DIFFERENCES IN CUS SEP SITES

GROUND MOTION AMPLIFICATION THROUGH THE FILL MATERIALS

A) SITE SPECIFIC SPECTRUM

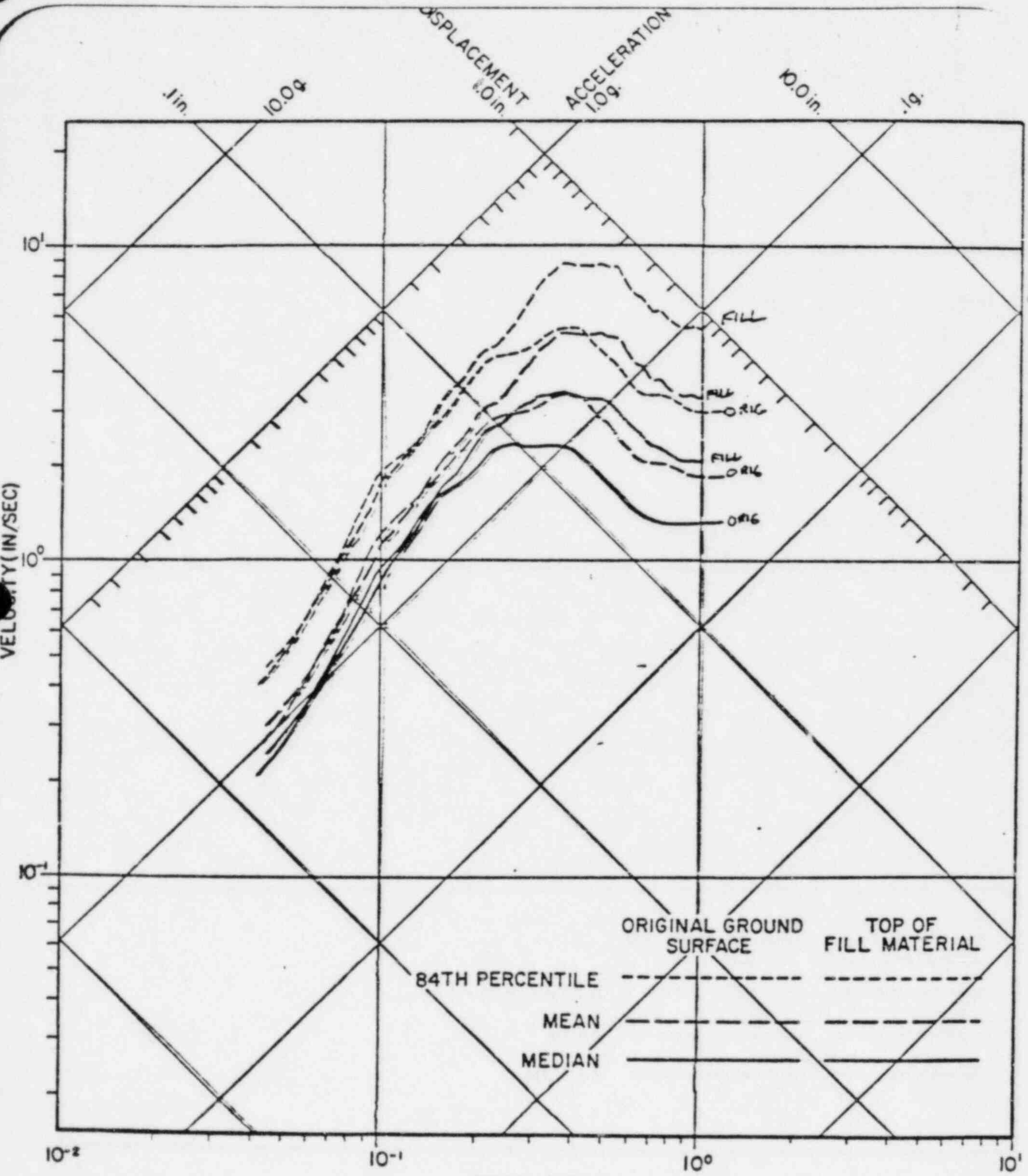
18 SETS OF ACCELEROGRAMS

M_L = 4.9 TO 5.6 (5.3)

R = 6 TO 31 (17.3)

B) SHAKE RUNS

SITE SPECIFIC SPECTRUM ENVELOPES THE SHAKE RUNS



MEDIAN, MEAN AND 84TH PERCENTILE RESPONSE SPECTRA
OF THE ORIGINAL GROUND SURFACE AND
THE TOP OF FILL MATERIAL AT
MIDLAND NUCLEAR POWER PLANT
5% CRITICALLY DAMPED

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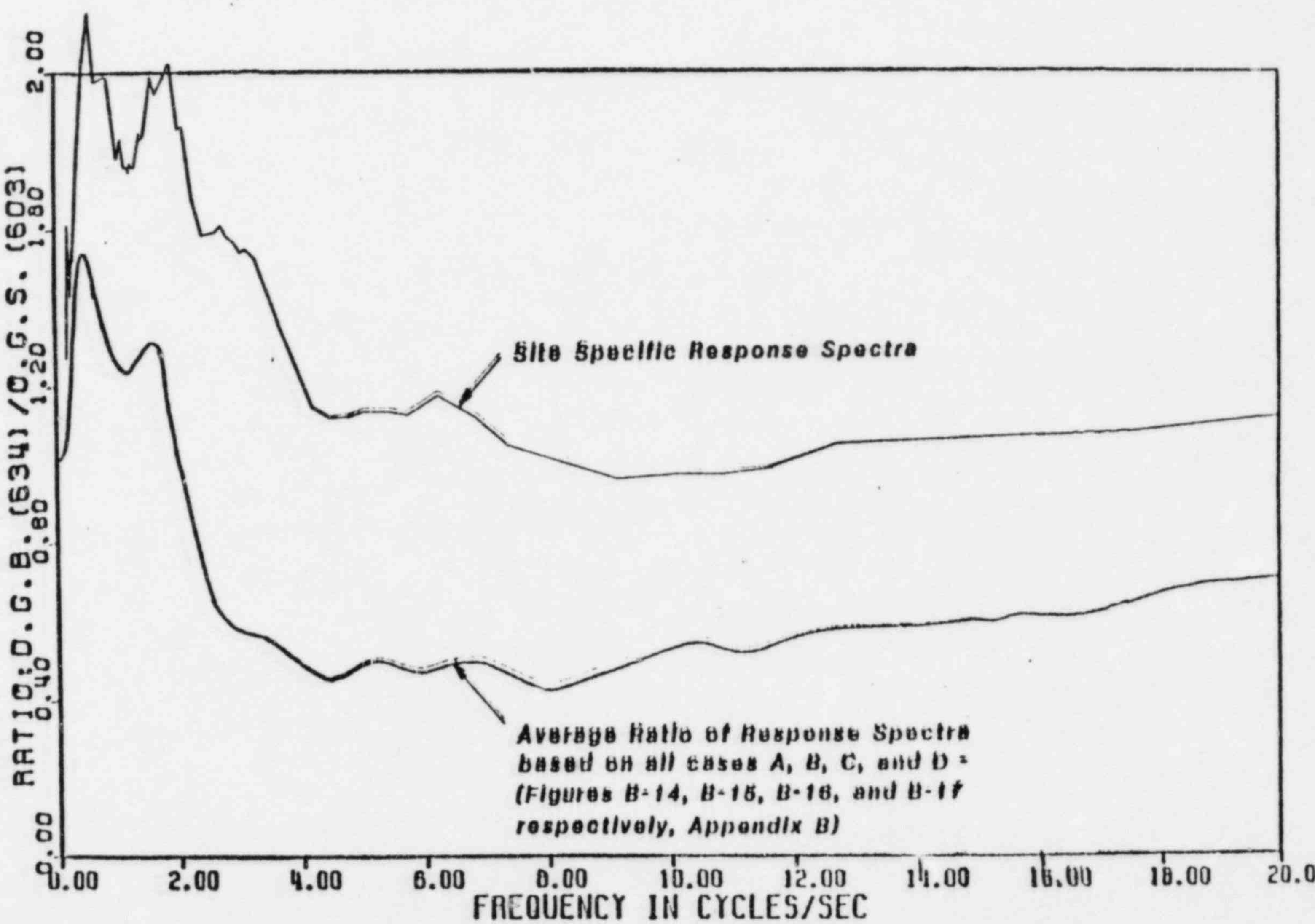


Ratio of response spectra for the soil at and near the Diesel Generator Building area (Figure B-25, Appendix B) and site specific response spectra.

FIGURE 5

Ratio of Response Spectra Top of Fill / Original Ground Surface

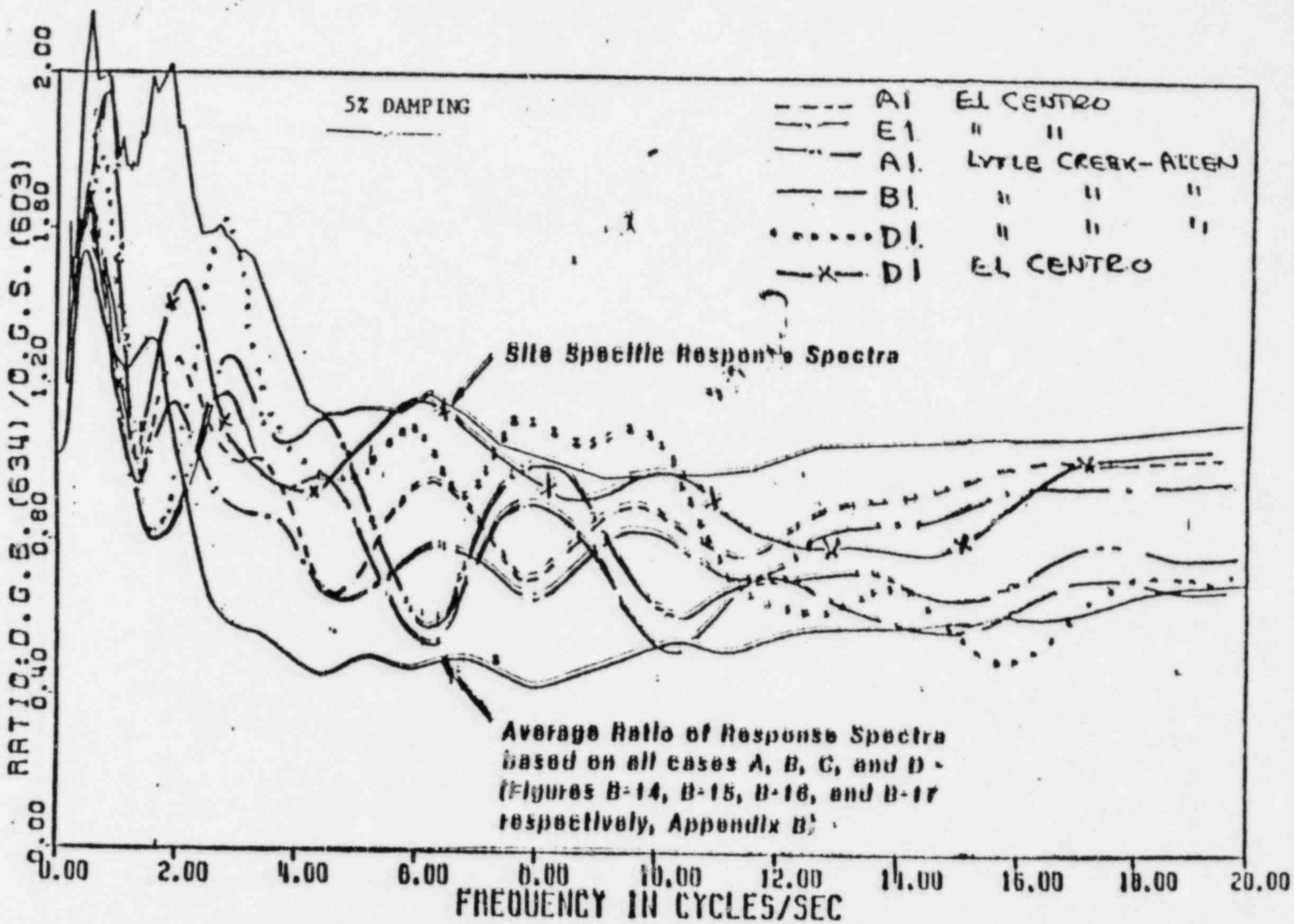


Figure 5. Ratio of response spectra for the soil at and near the Diesel Generator Building area (Figure B-25, Appendix B) and site specific response spectra.

ATTACHMENT 5)

QUESTIONS BY ACRS TO NRC STAFF

- WHAT IS PROBABILITY OF SSE?
- WHAT IS EARTHQUAKE OR ACCELERATION ASSOCIATED WITH PROBABILITIES OF 10^{-5} OR 10^{-6} PER YEAR?

DEFINITION OF OBE

- MIXED DEFINITION IN APPENDIX A
- "REASONABLY EXPECTED TO OCCUR"
- SIMPLE ESTIMATES, RETURN PERIODS USUALLY ON ORDER OF HUNDREDS OF YEARS
- IS OBE SAFETY-RELATED?

WHAT IS RETURN PERIOD ASSOCIATED WITH SSE?

MANY STUDIES (NON DEFINITIVE) SHOW RETURN PERIODS ON THE ORDER OF 1000 OR 10,000 YEARS

SEQUOYAH REVIEW

RELATIVE HAZARD ASSOCIATED WITH DIFFERENT LEVELS OF GROUND MOTION AT THE SAME SITE

SEP REVIEW

EQUIVALENT LEVEL OF HAZARD RELATIVE TO OTHER SITES

A-109

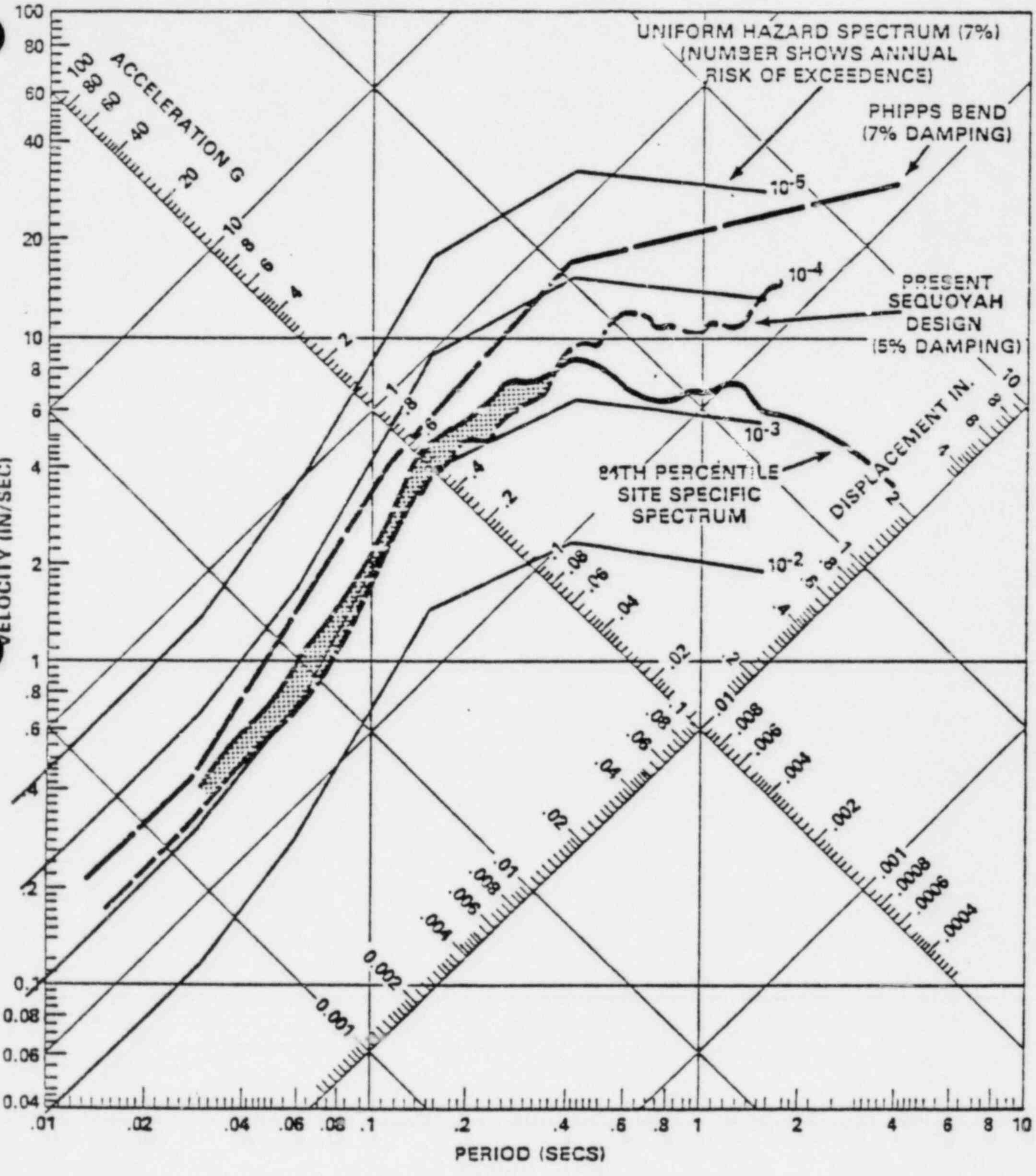
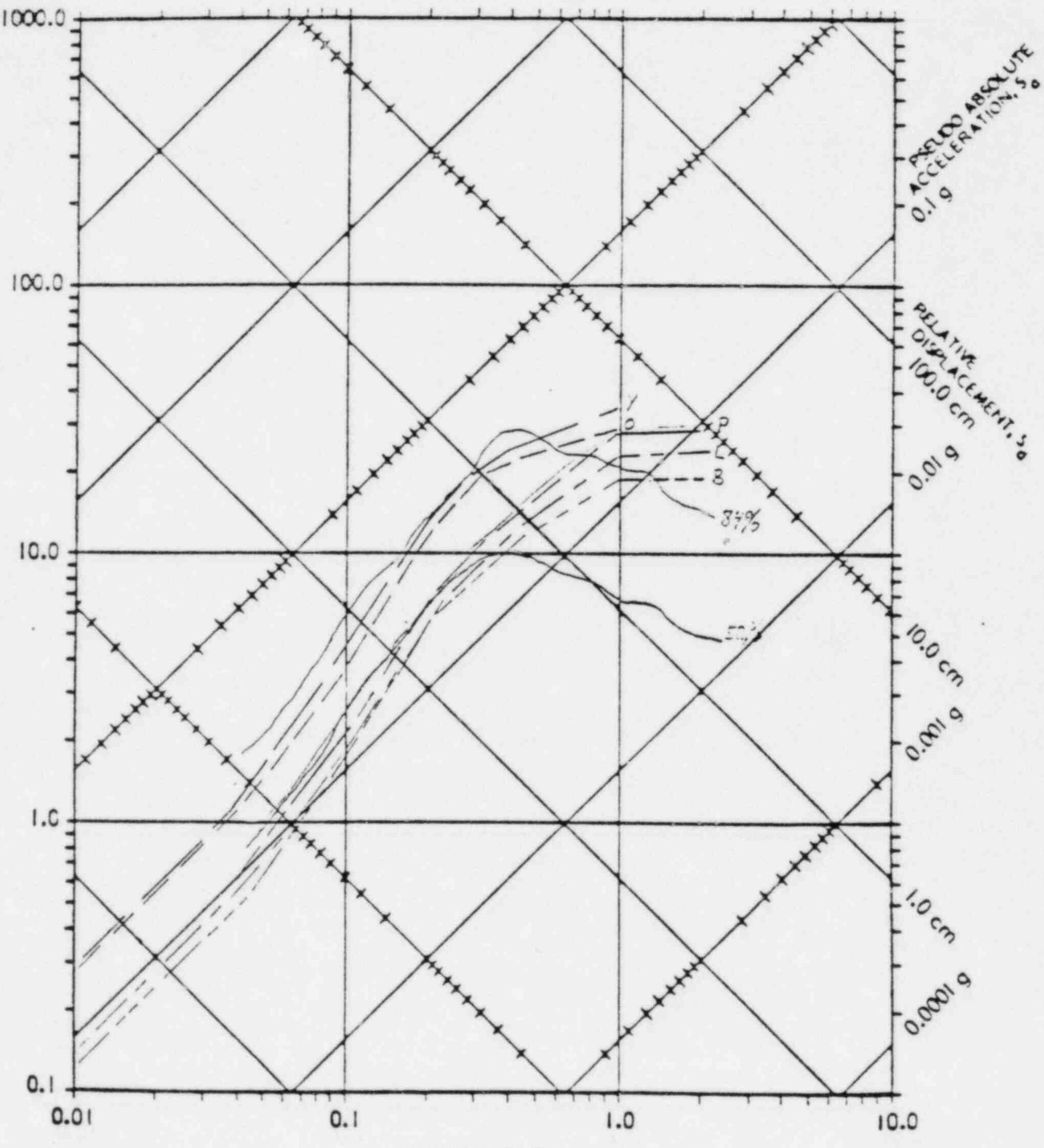


Figure 2-4

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.

Recommended Probabilistic Spectra at Soil Sites and
Recorded Spectra at Soil Sites



y - Yankee Rowe
O - Oyster Creek
P - Palisades
L - LaCrosse
B - Big Rock Point
84% - 84% spectra from nearby Mag. 5.3 ± .5 event
50% - 50% spectra from nearby Mag. 5.3 ± .5 event

A-111

Figure 15

GETR REVIEW

PLACE DETERMINISTIC ESTIMATES OF FAULT
OFFSET IN PROBABILISTIC PERSPECTIVE

MIDLAND REVIEW

UTILIZE PROBABILISTIC ESTIMATES TO HELP
MAKE A RATIONAL SEISMIC DECISION FOR A
"TECTONIC PROVINCE" THAT COVERS ONE THIRD
OR MORE OF THE U. S. AND HAS WIDELY VARYING
LEVELS OF SEISMIC ACTIVITY

STAFF EMPHASIS IN USE OF PROBABILITY

- RELATIVE RATHER THAN ABSOLUTE
- INSIGHT RATHER THAN DIRECT USE

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SENSITIVITY STUDIES

- INSIGHT INTO USEFULNESS OF COMPUTATIONS
- EFFECTS DUE TO VARIATIONS IN INPUT PARAMETERS AND THEIR UNCERTAINTY
- ZONATION, SEISMICITY, GROUND MOTION

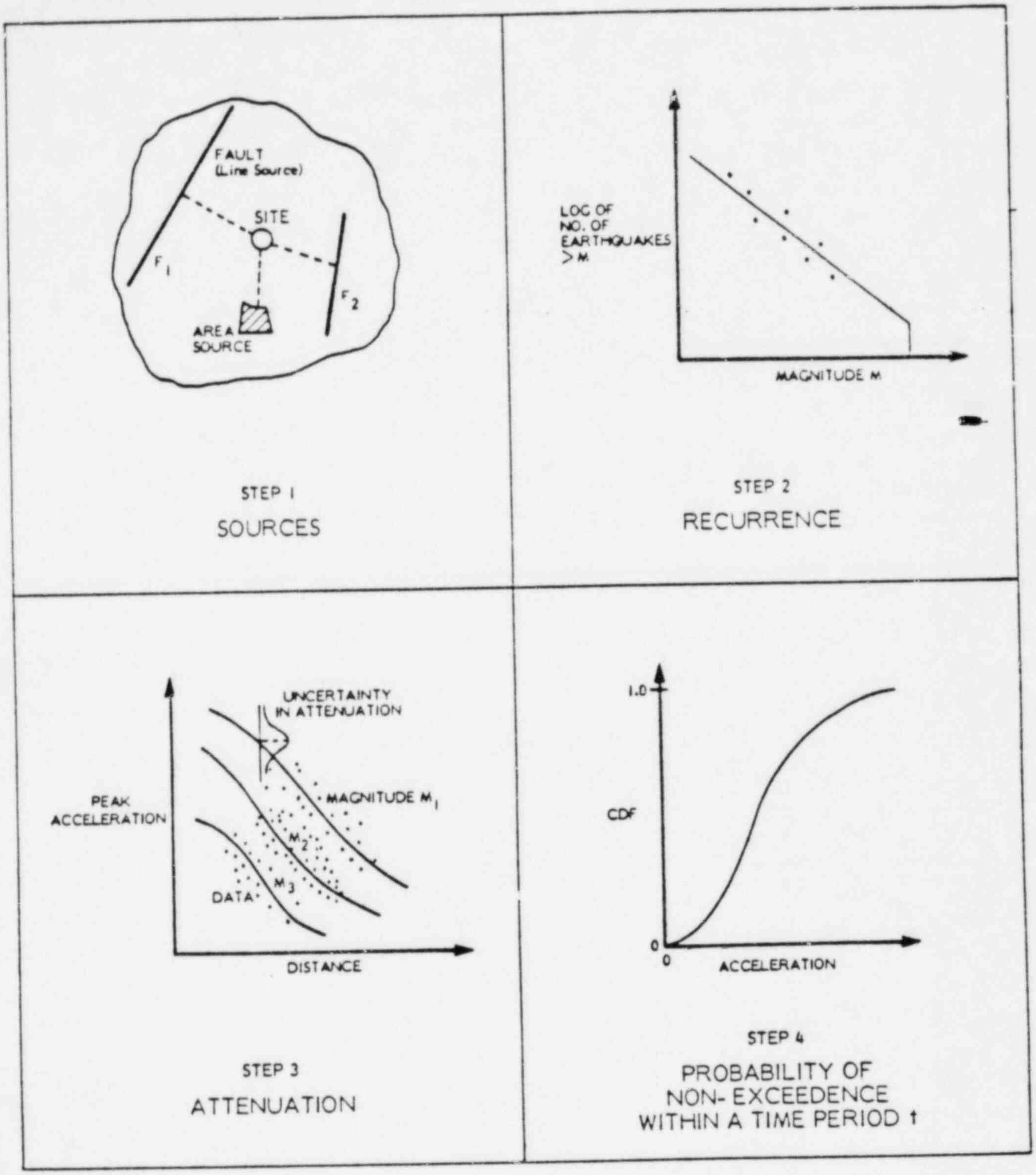


FIGURE 2

CURRENT APPROACH TO HAZARD MAPPING FOR PEAK VALUES

A-115

Dec. 8, 1978

TERA CORPORATION

NUREG/CR-1582 (LLNL-TERA)

- ZONATION - NEARBY ZONES IMPORTANT
- SEISMICITY

  A VALUE (SEISMICITY LEVEL) RELATIVELY LITTLE IMPACT

  B VALUE (RELATIVE SIZE DISTRIBUTION) IMPORTANT PARTICULARLY AT LONG RETURN PERIODS. INCLUDING UNCERTAINTY IN B VALUES INCREASES ACCELERATION 30 TO 50% AT RETURN PERIODS OF 4000 YEARS

- UPPER MAGNITUDE CUTOFF

  NOT TOO IMPORTANT FOR SHORT RETURN PERIODS.
  IMPORTANT AT LONG RETURN PERIODS.
  LOW B VALUE AND HIGH MAGNITUDE CUTOFF VERY IMPORTANT AT LONG RETURN PERIODS

- GROUND MOTION MODEL

  VERY IMPORTANT
  UNCERTAINTY EXTREMELY IMPORTANT AT LONG RETURN PERIODS

NUREG/CR/1582

- VARIATIONS INCREASE WITH RETURN PERIOD
- IN EXTRAPOLATION BEYOND EXISTING DATA, SIGNIFICANT DIFFERENCES CAN RESULT FROM SMALL VARIATIONS IN INPUT PARAMETERS
- MOST IMPORTANT IS INTERACTION OF DIFFERENT PARAMETERS

D. M. PERKINS - "EFFECT OF CHANGING RETURN PERIODS ON PROBABILISTIC GROUND MOTION"

- ROUGH RULES OF THUMB
- CAN ESTIMATES BE MADE FOR LONG RETURN PERIODS?
- SINGLE PARAMETER VARIATIONS

ROUGH RULES OF THUMB (PERKINS)

- UNCERTAINTY IN GROUND MOTION INCREASES PGA BY 25 TO 40% AT SHORT PERIODS (HUNDREDS OF YEARS OR LESS) AND BY 100% AT LONG PERIODS (THOUSANDS OF YEARS OR MORE)

- TO GET EQUIVALENT CHANGES AT LONG PERIODS WOULD REQUIRE LARGER CHANGES IN INPUT PARAMETERS:
  - . 10 FOLD INCREASE IN LEVEL OF SEISMICITY OR
  - . TWO FOLD INCREASE IN RELATIVE SIZE DISTRIBUTION OR
  - . INCREASE IN MAXIMUM MAGNITUDE FROM 5.0 TO 8.5

- AT LONG RETURN PERIODS UNCERTAINTY IN GROUND MOTION SWAMPS REASONABLE CHANGES IN SEISMICITY PARAMETERS

GROUND MOTION VARIATIONS (PERKINS)

- DUE TO VARYING "STRESS DROPS" (?)
- MOST IMPORTANT ASPECT IS CONFIGURATION OF TAIL OF DISTRIBUTION
- IF "STRESS DROP" IN EUS VARIES MORE THAN WUS, LONG PERIOD ACCELERATION MAY BE GREATER IN EUS.

