ACRS-1544

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PDR 12/22/78

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MINUTES OF THE
218TH ACRS MEETING
JUNE 1-2, 1978

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D.C. 20506, or call area code 202-724-0367.

STEPHEN J. MCCLEARY,
Advisory Committee Management Officer.

[FR Doc. 78-13377 Filed 5-16-78; 8:45 am]

[7536-01]

HUMANITIES PANEL

Meeting

MAY 2, 1978.

Pursuant to the provisions of the Federal Advisory Committee Act (Pub. L. 92-463, as amended), notice is hereby given that a meeting of the Humanities Panel will be held at 806 15th Street NW., Washington, D.C. 20506, in room 807, from 9 a.m. to 5:30 p.m. on June 8-9, 1978.

The purpose of the meeting is to review Elementary and Secondary Education Program applications submitted to the National Endowment for the Humanities for projects beginning after October 1, 1978.

Because the proposed meeting will consider financial information and disclose information of a personal nature the disclosure of which would constitute a clearly unwarranted invasion of personal privacy, pursuant to authority granted me by the Chairman's Delegation of Authority to Close Advisory Committee Meetings, dated January 15, 1978, I have determined that the meeting would fall within exemptions (4) and (6) of 5 U.S.C. 552b(c) and that it is essential to close the meeting to protect the free exchange of internal views and to avoid interference with operation of the Committee.

It is suggested that those desiring more specific information contact the Advisory Committee Management Officer, Mr. Stephen J. McCleary, 806 15th Street NW., Washington, D.C. 20506, or call area code 202-724-0367.

STEPHEN J. MCCLEARY,
Advisory Committee Management Officer.

[FR Doc. 78-13378 Filed 5-16-78; 8:45 am]

[7536-01]

HUMANITIES PANEL

Meeting

MAY 10, 1978.

Pursuant to the provisions of the Federal Advisory Committee Act (Pub. L. 92-463, as amended) notice is hereby given that a meeting of the Humanities Panel will be held at 806 15th Street NW., Washington, D.C. 20506, in room 1130, from 9 a.m. to 5:30 p.m. on June 15 and 16, 1978.

The purpose of the meeting is to review Youthgrants in the Humanities applications submitted to the National Endowment for the Humanities for projects beginning after October 1, 1978.

Because the proposed meeting will consider financial information and disclose information of a personal nature the disclosure of which would constitute a clearly unwarranted invasion of personal privacy, pursuant to authority granted me by the Chairman's Delegation of Authority to Close Advisory Committee Meetings, dated January 15, 1978, I have determined that the meeting would fall within exemptions (4) and (6) of 5 U.S.C. 552b(c) and that it is essential to close the meetings to protect the free exchange of internal views and to avoid interference with operation of the Committee.

It is suggested that those desiring more specific information contact the Advisory committee Management Officer, Mr. Stephen J. McCleary, 806 15th Street, NW., Washington, D.C. 20506, or call area code 202-724-0367.

STEPHEN J. MCCLEARY,
Advisory Committee Management Officer.

[FR Doc. 78-13379 Filed 5-16-78; 8:45 am]

[7590-01]

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Meeting

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b.), the Advisory Committee on Reactor Safeguards will hold a meeting on June 1-3, 1978, in Room 10[illegible] 1717 H Street NW., Washington, D.C.

The agenda for the subject meeting will be as follows:

THURSDAY, JUNE 1, 1978

8:30 a.m.-9 a.m.: Executive session (open). The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities including the appointment of new Committee members.

This session will be open to the public except for those portions which must be closed to protect information the release of which would represent an unwarranted invasion of personal privacy.

The Committee will hear and discuss the report of the ACRS Subcommittee and consultants who may be present regarding the request for operation at increased power of the Maine Yankee Atomic Power Station.

Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter and provisions for physical protection of this unit.

9 a.m.-11 a.m.: Maine Yankee Atomic Power Station (open). The Committee will hear and discuss presentations by representatives of the NRC staff and the applicant related to the request to operate this unit at increased power. Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter and provisions for physical protection of this unit.

11 a.m.-12:15 p.m.: Executive session (open). The Committee will hear and discuss reports of Subcommittees and Working Groups on a number of generic matters related to reactor safety including anticipated transients without scram and proposed revisions to NRC regulatory guides. The Subcommittee on the Vermont Yankee Nuclear Power Station will also report on operating experience at this facility.

1:15 p.m.-2:15 p.m.: Report on Intergovernmental Review of Nuclear Waste Management (open). The Committee will hear and discuss a report by representatives of the NRC regarding NRC participation in the program for reviews of nuclear waste management and disposal.

2:15 p.m.-2:30 p.m.: Executive session (open). The Committee will hear and discuss the report of the ACRS Subcommittee and consultants who may be present regarding the request for operation of the Indian Point Nuclear Generating Station, Unit 3, at full power. Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter and provisions for physical protection of this unit.

2:30 p.m.-4:30 p.m.: Indian Point Nuclear Generation Station, Unit 3 (open). The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the applicant regarding the request for operation of this unit at full power. Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter and provisions for physical protection of this unit.

4:30 p.m.-5:30 p.m.: Executive session (open). The Committee will discuss proposed ACRS positions and comments regarding generic matters related to nuclear powerplant safety including the use of Class 9 accidents for evaluation of alternate reactor sites and the source term used in reactor safety analysis.

The Committee will also discuss its proposed reports to the NRC on the Maine Yankee Nuclear Plant and the Indian Point Nuclear Generating Station, Unit 3.

FRIDAY, JUNE 2, 1978

8:30 a.m.-1:30 p.m.: Meeting with NRC staff (open). The Committee will hear presentations from and hold discussions with members of the NRC

218th

staff regarding recent licensing actions and operating experience including the seismic reevaluation of several nuclear powerplants and review of a proposed safe shutdown system for the Oconee Nuclear Plant.

Representatives of the NRC staff and its contractors will also report to the ACRS on generic matters related to nuclear powerplant safety including the bases for combination of seismic and other dynamic loads, the proposed use of Class 9 accidents for evaluation of alternate powerplant sites, and comparison of risks from nuclear powerplants with other societal risks.

The future schedule for ACRS activities and topics proposed for consideration by the Committee will also be discussed.

2:30 p.m.-6 p.m.: Executive session (open). The Committee will discuss proposed ACRS comments regarding the establishment of a quasi-judicial, statutory board to investigate reactor accidents. The Committee will also discuss proposed comments regarding generic matters discussed during this meeting and miscellaneous Committee activities including reorganization of ACRS Subcommittees and Working Groups and a proposed periodic report of ACRS activities.

The Committee will also discuss proposed reports to the NRC on the Maine Yankee Nuclear Plant and the Indian Point Nuclear Generating Station, Unit 3.

SATURDAY, JUNE 3, 1978

8:30 a.m.-12 noon: Executive session (open). The Committee will discuss its proposed reports to NRC regarding the Indian Point Nuclear Generating Station, Unit 3, and the Maine Yankee Atomic Power Station.

The Committee will complete discussion of generic matters and miscellaneous ACRS activities considered during this meeting.