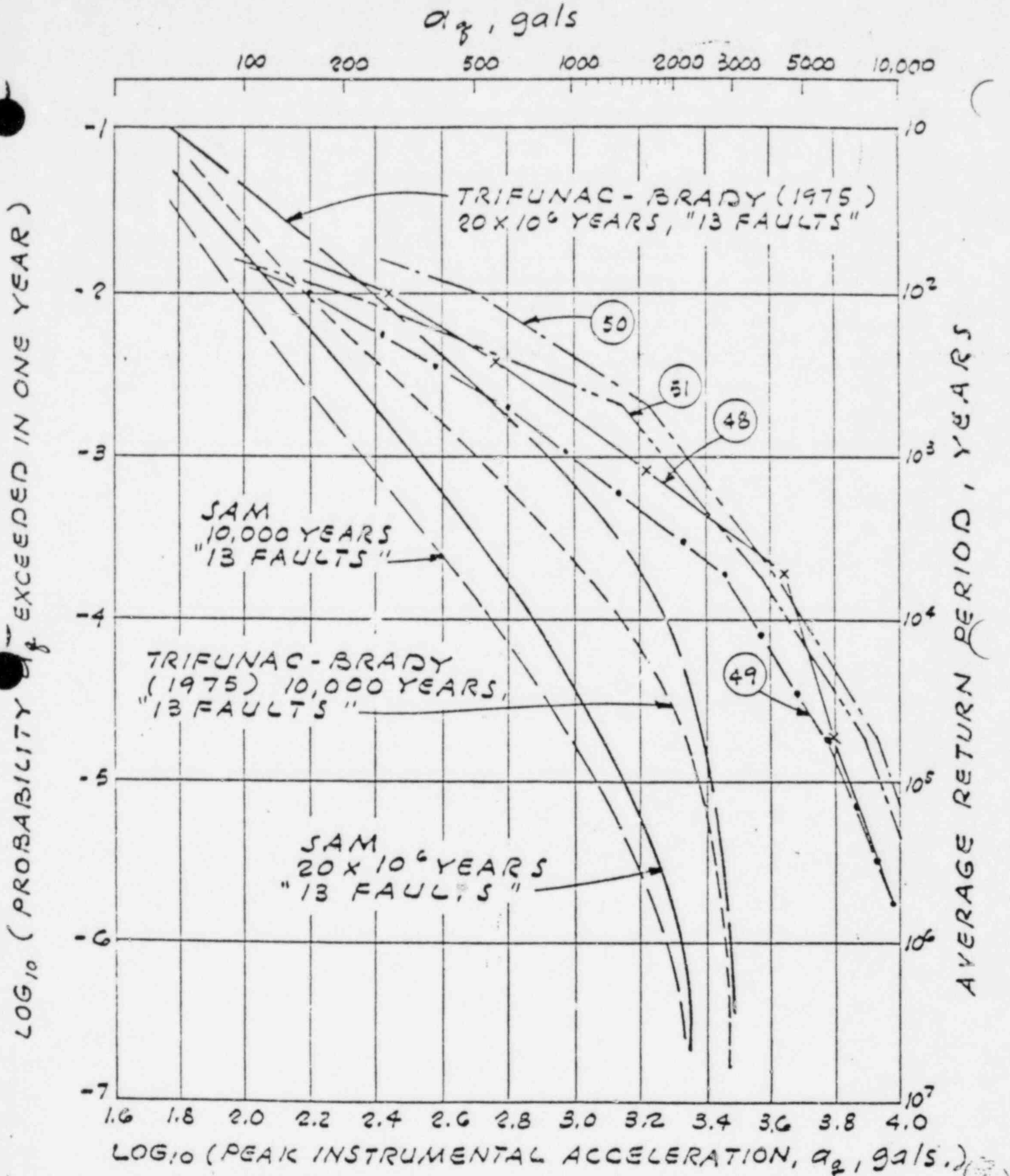
WORDS OF CAUTION

- VENEZIANO ---- "AVOID PLACING EXCESSIVE CONFIDENCE ESTIMATES OF HAZARD ESPECIALLY IN THE CASE OF RARE EVENTS"

- PERKINS -- "IF FOR SOME REASON SITE SPECIFICATION MUST CONSIDER RELATIVELY VARY LONG RETURN PERIODS, THEN THE SITE QUALIFICATION PROCESS IS FACED WITH A RESEARCH PROBLEM THAT LOOKS LIKE IT HAS A LONG TIME TO RUN"

NATIONAL ACADEMY OF SCIENCES PANEL

"AT PRESENT, BECAUSE OF THE MANY UNCERTAINTIES IN THE EXISTING GEOLOGIC AND GEOPHYSICAL DATA BASE FOR MOST PARTS OF THE NATION, EXTREME CAUTION MUST BE EXCERCISED WHEN USING THE RESULTS OF MOST COMPUTER-PRODUCED EARTHQUAKE RISK ANALYSES THAT ARE BECOMING AVAILABLE. THE PANEL BELIEVES THAT AT THIS TIME STATISTICAL PROBABILISTIC ANALYSES SHOULD BE USED FOR INSIGHT RATHER THAN FOR NUMERICAL RESULTS.



COMPARISON OF ANDERSON-TRIFUNAC AND BLUME RESULTS

PRESENT PROBABILISTIC GUIDELINES

- PROBABILISTIC ESTIMATES ARE POWERFUL TOOLS
- PROCEED SLOWLY WHILE EXPANDING UTILIZATION
- EMPHASIZE RELATIVE USE
- EMPHASIZE INSIGHTS
- RELIANCEUPON PROBABILISTIC ESTIMATES FOR VERY LONG RETURN PERIODS IS NOT THE WAY TO ALLEVIATE CONCERNS ABOUT EARTHQUAKES GREATER THAN SSE.
- ENCOURAGE RESEARCH TO FACILITATE INCREASING USE OF PROBABILISTIC ESTIMATES

SEISMIC MARGIN EARTHQUAKE
(SME)

- BASED ON SITE SPECIFIC EARTHQUAKE
- INCLUDES STRUCTURES AND EQUIPMENT
- SCREENING PROCESS USED TO IDENTIFY ELEMENTS AND COMPONENTS FOR REVIEW FOR SEISMIC ADEQUACY
- ALLOWS FOR DEVIATIONS FROM STANDARD REVIEW PLAN FOR FAILURE CAPACITY EVALUATION

DIFFERENCES BETWEEN SME AND DESIGN

- SEISMIC INPUT
- RANGE OF SOIL PARAMETERS
- PARAMETRIC VARIATION OF RELATIVE SOIL STIFFNESS UNDER AUXILIARY BUILDING PENETRATION WINGS
- DAMPING

	FSAR	SME
REINFORCED CONCRETE	5%	7%
WELDED STEEL	2%	4%
SOIL MATERIAL	3%	5%
COMPOSITE MODAL DAMPING	10% MAX	*

* DETERMINED FROM TIME HISTORY DIRECT INTEGRATION (UP TO 18% FOR AUXILIARY BUILDING)

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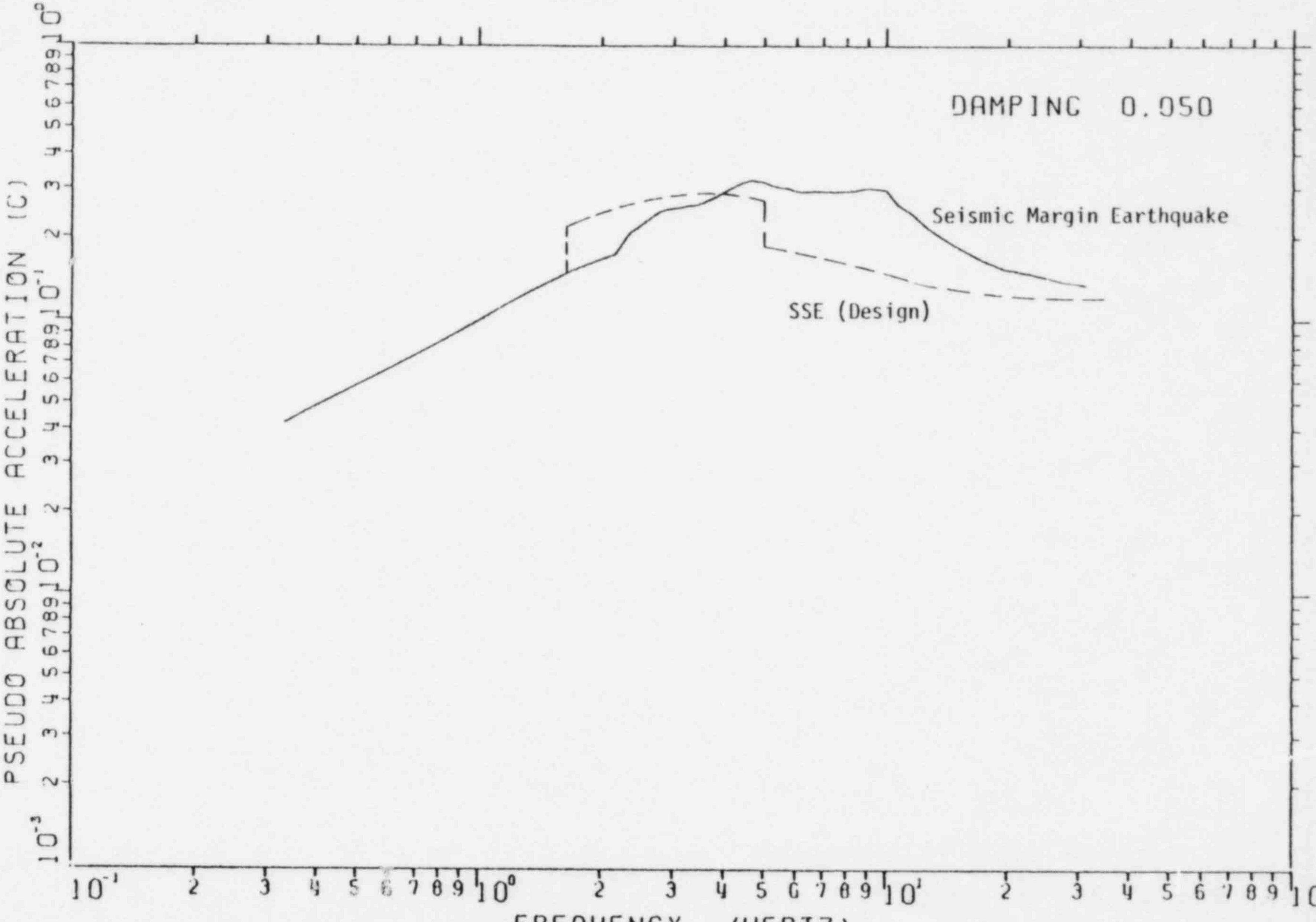
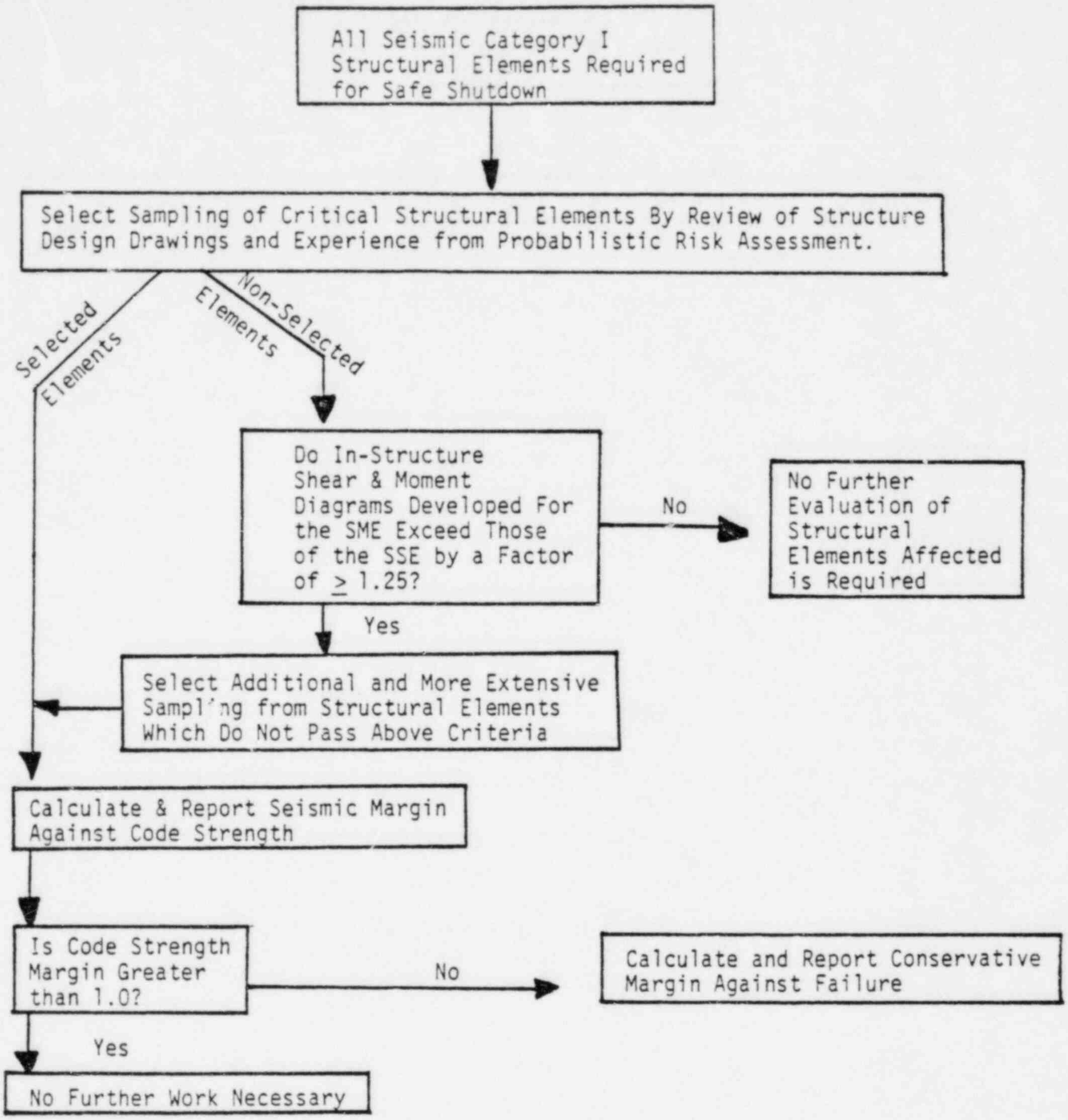
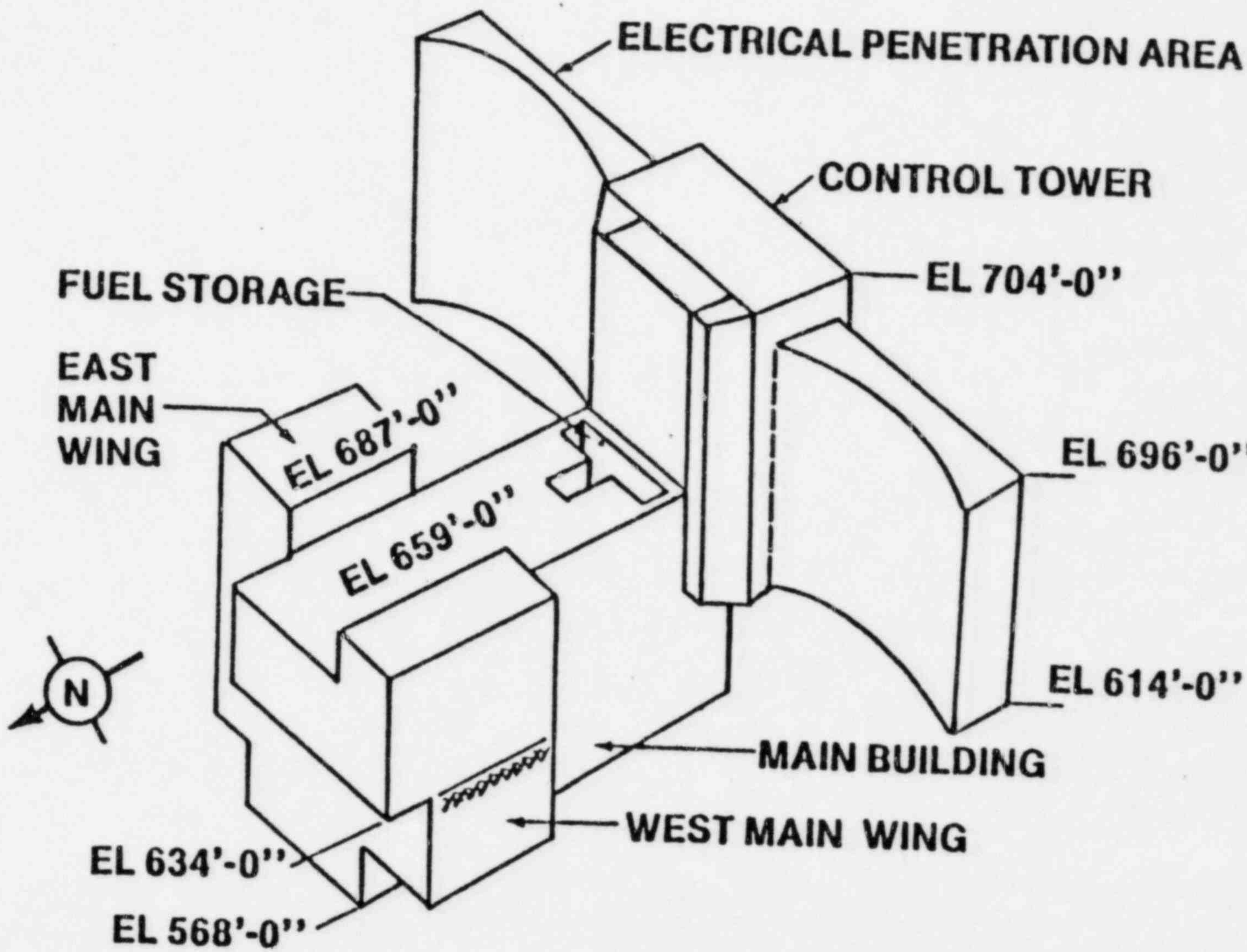


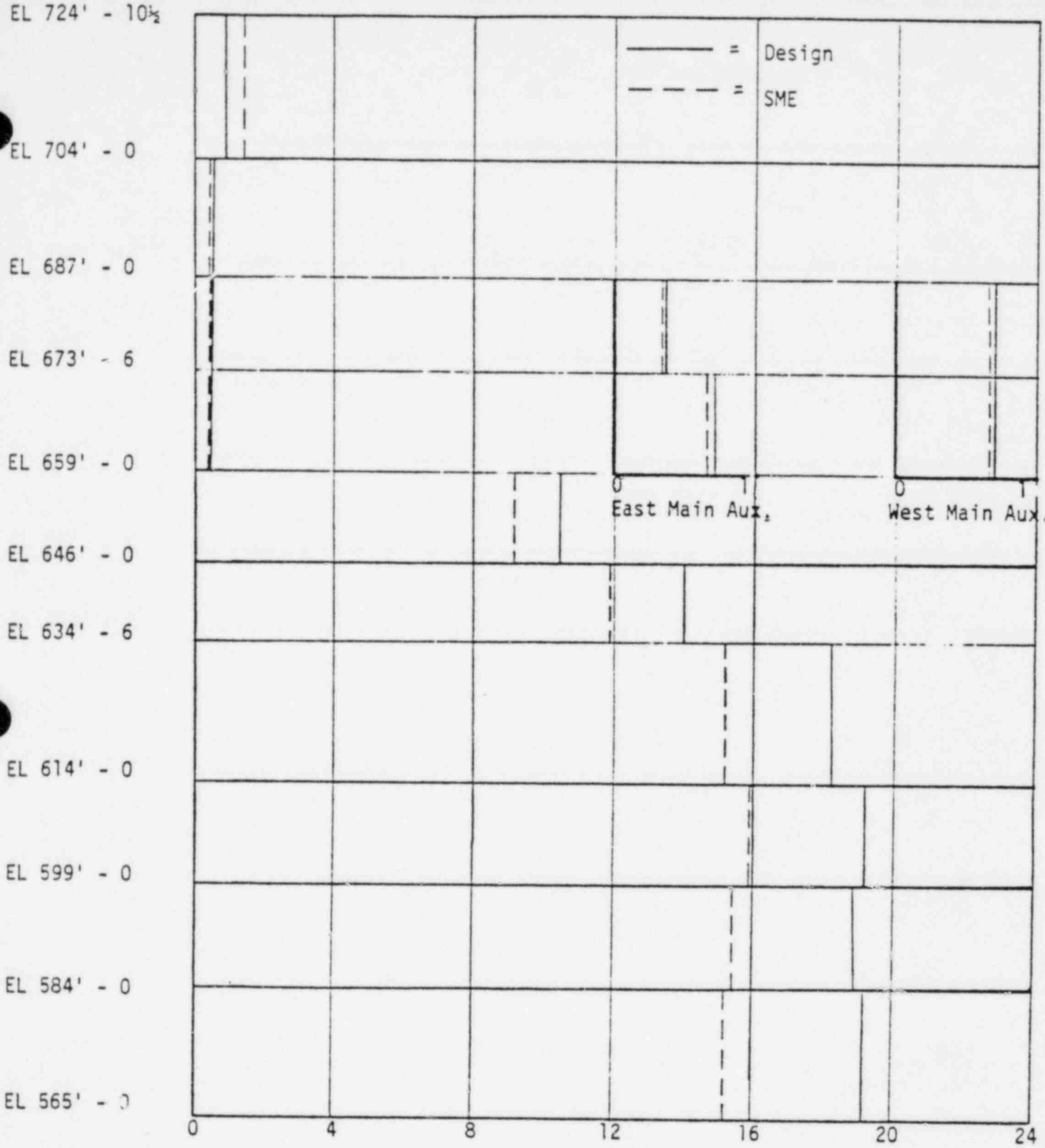
MIDLAND - ORIGINAL GROUND SURFACE ENVELOPE RESPONSE SPECTRA



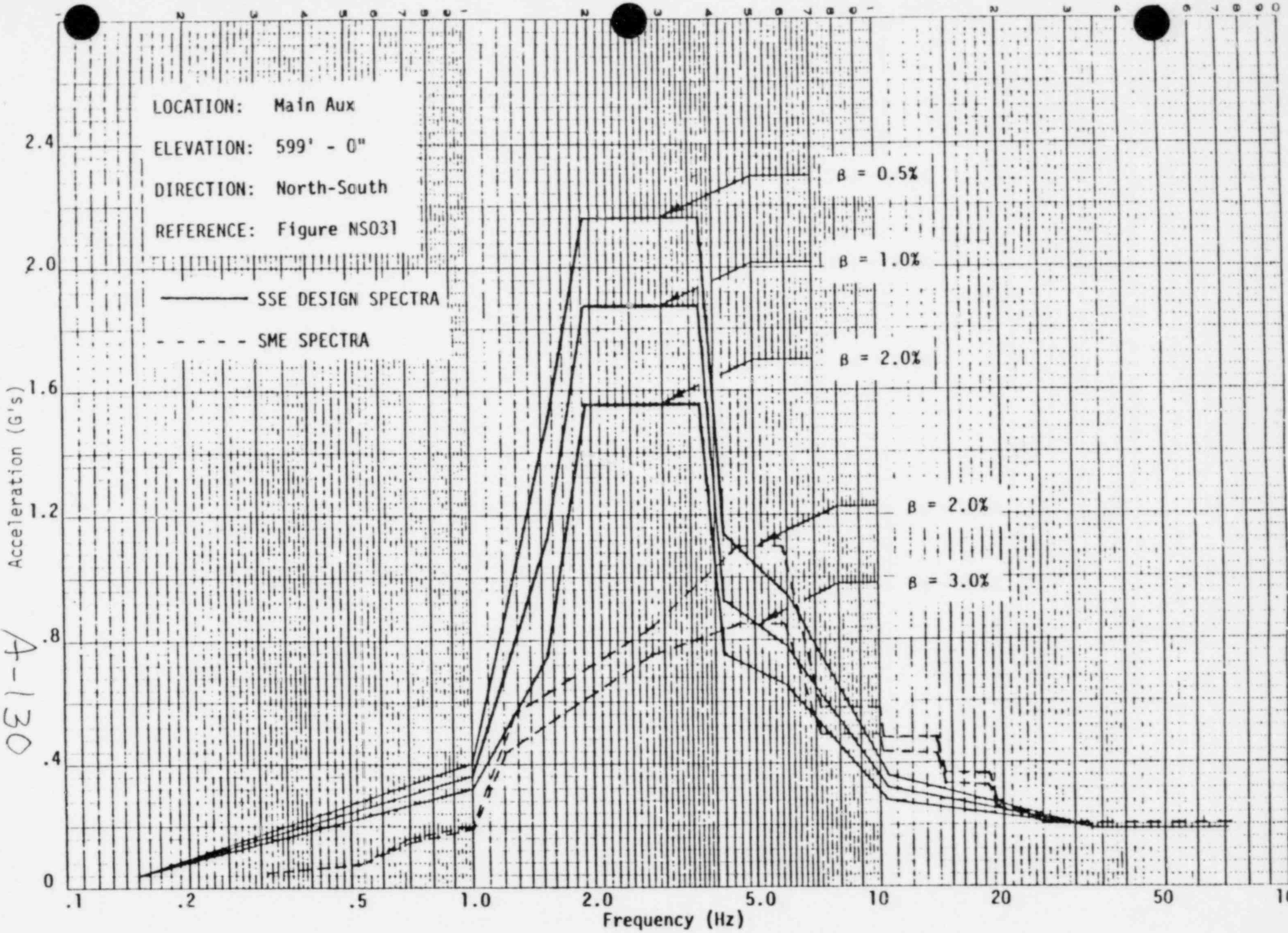
SCREENING PROCESS TO SELECT STRUCTURAL ELEMENTS FOR SEISMIC SAFETY MARGIN EVALUATION