Procedures for the conduct of and participation in ACRS meetings were outlines in the FEDERAL REGISTER on October 31, 1977, page 56972. In accordance with these procedures, oral or written statement may be presented by members of the public, recordings well be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

I have determined in accordance with section 10(d) of Pub. L. 92-463 that is is necessary to close portions of the meeting as noted above to protect proprietary information (5 U.S.C. 552b(c)(4)), to preserve the confidentiality of information related to safeguarding of special nuclear material and the physical protection of nuclear facilities (5 U.S.C. 553b(c) (1) and (4)), and to protect information the release of which would represent an unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)). Separation of factual information from information considered exempt from disclosure during closed portions of the meeting is not considered practical.

Background information concerning items to be considered during this meeting can be found in documents on file and available for public inspection in the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C. 205[illegible] and in the following public document rooms:

INDIAN POINT NUCLEAR GENERATING STATION, UNIT 3

White Plains Public Library, 100 Mar[illegible] Avenue, White Plains, N.Y. 10601.

MAINE YANKEE ATOMIC GENERATING STATION

Wiscasset Public Library, High Street, Wiscasset, Maine 04578.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley, telephone 202-634-1371, between 8:15 a.m. and 5 p.m. e.d.t.

Dated: May 15, 1978.

JOHN C. HOYLE,
Advisory Committee Management Officer.

[FR Doc. 78-13560 Filed 5-16-78; 8:48 am]

[7590-01]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Proposed Meetings

In order to provide advance information regarding proposed meetings of the ACRS Subcommittees and Working Groups and of the full Committee, the following preliminary schedule is being published. This preliminary schedule reflects the current situation, taking into account additional meetings which have been scheduled and meetings which have been postponed or canceled since the last list of proposed meetings published in the FEDERAL REGISTER on April 28, 1978. Those meetings which are definitely scheduled have had, or will have, an individual notice published in the FEDERAL REGISTER approximately 15 days (or more) prior to the meeting. Those Subcommittee and Working Group meetings for which it is anticipated that there will be a portion or all of the meeting open to the public are indicated by an asterisk (*). It is expected that the sessions of the full Committee meeting designated by an asterisk (*) will be open in whole or in part to the public. ACRS full Committee meetings begin at 8:30 a.m. and Subcommittee and Working Group meetings usually begin at 8:30 a.m. The exact time when items list[illegible] in the agenda will be discussed du[illegible] full Committee meetings and when Subcommittee and Working Group meetings will start will be published approximately 15 days prior to each meeting. Information as to whether a meeting has been firmly scheduled, canceled, or rescheduled, or whether changes have been made in the agenda for the June 1-3, 1978, ACRS full Committee meeting can be obtained by a prepaid telephone call to the Office of the Executive Director of the Committee, telephone 202-634/1374, Attn.: Mary E. V[illegible]erholt, between 8:15 a.m. and 5 p.m. [illegible]

SUBCOMMITTEE AND WORKING GROUP MEETINGS

**Davis Besse Nuclear Power Station*, Units 2 and 3, May 18, 1978, Washington, D.C. Rescheduled to June 30, 1978. Notices of this meeting were published in the FEDERAL REGISTER on May 3 and 11, 1978.

**Vermont Yankee Nuclear Power Station*, May 19, 1978, Vernon, Vt. The Subcommittee will review the operating history and fuel performance for this station. Notice of this meeting was published in the FEDERAL REGISTER on May 4, 1978.

**Fluid/Hydraulic Dynamic Effects*, May 23, 1978, Des Plaines, Ill. The Subcommittee will discuss items related to the Mark I, II, and III containment systems. Notice of this meeting was published in the FEDERAL REGISTER on May 8, 1978.

**Diablo Canyon Nuclear Power Station*, May 24-25, 1978 (rescheduled from May 17, 1978), Washington, D.C. Rescheduled to June 14-15, 1978.

**Maine Yankee Nuclear Plant*, May 25, 1978 (rescheduled from May 2, 1978), Washington, D.C. The Subcommittee will review the request of the Maine Yankee Atomic Power Corp., to operate this plant beyond the FSAR designated power of 2,560 MW(t) up to a power level of 2,630 MW(t). Notices of this meeting were published in the FEDERAL REGISTER on April 17, May 2, and May 11, 1978.

**Anticipated Transients Without Scram (ATWS)*, May 26, 1978, Washington, D.C. The Working Group will discuss various [illegible] pertaining to anticipated transients during reactor op-

Issue Date:

SEP 29 1978

Meeting Dates:
June 1-2, 1978

CERTIFIED

MINUTES OF THE
218TH ACRS MEETING
JUNE 1-2, 1978
WASHINGTON, DC

The 218th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC, was convened at 8:30 a.m., Thursday, June 1, 1978.

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

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

I. Chairman's Report (Open to Public)

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

A. Reviewers

The Chairman named Messrs. Bender and Isbin as reviewers for the 218th ACRS meeting.

B. Illness of Member

The Chairman noted that Mr. Ebersole is currently ill, and will not be available to participate in ACRS activities for an indefinite time.

C. New Members

The Chairman noted that the Committee's nominations for new members to be appointed by the Commission has been submitted to the Commissioners. As of the beginning of this meeting, all the Commissioners had not voted yet.

D. ACRS Fellowship Program

The Chairman noted that funds have been appropriated for the ACRS Fellowship Program. A recruitment announcement has been prepared and released (see Appendix III). The Ad hoc Working Group that has been developing the scope of the Fellowship Program, will consider assignments of fellows, and present its recommendations to the full Committee for its concurrence.

E. Service Award

The Chairman awarded a 25-year length-of-service pin and citation to R.F. Fraley, Executive Director.

II. Maine Yankee Nuclear Power Station (Increase of Power Level) (Open to Public)

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

A. Subcommittee Report

Mr. Kerr, Subcommittee Chairman, reviewed the chronology of the licensing actions regarding the Maine Yankee Atomic Power Station, and noted that the original safety evaluation was carried out at a power level of 2560 MWt, but that the ACRS report associated with the operating license was written on the basis of 2440 MWt. He noted that Maine Yankee has requested an increase of power to 2630 MWt, and that this increased power level was based primarily on a measured reactor core flow which turned out to be approximately 10% greater than the design basis flow on which the original safety evaluation had been carried out. He noted that both the Licensee and the NRC Staff have performed safety evaluations for the 2630 MWt level, and that these evaluations were made under current rules and criteria rather than those which were in effect when the original SER was written. Because of the different bases involved in the original versus the current evaluations, it is difficult to compare the safety margins calculated. However, both the Licensee and the NRC Staff conclude that the plant can be operated at the new higher power level, and that current safety criteria and rules will be satisfied. He noted that the NRC Staff's current evaluation uses meteorological data gathered during 1977, and with those data and the new criteria used, the NRC Staff obtained results which require that the technical specification leakage be reduced from 0.15% per day, which was used in the old evaluation, to 0.1% per day. The Licensee has accepted the proposed leak rate in the new Technical Specifications. However, the NRC Staff questions whether the 1977 meteorological data is in fact representative. Therefore, the NRC Staff is requiring that additional meteorological data be collected over the next year, and that the dose calculations be redone. ACRS consultants disagreed with this conclusion, and believe that the meteorological model being used by the NRC Staff is extremely conservative, and that any change of data resulting from the new collection would be insignificant. (For project status report, see Appendix IV; for consultants' report, see Appendix V.)