A-128


ELECTRICAL PENETRATION AREA
CONTROL TOWER
EL 704'-0"
FUEL STORAGE
EAST MAIN WING
EL 687'-0"
EL 659'-0"
EL 696'-0"
N
EL 614'-0"
MAIN BUILDING
WEST MAIN WING
EL 634'-0"
EL 568'-0"

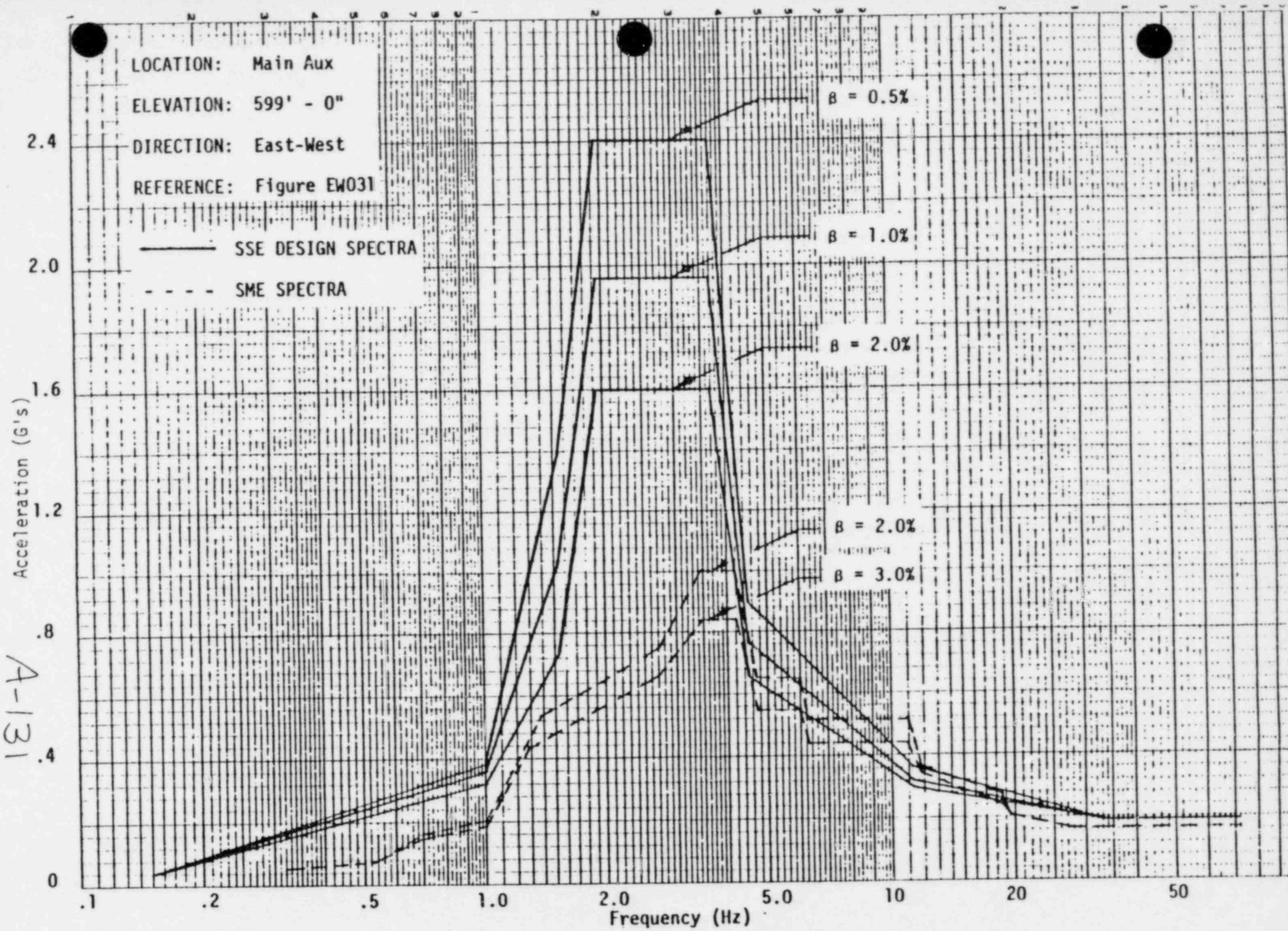


SRSS N-S SHEAR (10^3 k)
MAIN AUXILIARY BUILDING

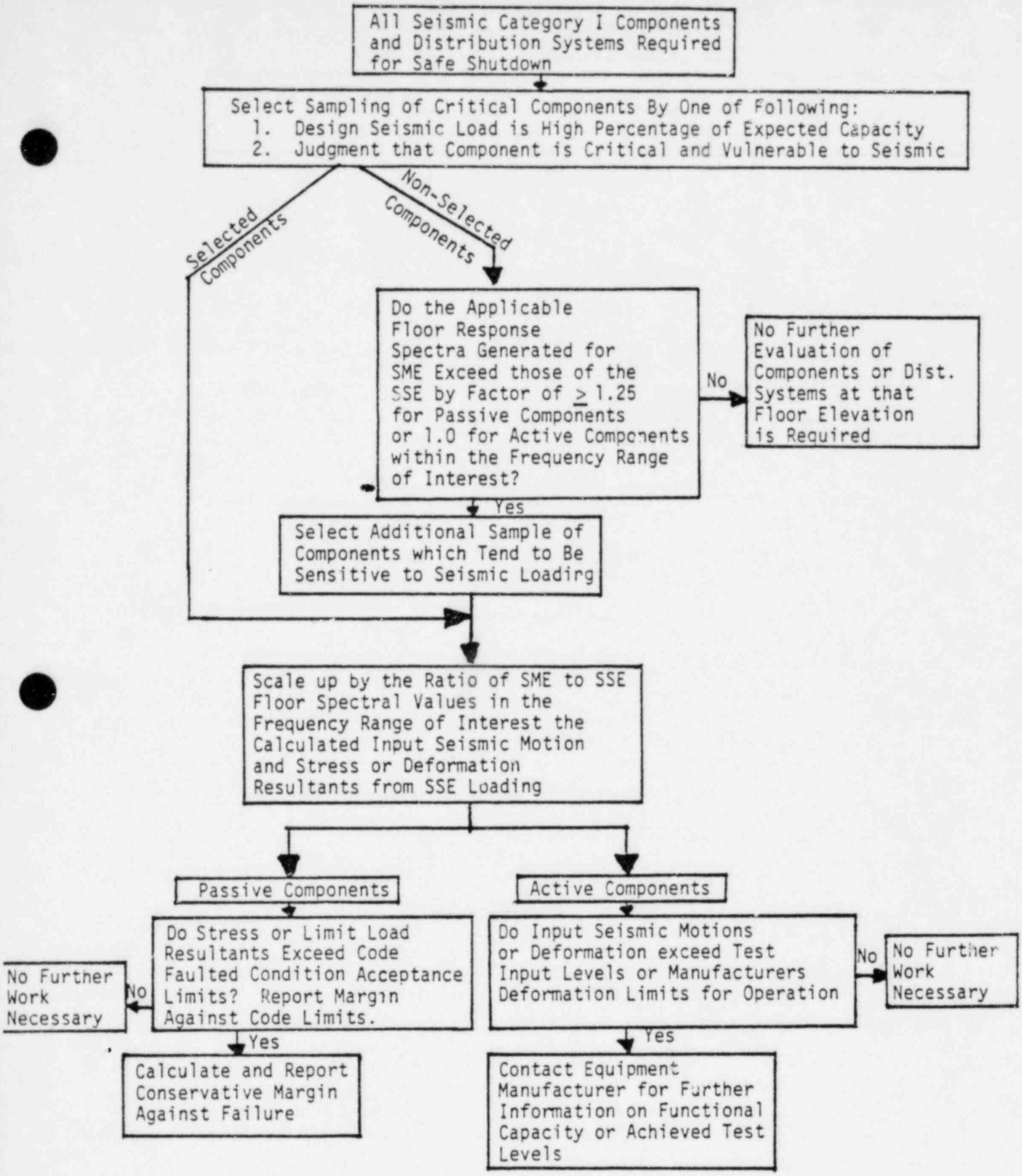


Comparison of SSE Design and Enveloped SME Spectra
Main Auxiliary Building, Elevation 599' - 0",

A-130



Comparison of SSE Design Spectra and Enveloped SME Spectra
Main Auxiliary Building, Elevation 599' - 0",
East-West Direction



SCREENING PROCESS TO SELECT COMPONENTS AND DISTRIBUTION SYSTEMS FOR SEISMIC SAFETY MARGIN EVALUATION

<u>SAMPLING CRITERIA FOR EQUIPMENT</u>

SAMPLING LIMITED TO EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

SAMPLING BASED ON SEVERAL CRITERIA WHICH INCLUDE:

A) HIGH SEISMIC STRESS FOR FSAR EARTHQUAKE

B) CRITICALITY OF FUNCTION

C) EQUIPMENT DEEMED MOST VULNERABLE BASED UPON PLANT WALKDOWN, AND JUDGMENT

D) EQUIPMENT LOCATION, IE, HIGHER ELEVATIONS EXPERIENCE GREATER ACCELERATIONS

<u>EXAMPLES</u>

- PIPING - SELECTION OF PIPE LINES FOR ANALYSIS MADE FROM BECHTEL PIPE STRESS SUMMARIES. ONLY HIGH STRESS LINES THAT CONTAIN ACTIVE VALVES ARE CONSIDERED

- PIPE SUPPORTS - ALL SUPPORTS FOR PIPE LINES SELECTED FOR RE-ANALYSIS ARE EXAMINED. IF SME LOAD EXCEEDS FSAR SSE DESIGN LOAD, SUPPORTS ARE ANALYZED FOR MARGIN

- ACTIVE VALVES - ALL ACTIVE VALVES IN PIPE LINES SELECTED FOR ANALYSIS ARE INCLUDED. SME ACCELERATIONS ARE COMPARED TO VALVE QUALIFICATION ACCELERATIONS

EXAMPLES (CONTINUED)

- SWITCHGEAR, MOTOR CONTROL CENTERS, INSTRUMENTATION & CONTROL EQUIPMENT SELECTED ON BASIS OF FUNCTION CRITICALITY AND LOCATION. MOST, EXCEPT SWITCHGEAR, IS MOUNTED HIGH IN STRUCTURE.

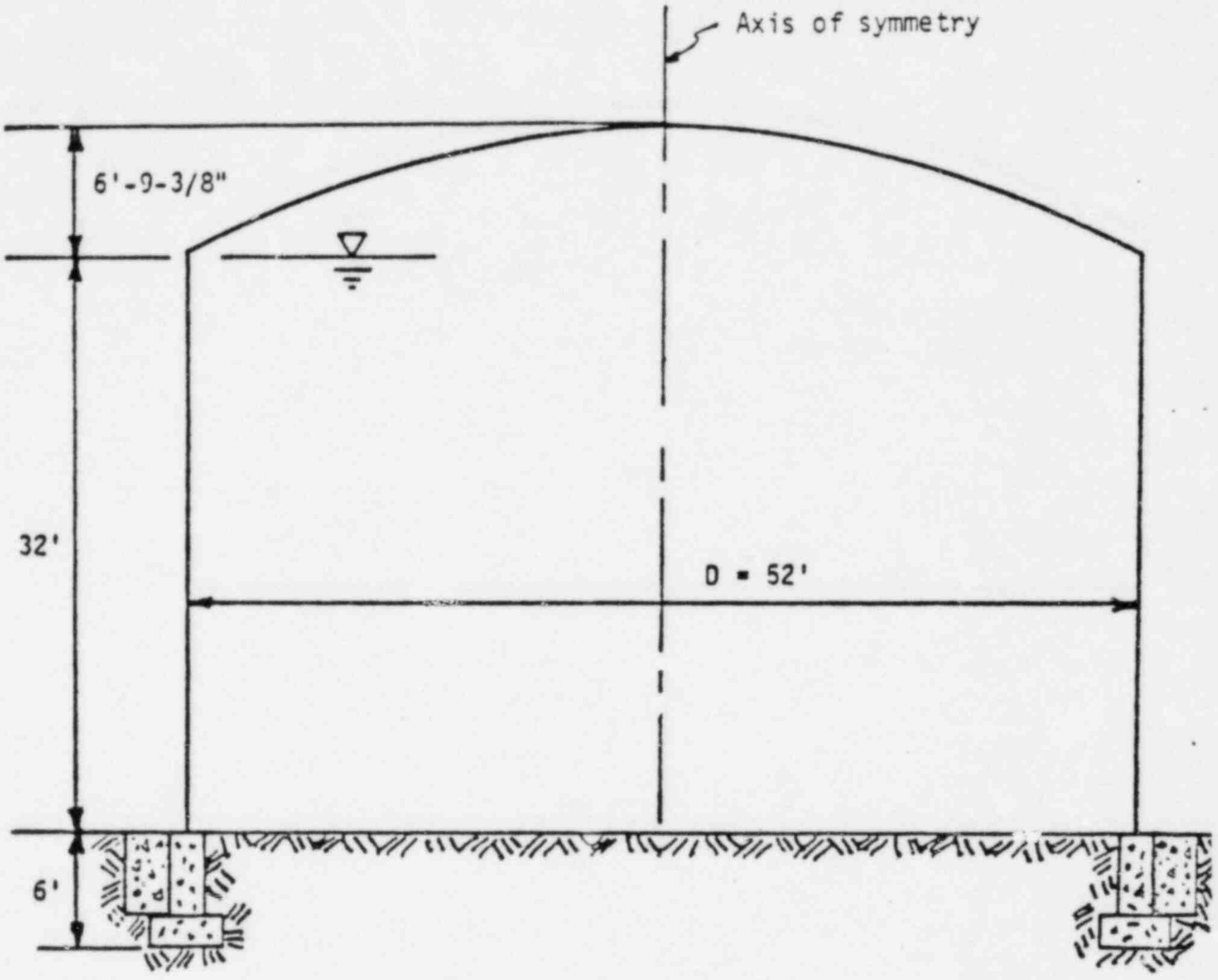
- PUMPS SELECTED ON BASIS OF FUNCTIONAL CRITICALITY AND POTENTIAL VULNERABILITY TO SEISMIC EXCITATION. VERTICAL AND HORIZONTAL PUMPS ARE INCLUDED.

- HEAT EXCHANGERS-SELECTED ON BASIS OF CRITICALITY AND POTENTIAL VULNERABILITY. VERTICAL AND HORIZONTAL HEAT EXCHANGERS ARE INCLUDED.

- VESSELS - LARGE VESSELS SELECTED BECAUSE OF LARGE REACTIONS ON SUPPORTS AND GREATEST POTENTIAL FOR SEISMIC DAMAGE.

- BATTERIES AND DIESEL GENERATOR SELECTED ON BASIS OF CRITICALITY TO SAFETY SYSTEM OPERATION.

- CABLE TRAYS AND HVAC DUCTING SELECTED FROM EACH BUILDING AT UPPER ELEVATIONS WHERE INERTIAL LOADING WILL BE GREATEST



BORATED WATER STORAGE TANK CONFIGURATION

BORATED WATER STORAGE TANK

SME CODE MARGINS

(SME + DW + SETTLEMENT)

Stress or Load Parameter		CM	F_{SME}
Ring Wall Moments and Shear Capacity		1.57*	NA**
Soil Bearing Capacity		1.57	2.28
Tank Sliding Capacity		3.00	3.00
Uplift Capacity of Foundation		1.95	1.75
Anchor Bolt Uplift Capacity		2.65	3.27
Bolt Chair Uplift Capacity		1.50	1.69
Tensile Hoop Stress	3/8" Shell	2.44	12.1
	1/4" Shell	2.13	8.40
Compressive Buckling Stress	3/8" Shell	2.22	3.74
	1/4" Shell	1.53	2.44
Local Membrane Stresses of Bolt Chair		1.71	2.89

* Very conservatively evaluated
** Not computed

APPENDIX XIV
MIDLAND PLANT - LTR FROM H. DENTON TO HENRY MYERS DTD 5/7/82 RE. NRC STAFF COMMENTS TO SUBCMTE ON ENERGY AND ENVIRONMENT

MAY 7 1982

Dr. Henry Myers
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Dr. Myers:

In early April you requested the NRC staff to comment on a statement in paragraph 619 of the TMI Restart Partial Initial Decision. The statement read as follows:

> "If, however, the voids are steam, as would be expected in a small-break LOCA, the bubble in the hot leg should be compressed and condensed as the primary system pressure is increased by operation of the HPI system."

The context of this statement in paragraph 619 is a discussion of the recovery from a small-break LOCA when ECCS injection flow exceeds break flow so that the reactor system is filling. The discussion addressed a Union of Concerned Scientists (UCS) concern that for this condition a steam bubble in the top of the hot leg U-bends might prevent the reestablishment of single-phase natural circulation.

Mr. Kammerer of NRC replied to your request in a letter of April 8, 1982. The response included the following comments:

> "If a steam bubble exists and primary system pressure is raised, the bubble will be compressed and there will be condensation. <u>The condensation occurs because as you raise system pressure system temperature drops below saturation.</u> Condensation must occur to reach saturation conditions again. The bubble will not necessarily be completely condensed. This depends on bubble size and pressure change." (Sentence was underlined in your April 19, 1982 letter.)

In a letter of April 19, 1982, you asked if the quote underlined above from our April 8, 1982, response was correct. Specifically, you questioned "... whether, in the case of a small break LOCA, the heat transfer would be sufficient to keep the steam space from expanding into the core." The underlined quote is not correct because there must be heat transfer to a heat sink to absorb the heat of vaporization associated with the condensation of steam, and the heat transfer rate is important. Also, the system temperature does not actually drop; rather, the saturation temperature increases with increasing pressure. However, while the condensation rate of the hot leg bubble will affect the overall recovery behavior, it does not, by itself, govern the overall adequacy of decay heat removal during a small break LOCA in a plant designed by B&W. To be certain that there is no similar misunderstanding regarding this matter, we are providing a copy of this letter to the Licensing Board and will inform the Appeal Board of this issue during the upcoming appellate process.

A137

The response of a B&W designed reactor to steam accumulation at the top of the hot leg U-bends under off-normal conditions is complex. The detailed behavior of the reactor under these conditions is still being actively studied by B&W, the B&W plant owners, and the staff. Although uncertainties remain, including heat transfer rates for steam condensation, we have reached some general conclusions about the effect of these uncertainties and the potential for producing unacceptable core heatup. These conclusions are based on calculations by B&W and/or by a staff contractor of a number of postulated small break LOCAs. We have concluded that the uncertainties can be physically bounded and that these bounding assumptions do not produce unacceptable core heatup.

In the enclosure, the staff addresses your questions regarding the impact of steam condensation rates and supporting analyses and describes how the use of bounding assumptions would not result in unacceptable core heatup. Calculations related to these conclusions are identified.

There are still uncertainties in the thermal-hydraulic response of the B&W design under some of these conditions. We are pursuing with B&W and B&W plant owners the details of the thermal and hydraulic phenomena involved, including the possible need for additional sensitivity analyses and a small scale, integral systems test. Our objective is to confirm the analytical results described in the enclosure and to aid in our further analysis of more complex, multiple failure events being studied in the context of the new symptom-oriented, emergency procedure guidelines.

Sincerely,
Original Signed by
H. R. Denton
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

Distribution
ECase
HDenton
PPAS
Deisenhut
RVollmer
HThompson
PCheck
WDircks
KCornell
TRehm
RDeYoung
GCunningham
LUnderwood
RMattson
TSpeis
BSheron
Central File
RSB R/F

RSB Subject File
BSheron R/F
SCavanaugh-11793
RCapra-11793
DMeyer-11793
PBrandenberg-11793
OCA
SECY #82-0416

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SEE PREVIOUS CONCURRENCE SHEET FOR INITIALS.

OFFICE	DSI:RSB	DSI:AD:RS	DSI:DIR	NRR:DEP:DIR	NRR:DIR	EDO	
NAME	BSheron*	TSpeis*	RMattson	ECase	HDenton	WDircks	
DATE	5/3/82	5/3/82	5/6/82	5/6/82	5/6/82	5/7/82	

ENCLOSURE

DYNAMIC RESPONSE OF B&W REACTORS TO SMALL BREAK LOCAS

Analyses have been performed by B&W or by staff contractors for the two classes of small break LOCAs: (1) those that can be subsequently isolated (e.g., letdown lines, PORVs), and (2) those that are not isolatable. They are discussed in that order, below.

Small Breaks Which Are Subsequently Isolated

We have had our contractor, the Los Alamos National Laboratory (LANL) perform an analysis of a small break in the cold leg of the B&W reactor coolant system that is subsequently isolated. The calculations were performed with the advanced TRAC computer code, and we have some preliminary results.

In our analysis, the system was assumed to lose primary coolant from the break until the upper vessel head region, pressurizer, and hot leg U-bends were filled with steam. At that time, the break was assumed to be isolated by the operator. Approximately 1,000 seconds later, it was assumed the operator began a controlled secondary system depressurization. The analyses showed that the flow of cold water from the two high pressure injection pumps, coupled with controlled secondary system depressurization, would condense the hot leg bubbles and restore natural circulation. We are still evaluating the ability of the TRAC code to model the heat conduction in the liquid near the steam-liquid interface which is required to accurately calculate the primary system refilling process. However, steam condensation rates are not expected to influence the overall conclusion that no unacceptable core heatup would occur, as explained in the following paragraphs.

If the steam was condensed at 100% efficiency by the cold HPI water, the top of the hot leg U-bends would refill with liquid and single phase natural circulation would be restored. If, however, the steam condensation rate is very low, and in fact, in the limit the steam is assumed not to condense at all, two possibilities would result.

The first possibility is that the HPI pumps would repressurize the system sufficiently to compress the steam to a small enough volume to allow liquid on the upstream side of the hot leg U-bend to spill over into the downstream side and resume natural circulation. If this did not occur, the HPI would continue to inject ECC water and repressurize the reactor coolant system until the pressure reaches either the PORV or the safety valve set pressure. We would expect that core cooling would be maintained by a "feed and bleed" process until the steam bubble at the top of the U-bend eventually condensed by heat transfer to the pipes and across the liquid vapor interface. Once the bubble condenses, single phase natural circulation throughout the steam generators would be resumed.

The other possible behavior if the steam condensation rate is low is associated with a design difference in B&W reactors. One plant designed by B&W, the Davis-Besse plant, does not have a "high head" HPI pump (one that can pump water into the primary system at or above the safety valve set point). The shutoff head of this pump is about 1700 psi. In the event the cold water from the HPI pumps does not condense the steam in the hot leg U-bend, the system may repressurize to above the shutoff head of the HPI pump, and eventually reach the PORV or the safety valve setpoint. The system will begin to lose primary coolant through the PORV or the safety valves and drain down. Once the primary coolant level on the downstream side of the hot leg U-bend extends into the steam generator tube region below the condensing surface of the secondary coolant, steam in the primary system will begin to condense, lowering the pri ary system pressure and closing the PORV or the safety valve. This mode of decay heat removal is referred to as the boiler condenser mode of two-phase natural circulation.

As the system pressure decreases below 1700 psi, the HPI will actuate and begin to fill the primary system. This would result in covering the condensing surface in the steam generator, and producing another repressurization, which, in turn, could stop HPI flow and cause the PORV or the safety valve to open and release enough coolant to reestablish a condensing surface. A number of these cycles may occur before the charging system completely refills the system or before the steam bubble is condensed by heat transfer to pipes and across the liquid vapor interface. Since the condensing surface in the steam generators is above the elevation of the top of the core, natural circulation should be established before the hot leg steam bubble extends into the core.

In addition to our recent contractor analyses, there are some B&W evaluations that support our current understanding of system performance for an isolated small break. In their May 7, 1979 submittal on small break LOCA behavior, B&W evaluated a small break LOCA which is subsequently isolated. This evaluation did not involve any thermal-hydraulic computer code analyses. It showed that no unacceptable core heatup would result. In addition to this B&W evaluation, the staff recommended in section 2.6.2.C of NUREG-0565 (Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants) that licensees with B&W designed reactors perform an analysis of a small break which is isolated and the PORV fails open upon repressurization. In response to this recommendation, B&W licensees submitted an analysis of a small break (0.01 square feet) with a concurrent loss of all feedwater. The break size selected was insufficient to remove all of the decay heat and thus re-ressurization to the PORV setpoint results due to the assumed loss of all feedwater. Once opened, the PORV in this analysis was assumed to stick open producing a second "small break" in the system. The analyses showed that the HPI water injected by two HPI pumps was sufficient to keep the core from uncovering. This event was presented since it produced a repressurization similar to that expected for the isolated break scenario, yet produced an inventory loss more severe than the isolated break case.

Small Breaks Which Are Not Isolated

Small break LOCAs which are not subsequently isolated have been calculated by both B&W and by the staff's contractor, LANL. The analysis performed by B&W was for a 0.01 ft^2 cold leg break. They used a computer code that conforms to the requirements of Appendix K to 10 CFR 50. With the exception of the assumed number of HPI pumps available, the modeling assumptions required by Appendix K do not affect the thermal-hydraulic models of interest for the small break LOCA. Thus, the results of the hydraulic analysis should be realistic. The 0.01 square foot break was selected since this size is insufficient to remove decay heat via break flow (thus requiring the steam generators for decay heat removal). It was also predicted to result in the repressurization phenomenon for the reactor coolant system.

From this analysis, B&W concluded that a range of small break sizes could be postulated in which steam generated in the core would accumulate at the top of the hot leg U-bends and cause an interruption of natural circulation flow. The interruption of natural circulation flow would isolate the steam being produced in the reactor core from the steam generator heat sink. The net steam accumulation in the system was calculated to cause the primary system to repressurize. This repressurization was calculated to continue until the primary system coolant loss through the break was sufficient to uncover a steam condensing surface in the steam generators. It is expected that some steam generated in the core would flow into the upper elevations of the downcomer annulus via the vent valves and condense in the colder water in that region. However, cold water from one HPI pump was not calculated to be sufficient to condense all of the steam generated in the core.

Similar to the isolated break case previously discussed, the repressurization of the reactor coolant system caused by interruption of natural circulation would lead to a boiler-condenser mode of two-phase natural circulation, and subsequently reduce system pressure. Once the HPI flow is calculated to exceed the break flow, the system coolant inventory will stop decreasing and begin to increase. In figures 1, 2, and 3, the temporal behavior of system pressure, liquid level in the hot leg piping, and liquid level in the reactor vessel are shown for this case as calculated by B&W. The B&W analyses submitted to the staff terminate at about the time system inventory begins to increase. However, the continued recovery of the event is considered relevant to your concern, and is described further below.

As the system refills, the steam condensing surface in the steam generator will again be recovered by liquid, and a steam bubble will be trapped at the top of the hot leg U-bend. The scenario is now expected to proceed similarly to the isolated break case previously described. That is, if the steam is rapidly condensed by cold HPI water during the refilling process, single phase natural circulation will be reestablished and primary system pressure will remain low with no significant repressurization. If the steam trapped at the top of the hot leg U-bend is not rapidly condensed, the system would repressurize until (1) the break flow exceeded the HPI flow and a condensing surface was reestablished, (2) the system repressurized and compressed the steam to a sufficiently small enough volume so that water upstream of the hot leg U-bend could spill over into the down-side of the hot leg U-bend and reestablish natural circulation, or (3) the PORV/safety valve setpoint was reached. For the Davis-Besse plant, we believe only the first three cooling modes would occur since the HPI pumps are not sufficient to pump water into the system above 1700 psia, a pressure well below the safety valve setpoint.

The staff has also been calculating and analyzing the response of B&W-designed reactors to small break LOCAs in which natural circulation is predicted to be interrupted by steam accumulation at the top of the hot leg U-bends. Our contractor at LANL has recently completed a few small break analyses and has looked at four recovery enhancement actions presently either proposed by B&W or being considered by the staff. These four options are: (1) high point vent operation, (2) momentary pump restart, (3) secondary side depressurization, and (4) ECC spray at the top of the hot leg U-bends. All of these options are being investigated to determine their ability to enhance the reestablishment of single phase natural circulation during the recovery portion of the accident. Initial results of our contractor's calculations show that for a realistically calculated small break (i.e., nominal decay heat, two HPI pumps available, etc.) in a B&W plant, with a break size in the range in which B&W predicts repressurization would occur, the system did not repressurize once the hot leg U-bends filled with steam. Although our evaluation is not yet complete, we believe that the reason the LANL calculation did not show a repressurization is because steam generated in the core vented to the upper reaches of the vessel annulus via the vent valves, and the flow of cold water from two HPI pumps was sufficient to condense all of the steam produced in the core.

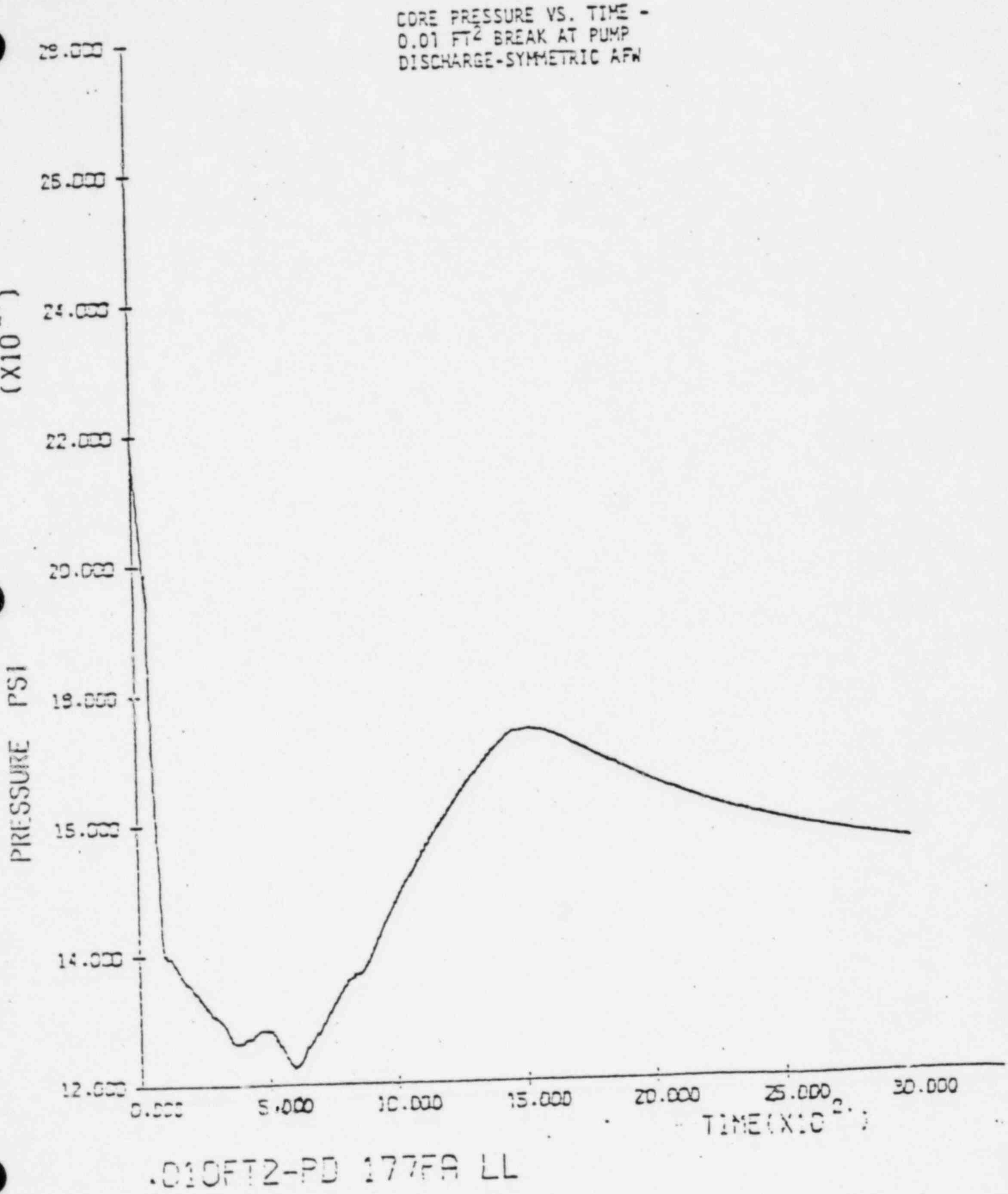
The results of the TRAC analyses are shown in figures 4 through 7. In figures 4 and 5, the B&W results are overlayed to show the differences. We believe the differences can be attributed to the assumption of two versus one HPI pump being available.

The results of the analyses to investigate natural circulation recovery enhancement methods were only recently presented to the staff at a meeting with LANL. These analyses have not been documented by LANL in a formal report, and we have not reviewed them in any detail. However, based on information received at the meeting with the contractor, the results show that the hot leg U-bend was not refilled following recovery from the small break LOCA (1.75 inch diameter), and that none of the natural circulation recovery enhancement methods previously listed were effective. However, LANL reported that the core remained covered and decay heat was continuously removed from the core. They attributed the heat removal to internal recirculation (steam exiting the core is vented to the vessel upper annulus and condensed by the cold HPI water entering the downcomer). This situation physically could only persist until the decay heat was eventually removed entirely by the break flow or the system eventually was refilled and natural circulation reestablished.

Before widely disseminating the results of the LANL calculations, we have asked our Office of Regulatory Research to carefully document and evaluate them and assist us in confirming their validity. Although this careful approach will take time, we believe it is justified because the analyses showed that core decay heat was continuously removed and that no core uncovery or heatup was predicted.

In summary, although we are continuing our evaluation of the rate of hot leg steam bubble condensation in the recovery from both isolated and unisolated reactor coolant system small-break LOCAs in reactors like TMI-1, we do not believe that steam bubbles present in the reactor coolant system resulting from small break LOCAs (either isolated or unisolated) in either the cold or hot legs of the primary system will, by themselves, result in unacceptable heatup of the core.

Figure 1



A-147

Figure 2

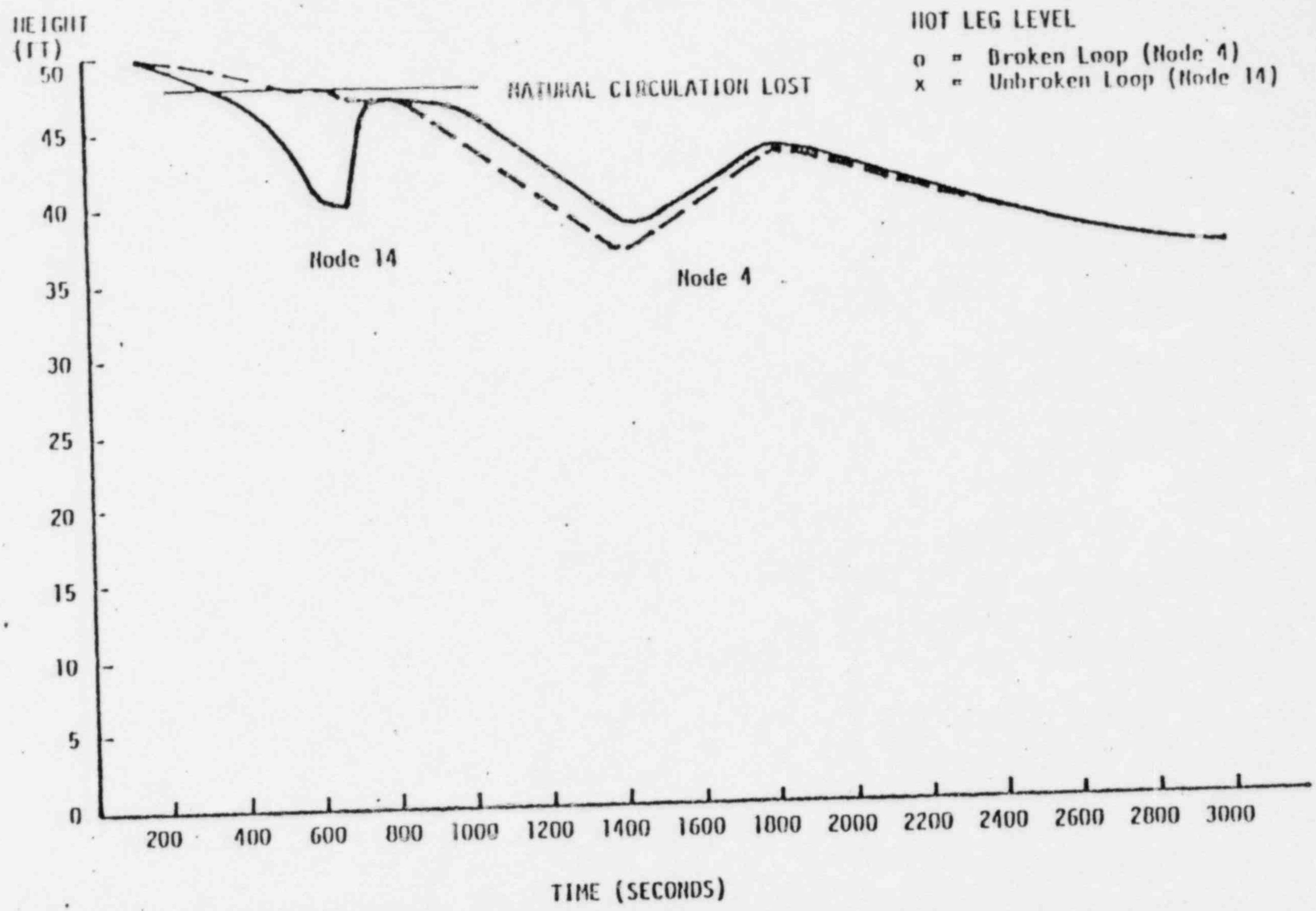
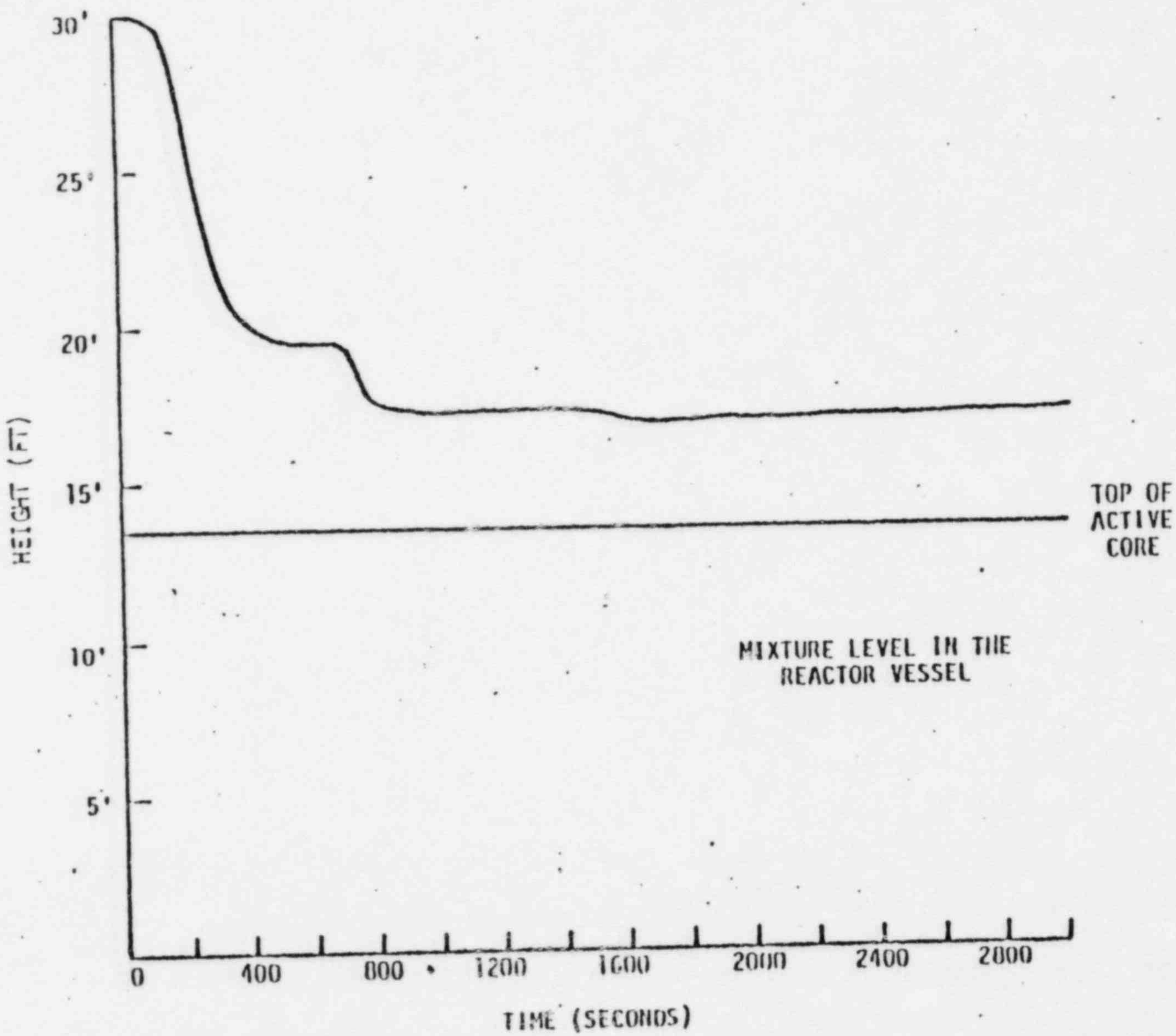

HEIGHT (FT)
50
45
40
35
30
25
20
15
10
5
0
NATURAL CIRCULATION LOST
HOT LEG LEVEL
o = Broken Loop (Node 4)
x = Unbroken Loop (Node 14)
Node 14
Node 4
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
TIME (SECONDS)

Figure 3


30'
25'
20'
15'
10'
5'
HEIGHT (FT)
TOP OF
ACTIVE
CORE
MIXTURE LEVEL IN THE
REACTOR VESSEL
0
400
800
1200
1600
2000
2400
2800
TIME (SECONDS)

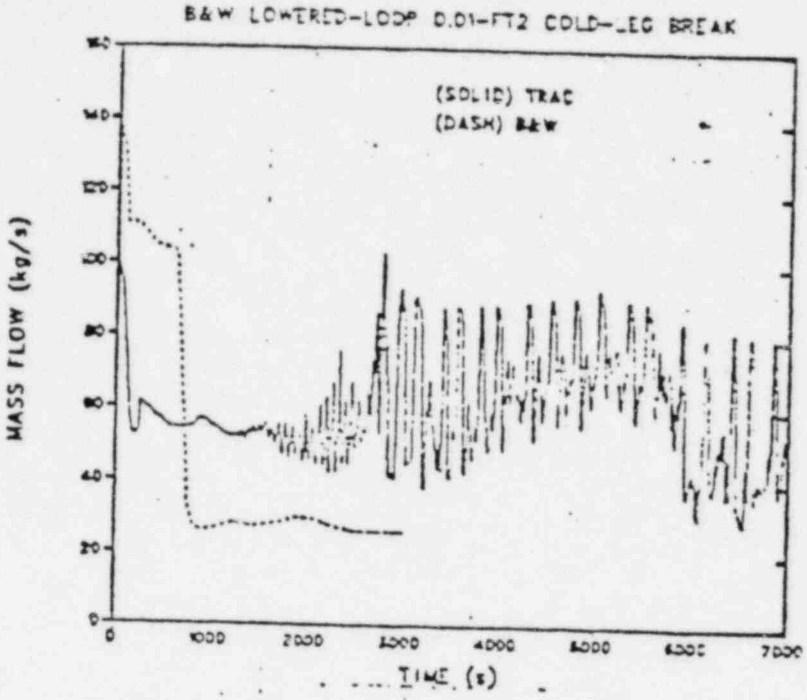


Figure 4
Comparison of TRAC and B&W break flows.

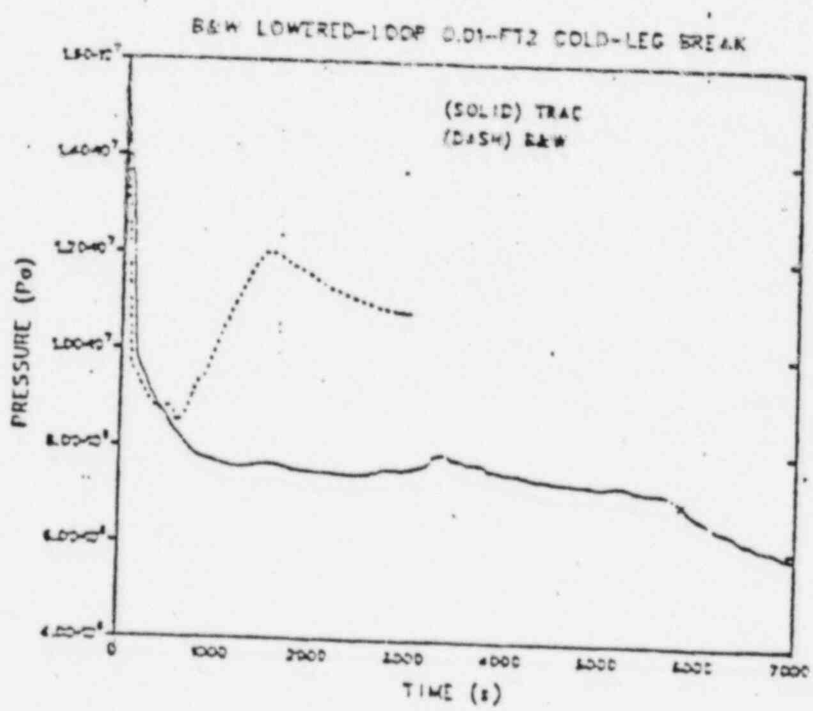


Figure 5
Comparison of TRAC and B&W primary pressures.

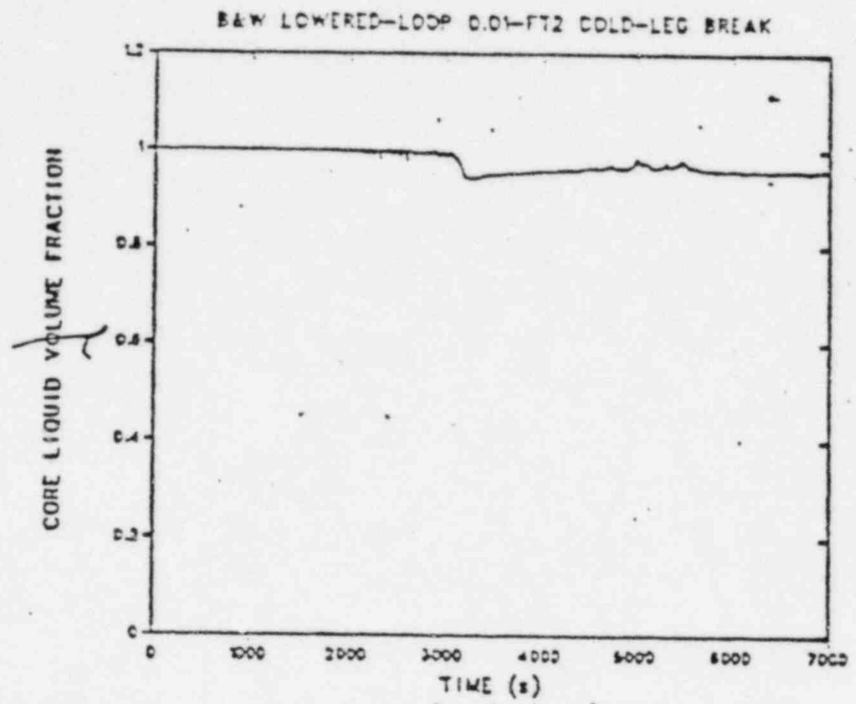


Figure 6
TRAC-calculated core liquid volume fraction.

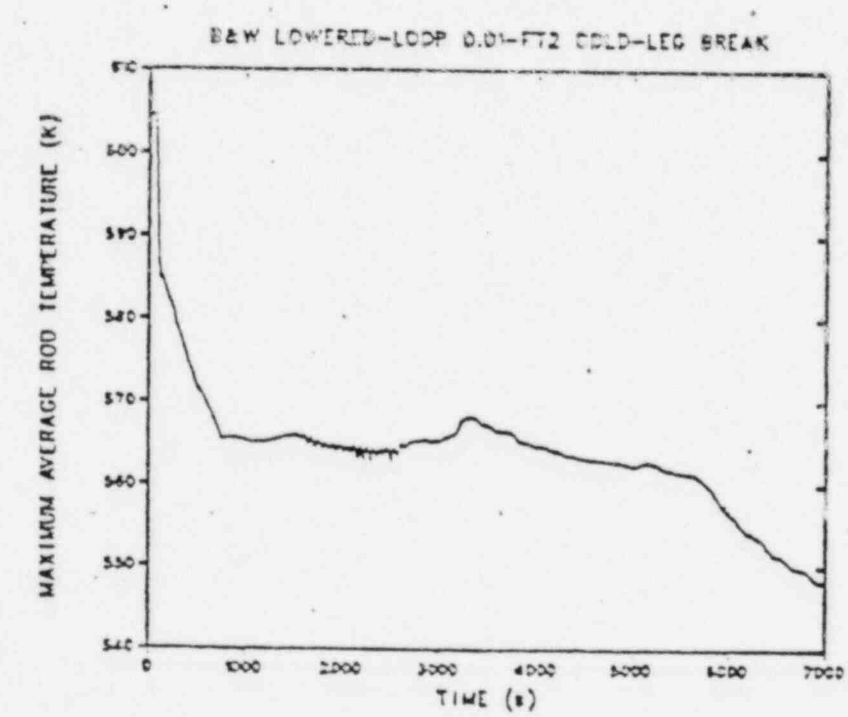


Figure 7
TRAC-calculated maximum average rod cladding temperature.

266th mtg
Press Thermal
Shock

APPENDIX XV
LTR FROM T. G. THEOFANOUS TO M. BENDER
RE. PRESSURIZED THERMAL SHOCK

To : M. Bender

From: T. G. Theofanous

Re: Vessel Wall Temperature Response in Perspective.

This report is in response to your suggestion for providing a "feel" for the wall temperature response under PTS. I think a broad perspective may be gained by considering the response of the wall to three kinds of coolant temperature transients: (a) a constant cooldown rate i.e. linear decrease of coolant temperature with time, (b) an instantaneous decrease from T_0 to a new value of T_∞ and (c) an exponential decrease from T_0 to T_f with a time constant of t_c.

For each case the results may be provided in the form generalized (dimensionless) plots such that one can read off the temperature for any point and any time.

Case a: Constant cooldown rate (of downcomer water).

i.e. $T_a = T_0 - Ct$ where C is a constant.

Temperatures may be read off Figure 1.

Example: Take $C = 300°F/hr$. Say we want the temperature at a point 30% away from the inner wall (i.e $\xi = 0.7$) and at time $t = 15$ min after the start of the transient. Since $\tau = 1$ corresponds to $t = 44$ min, time $t = 15$ should correspond to $\tau \sim 0.3$. We also need L. Say $h_c = 100$ BTU/hr ft²°F. The cladding corresponds to ~ 650 BTU/hr ft²°F Hence overall h is ~ 87 BTU/hr ft² °F and (since $L = 1$ corresponds to $h \sim 50$ BTU/hr ft² °F) we have $L = 1.73$.