[Note: W. Johnson, Yankee Atomic Electric Co. (YAEC) coordinated presentations for the Licensee; C. Nelson, for the NRC Staff.]

B. Licensee's Presentations

1. Introduction

W. Johnson, YAEC, stated that his company is developing an in-house capability for performing safety analyses in their plants. The Company views its responsibility for safe operation as a serious obligation, and is currently expending upwards of $1 million on the current safety-related plant analysis.

He noted that the Maine Yankee Plant is currently beginning its 349th day of continuous full-power operation.

2. Site Description and Operating Parameters

T. Bergeron, YAEC, described the site and its location, the area demography, the cooling system layout, the core configuration, the licensing and operating history, operating parameters, and the postulated accidents considered in the current safety analysis (see Appendix VI).

3. Analyses and Performance

J. Di Stefano, YAEC, discussed the basis for reevaluation of the design basis accidents for radiological dose assessment, identified the design basis accidents that were reevaluated for the 2630 MWt stretch power submittal, compared the analysis of steam generator tube rupture for the final safety analysis report vs. the stretch power submittal, discussed the off-site doses from a steam generator tube rupture, compared the main steam line failure outside containment evaluation for the final safety analysis report vs. stretch power submittal, discussed the off-site doses from steam line break, compared the fuel handling incident analysis for the FSAR vs. the stretch power submittal, discussed the doses from the fuel handling incident, compared the loss of coolant accident doses calculated for the FSAR vs. the stretch power submittal, and discussed the doses from a loss of coolant accident, the revised LOCA analysis, and the revised doses from a LOCA (see Appendix VII).

4. Safety Comparisons

T. Bergeron discussed the safety analysis, deviations from the final safety analysis report, steady state DNBR comparisons, fuel performance, dose measurements, and a comparison of the calculated operating parameters with the design values of the major equipment in the plant (see Appendix VIII). He noted that the Maine Yankee Plant has never tripped from exceeding peaking factor limits.

C. Status of NRC Staff Review

C. Nelson, NRC Staff, discussed the NRC Staff Safety Review, and noted that the NRC Staff has concluded that it is acceptable for the Maine Yankee Atomic Power Station to operate at power levels up to 2630 MWt. He noted certain changes which will be made to the Technical Specfications, and also noted that an additional meteorological review will be made of the plant when additional data is obtained. He said that the NRC Staff has reviewed the Licensee Event Reports, and has found the performance of the plant to date to be satisfactory. He noted that there has been no operator-generated scrams since 1975.

In answer to a question, R. Shome, YAEC, said that the plant is operated in conformance with Branch Technical Position 18 regarding the locking out of certain ECCS valves.

In answer to a question, P. S. Littlefield, YAEC, said that the Licensee is evaluating hydrazine sprays as an alternative to sodium hydroxide sprays for use inside containment. The current hydroxide spray systems are now gravity feed systems, but if a hydrazine system is adopted, a positive pump injection system will be required. The difference between the use of hydrazine and hydroxide as a means of controlling iodine is the limitation on long-term retention.

D. Caucus

The Committee indicated unanimously that it believed it could write a favorable report on the matter of increasing operating power up to 2630 MWt for the Maine Yankee Nuclear Power Station.

III. Meeting with NRC Staff on the Use of Class 9 Accidents for Alternate Site Evaluation (Open to Public)

[Note: Thomas G. McCreless was the Designated Federal Employee for this portion of the meeting.]

(For background information regarding this discussion, see Appendix IX.)

A. Introduction

D. Bunch, NRC Staff, briefly reviewed the recent work to identify better ways to compare alternate nuclear power plant sites. He noted that in a letter dated December 10, 1975, the Committee suggested that the NRC Staff explore the use of the reactor safety consequence model in an attempt to devise better figures of merit for the site evaluation process. The NRC Staff's objective was to describe at this meeting the results of the efforts which were initiated after receipt of the letter.

B. Reactor Safety Study (RSS) Consequence Model

R. Blond, NRC Staff, discussed the applicability of the RSS Consequence Model, CRAC-code, for evaluation of the environment impact of potential reactor accidents. He discussed the limitations of the CRAC-code, described the code, discussed the parameters by which environmental impact is described, presented the schematic outline of the CRAC-code, discussed the site data requirements necessary for analysis by the CRAC-code and the areas in which the CRAC-code is effective, and outlined the research program for improvement of this method (see Appendix X).

Mr. Isbin asked whether the water pathway in some cases, that is through a flowing aquifer, does not produce consequences at least as serious as an air release of radioactive material. Members of the NRC Staff offered their opinion that the consequences from the air release are more serious, but members of the Committee disagreed with this characterization.

C. Use of RSS Consequence Model in Site Reviews

D. Bunch discussed the environmental reviews required under the National Environmental Protection Act (NEPA). He discussed current NRC practices with respect to NEPA reviews, the safety aspects of alternative site reviews, the major factors influencing consequences of accidental releases, examples of methods of comparing alternative sites, estimates, at various probabilities, of early fatalities for one reactor, comparisons of relative

Class-9 consequences at five alternate sites, an economic comparison of residual accident risks, and possible applications of the CRAC-code (see Appendix XI).

In answer to a question of why core-melt is not considered by the NRC Staff in its safety analyses, H. Denton said that the safety analyses and reviews followed the standard review plans. The NRC Staff is not presently addressing this question in the safety review, but does consider total risk in the environmental review.

In answer to a question, D. Bunch said that the presentation given here to the Committee was essentially the same as that given recently to the Commissioners.

IV. Meeting with the NRC Staff on the Interagency Nuclear Task Force (Open to Public)

[Note: Ragnwald Muller was the Designated Federal Employee for this portion of the meeting.]

S. Meyers, NRC Staff, discussed the history and organization of the Interagency Nuclear Management Task Force, noting that this effort was initiated by the President, and has been directed to provide a policy statement to the White House by September 15, 1978. Initially, the following departments and agencies were involved: Dept. of State, Dept. of the Interior, Dept. of Transportation, Dept. of Energy, Office of Management and Budget, Council on Environmental Quality, Environmental Protection Agency, Office of Science and Technology Policy, Domestic Policy Staff, and National Security Council. NRC was purposely omitted from this task force on the basis that a regulatory agency should not be involved in the development of a national policy for the disposal of waste. NRC disagreed with this concept, contacted the Dept of Energy, and has since been included in the task force on a nonvoting basis. Three additional organizations, NASA, Arms Control and Disarmament Agency (ACDA), and the National Governors Conference requested inclusion in the task force; NASA and ACDA were added, but it was believed that the presence of the National Governors Conference would be inappropriate for the setting of national policy.

S. Meyers stated that as of this date, three meetings have been held, April 5, 20, and June 1. Three goals have been enunciated:

- development of a Presidential policy, similar to that on spent fuel,

- a public announcement explaining the problems and proposals, and
- a legislative package.

Six active working groups have been established to address what were considered to be major issues, and agencies have been assigned to take the lead in these areas:

- Alternative Strategies -- Office of Science and Technology Policy
- Federal Involvement, including standards and licensing, ownership, management, etc. — EPA,
- Transportation — DOT,
- Defense Waste — DOE,
- Spent Fuel Charges — OMB, and
- International Aspects — Dept. of State.

The Standards and Licensing Group noted above was further broken down into the following subgroups:

- Low Level Waste,
- Mill Tailings,
- Decommissioning and Decontamination.

NRC personnel are involved in all three of these subgroups, as well as with high level waste which is being considered by the Alternative Strategies Working Group.

Meetings are planned with interest groups, and discussions are planned to inform the public, including scheduled meetings in Boston, Denver, and San Francisco. NRC is taking the lead in developing relationships with State governments. The nonvoting status of the NRC and the Task Group, has presented no problems as yet, since no votes have been taken. NRC is participating as fully in the Task Force as any other group.

Mr. Moeller suggested that the Committee be provided with the reports of the Task Group.

It was the consensus of the Committee that the impressions it had received at the 217th ACRS meeting, at which time it was concluded that the participation of the NRC in the Task Force was inadequate, was erroneous.

V. Meeting with the NRC Staff on Recent Licensing Actions, Recent Operating Experience, Generic Matters Related to Light-Water Reactors, and Future Agenda (Open to Public)

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

A. Oconee: Safe Shutdown Heat Removal System

M. Fairtile, NRC Staff, said that he had kept proprietary information to a minimum in his presentation, and had limited it to a single slide. [Note: The meeting was closed to the public for the presentation and discussion of this slide. This one page of the handout will not be provided with the Appendixes made available to the Public Document Room or to copies provided outside of ACRS.] He discussed a Licensee-proposed safe shutdown system conceived as a backup core heat removal system, independent of other heat removal systems, for the three units at Oconee. The system functions by maintaining adequate reactor coolant system coolant volume and steam generator secondary side cooling volume. The system will have a dedicated diesel generator multi-volume tank, a - d-c power distribution system and the needed d-c power supply, all housed in a building separate from the rest of the plant. The existing reactor protection system, including the trip function control systems and related instrumentation, all remain unaffected by, and independent of, the proposed safe shutdown system. The proposed system is still in a conceptual stage, and the status of the NRC Staff review is as follows:

- The proposal was presented by Duke Power Co. to the NRC Staff on January 18, 1977.
- The system was formally submitted to the NRC Staff on February 1, 1978, and requested that this submittal be withheld from public disclosure on proprietary grounds.
- Detailed requests for additional information was transmitted to the Licensee on May 18, 1978.
- The NRC Staff anticipates that it will complete the review in late June 1978.

- Upon completion of the review of the concept, the Licensee will proceed to a final design of the system, construction, and testing, which will be accomplished approximately 30 months after approval of the concept.

- The NRC Staff anticipates that, assuming no problems and NRC Staff final approval, the system can be put into use approximately December, 1981.

The NRC Staff believes that this system is a desirable augmentation of existing plant systems with regard to security, fire protection, and turbine building flooding, and that the Licensee should be encouraged to complete the system. (For details of the system, and background information, see Appendix XII.)

In answer to a question, M. Fairtile said that the Licensee plans to install a separate, independent, augmented safe shutdown system for each of the three units at Oconee.

In answer to a question, M. Fairtile said that the Office of Standards Development would be consulted with regard to the consistency of this proposed system with respect to current Regulatory Guides and standards.

In answer to a question regarding some design details, M. Fairtile indicated to the Committee that at this time the design is merely conceptual, and that the details of the design are not available. However, he assured the Committee that the NRC Staff would not permit a piping arrangement with respect to the spent fuel pool, from which the makeup water supply for the proposed safe shutdown system will be obtained, that could permit the drawdown of the spent fuel pool below the level of the tops of the stored spent fuel assemblies.

M. Fairtile said that this system is being proposed to provide an alternate means of maintaining a safe shutdown for all three units without taking any damage control measures for accidents or failures within the plant. However, this system is not designed to handle LOCA conditions.

B. Skagit: Reevaluation of Related Geological Faults

C. Stepp, NRC Staff, informed the Committee that this presentation would deal with the NRC Staff's interpretation of the Shuksan Thrust Fault, the earlier interpretation of which is now being challenged. He said that in the earlier interpretation, the

Thrust was considered to have been cut off from its roots which are on the eastern side of the Cascade Mountains, and therefore are no longer involved in the current tectonism. He said that further investigations have clearly shown that this fault is not capable within the meaning of 10 CFR 100. The NRC Staff believes that it presents no hazard to the plant site from earthquake or fault movement activity. He discussed the surface rocks and subsurface geology of the area to support his conclusions. The NRC Staff concluded that the Devils Mountain Fault is the major tectonic item in this area, and that it is controlling for the Skagit site. He also offered the opinion that J. Whetten's analysis and interpretation of the existence of the Shuksan Thrust Fault is in fact merely academic. Whether his interpretation is correct or not does not affect the conclusion that the Shuksan Thrust does not provide an earthquake potential. The current additional NRC Staff study is being carried out to satisfy certain legalisms.

In answer to a question, C. Stepp stated that the Skagit plants are designed to meet Regulatory Guide 1.60, and are designed to accept a safe shutdown earthquake with ground acceleration of 0.35g. That acceleration embraces an earthquake of Richter magnitude 7.5 with an epicenter approximately 20 km from the site, which translates to an intensity of MM VIII to IX. This design also embraces earthquakes of lesser intensity and magnitude, so that if some new small faults were found active, it would not necessarily imply that a design to a higher ground acceleration would be required. He was of the opinion that the dimensions of the Devils Mountain Fault are such that it is unlikely that faults will be found in that region that would require a change in the current Skagit design. (For geological maps used in this presentation, see Appendix XII.).

C. Watts Bar, Sequoyah, and Bellefonte: Review of Seismic Design

(For background material leading up to the evaluation of Sequoyah, Watts Bar, and Bellefonte, see Appendix XIII.)

W. Gammill discussed the NRC Staff's reevaluation of the seismic adequacy of the designs at Sequoyah, Watts Bar, and Bellefonte, including the objectives of the Reevaluation Working Group, its plan of action, the possible approaches to the seismic issue, methods to reevaluate the intensity of the SSE, reevaluation of the response spectra associated with the SSE, reevaluation of the design margins for the SSE, reevaluation of the OBE, probabilistic evaluation of the seismic risks, and recommended approaches (see Appendix XIV). He noted that this reevaluation was initiated because of the general increase in seismic design requirements that have been adopted since the safety analyses were made for these three plants.

After much questioning, W. Gammill offered the opinion that the extrapolation of a historical earthquake record to larger than historical earthquakes is a matter of judgement.

Mr. Siess requested that the NRC Staff provide him with a list of additional plants, if any, for which it was requiring similar seismic design reevaluations.