From Fig. 1 we read $F_1 = 0.21$. Take $T_0 = 550°F$.

Therefore:

$$F_1 = \frac{550 - T}{300 \times \frac{1}{4}} = 0.21 \quad \therefore \; T = 550 - 15 = \underline{\underline{535°F}}.$$

$\frac{1}{4}$ hr = 15 min.

Note: Even if $h_0 \to \infty$ $h \to$ to the cladding value of 650 BTU/hr ft² °F (based on 1/8" stainless steel cladding) i.e. $L_{max} = 13$ and this is why I cover up to $L = 10$.

Case b: Instantaneous cooling of downcomer water to T_∞.

Example: Take $T_\infty = 300°F$. Everything else specified as in example of case a. From Fig. 2 we read $F_2 = 0.63$

Hence:

$$F_2 = \frac{T - 300}{550 - 300} = 0.63 \quad \therefore \; T = \underline{\underline{457°F}}$$

Case c: Exponential decrease given by:

$$T_c = T_0 + (T_f - T_0)(1 - e^{-t/t_c})$$

where T_f is final coolant temperature and

t_c time constant also specified.

Now:

$$\frac{T - T_0}{T_f - T_0} = F_3 (\xi, T ; L, T_c) \quad \text{where} \quad T_c = \frac{\alpha t_c}{\ell^2}$$

The function F_3 now depends on two parameters so we need to combine L and T_c values to get Figures similar to those of Fig. 1 & 2. This Fig. 3 will be provided shortly. Also I can get more complete computer plots than those sketched in Figs 1 & 2.

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Note: It is interesting also to consider the coolant temperature response in the event that the source of cooling was interrupted. i.e. discontinue feedwater in an overfeed transient. Assuming that primary and secondary sides are well coupled (a total of 19,500 ft^3 of water) and 1% of nominal power decay heat equal the heatup rate would not exceed 60°F/hr. This means that recovery of coolant temperature would be very slow i.e. overcooling transient will continue even beyond the point that its cause has been removed. Therefore the monotonic T_o slopes chosen above should cover well a broad range of actual transients.

$J = \frac{T - T_\infty}{T_0 - T_\infty}$ where: T_0 = initial temperature; T_∞ = coolant temperature at $t = 0^+$

All other symbols same as Fig. 1.

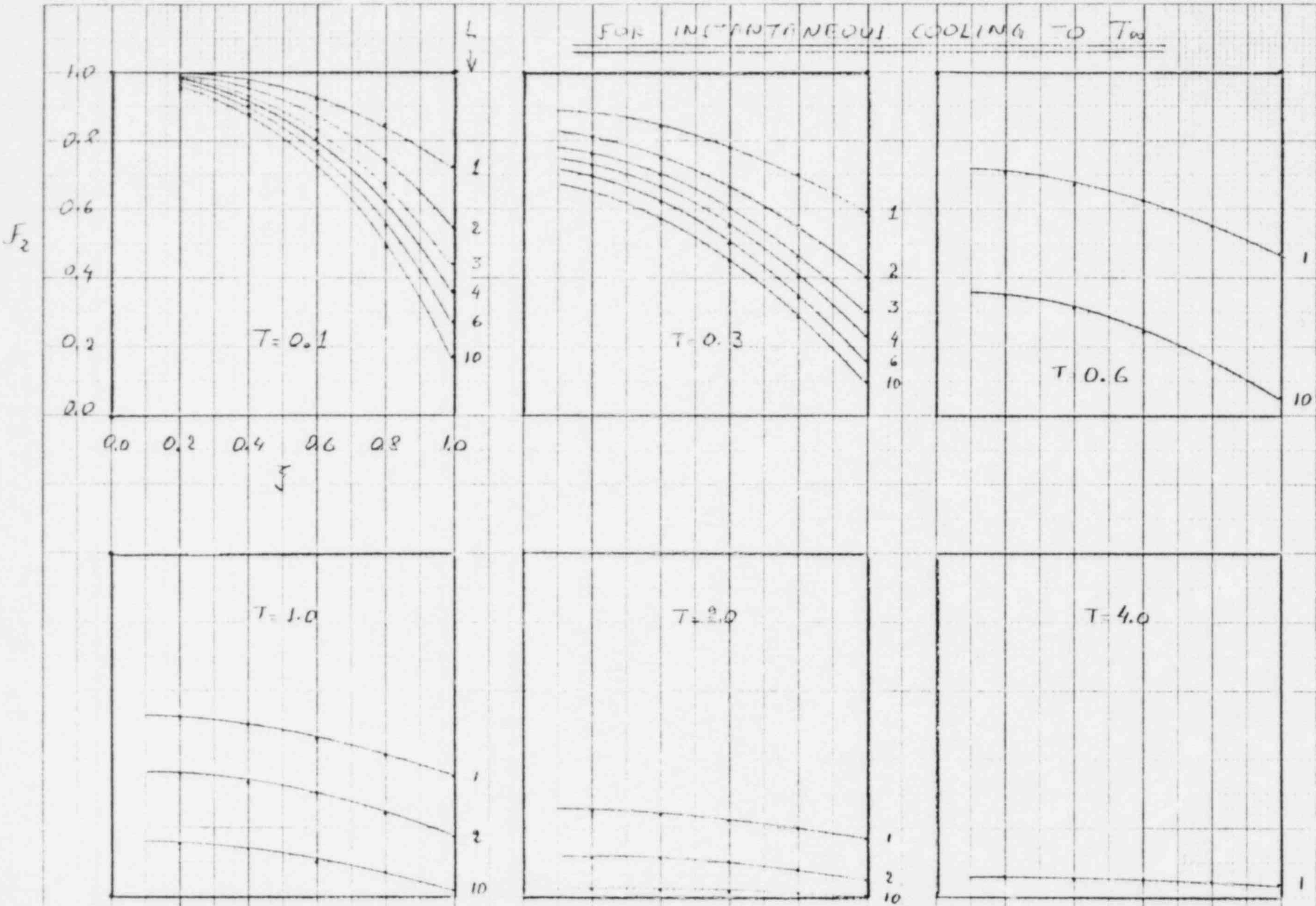

Figure 2.

$F_1 = (T_0 - T)/ct$ where: T_0 = initial temperature, t = time, C cooling rate of coolant (i.e. °F/hr)
$\xi = x/\ell$ dimensionless position, $T = \alpha t/\ell^2$ dimensionless time, $L = h\ell/k$ dimensionless

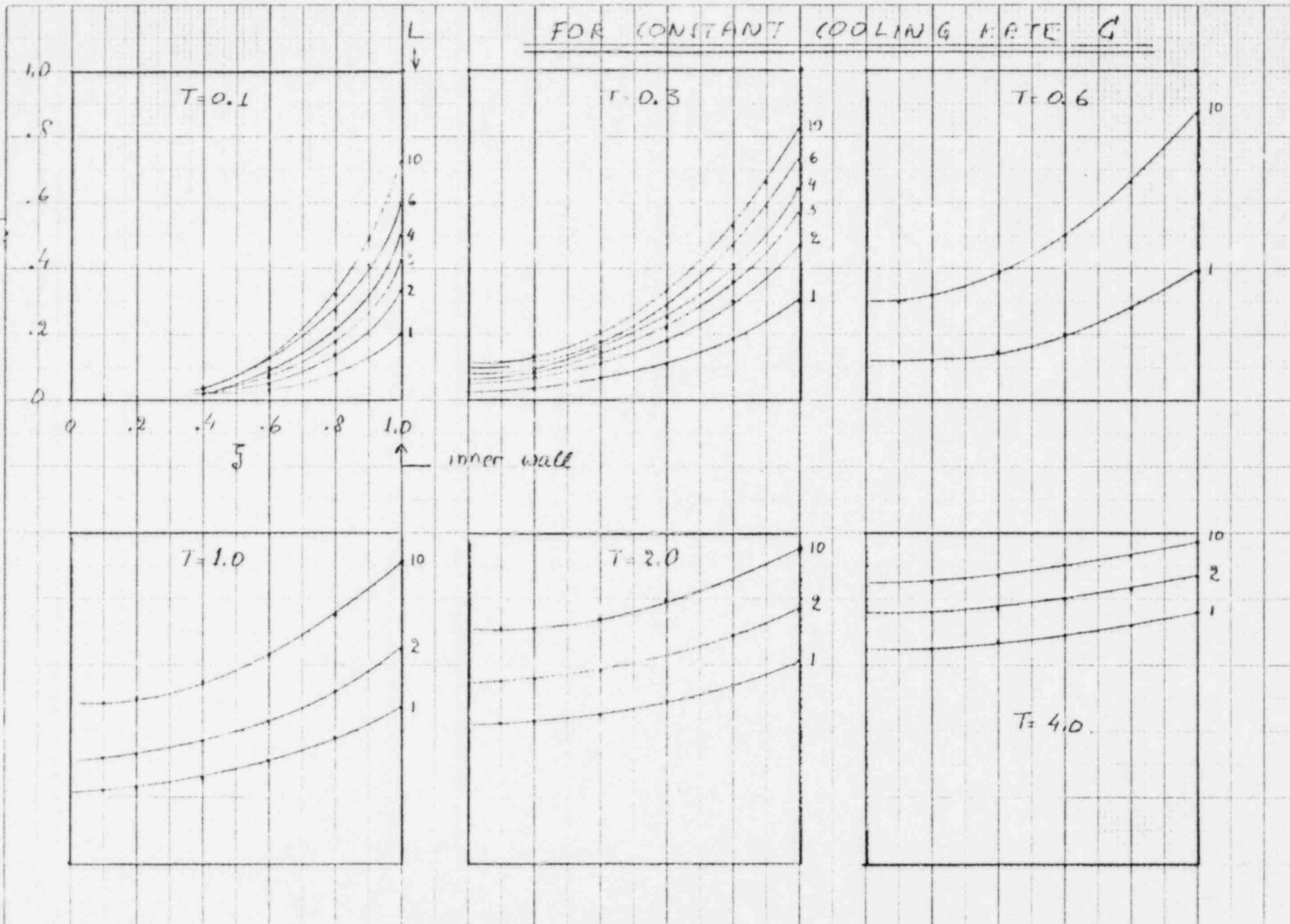

Figure 1.

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ℓ : wall thickness, α : wall thermal diffusivity, k : wall thermal conductivity, h : heat transfer coefficient.
t : time since start of cooling.

KEY VALUES $T = 1$ corresponds to $t \sim 44$ min
$L = 1$ corresponds to $h \sim 50$ Btu/hr ft² °F ~ 270 W/m²

46 1240

K+E 20 X 20 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

#1

APPENDIX XVI
CURRENT NRC STAFF CONSIDERATION OF POSSIBLE RECOMMENDATIONS FOR PTS REQUIREMENTS

Current NRC Staff Consideration of Possible Recommendations for PTS Requirements

June 3, 1982 (ACRS)

Summary of Actual Plant Events

NSSS	Plant Description	Date	Beta (1/m)	T-final (deg F)	Duration	T-init. (deg F)	Pressure (psig)	Comments
W	Robinson 2 Pre-op small SLB	04/28/72	0.097	320.	1 hr	530.	2050.	
W	Robinson 2 Stuck S.G. Valve	11/05/72	0.043	389.	2 hr	550.	1700.	P-reading off scale
B+W	Rancho Seco Loss of NNI/ICS	03/20/78	0.104	285.	1 hr	600.	2000.	
W	Robinson 2 RCP seal SBLOCA	05/01/78	0.172	310.	30 m	450.	1000.	P-reading off scale
B+W	Three Mile-2 PORV SBLOCA	03/28/79	0.098	250.	1 hr	450.	1800.	3-4 hrs in transient
B+W	Crystal River-3 Loss of NNI/ICS	02/26/80	?	250.	?	560.	2300.	T-range 100 -400 deg F
W	Ginna SGTR/Stuck PORV	01/25/82	0.127	265.	45 m	550.	1400.	Coldest T measured
CE	Ft. Calhoun			440				
CE	ANO-2			450				

#2

Foreign experience is not included in our data base. Example of experience not included:

NSSS	Plant Description	Date	Beta (1/m)	T-final (deg F)	Duration	T-init. (deg F)	Pressure (psig)	Comments
W	Borselle S.G. Blowdown	03/02/81	0.253	285.	45 m	440.	2100.	2-loop PWR European

Totals: 4 W 2 CE 3 B+W (9 Total)

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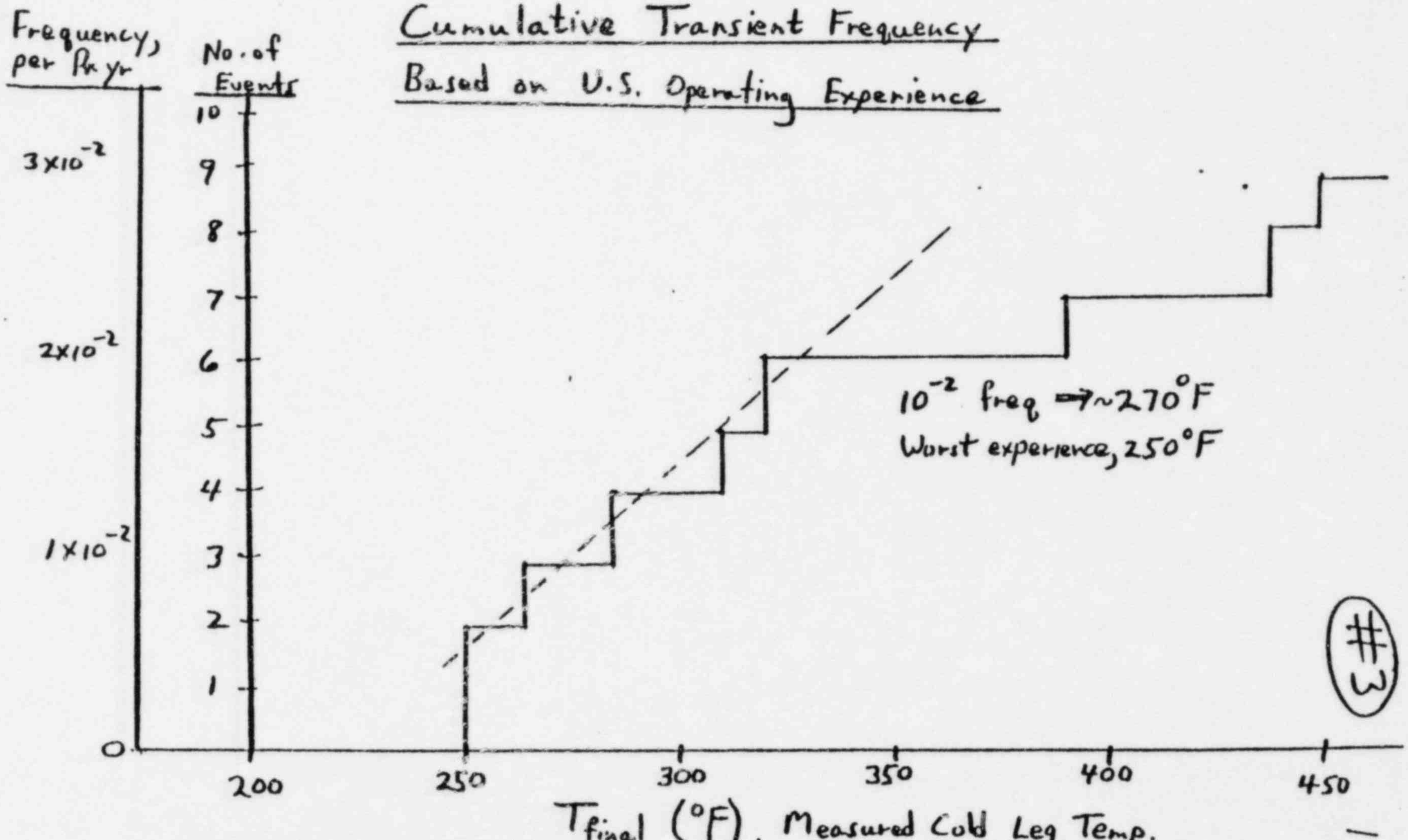
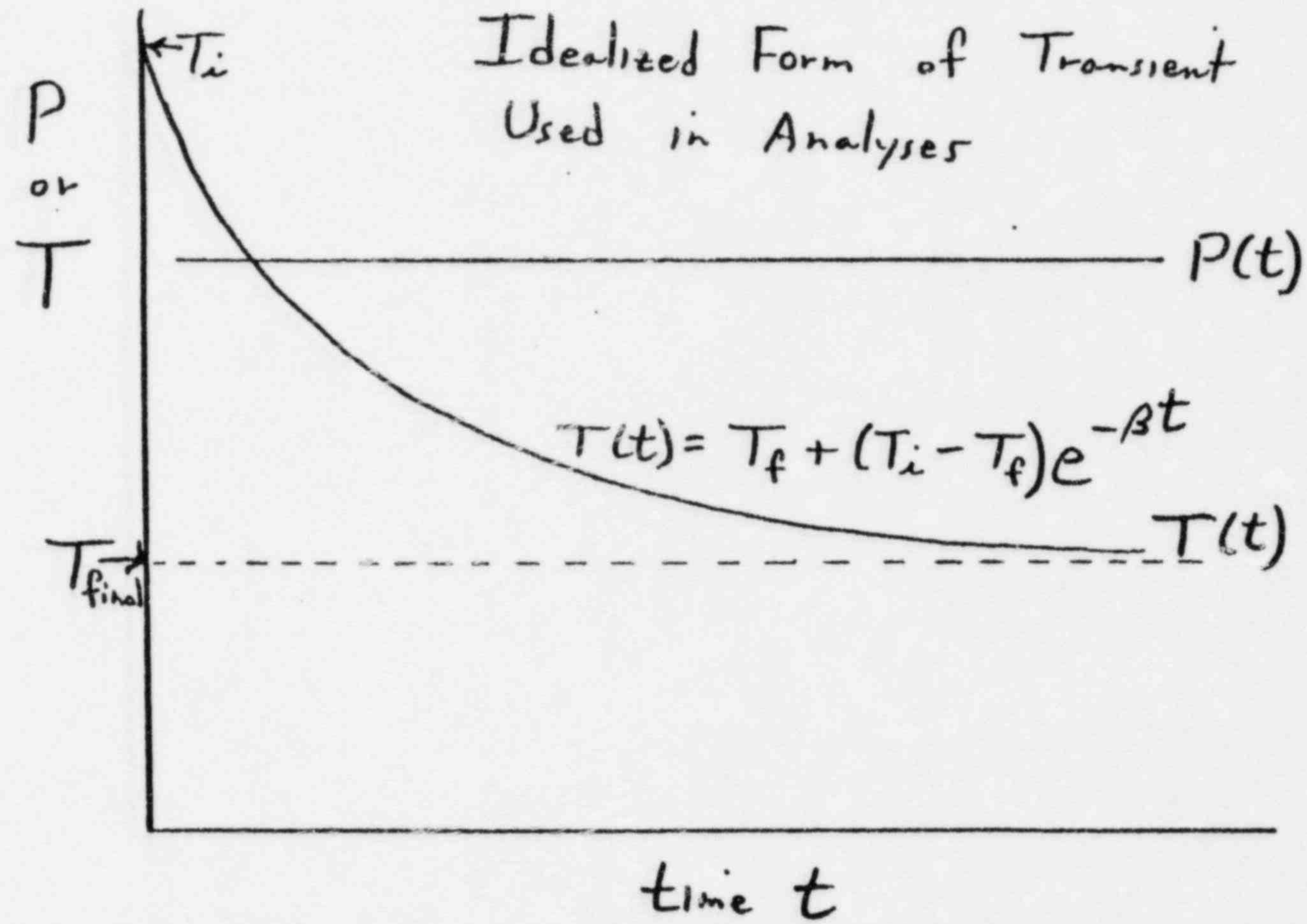
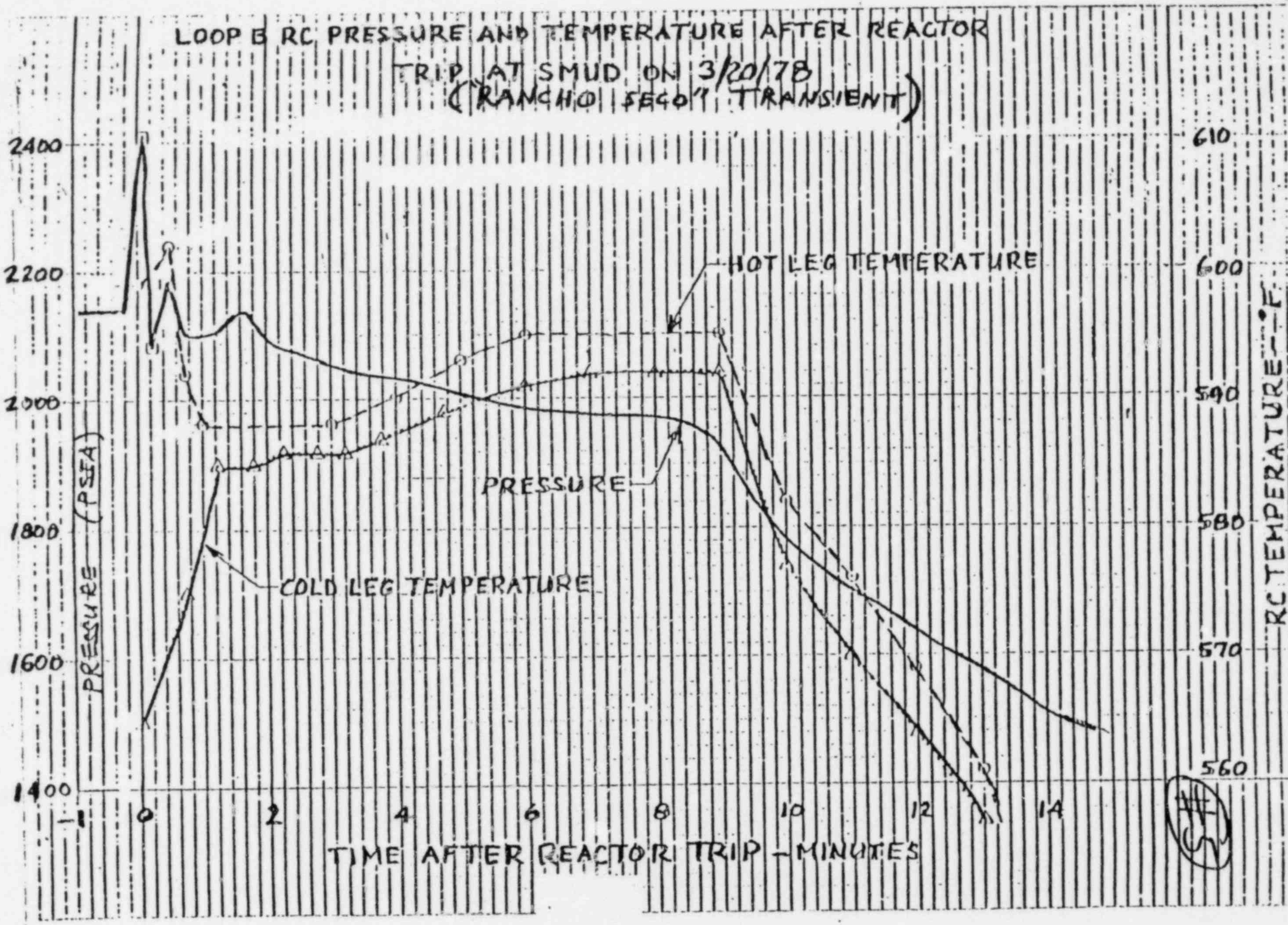
Cumulative Transient Frequency
Based on U.S. Operating Experience



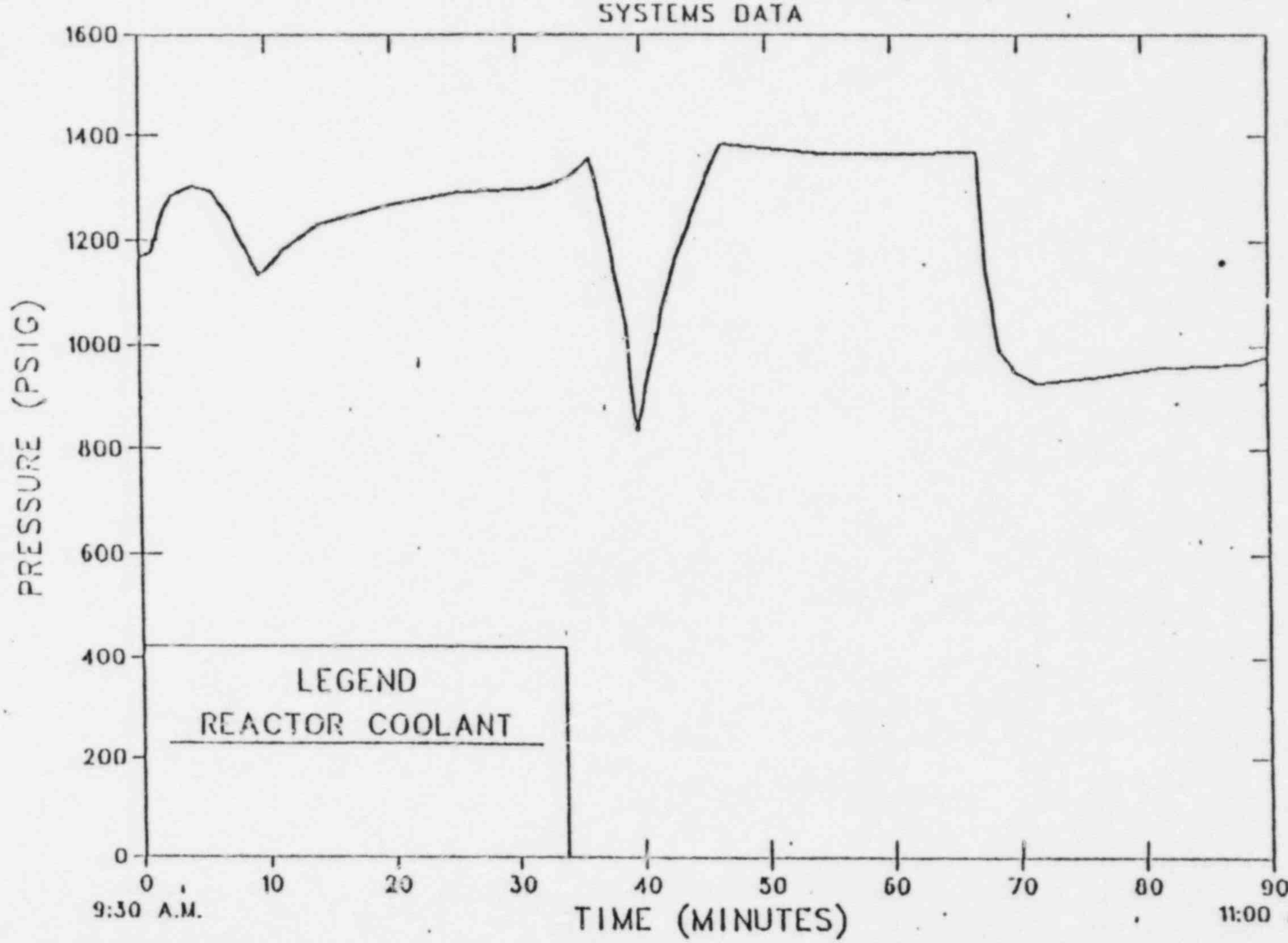
#4


Idealized Form of Transient
Used in Analyses
P or T
T_i
P(t)
$T(t) = T_f + (T_i - T_f)e^{-\beta t}$
T(t)
T_{final}
time t