D. Indian Point: Seismic Reevaluation

C. Stepp, in his introduction to this discussion, placed the Ramapo Fault question into its historic perspective. He said that the question concerning the earthquake activity or capability of the Ramapo Fault was raised as early as 1974 by the State of New York. The NRC Staff responded at that time by writing a supplement to the SER for Indian Point, which is Appendix C to the Final Safety Analysis Report for Indian Point, Unit 3. The NRC Staff concluded in that report that there was no evidence that the Ramapo Fault is capable within the meaning of 10 CFR 100. However, the NRC Staff recognized that some microearthquakes have been associated, at least geographically, with the Ramapo Fault in the historic record, and it was suggested that the question was still open. Subsequently, a consultant to the Licensee, Dr. Radcliff, in his mapping of the Ramapo Fault determined that the Indian Point Unit 3 is located very close to the fault. That mapping precipitated a very extensive additional investigation, including the establishment of a microearthquake monitoring network consisting of 11 stations, closely centered around the plant. The second part of that study was a geologic structural investigation of the fault itself, to determine all of the elements of the fault zone to establish which of the faults had the most recent movement. The elements of these faults were dated. Movement on the faults was dated by taking mineral assemblages from the fault zones, and dating these assemblages. The NRC Staff concluded on the basis of this examination that none of the faults have moved in the last seventy million years. This seventy million year period corresponds approximately to a known generalized thermal event for the eastern United States, which brought hot mineralized waters to the near surface. These waters penetrated fault zones, and the minerals that formed in the fault zones, because of the event, formed a basis for dating the final movement. No evidence has been found, based on geological data, that this fault is active. Nevertheless, there is a higher level of earthquake activity in the vicinity of the Hudson Highlands than in the nearby adjoining Appalachian Mountain region. The explanation for this activity is currently unknown. This whole question was litigated at length by the Atomic Safety and Licensing Appeal Board, and it was concluded that the fault is not capable within the meaning of 10 CFR 100, and the extension of that conclusion is that there is no basis for continuing the monitoring network.

J. Kelleher, NRC Staff, discussed the paper published in Science, Vol. 200, April 28, 1978, by Y. P. Aggarwal and L. R. Sykes. (For the Aggarwal and Sykes report, and additional comments on the seismicity of Southeastern New York and Northern New Jersey, see Appendix XVII; for ACRS consultant's comments on the Aggarwal and Sykes paper, see Appendix XVIII; for NRC Staff data on the Ramapo Fault, See Appendix XIX.)

E. Davis-Besse: Orifice Rod Assembly Performance

S. H. Weiss, NRC Staff, discussed the general problems encountered in Babcock and Wilcox reactors with respect to burnable poison rod assemblies and orifice rod assemblies. In March, 1978, the NRC Staff was notified that at Crystal River, Unit 3, two burnable poison rod assemblies failed. He described the core configuration, identified the s[illegible]en locations where control rod guide tubes are placed, and n[illegible]ed that in some of these locations, burnable poison rods were substituted for control rods. He described the control rod design and the burnable poison rod design, noting that there was a different coupling mechanism used for the burnable poison rods, because these rods are not movable. The burnable poison rods are used at the beginning of a fuel cycle, and then are replaced with orifice rods. These orifice rods are inserted to control flow through the control rod guide tubes. He then described the design of the orifice tubes. When notified of the problems with the burnable poison rods encountered at Crystal River, Davis-Besse proposed that they inspect their burnable poison rods, which at this time were approximately one-third consumed, and offered to replace worn burnable poison assemblies with orifice rods. On May 22, the NRC Staff was notified that Davis-Besse was encountering handling problems in the removal of the burnable poison rod and orifice rod assemblies, and they found wear on upper end fittings of fuel assemblies that held either the burnable poison rod or orifice rod assemblies. Babcock and Wilcox anticipates that it will complete the inspection at Davis-Besse by June 15. In the interim, Babcock and Wilcox has recommended to operators of their plants that two pump operation be minimized, and that the plants be monitored closely for loose parts. This matter is being reviewed by the NRC Staff. (For background material on the B&W orifice rod assembly failures, see Appendix XX; for details of the NRC Staff presentation, see Appendix XXI.)

F. Connecticut Yankee: Behavior of Boron Carbide Burnable Poison Plates in the Storage Rack of the Spent Fuel Pool

W. Russell, NRC Staff, stated that the current problem is caused by a buildup of gas pressure in the double wall can in which spent fuel elements are stored in the spent fuel pool. In this double wall, boron carbide plates are inserted. Gas pressure caused the canning to swell, reduced the clearances for the stored fuel, and three fuel assemblies became stuck. Measurements were taken in the fuel pool, and swelling was found occurring in assemblies which had been discharged earlier. It appears that the amount of swelling is directly related to the radiation exposure in the spent fuel pool. The Licensee plans to fix the problem by drilling vent holes in the can walls. (For details of the configuration of the spent fuel pool racks, and the design of the canning, see Appendix XXII.)

G. Basis for the Combination of Seismic and Other Dynamic Loads

R. Mattson, NRC Staff, said that he would attempt to give the Committee a status report on the NRC Staff's endeavor to write a White Paper on where, how, and why loads are combined for safety analyses. He suggested that progress on this White Paper would be slow because of the shortage of manpower resources. He said, however, that the NRC Staff has many ongoing activities in which load combinations are an important element. He suggested that the Committee permit him to treat these activities as they bear on the load combination question, rather than restructuring the NRC Staff to answer the general question on load combination. He noted that he was not prepared to provide any real answers at this time. Rather, he identified the pertinent items in the General Technical Activities Program, licensing activities, topical report reviews, and similar areas of endeavor to show where there are interactions with this question. He discussed the matter of load combinations from the historical point of view, stating that they were developed from an interpretation of General Design Criterion 2. GDC 2 requires that loads be combined appropriately. A second reason for combining loads may be to provide conservatism to cover failure caused by undetected flaws during the design earthquake, or LOCA. In combining loads, responses are also being combined. One way to combine loads is the absolute or linear summation of peaks. Another means is the use of the square root sum of the square method, which method is currently being argued by General Electric.

Mr. Bender questioned the logic of the approaches described, noting that loads were being combined for certain systems, but not for others. He noted, for example, that failures were not considered for seismically qualified electrical structures.

R. Mattson replied that the basic reason underlying that logic is that the failure of mechanical equipment in the primary system leads directly to a LOCA, whereas the failure of electrical equipment because of an undetected flaw does not lead to a LOCA. He said that it is not a question of agreement with the method, that what he is explaining is merely the rationale for what is being done. Most of the arguments for combining the loads, and determining where the combinations will take place, are matters of engineering judgment. He noted that GE has proposed, and the NRC Staff is proceeding with a review, a method for demonstrating the time phase relationship of the dynamic loads occurring within the pressure suppression pool. Their review will be occurring over a one or two year period consistent with the Shoreham, LaSalle, and WPPSS reviews. He said that in the case of Diablo Canyon, the NRC Staff is applying the interim approach, which the staff believes to be defensible on the basis of safety. In this seismic reevaluation, the absolute summation of the peaks was required for combining the SSE and LOCA loads. Mr. Bender questioned both the logic and the methods which were being described.

In answer to a question regarding the level of attention being given to the criteria for failure, J. Knight, NRC Staff, said that if the question deals with the physical realities when postulating failures, no attention is being given to the criteria for failure. This leads to extremely conservative positions.

H. Skagit: Destruction of the Meteorological Tower

R. Woodruff, NRC Staff, informed the Committee that on the evening of May 29, the meteorological tower at Skagit collapsed. The Applicant believes that this was an act of sabotage, and was directly caused by the loosening of four turnbuckles in the guy wire. At this time, there has been no construction at the site, there are no security measures being taken at the site, other than a caretaker who resides at the site. The FBI has been notified, as has the City of Seattle and the State of Washington.

I. Future Agenda

The Committee agreed upon a tentative future schedule for the review of cases (see Appendix II).

The Committee and the NRC Staff agreed, because of the originally proposed heavy load for the 219th ACRS meeting, and because such slippage would not affect the schedule of the licensing procedures, to defer consideration of Davis-Besse, Units 2 and 3, until the 220th ACRS meeting in August.

VI. Executive Sessions (Open to Public)

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

A. Subcommittee Reports

1. Regulatory Activities

Mr. Siess, Subcommittee Chairman, said that the Subcommittee has reviewed proposed Regulatory Guide 1.133 (Rev. 1), Loose Parts Detection Program for the Primary Systems of Light-Water-Cooled Reactors, and recommends that this Guide be referred to the Power and Electrical Systems Subcommittee (formerly the Electrical Systems, Control, and Instrumentation Subcommittee) for further review. They further recommended that upon such review, the Guide be referred back to the Committee for its comments. The Committee concurred.

Mr. Siess also noted that the Subcommittee recommended that the Committee concur in the regulatory position of Regulatory Guide 1.136 (Rev. 1), Material for Concrete Containments. The Committee so concurred.

2. ATWS

Mr. Kerr, Subcommittee Chairman, reported on the Subcommittee meeting held on May 26, 1978, at which discussion of the NRC Staff report, NUREG-0460, Anticipated Transients Without Scram for Light-Water Reactors, was discussed. He noted that additional meetings have been scheduled for July 13 and August 1, to continue discussions on this subject with vendors, plant operators, and probably the Atomic Industrial Forum and Edison Power Research Institute. In addition, the NRC Staff has been requested to make a presentation at the 220th ACRS meeting; and plans have been made to schedule presentations by representatives of vendors and perhaps others at the 221st ACRS meeting.

Mr. Kerr recalled that in NUREG-0460, the NRC Staff concludes that an appropriate value to assume for the frequency of an ATWS event is about $2x10^{-4}$ per year. This is based on the NRC Staff's conclusion that the unreliability of shutdown systems is about $3x10^{-5}$, and that there are about 6 anticipated transients of concern per year. The NRC Staff has established a goal of 10^{-6} per year for ATWS events that contribute to core-melt. Based on this goal, and the NRC Staff's assumption about ATWS frequency, they have concluded that a mitigating system or systems is needed in order to resolve the problem. In order to achieve the goal, assuming about 10 transients a year, requires a frequency of approximately 10^{-7} per year, if one depended upon the shutdown system alone for safety. Mr. Kerr noted, however, that the NRC Staff's conclusion that the frequency of an ATWS would be in the range of 10^{-4} per year, is based upon only one data point, and that number is not consistent with the one data point. This data point was based upon the Kahl reactor in Germany. While the Staff reaches these conclusions using probability calculations, it does not propose that the fix be judged by probability considerations. It proposes, rather, that the ATWS become a design basis accident. Mr. Kerr concluded that this form of designation will require a rule making hearing and proceeding, and that the NRC Staff is recommending that this hearing be held, and that an approved evaluation model, which describes scenarios that would accompany the ATWS events, be considered. Mr. Kerr offered the opinion that this problem is becoming more difficult. It involves something of a logical inconsistency in that one demonstrates that a problem exists by using probabilistic considerations, but does not use these same considerations to conclude that a fix has been reached. This does not cause difficulty if it can be demonstrated that a fix is indeed a fix. (For additional details, see Appendix XXIII.)

3. Vermont Yankee

Mr. Isbin, Subcommittee Chairman, recalled for the Committee that the Vermont Yankee Nuclear Power Station utilized a Mark I containment, and that the operation of this containment was considered in the ACRS letter of March 12, 1976. Progress in the long term containment program is being monitored by another ACRS subcommittee. During the interim period, the commitment by Vermont Yankee remains that no structural part has a factor of safety of less than 2, and by the time the long term program is completed, in 1981, the factor of safety will be 4.

Mr. Isbin said that the Committee had been provided an historical account of the position of the Committee regarding the desirability of having the containments of the BWRs of this vintage inerted. An Atomic Safety and Licensing Appeal Board Hearing resulted in the ruling requiring that the Vermont Yankee containment not be inerted. The claimed advantage of a non-inerted containment has been that it would provide an increased ability to enter the containment for the location, evaluation, and isolation of system leaks. For the first time, the ACRS has received a report of the experiences where such entrances into the containment, even with the reactor operating at a rather high power level (75%) were made. Some ten entries made since 1975 were summarized, and the beneficial effect of having a deinerted containment was also summarized. (Mr. Isbin could not say with certainty whether the Dresden 2/3 and Quad Cities 1/2 have qualified for deinerting, based on the latest interpretation of Regulatory Guide 1.7.) He recommended that the Committee should schedule a report from the NRC Staff on this matter.

Mr. Isbin noted that the Licensee does not consider a loose-parts monitoring system as an aid in safety. He suggested that the Committee request the NRC Staff to provide a report on the status of their review of the GE Topical Report, NEDO-10780-5, Development of Vibration Monitoring Systems for Light-Water Nuclear Reactors.

Mr. Isbin reported that during the 1977 inspection no cracking was observed in the feedwater nozzles and control rod drives. The NRC Staff does not plan an inspection at the next fuel outage in September or October, 1978, but does plan an inspection in 1979. He noted that there have been no residual problems with snubbers. Mr. Isbin noted that the generic items of the BWR recirculation pump potential for overspeed is still under review by the NRC Staff. General Electric submitted a report in May, 1978, which claims that a decoupler is not needed. Mr. Isbin recommended that a report from the NRC Staff be scheduled regarding their evaluation of this GE report.

Vermont Yankee is in its fifth fuel cycle, and therefore has conducted five initial startups. The NRC Staff reported its monitoring of the startups to the Subcommittee. Mr. Isbin noted that for many years ACRS has been requesting the NRC Staff to prepare criteria relating to fuel reloads. He suggested that the Committee request that the NRC Staff brief the Committee on this matter. With respect to Vermont Yankee, no unusual problems or concerns were identified. A control rod has been removed from the reactor for testing and examination, and the Committee should request a report on the results of this examination.

Mr. Isbin noted that the Licensee has not yet provided a recirculation pump trip for additional protection in the unlikely event of an ATWS. While the Licensee has made a commitment to install such a system if it is proven necessary, they remain unconvinced that such an installation is needed.