LOOP B RC PRESSURE AND TEMPERATURE AFTER REACTOR
TRIP AT SMUD ON 3/20/78
("RANCHO SECO" TRANSIENT)
HOT LEG TEMPERATURE
PRESSURE
COLD LEG TEMPERATURE
PRESSURE (PSIA)
RC TEMPERATURE - °F
TIME AFTER REACTOR TRIP - MINUTES
2400
2200
2000
1800
1600
1400
610
600
590
580
570
560
-1
0
2
4
6
8
10
12
14
#5

R.E. GINNA SGTR EVENT 01/25/82

SYSTEMS DATA



EDT/RSB 05/03/82

#6

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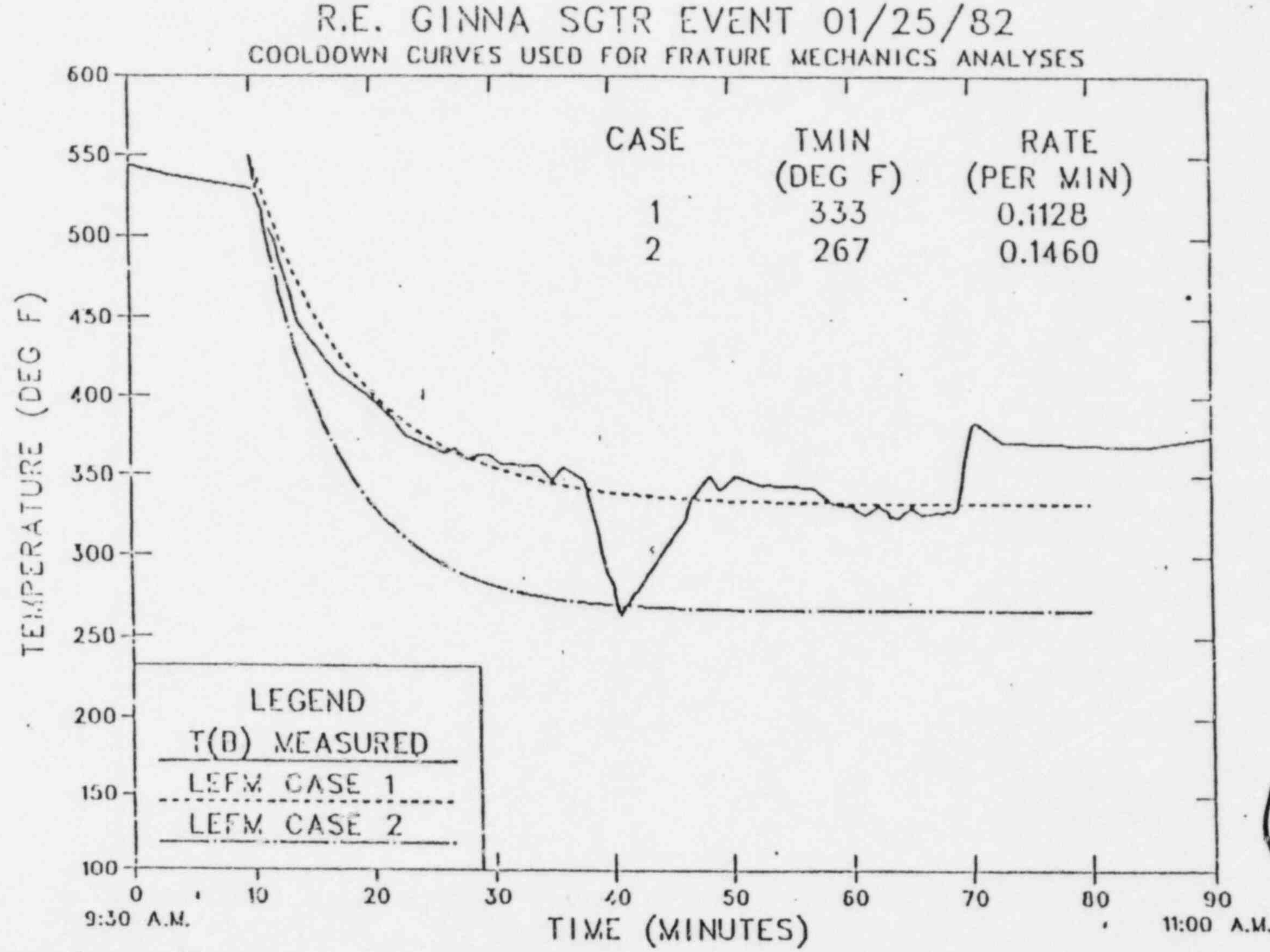

EDT/RSB 05/03/82

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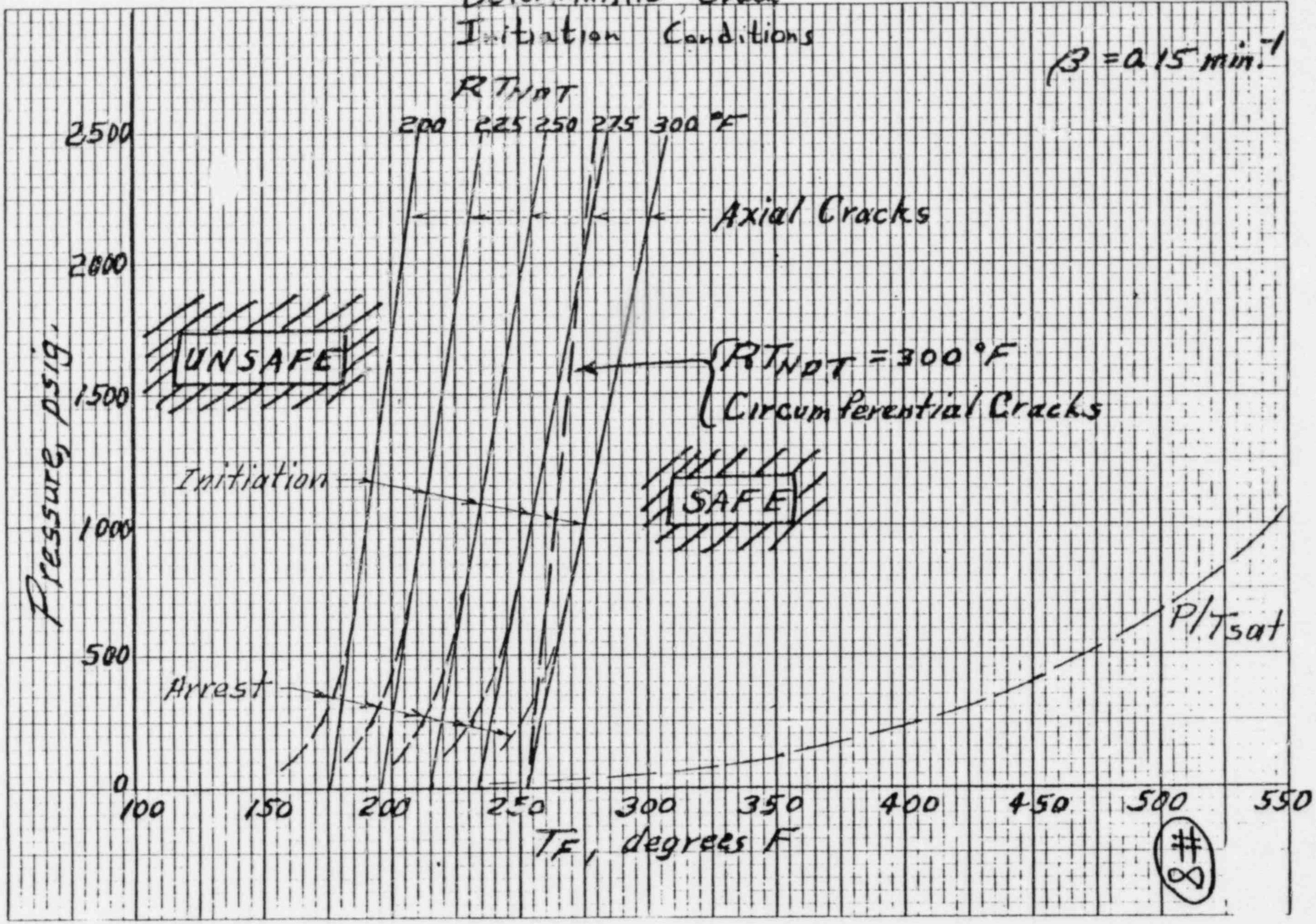
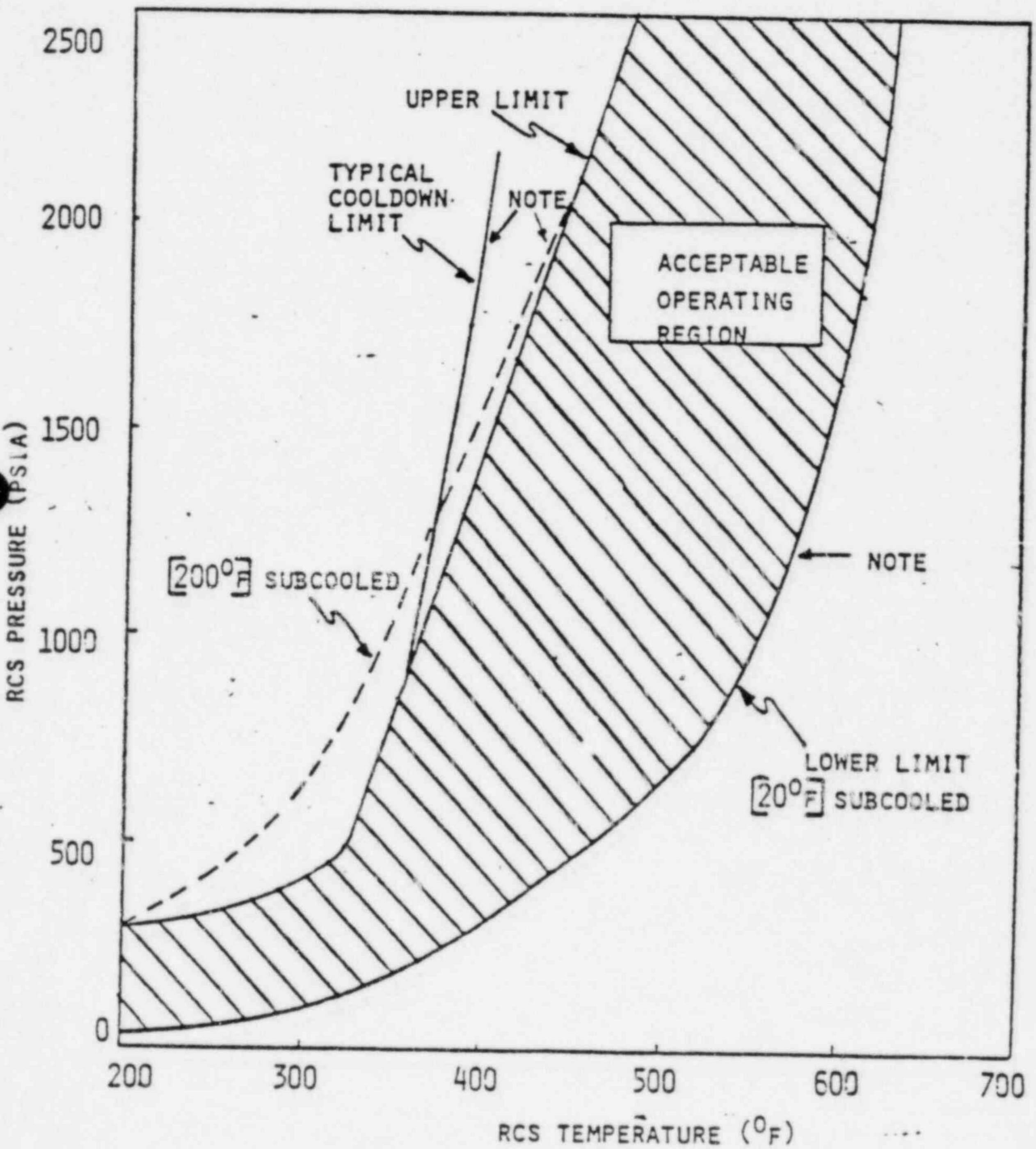
#7

Deterministic Crack Initiation Conditions



POST ACCIDENT RCS PT LIMIT CURVES

#9



(Provided to NRC by CEOG at 3/3/82 meeting)

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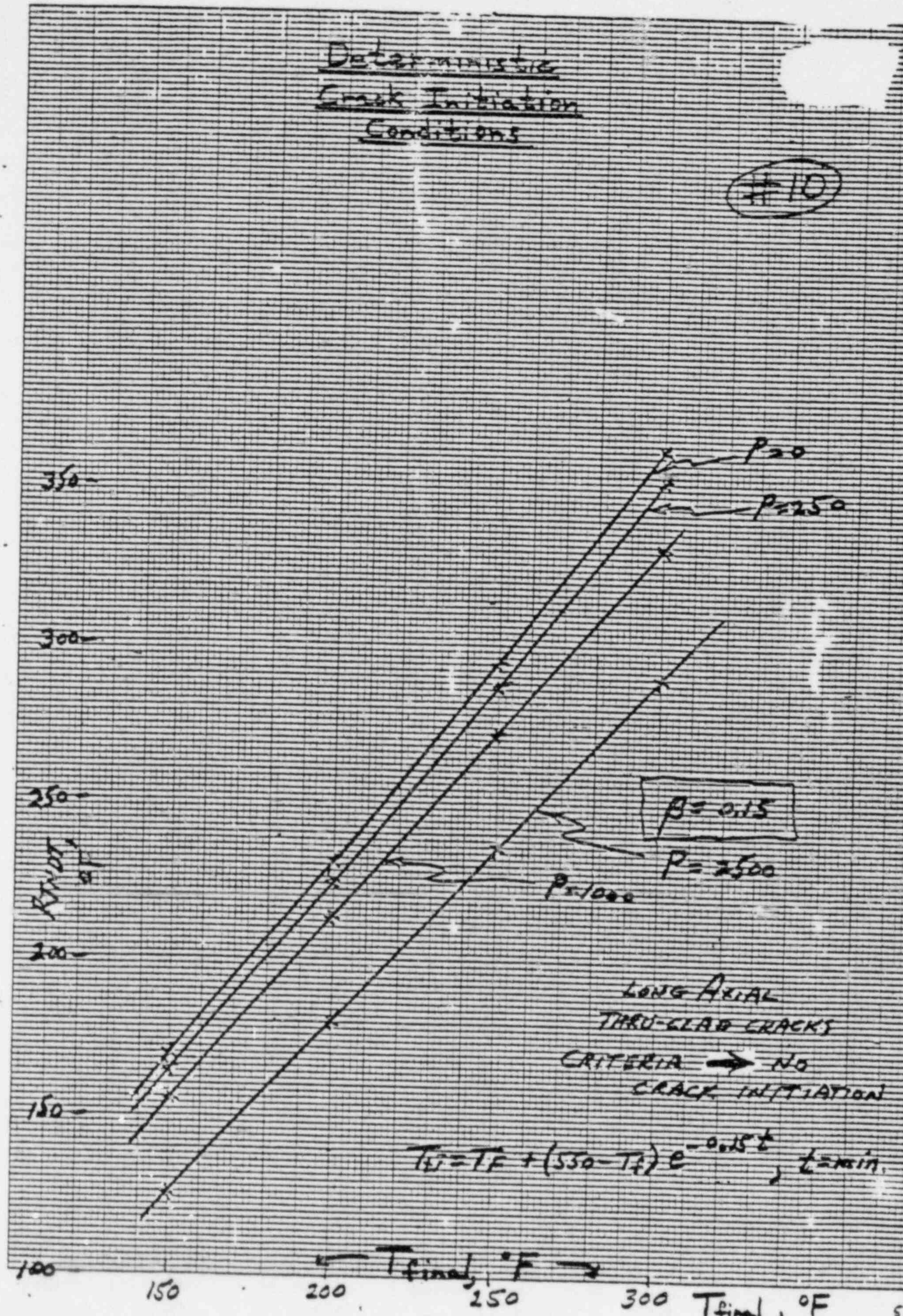


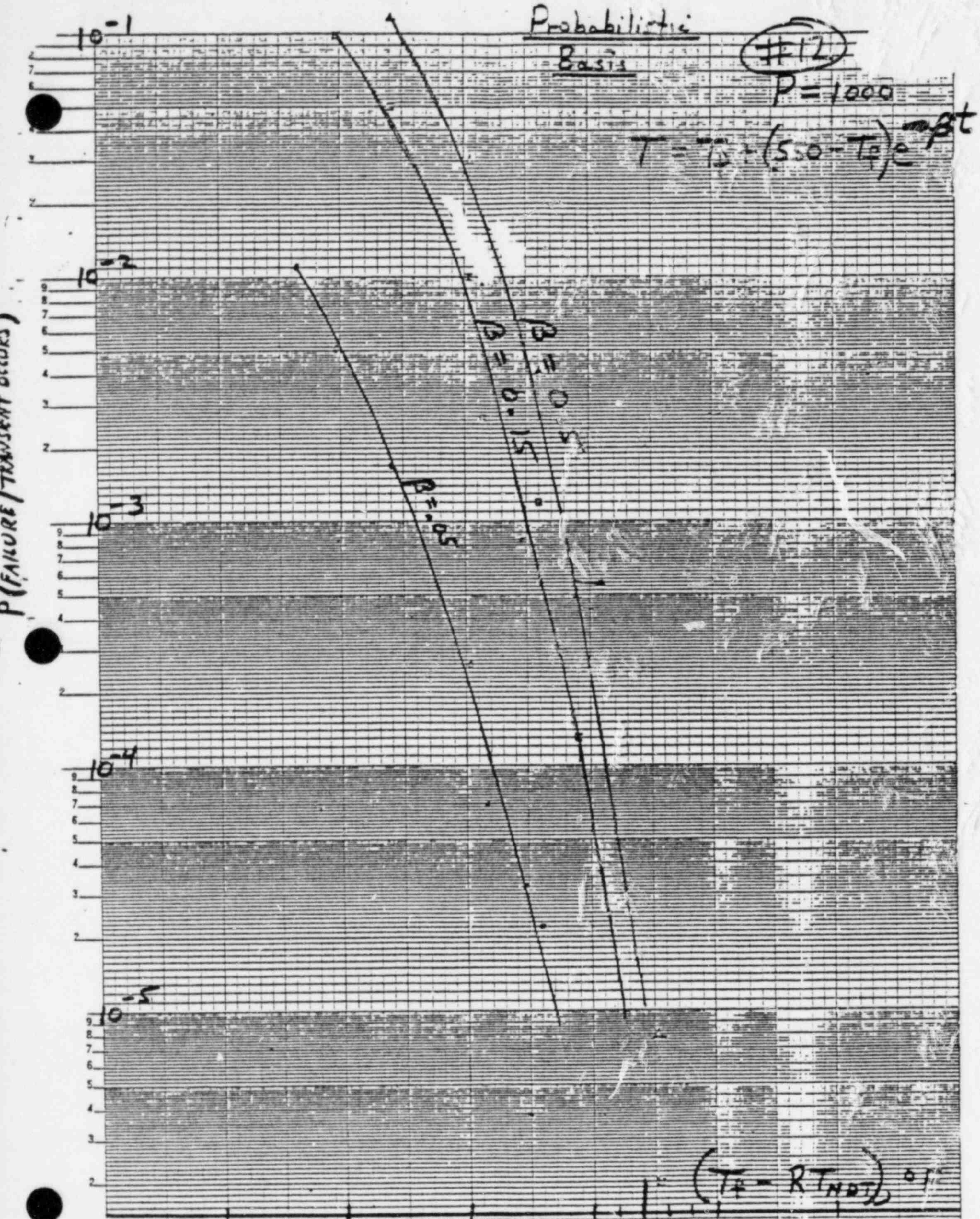
Deterministic
Crack Initiation
Conditions
#10
350
300
250
200
150
100
P=0
P=250
P=1000
P=2500
β= 0.15
LONG AXIAL
THRU-CLAD CRACKS
CRITERIA → NO
CRACK INITIATION
$T_{f1} = T_F + (550 - T_f) e^{-0.15t}$, t=min.
150
200
250
300
← T_final, °F →
T_final, °F

#11

Deterministic Method for Crack Initiation

(based on transient with $T_{final} = 250°F$ and $\beta = 0.15\ min^{-1}$)

RT_{NDT}, °F	Pressure, psig
230	> 2500
275	< 1000
295	< 300



Probabilistic Basis
#12
P=1000
T = Tf + (550 - Tf) e^-βt
P (FAILURE / TRANSIENT OCCURS)
10^-1
10^-2
10^-3
10^-4
10^-5
β = 0.5
β = 0.15
β = .05
(Tf - RTNDT)o °F
-170
-120
-70
-20
0
+30
+80

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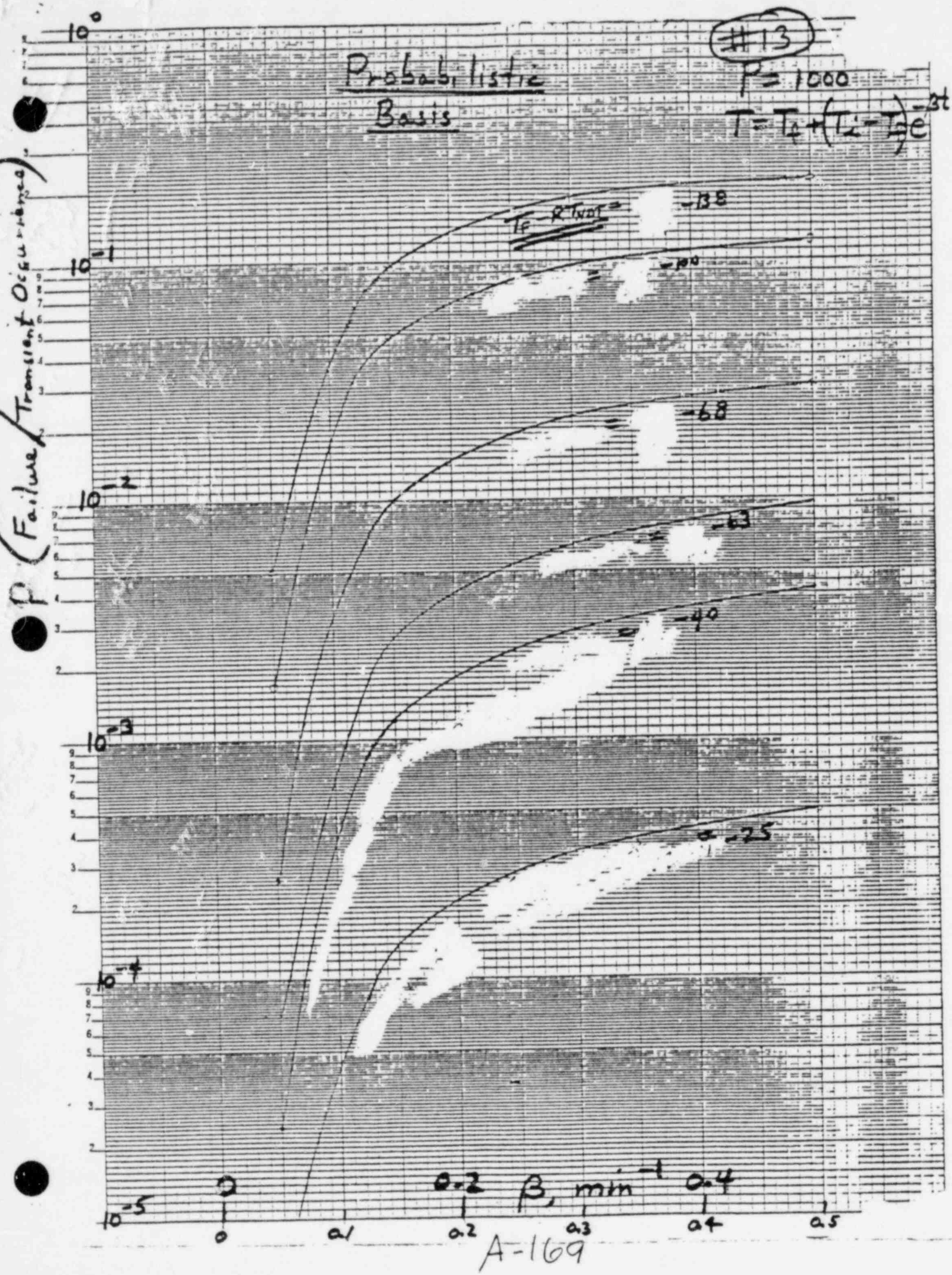
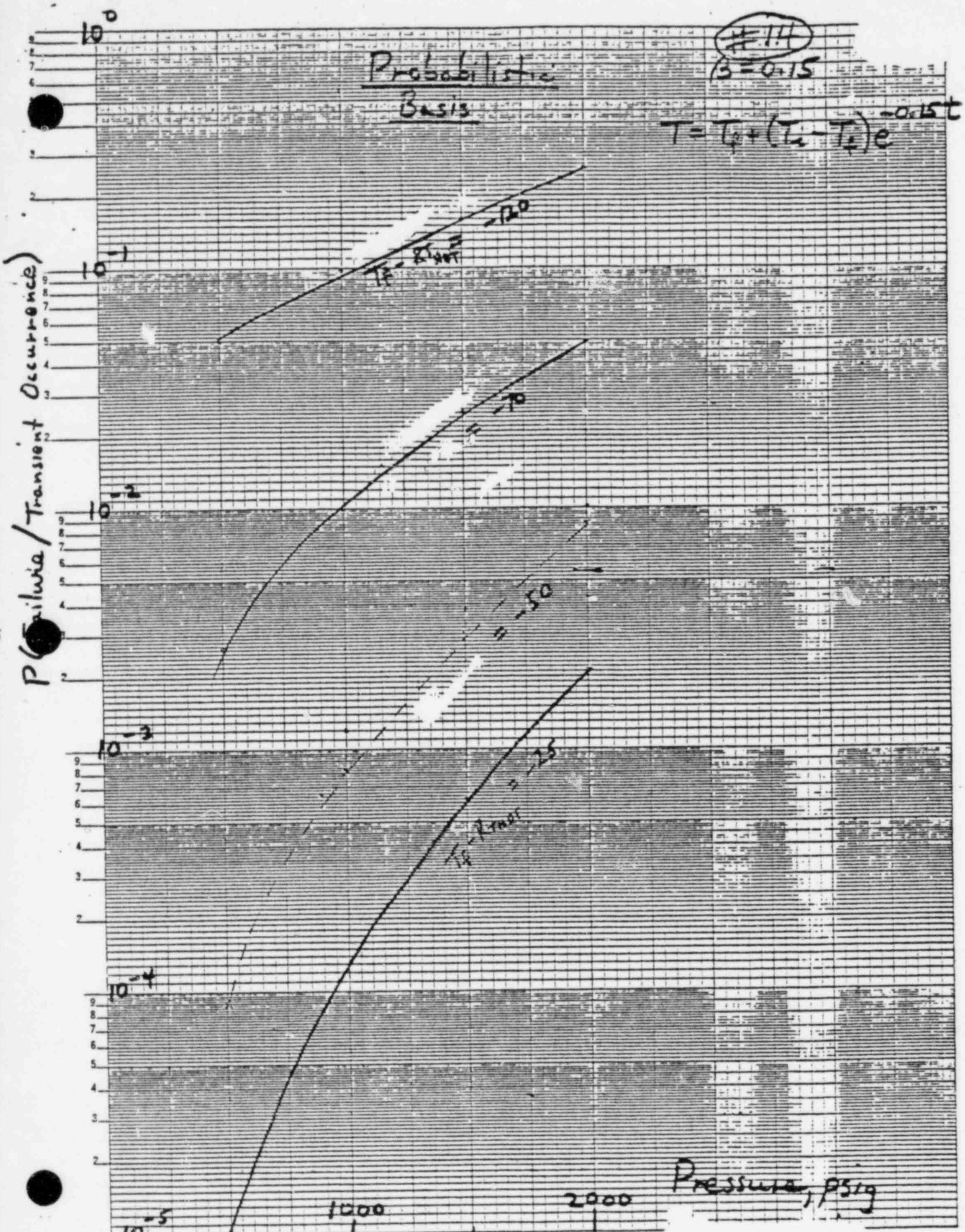