Mr. Isbin recalled that the original 7x7 GE fuel had a number of problems which were overcome by the improved 7x7 design, and later by the 8x8 design. Currently there is no indication of any fuel leaks. Fuel channel performance has not been a problem, but there were earlier problems of bypass flow-induced vibration of the poison curtains, which are no longer used. Bypass flow has now been provided by small holes in the lower tie plate, which has reduced flow vibration of the low-power radiation monitors (LPRMs). He noted that there is a continuing program of inspection of fuel channels; and the ultimate lifetime has not yet been established.

With respect to in-core instrumentation, Mr. Isbin noted that the traversing in-core probes (TIPs) have an average life time of about two years, and that the Licensee replaces the LPRMs at a rate of about 13 out of 20 per fuel cycle.

Mr. Isbin noted that the Vermont Yankee has doubled the capacity for spent fuel storage, and additional expansion will be completed before 1980, providing approximately 2000 cavities, representing an expansion of about 4 times. The safety aspects have been extensively reported by means of an NRC Safety Evaluation Report, and in the ASLB proceedings. The Subcommittee received an appeal from Diana P. Sedebotham on behalf of a coalition that had been an intervenor to the ASLB hearing. The coalition is concerned about the ultimate disposal of the spent fuel. Mr. Isbin said that some additional remarks were made in conjunction with the Mark-I containment. He requested that the ACRS Staff review the statements and determine whether follow-up through the Committee is warranted.

Mr. Isbin noted that some recommendations were made by the Licensee at the Subcommittee meeting for increased participation in licensing reviews (see Appendix XXIV).

4. Radiological Effects and Site Evaluation

Mr. Moeller, Subcommittee Chairman, noted his concern over the tone of the reported NRC Staff's briefing of the Commission relating to the CRAC-code, and said for that reason, he had requested that the NRC Staff brief the Committee on the CRAC-code (see item III preceeding).

[Note: After the NRC Staff presentation to the Committee, Mr. Moeller noted that his concerns have been satisfied.] He also noted the inadequacies of the CRAC-code:

- o inadequate for distances beyond 25 miles,
- o inadequate for sea coast sites, and
- o inadequate for river valley sites.

Mr. Isbin suggested that the Committee should consider, and attempt to develop a collegial opinion, at the 219th ACRS meeting, regarding the use of Class-9 accidents in safety and environmental evaluations.

Mr. Moeller agreed to redraft a proposed letter, regarding the use of Class-9 accidents for comparative site analysis, for consideration by the Committee at the 219th ACRS meeting.

B. Report on Meeting of the International Commission on Radiological Protection, Stockholm, May 22-7, 1978

Mr. Moeller briefly discussed his report on the meeting of the International Commisssion on Radiological Protection, held in Stockholm, Sweden, May 22-7, 1978 (see Appendix XXV).

C. Reorganization of ACRS Generic Subcommittees

The Committee approved the proposal to reorganize the ACRS Subcommittees and Working Groups into ACRS Standing and Ad hoc Subcommittees as proposed by Mr. Siess (see Appendix XXVI) The Committee agreed that the subcommittee chairmen assignments, for the Standing and Ad hoc Subcommittees as appropriate, be reviewed annually to consider rotation of these assignments. It was further agreed that the ACRS Executive Director and the Chairman may review the actual assignments on the Generic Subcommittees and make adjustments as appropriate.

D. Amendment to ACRS Bylaws

The Committee agreed to consider, at the 219th ACRS meeting, a proposed change to the ACRS Bylaws providing that the presence of at least two ACRS Members would meet quorum requirements to begin ACRS subcommittee meetings.

E. Reevaluation of ACRS Generic Matters Applicable to Light-Water Reactors

The Committee agreed to a review, as proposed by Mr. Bender, of those generic items listed as resolved and resolution pending to determine if changes are appropriate (see Appendix XXVII). It was also agreed that the Generic Items Subcommittee will review the implementation that has been made of resolved generic items. Unresolved generic issues will be assigned to the appropriate generic subcommittee to work toward resolution of the matter.

F. Subcommittee Activities

1. Indian Point — Seismic

The Committee requested that the ACRS Staff should try to arrange for presentations by Drs. Aggarwal and Sykes to explain their papers regarding the seismicity of the Ramapo Fault. A joint meeting of the Seismic and Indian Point 3 Subcommittees is scheduled for June 16, 1978.

2. General Electric Test Reactor — Seismic

A joint meeting of the General Electric Test Reactor and the Seismic Subcommittees is scheduled for July 21 and 22, 1978 in San Jose, California. Seismic matters relating to the GETR will be considered. (For background, see Appendix XXVIII.)

3. Future Schedule

A schedule of future ACRS subcommittee meetings and tours was distributed (see Appendix XXIX).

4. Reactor Operations

The Committee agreed that the Reactor Operations Subcommittee should consider, at its June 8, 1978 meeting, the NRC system for review of LERs as well as the Nuclear Power Reactor Data System (NPRDS).

G. Source Terms Used in Accident Analyses

The Committee agreed to continue, during the 219th ACRS meeting, its consideration of a letter to NRC regarding the source terms to be used in accident analyses.

H. Review of Department of Energy Facilities

Mr. Moeller suggested that the ACRS should give more active consideration to its periodic review of Hanford and Savannah River facilities. It was noted that former NRC Chairman Anders had made arrangements with ERDA (DOE) for the ACRS review of their facilities. [Note: An update of the SAR for the Hanford-N reactor will be provided to the Committee by June 30, 1978. This will provide an opportunity to review the operating experience at the Hanford-N reactor since the last ACRS review.]

I. ACRS Reports and Letters

1. Maine Yankee Atomic Power Station

The Committee advised the Commission that it believes that there is reasonable assurance that the Maine Yankee Atomic Power Station can be operated at power levels up to 2630 MWt without undue risk to the health and safety of the public (see Appendix XXX).

2. Letter to Representative Udall

The Committee approved a letter to Rep. M. K. Udall, U.S. House of Representatives, regarding the proposal for establishing an independent quasi-judicial board to review accidents at nuclear facilities (see Appendix XXXI).

3. Regulatory Guides

The Committee prepared a memorandum to the Executive Director for Operations informing him that the Committee concurred in the regulatory position of Regulatory Guide 1.136 (Rev. 1), Material for Concrete Containments (see Appendix XXXII).

The 218th ACRS meeting was adjourned at 4:01 p.m., Friday, June 2, 1978.

APPENDIXES
TO
MINUTES OF THE 218TH ACRS MEETING
JUNE 1-2, 1978

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Appendix I

218TH ACRS Meeting

Meeting Dates:
June 1-2, 1978

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Stephen Lawroski, Chairman
Max W. Carbon, Vice-Chairman
Myer Bender
Harold Etherington
Herbert S. Isbin
William Kerr
J. Carson Mark
Dade W. Moeller
Milton S. Plesset
Chester P. Siess

ACRS STAFF

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

NRC ATTENDEES

218TH ACRS MEETING - 6/1/78

Nuclear Reactor Regulation

L. Olshan
G. Cwalina
D. Kers
J. Roe
F. J. Williams, Jr.
J. A. Martin
C.R. Van Niel
R. H. Vollmer

RESEARCH
R. D. Salva
R. Blond

Div. of Systems Evaluation
W. Nischan
D. Bunch
H. Denton

Div. of Operating Reactors

E. G. Adensam
S. H. Wiess
B. Grimes
R. W. Reid

ORB 4
C. Nelson

HMB
J. E. Fairobent
E. H. Markee

DFC
S. Meyers

A-1a

APPLICANT ATTENDEES

218TH ACRS MEETING - 6/1/78

Yankee Atomic Electric

Joseph Laznow
Peter S. Littlefield
Stephen P. Schultz
Donald B. Pounder
John H. Garrity
Paul A. Bergeron
James Brinkler
John Hanour
George M. Solan
Wendell Johnson
Robert Shome
John D. Stefano

PUBLIC ATTENDEES

218TH ACRS MEETING
JUNE 1-3, 1978

Thursday, June 1, 1978 A.M.