#13
Probabilistic
Basis
P = 1000
$T = T_f + (T_i - T_f)e^{-\beta t}$
$T_f - RT_{NDT} =$
-138
-100
-68
-63
-40
-25
P (Failure / Transient Occurrences)
β, min^{-1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
0
0.1
0.2
0.3
0.4
0.5



#14
Probabilistic
Basis
$\beta = 0.15$
$T = T_f + (T_i - T_f)e^{-0.15t}$
P(Failure/Transient Occurrence)
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
$T_f - RT_{NDT} = -120$
= -70
= -50
$T_f - RT_{NDT} = -25$
Pressure, psig
1000
2000
0
500
1000
1500
2000

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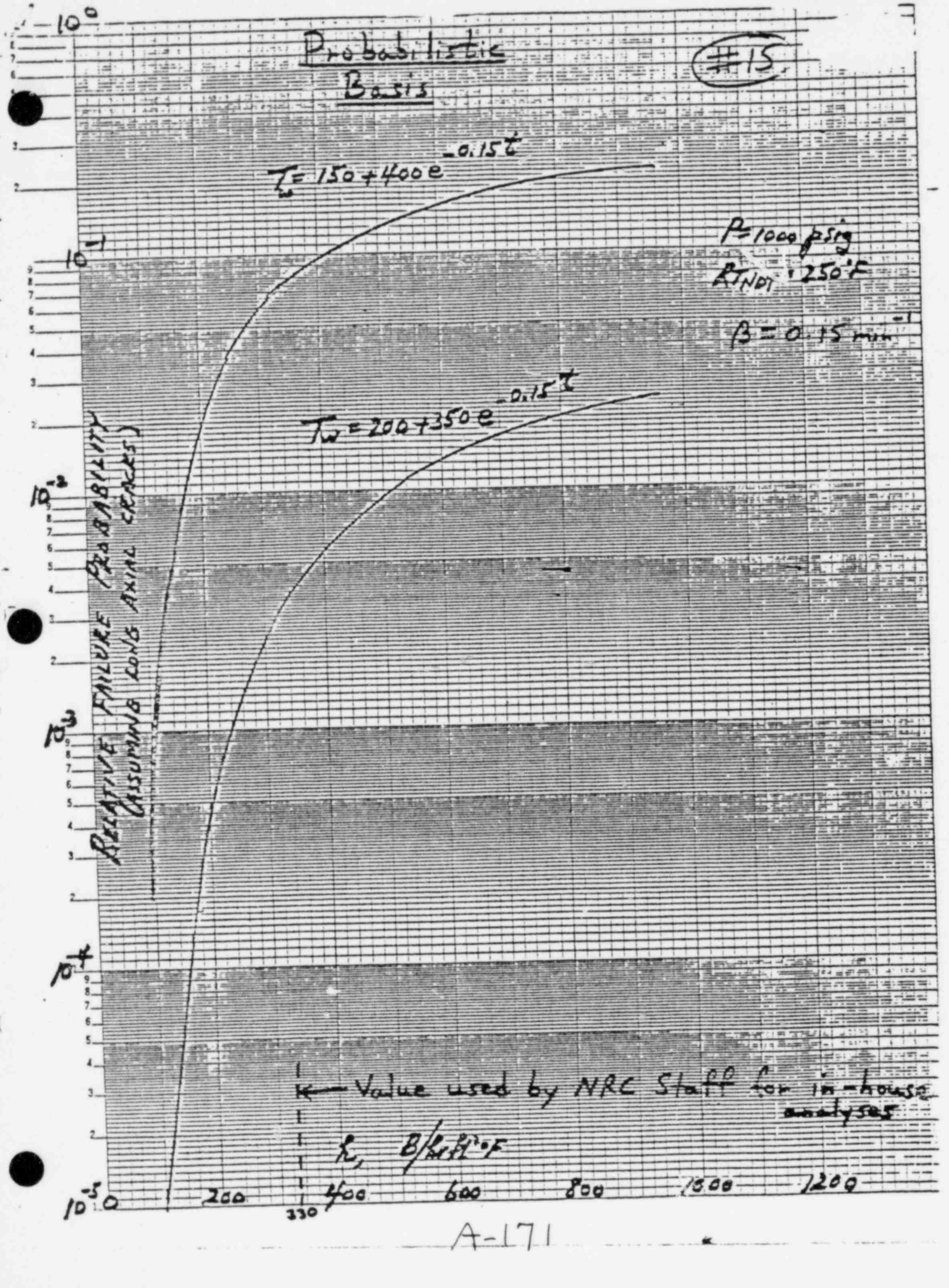


Probabilistic Basis
#15
$T_w = 150 + 400e^{-0.15t}$
$P = 1000$ psig
$RT_{NDT} = 250°F$
$\beta = 0.15\ min^{-1}$
$T_w = 200 + 350e^{-0.15t}$
RELATIVE FAILURE PROBABILITY (ASSUMING LONG AXIAL CRACKS)
← Value used by NRC Staff for in-house analyses
h, B/hr ft² °F
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
0
200
330
400
600
800
1000
1200

#16

Deterministic Method for Crack Initiation

(based on transient with $T_{final} = 250°F$ and $\beta = 0.15 \ min^{-1}$)

RT_{NDT}, °F	Pressure, psig	*P(failure/trans)*
230	> 2500	10^{-5} to 10^{-4}
275	< 1000	10^{-4} to 10^{-3}
295	< 300	10^{-4} to 10^{-3}

* Probability of vessel failure <u>given</u> occurrence of the transient described and with vessel RT_{NDT} shown

#17

RT_{NDT} Limits Under Consideration

- Using deterministic curves, with $\beta = 0.15$, $T_f = 250$:

  for full pressure capability, then

  RT_{NDT} (Actual) = 230°F for longitudinal welds

  RT_{NDT} (Actual) = 255°F for circumferential welds

- Using probabilistic curves for above β, T_f, and RT_{NDT}:

  P (failure / transient occurrence) = $\sim 10^{-5}$

  P (transient occurrence) = $\sim 10^{-2}$

  ∴ P (failure) (For the given transient) = $\sim 10^{-7}$

  Note, however, that the total failure probability would be higher to account for all transients including those with $T_f < 250°F$ and/or $\beta > 0.15 \text{ min}^{-1}$

#18

Determination of RT_{NDT}

- Limit under consideration was derived assuming actual RT_{NDT} is known

- How should RT_{NDT} be determined for regulation?
  - "Best Estimate" value
  - Conservative value

(#19)

Unquantified Uncertainties

- Local temperature in downcomer vs. measured temp.
- Changes being made:
  - better instrumentation (bus seperation in B&W plnt)
  - interim training and procedure modifications
- Consequences of vessel failure
- Failure criteria
- Heat transfer coefficient
- Cladding effects (increases K_{IC} but may inhibit growth)
- Use of K_{IC} 200 $ksi\sqrt{in}$ for crack arrest
- Best estimate vs. lower bound K_{IC}
- Method used to determine RT_{NDT}
- More severe transients may occur
- We used normal distributions with "tails" that may not exist in reality
- Assumed crack shape, size, and probability

#20

Possible Actions for Plants that Do Not Currently Meet Criteria

- Justification of lower RT_{NDT}, for example through lower initial RT_{NDT}

- Operations improvements
  - licensee would have to prepare and justify
  - licensees would have to show that no other safety problems were created

- Instrumentation improvements for use by operator

- Credit for pressure limiting control system
  - licensee would have to design reliable system
  - licensee would have to determine and justify that system would not create safety problems

- Full volumetric NDE of vessel

- Immediate shutdown for annealing

#21

Flux Reduction Considerations

- Are we willing to accept EOL RT_{NDT} equal to limit for many older operating plants?
- Does this mean limiting RT_{NDT} should be lowered?
- Should we require flux reduction methods to slow down toughness degradation?
- We propose that these questions should be answered when the safety goal is finalized.

#22

Defense in Depth

Should we require:

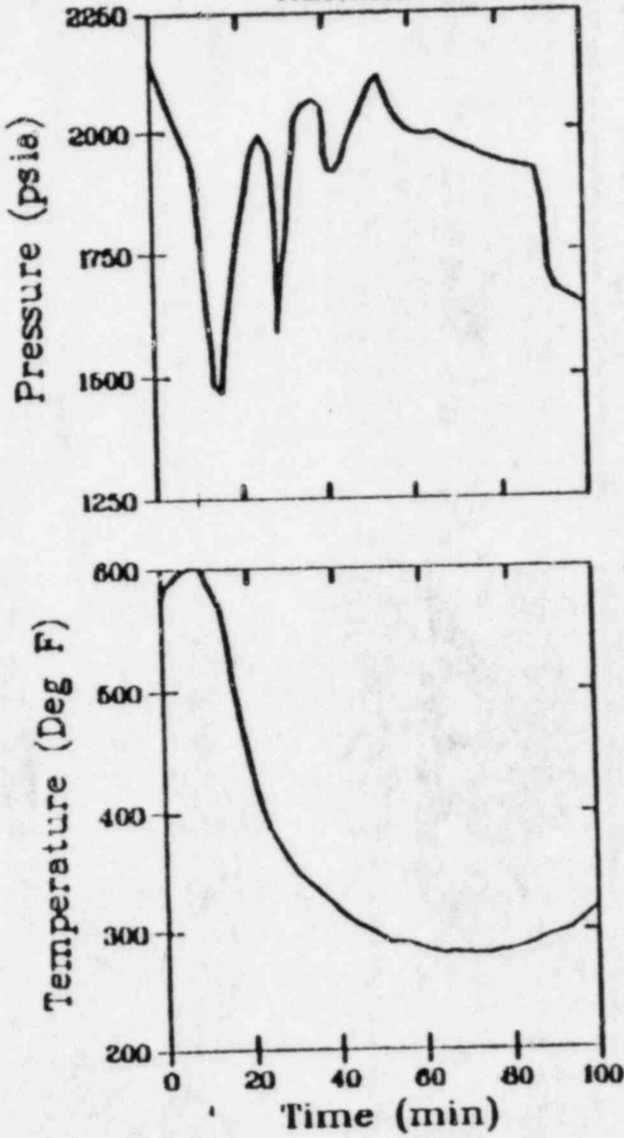
- upgraded operator procedures and training at all plants

- improved instrumentation at some RT_{NDT} threshold

- hardware improvements at some RT_{NDT} threshold:
  Auto Pressure Control
  Warmer Feedwater
  Warmer ECCS Water

- required flux reduction at some RT_{NDT} threshold

Rancho Seco 1 03/20/78 Overcooling Transient



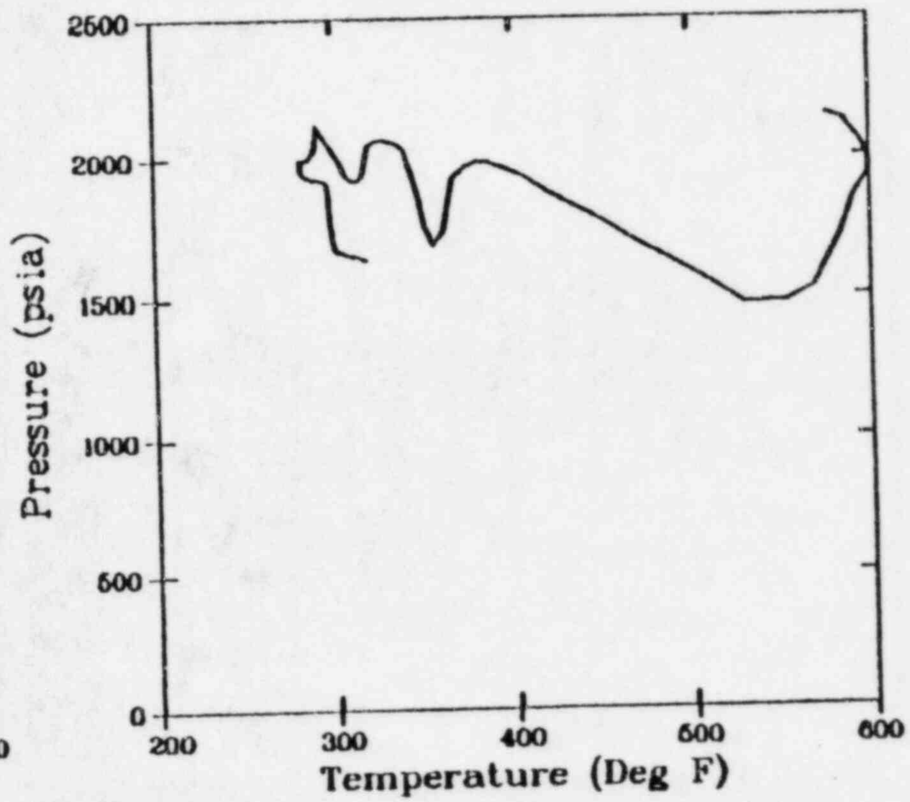
NOTES:

B&W NSSS

Loss of NNI/ICS Indicators

SG refill with MFW

Pressure Controlled with SI



OBJECTIVES OF WOG PROGRAMS

- DEMONSTRATE THAT THERE ARE NO NEAR-TERM SAFETY ISSUES IN W OPERATING PLANTS.
- GENERICALLY DEVELOP METHODOLOGIES AND TECHNIQUES FOR ADDRESSING PTS

  E.G. - ANALYTICAL IMPROVEMENTS FOR FUTURE PLANT - SPECIFIC ANALYSES
  - TEMPERATURE - LIMIT TECHNIQUE FOR OPERATOR ASSESSMENT
  - OPERATOR TRAINING
  - TECHNIQUES FOR EVALUATING REMEDIAL ACTIONS
- PROVIDE INPUT TO ECONOMIC DELIBERATIONS BY UTILITY MEMBERS.

ELEMENTS OF WOG PROGRAM

- WCAP-10019 - SAFETY EVALUATION
  "GENERIC" LEFM
  USES DESIGN BASES TYPE TRANSIENTS

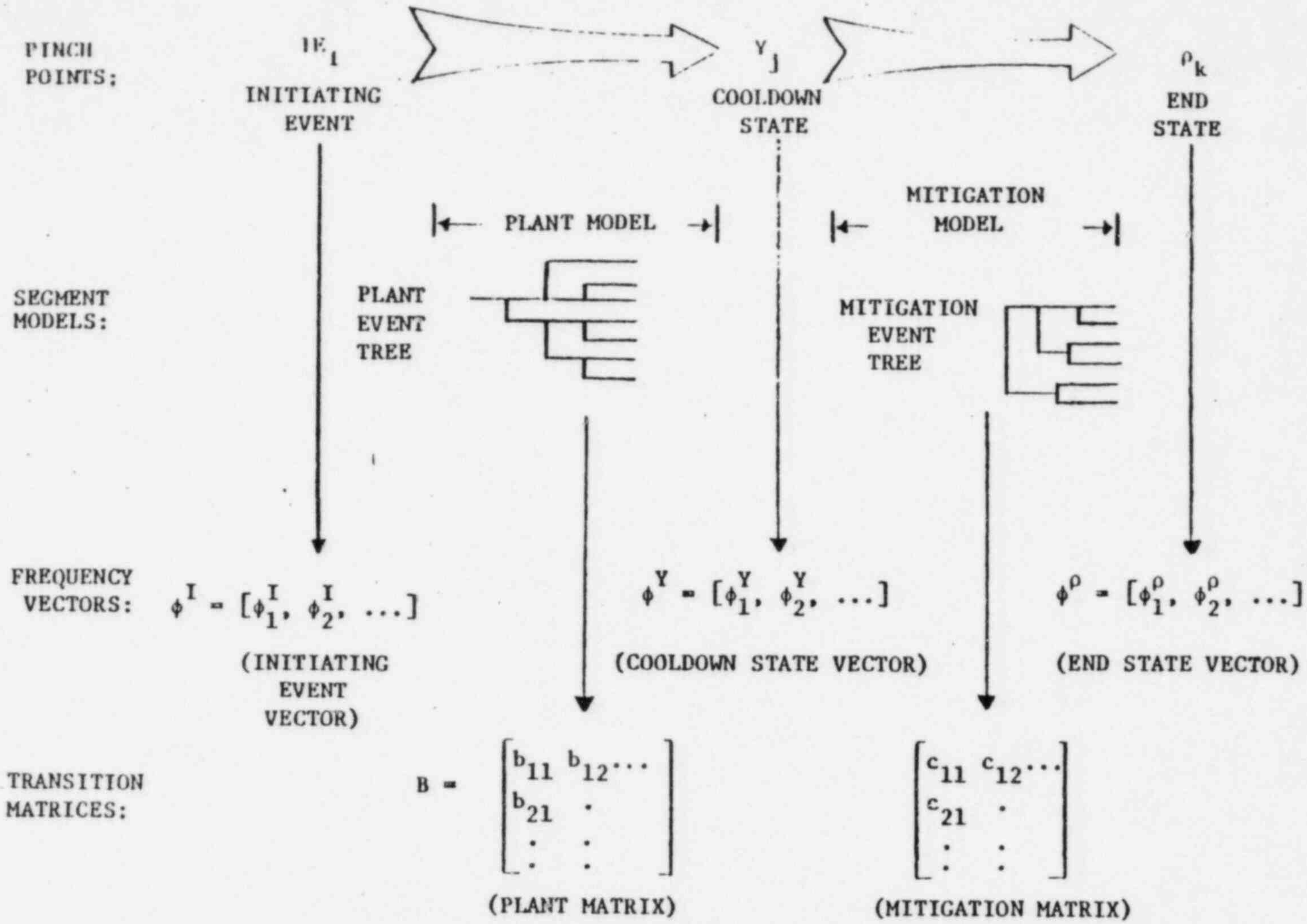
- WOG 5/28 REPORT - SUPPORTS WCAP-10019
  UTILIZES EVENT SEQUENCE ANALYSIS
  (PRA) ESTABLISHES TEMPERATURE
  CRITERIA FOR POTENTIAL INITIATION.

  CONCLUSION: TRANSIENTS IN WCAP-10019
  REPRESENT LIMITING CLASSES
  OR COOLDOWN TRANSIENTS.

- WOG PROCEDURES REVIEW AND MODIFICATION

- CLAD EFFECTS PROGRAM

- SMALL STEAMLINE BREAK RE-ANALYSIS



ASSEMBLY PROCESS:

$$\phi^Y = \phi^I B$$

$$\phi^\rho = \phi^Y C = \phi^I BC$$

OVERVIEW OF THE ASSEMBLY PROCESS, SHOWING RELATIONSHIP OF PINCH POINTS, FREQUENCY VECTORS, EVENT TREES, AND TRANSITION MATRICES

Figure II-1

FIGURE II-5

PRESSURE-TEMPERATURE LIMITS FOR SORTING PROBABILISTIC TRANSIENTS FOR REPRESENTATIVE "LEAD" VESSELS AT 5 EFPY FROM PRESENT

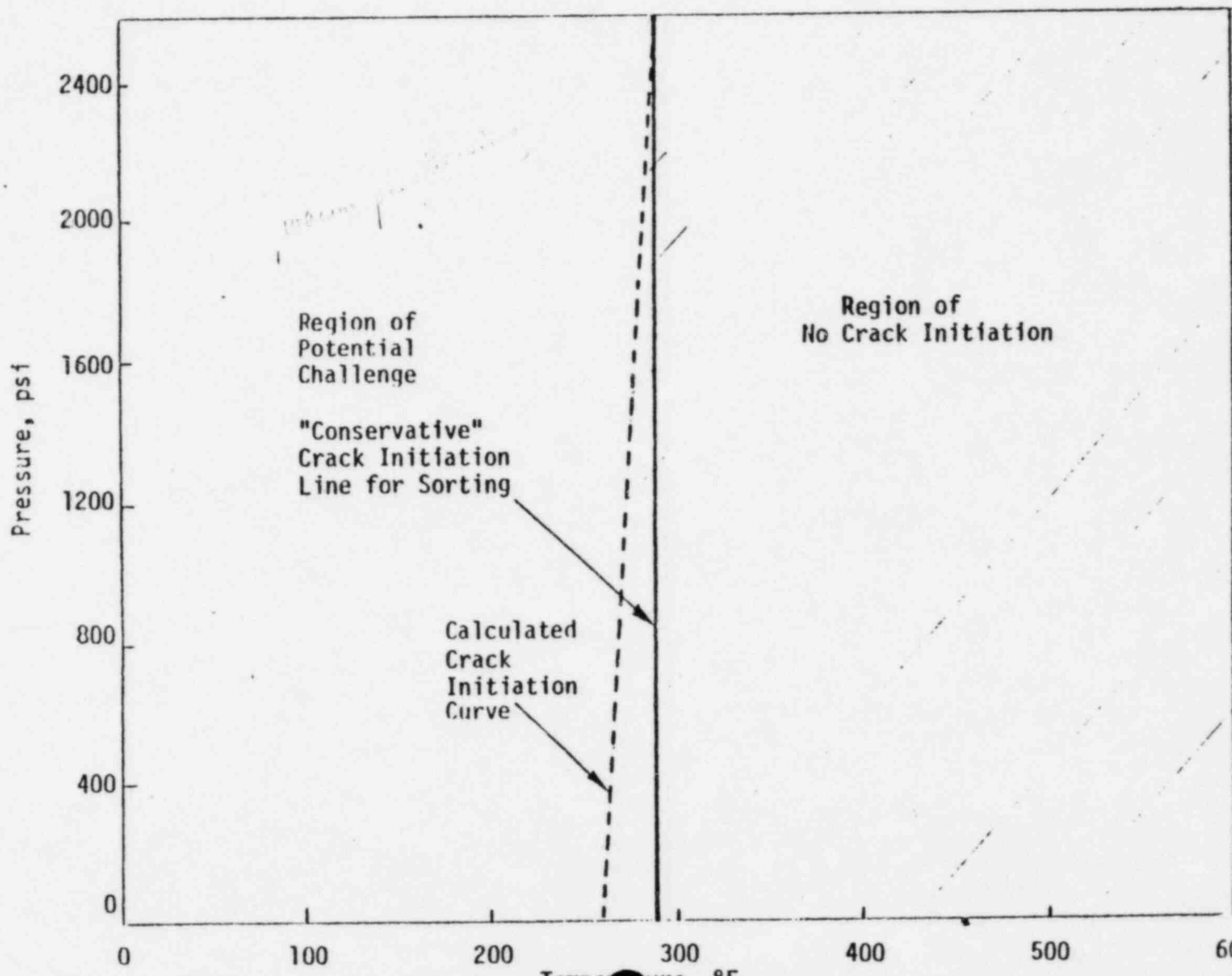


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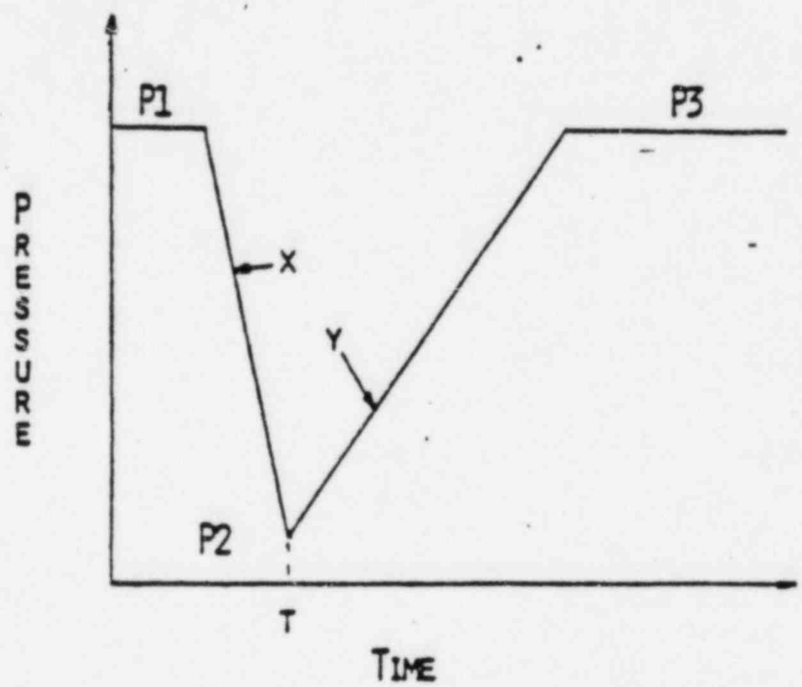
TABLE II-2

FREQUENCY OF TRANSIENTS WHICH POTENTIALLY INITIATE
A CRACK BY CLASS OF COOLDOWN
TRANSIENT (IN OCCURRENCES PER REACTOR YEAR)

Class of Cooldown Transient	Sort A 210°F Perfect Mixing	Sort B 290°F Perfect Mixing	Sort C 290°F No Mixing Under Stagnant Loop Conditions	Sort D 290°F No Manual RCP Trip During Non-LOCA Events
Secondary Depressurization	6.0×10^{-7}	2.8×10^{-4}	2.8×10^{-4}	9.9×10^{-5}
Excessive Feedwater	$<1.0 \times 10^{-7}$	$<1.0 \times 10^{-7}$	$<1.0 \times 10^{-7}$	$<1.0 \times 10^{-7}$
Loss of Primary Coolant Accident	2.1×10^{-3}	3.1×10^{-3}	3.1×10^{-3}	3.1×10^{-3}
Steam Generator Tube Rupture	9.8×10^{-5}	3.7×10^{-4}	2.0×10^{-2}	3.9×10^{-4}

Figure III-3

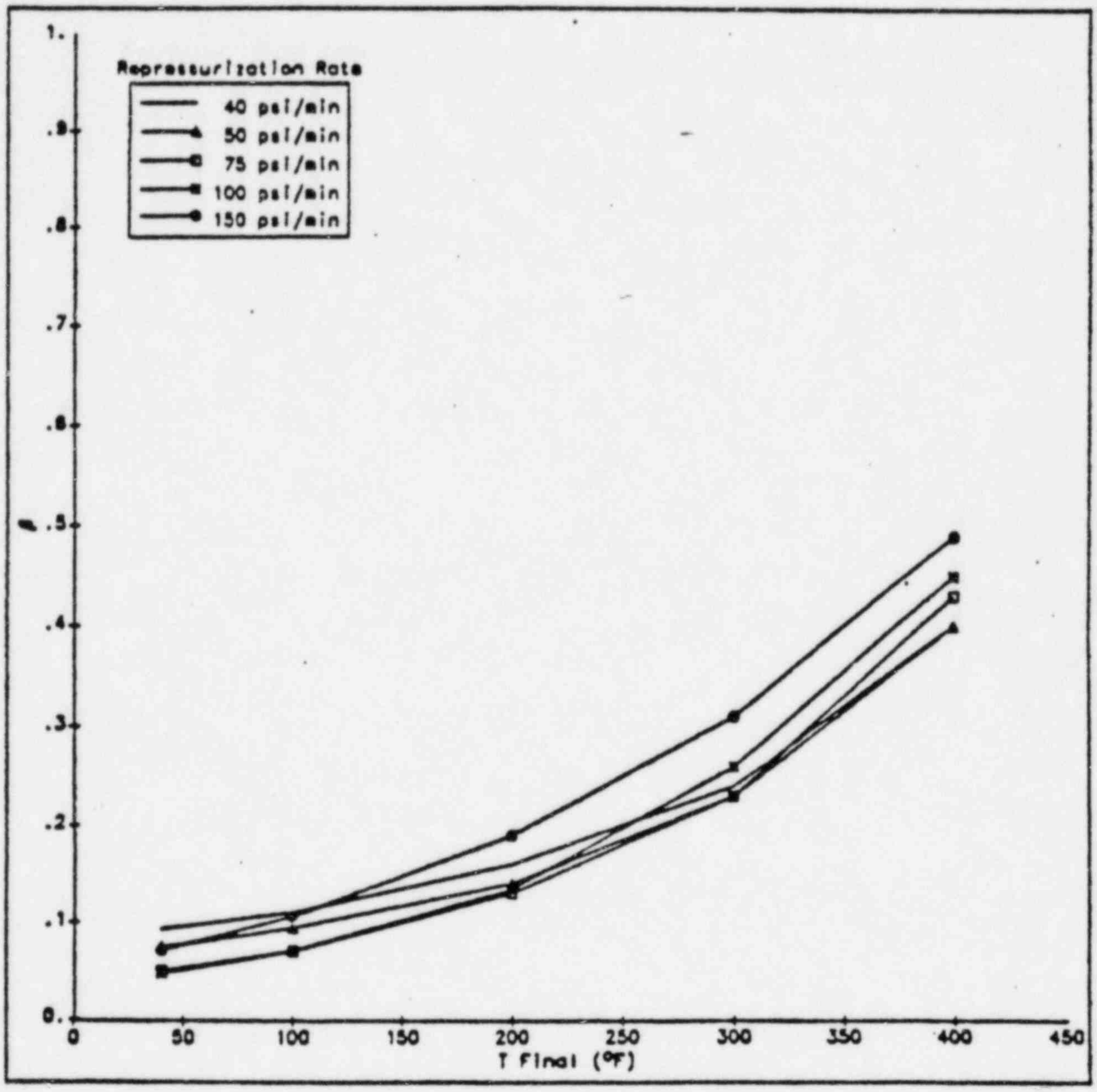
GENERAL PRESSURE PROFILE AND PRESSURE TRANSIENTS



TRANSIENT	P1	X (PSI/MIN)	T (SEC.)	P2	Y (PSI/MIN)	P3
A	2575	5000	500	15	200	VARIED
B	2575	5000	2000	15	200	VARIED
C	2575	5000	500	1000	200	VARIED
D	2575	5000	2000	1000	200	VARIED
E	2575	50	3400	15	200	VARIED
F	15	-	500	15	200	VARIED
G	2575	5000	500	15	VARIED	2575

Figure III-8

RESULTS FROM EVALUATING PRESSURE TRANSIENT G FOR $\beta > 0$



CONCLUSIONS TO DATE

- THE DESIGN BASES ANALYSES PRESENTED IN WCAP-10019 ARE REPRESENTATIVE OF LIMITING TRANSIENTS WITHIN THE CLASSES OF COOLDOWN TRANSIENTS WHICH CAN CHALLENGE VESSEL INTEGRITY. THE BASIC CONCLUSION OF WCAP-10019 HAS BEEN REAFFIRMED -- THERE ARE NO NEAR TERM SAFETY CONCERNS ON OPERATING WESTINGHOUSE PWR'S DUE TO PTS.

- WHILE NOT QUANTIFIED IN THIS REPORT, BASED UPON THE LIKELIHOOD OF AN INITIATING EVENT BEING OF THE ORDER OF 10^{-3}, THE CONTRIBUTION TO THE OVERALL RISK TO PUBLIC HEALTH AND SAFETY FROM PTS IS ACCEPTABLY SMALL. HOWEVER, UTILITY INTEREST IN RESOLUTION OF THE ISSUE REMAINS HIGH DUE TO THE ECONOMIC IMPLICATIONS OF A PTS EVENT IN TERMS OF OUTAGE TIME, PLANT AVAILABILITY, ETC.

- AN APPROPRIATE PROBABILISTIC METHODOLOGY HAS BEEN DEVELOPED AND PRESENTED IN THIS REPORT WHICH CAN BE USED TO EVALUATE THE IMPACT ON PTS OF POTENTIAL MODIFICATIONS IN A SYSTEMATIC MANNER.

- UTILIZATION OF WARM PRESTRESSING HAS BEEN SHOWN TO BE APPLICABLE FOR THE TRANSIENTS ANALYZED IN WCAP-10019. FURTHER, THE BASIS FOR APPLYING WARM PRESTRESSING TO TRANSIENTS WITH FLUCTUATIONS IN PRESSURE AND/OR TEMPERATURE HAS BEEN PROVIDED.

CONCLUSIONS TO DATE (CONTINUED)

- THE ANALYTICAL MODELS, CODES AND ASSUMPTIONS UTILIZED IN WCAP-10019 ARE APPLICABLE FOR PTS EVALUATIONS.
- CONTROL SYSTEM FAILURES ARE NOT MAJOR CONTRIBUTORS TO THE FREQUENCY OF OCCURRENCE OF CHALLENGING TRANSIENTS ON WESTINGHOUSE PWR'S (APPROXIMATELY 10^{-4} OCCURRENCES/REACTOR-YEAR).
- A TEMPERATURE-LIMIT METHODOLOGY HAS BEEN DEVELOPED WHICH CAN BE APPLIED ON A PLANT-SPECIFIC BASIS TO PROVIDE DIRECTION TO THE PLANT OPERATOR IN ASSESSING THE ONSET OF PRESSURIZED THERMAL SHOCK CHALLENGE.
- EXCESSIVE FEEDWATER TRANSIENTS DO NOT REPRESENT ANY SIGNIFICANT CONTRIBUTION TO THE RISK OF PTS ON A WESTINGHOUSE PWR.
- WARM PRESTRESSING MAY BE APPLIED TO EVEN SEVERE REPRESSURIZATION TYPE TRANSIENT SCENAPIOS.

RECOMMENDATIONS FOR FUTURE R&D

- THREE-DIMENSIONAL MODEL DEVELOPMENT
  - T&H ANALYSES
  - LEFM
- FULL PROBABILISTIC RISK ASSESSMENT
- REALISTIC TREATMENT OF WARM PRESTRESSING (B>0).
- REALISTIC VESSEL TOUGHNESS
  - MATERIAL PROPERTIES (ACTUAL)
  - ELASTIC/PLASTIC ANALYSES TECHNIQUES
- INSPECTION TECHNIQUES

FUEL MANAGEMENT TECHNIQUES

- WOG SUBMITTING REPORT TO NRC BY 6/15 DISCUSSING COST/BENEFIT
- IMPLEMENTING FUEL MANAGEMENT TO REDUCE FLUENCE THROUGH LLLP
  - FOR OLDER VESSELS, IMPACT MAY BE MINIMAL
  - UTILIZATION REQUIRES DETAILED POWER SHAPE ANALYSIS
- UTILIZING DUMMY FUEL ASSEMBLIES IS UNACCEPTABLE DUE TO NEGATIVE IMPACT ON SAFETY (REDUCTION IN LOCA MARGIN).
- CONCLUSION: IMPLEMENTING L^3P SHOULD BE PREROGATIVE OF OPERATING UTILITIES.

HUMAN FACTORS CONSIDERATIONS

- OPERATOR IS AN IMPORTANT FACTOR IN MITIGATING PTS EVENTS. BALANCE MUST BE GIVEN TO MAINTAINING CORE INTEGRITY (ADEQUATE CORE COOLING) AND MAINTAINING SYSTEM INTEGRITY.

- WOG POSITION

  BASED UPON STUDIES PERFORMED TO DATE --

  - OPERATOR PERFORMANCE CAN BEST BE ENHANCED THROUGH IMPROVED PROCEDURES AND TRAINING.

  - THE USE OF THE TEMPERATURE-LIMIT METHODOLOGY DESCRIBED IN WOG MAY 28 REPORT IN STATUS TREE FOR SYSTEM INTEGRITY WILL PROVIDE IMPROVED GUIDANCE FOR OPERATOR.

  - CONTROL OR PROTECTION SYSTEM MODIFICATIONS TO IMPROVE PROTECTION HAVE NOT BEEN IDENTIFIED.

  - PRESENT INSTRUMENTATION APPEARS ADEQUATE, HOWEVER, CONSIDERATION TO IMPROVED INFORMATION DISPLAYS MAY BE WARRANTED.

SUMMARY

- $\underline{W}$OG PROGRAMS DEMONSTRATE NEAR-TERM SAFETY OF $\underline{W}$ OPERATING PLANTS.
- PROGRAMS HAVE LED TO IMPROVED METHODOLOGY FOR EVALUATING FUTURE PTS ACTIONS.
- PROBABILISTIC RISK ASSESSMENT WOULD DEMONSTRATE NO LONG-TERM SAFETY CONCERNS.
- UTILITIES IN $\underline{W}$OG ARE COMMITTED TO SYSTEMATIC RESOLUTION OF PTS ISSUE BASED UPON ITS FINANCIAL IMPACT (DOWN-TIME, AVAILABILITY), AND TO FULLY ADDRESS THE SAFETY QUESTION.

ACRS MEETING ON PTS

(JUNE 3, 1982)

- CE APPROACH TO PTS
- OPERATOR ACTIONS IN CURRENT EVALUATIONS
- FLUENCE REDUCTION BY FUEL MANAGEMENT
- FUTURE R+D WORK

CE APPROACH TO PTS

- NO CONCERN ON NEWER VESSELS DUE TO LOW COPPER MATERIALS

- NO NEAR TERM CONCERN ON OLDER VESSELS

  1) VERY FEW OVERCOOLING EVENTS EXPERIENCED AT CE NSSS PLANTS.
  2) EMERGENCY PROCEDURE GUIDELINES.
  3) EVALUATIONS OF MSLB, SBLOCA AND LOFW SCENARIOS.

SUMMARY OF PTS EVALUATIONS

PLANT	TRANSIENT SCENARIO		
	II.K.2.13	MSLB	AOO
CALVERT CLIFFS-1	EOL	+21	EOL
FORT CALHOUN	EOL	EOL	EOL
MAINE YANKEE	EOL	EOL	EOL
PALISADES	EOL	EOL	EOL
MILLSTONE-2	EOL	EOL	EOL
ST. LUCIE-1	EOL	EOL	EOL
CALVERT CLIFFS-2	EOL	EOL	EOL
ARKANSAS-2	EOL		
ST. LUCIE-2	EOL		
WATERFORD-3	EOL		
SAN ONOFRE-2&3	EOL		
PALO VERDE-1,2&3	EOL		

OPERATOR ACTIONS IN CE EVALUATIONS

SBLOCA + LOFW (CEN-189)

1) RCP TRIP ON SIAS
2) A) OPEN PORV'S
   B) RECOVER FW

MSLB (150-DAY LETTERS)

1) RCP TRIP ON SIAS
2) HPSI TERMINATION
   A) PRIOR TO REPRESSURIZATION
   B) AFTER REPRESSURIZATION

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VESSEL FLUENCE REDUCTION BY FUEL MANAGEMENT

PLANT SPECIFIC EVALUATIONS

- VESSEL FLUENCE HISTORY
- CRITICAL WELD LOCATIONS
- TRANSITION LOADS AND RELOADS
- RELOAD FREQUENCY
- FUEL MARGIN
- STRETCH POWER CAPABILITY
- OVERALL PTS ACCEPTANCE CRITERIA

PTS R+D

- RG-1.99 SHIFT PREDICTION CURVES
- ELASTIC PLASTIC FRACTURE MECHANICS
- HPSI MIXING DURING STAGNANT LOOP FLOW

NEUTRON FLUX REDUCTION AT VESSEL WALL
(Fort Calhoun Station)

- PRACTICAL FUEL MANAGEMENT CHANGES (Octant Symmetric, Minimum Use of Shims)

  - Reduce flux to critical longitudinal welds by a factor of two

  - Peak pin power increases accommodated by improved safety analysis methodology

  - Requires increased reload analysis ($100,000/cycle)

- POTENTIAL FUEL MANAGEMENT CHANGES (Half Core Symmetric, Heavy Shim Utilization)

  - Potential flux reduction to critical longitudinal welds by a factor of four or five

  - Complex fuel management and safety analysis

  - Advanced analytical techniques, possible Reactor Protective System changes, and utilization of shims required to maintain full power capability

  - Required Studies

    - Fuel management strategy
    - Safety analysis scoping
    - Identify changes required in safety analysis
    - Identify Station modifications

  - Significantly increase reload analysis costs

  - Adversely impacts fuel cycle costs

NEUTRON FLUX REDUCTION AT VESSEL WALL (Continued)

- REACTOR VESSEL WALL SHIELDING
  - Use of hollow stainless steel dummy rods or highly depleted uranium assemblies has the potential to accomplish an estimated factor of ten reduction
  - Only neutronics calculations have been done
  - Probably cannot maintain full power capability
  - Detailed feasibility analyses on core hydraulics, power distributions, accident analysis, ECCS, etc. are required
  - Not considered practical or necessary

NEED FOR FLUX REDUCTION

- MSLB is bounding transient

- Analysis shows on the order of 20 additional EFPY before crack extension is calculated for MSLB transient using NRC initial RT_{NDT} (-20°F)

- Plastic fracture mechanics can probably show the vessel will not suffer a through wall crack

- Operator action can minimize the potential for repressurization

- A factor of two reduction in flux to the reactor vessel wall currently provides assurance of no crack extension at end of vessel life with a significant safety margin

EXCESS FEEDWATER TRANSIENT
(Fort Calhoun Station Unit No. 1)

- No dependence on operator action

- Large "hot" water inventory in steam generators combined with control and safety systems provides the operator with a significant amount of time (20-30 minutes) to diagnose and take action if necessary

- Existing EOP instructions are to maintain RCS pressure and temperature within the 50°F subcooling and 100°F/hr cooldown curve limits

- Control Systems - Main Feed Water (MFW)

  - MFW regulating valves ramp to 10% flow on reactor trip

  - MFW regulating valves close on high Steam Generator (SG) downcomer level

  - Single overcooling transient (ΔT = 107°F) caused by regulating valve failure in 42 years of operation for CE plants

EXCESS FEEDWATER TRANSIENT (Continued)

- Safety Systems - MFW

  - MFW isolation valves close on Pressurizer Pressure Low Signal (Current) or Steam Generator Pressure Low Signal (Future) terminating the transient at an RCS cold leg temperature of approximately 465^{o}F.

- Safety Systems - Auxiliary Feed Water (AFW)

  - Activates only on low-low level in a SG

  - AFW flow prohibit signals based on low SG pressure and low differential pressure between SG's

  - AFW flow terminated when level is restored

- Improved Diagnostic Capability

  - Pressure-temperature envelope CRT displays to be provided on new computer during 1983

  - Exploring installation of wide range RTD's on all cold legs (currently installed on one cold leg in each loop)

WHY IS THE APPROACH THE RIGHT ONE?

- B&W OWNERS GROUP APPROACH IS REALISTIC, YET CONSERVATIVE.
- NUREG-0737 REQUIREMENT
  - GENERIC, BOUNDING ANALYSIS PERFORMED TO MEET SCHEDULE
  - SHORT LIFETIMES CALCULATED (BAW-1648 SUBMITTED IN JAN. 1981)
  - LEARNED A LOT (CAREFUL INTEGRATION OF MANY TECHNICAL AREAS IMPORTANT)
- PLANT-SPECIFIC PROGRAM INITIATED
  - REALISTIC PLANT CONFIGURATION AND PERFORMANCE
  - CONSERVATIVE ANALYTICAL TECHNIQUES
  - OCONEE-1 REPORT SUBMITTED IN JAN. 1982

R&D WORK NEEDED?

- UNCERTAINTIES ASSOCIATED WITH EXTRAPOLATION OF TODAY'S BEST ENGINEERING INFORMATION DO EXIST

- CONSERVATIVE ANALYSES USING NOMINAL VALUES SHOW NO NEAR-TERM SAFETY CONCERN

- HOWEVER, PROGRAMS SHOULD CONTINUE SO WE STAY AHEAD OF THE CONCERN E.G.:

  -MATERIAL (E.G., RVSP, FRACTURE TOUGHNESS)

  -INSPECTION TECHNIQUES (E.G., ENHANCED NEAR SURFACE; FLAW SIZING)

  -FLUENCE ANALYSES (E.G., BENCHMARK CODES)

  -THERMAL MIXING (E.G., BENCHMARK CODES)

  -PLANT OPERATIONS (E.G., CODE BENCHMARKING)

- FEEDBACK OF R&D TO ANALYSIS IS PRUDENT: PERIODIC REEVALUTIONS OF PTS SHOULD BE REQUIRED BY APPENDIX G OF 10CFR50

REDUCE FLUENCE AT VESSEL WALL?

- CALCULATED REDUCTIONS - B&W OWNERS GROUP PROGRAM BENCHMARKED AND REFINED CALCULATIONS TO REMOVE ABOUT 30% CONSERVATISM (BAW-1485)

- REAL REDUCTIONS - MOST B&W PLANTS HAVE IMPLEMENTED A LOW LEAKAGE FUEL CYCLE (IN-OUT-IN) FOR REAL REDUCTION OF 30%

- FURTHER REDUCTIONS MAY BE POSSIBLE (E.G., IN-IN-OUT) BUT REQUIRE EVALUATION AND TRADEOFFS

  - CONCEPTUAL STUDIES
  - ANALYSES (POWER DISTRIBUTION, FLUENCE CALCULATIONS, ETC.)
  - HARDWARE LEAD TIME

- BASED ON RESULTS TO DATE FOR B&W PLANTS, FURTHER REDUCTIONS ARE NOT NECESSARY

HUMAN ACTION DEPENDENCE TO AVOID RANCHO SECO TYPE TRANSIENT?

- ANALYSES ASSUME THAT OPERATORS WILL TAKE ACTIONS PER PROCEDURES
- RANCHO SECO (MARCH 1978) INVOLVED EQUIPMENT FAILURES
- CHANGES HAVE BEEN MADE AT PLANTS SINCE THEN, E.G.:
  - ICS/NNI UPGRADES PER IEB 79-27
  - EFW SYSTEM UPGRADES
  - REVISED SB LOCA OPERATING GUIDELINES
  - ATOG DEVELOPMENT (SYMPTOM ORIENTED PROCEDURES)
- B&W BELIEVES THAT A PRESSURE/TEMPERATURE DISPLAY WOULD PROVIDE IMMEDIATE INFORMATION TO ENHANCE OPERATOR ACTIONS
- ADDITIONAL CONTROL SYSTEMS MUST BE CAREFULLY CONSIDERED - TRADEOFFS INVOLVED; OBJECTIVE MUST BE TO INCREASE OVERALL RELIABILITY

GPUN APPROACH TO PTS

1. PLANT SPECIFIC PTS EVALUATION. DUE AT END OF JUNE 1982.

2. GPU BELONGS TO B&W OWNERS GROUP.
   GPUN SUPPORTS EPRI EFFORTS
   GPUN/B&W OWNERS GROUP HAS ACTIVE MATERIALS SURVEILLANCE PROGRAM.

3. GPUN IS PRESENTLY CONSIDERING LOW LEAKAGE FUEL MANAGEMENT SCHEME, I.E., 18 MO LBP "IN-OUT-IN" CYCLE.

4. GPUN HAS MADE FOLLOWING MODIFICATIONS:
   A. ENHANCED POWER SUPPLY RELIABILITY FOR ICS
   B. UPGRADING OF EFW SYSTEM TO SAFETY GRADE (PARTIALLY COMPLETED)
   C. INSTALLATION OF VENTURI IN EFW AND HPI TO LIMIT FLOW.
   D. PLANT OPERATOR TRAINING EFFORT.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

5-25-82

APPENDIX XXII
POST-HEARING QUESTIONS ON ENERGY RESEARCH AND PRODUCTION

To: Mr. Fraley, ACRS

FROM: John Lucemann, OCA

Ray:

Enclosed are the post-hearing questions from Dr. Okrent at the subcommittee. There is no formal deadline but it is requested that the responses be provided at your earliest convenience.

John

QUESTIONS FROM
ENERGY RESEARCH & PRODUCTION SUBCOMMITTEE ON
NRC RESEARCH PROGRAM

Dr. Siess

1. Would you explain the basis for ACRS' contention that the distinction between confirmatory research and research to improve reactor safety is no longer useful?

2. Would you cite examples of the kind of research ACRS was referring to in saying that the NRC should reduce sharply confirmatory research in areas where current regulatory requirements provide adequate protection to the public?

3. Your report also stated that the NRC research response to the TMI-2 accident has largely been the implementation of remedies to specific issues subject to early identification and resolution, but that there has been a lagging incoherent approach to more general and probably more important issues. Would you elaborate on these more general issues and suggest how they should impact NRC's research?

4. ACRS singled out the "Accident Evaluation and Mitigation" decision unit as warranting lower funding than proposed due to the absence of a program that is focused to meet NRC needs. Is this recommendation based on a lack of confidence in particular program elements? What would you include in the "substantial portion of the longer-term experimental and code development work" which ACRS said could be eliminated in a better-focused program?

5. Why does ACRS feel that the separate effects experiments on the behavior of damaged fuel in the Annular Core Research Reactor are too detailed?

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6. ACRS expressed some concern about the goals of the German Upper Plenum Test Facility and their priority in NRC's safety research. In what ways do you recommend that NRC attempt to redirect its portion of the expenditures in this program?

7. What concern led to the ACRS recommendation that the program on Fission Product Release and Transport be subjected to careful peer review?

8. ACRS suggested that even with a complete set of input data to describe a given plant, an evaluation using the methods of the Seismic Safety Margins Research Program may be so beset with uncertainties as to provide little basis for licensing decisions beyond the judgments already being made. Would you explain the basis for this concern?

9. The ACRS recommended that human factors research ought to be redirected from control room design to those areas already identified as significant contributors to risk. What specific areas should receive this redirected emphasis?

10. The ACRS cited a need for more formal interaction with EPA on uranium mining and milling. What led to this concern?

11. In reference to Earth Sciences research, ACRS recommended examination of the state of knowledge of seismology and geology, keeping in mind problems which arise in the application of past regulatory practice. Would you explain what problems are referred to here?

12. The ACRS recommended that NRC review the risk-limitation effectiveness of general design criteria, regulatory guides, and standard review plan. Would you clarify the effort that is being suggested here?

13. You note in your testimony that the Commissioners should help establish research priorities by addressing questions relating to the respective roles of NRC, DOE and industry in safety research. Would you describe ACRS' view of the appropriate roles of these segments?

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APPENDIX XXIII

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Summary of Evaluations Related to Reactor Vessel Integrity, Westinghouse Electric Corporation, Nuclear Technolocy Division for the Westinghouse Owners Group, May 1982

2. Memorandum, H. Etherington to M. Bender, Chairman, Ad Hoc Metal Components Subgroup, Question Concerning the Effect of Cladding On Crack Propagation in a PTS, May 31, 1982

3. DRAFT, Pressurized Thermal Shock (PTS) in Nuclear Power Plant Reactor Vessels, Prepared by M. Bender as a Written Subcommittee Report for Internal Committee Use Only, May 28, 1982

4. Memorandum, H. R. Denton, Director, NRR, to R. B. Minogue, Director, RES, RES FY 1984-85 Budget-Internal Review, June 2, 1982

5. NRC Staff Responses to Questions by the ACRS Subcommittee During Meeting of May 20-21, 1982 on Midland Plant, Units 1 and 2, May 2, 1982