R. B. Borsum - B&W - Derwood, MD
Andrea Fleischer - Capital Broadcast News - Washington, DC
David Gitlitz - Business Publisher, Inc. - Silver Spring, MD
Harry H. Hersey - UCS - Alexandria, VA
Steven H. Hueberle - Senate, Environmental and Public Work Committee
B. A. Maguire - Ramco - Vienna, VA
T. E. Potter - Pickard,Lowe, Garrick, Inc. - Washington, DC
Michael Rankin - Capital Broadcast News - Washington, DC
R. E. Schaffstall - GE - Reston, VA
Gary Thomas - Bangor Daily News - Washington, DC
Keith Woodard - PLG - Bethesda, MD

P. B. Haga - Offshore Power Systems

A-1-c

NRC ATTENDEES

218TH ACRS MEETING - 6/2/78

Div. of Project Management

D. B. Vassallo
F. J. Williams
R. P. Denise
J. F. Stolz
H. Rood
H. Silver

Div. of Systems Safety
E. Brooks
J. Knight
D. Jeng
M. Houston
S. Chan
I. Sihweil
J. Rajan

Standards Development
G. Robbins

Inspection & Enforcement
R. W. Woodruff

ORB-1
W. T. Russell

LPM
I. A. Peltier

DSE
J. Kelleher
J. C. Stepp
D. R. Muller
T. Bennett
W. P. Gammill
L. Retler

Div. of Operating Reactors
R. M. Reid
H. J. George
M. B. Fairtile
A. Schwencer
S. Weiss
J. Knight
B. C. Buckley
T. V. Wambach

Office of Standards Development
E. Imbro
P. Sobel

PUBLIC ATTENDEES

218TH ACRS MEETING
JUNE 1-3, 1978

Friday, June 2, 1978 A.M.

C. B. Brinkman - CE - Bethesda, MD
K. S. Kanady - Duke Power - Charlotte, NC
Joyce P. Davis - Con Edison - New York, NY
Victor C. Gonnella - Con Edison - New York, NY
Gustaaf V. Giese-Koch - TVA - Knoxville, TN
W. Walter LaRoche - TVA - Knoxville, TN
J. E. McEwen, Jr. - KMC, Inc. - Washington, DC
Jim McIlvaine - Bechtel - Frederick, MD
T. C. McMeekin - Duke Power - Charlotte, NC
C. R. Morgan - TVA - Knoxville, TN
James E. Mecca - Puget Sound Power & Light - Kirkland, WA
Robert Maum - Carborundum Co. - Lewiston, NY
Tom Statton - Woodward Clyde Consultants - Upper Montclair, NJ
R. E. Schaffstall - GE - Reston, VA
Ralph H. Talmage - Bechtel - Petaluma, CA
Don Tocher - Woodward-Clyde Consultants - Berkeley, CA
James L. Woods - Power Authority - New York, NY
George Wilverding - Power Authority-State of NY - Flanders, NY
Mark R. Wisenburg - TVA - Chattanooga, TN

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Appendix II

ACRS FUTURE AGENDA

5/30/78

ACRS MEETING PROJECT	TYPE OF REVIEW REVIEW	REACTOR VENDOR	SER ISSUE DATE
JULY			
NEW ENGLAND 1 & 2	CP	W	6/1/78
DIABLO CANYON 1 & 2	OL	W	6/1/78
S8G	SP	-	6/1/78
INDIAN POINT 3	FULL POWER	W	
AUGUST			
DAVIS BESSE 2 & 3	CP	B&W	6/1/78
ERIE 1 & 2	CP	B&W	7/3/78
RESAR-414	STD NSS	W	7/3/78
GETR	RETURN TO POWER	-	7/3/78
SEPTEMBER			
NORTH COAST	ESR	-	8/1/78
FFTF	SP	-	8/1/78
OCTOBER			
ZIMMER	OL	GE	9/1/78
SEQUOYAH	OL	W	9/1/78
NOVEMBER			
SHOREHAM	OL	GE	10/2/78
BOPPSAR/BSAR-205	PDA	B&W	10/2/78

Appendix III

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

APPENDIX III

Dear

The attached brochure describes a new fellowship program recently authorized by Congress to assist the Advisory Committee on Reactor Safeguards in carrying out its functions.

We would appreciate your assistance in bringing this program to the attention of your doctoral candidates and post doctorals who may be interested. As you will note, most of the fellows will be working in Washington under guidance of the ACRS Members and Senior Technical Staff. Those fellows located at colleges and universities, which can provide the necessary facilities, will require extensive guidance and direction in order to provide a useful contribution to the activities of the Committee.

In this connection, work on the review of specific nuclear power plant license applications will be limited for the most part to analysis of the analytical methodology and technology applied to the evaluation of nuclear facility safety.

If you need further information on the program, please call Mr. Marvin C. Gaske at the ACRS Washington office, tel. 202-634-1371.

R. F. Fraley
R. F. Fraley
Executive Director

U.S. NUCLEAR REGULATORY COMMISSION

Opportunities in Reactor Safety Research; the study of Generic Issues related to nuclear power plant safety; and the review of specific nuclear power plant license applications

Appendix IV

PROJECT STATUS REPORT
MAINE YANKEE
DOCKET NO. 50-309

FACILITY DESCRIPTION

1. Location: Lincoln County, Maine
2. Nearest population center: Wiscasset, Me which is about 3.9 miles south of the plant.
3. Type of Reactor: PWR - Combustion Engineering.
4. Licensed power level: 2440 MW(t).
5. Date Initial Criticality: October 23, 1972.
6. Date of Commercial Operation: December 28, 1972.
7. Condenser cooling water source: Back River (Atlantic).
8. Condenser cooling method: once through.
9. Licensee: Maine Yankee Atomic Power Company.
10. A/E & Constructor: Stone & Webster.

OPERATING STATUS

Plant continues at essentially 100 % power. Continuous power run has now passed about 225 days with capacity factor of 100%. Currently, the reactor core contains cycle 3 reload.

PURPOSE OF SUBCOMMITTEE MEETING

The purpose of the subcommittee meeting is to review the Licensee request to operate beyond the FSAR power of 2560 MW(t) to 2630 MW(t). The current power level of Maine Ynakee is 2440 MW(t).

Note: . 2560 is 2.74% above the current power level.

. 2630 is 7.4 % above the current power level.

EVALUATION

The NRC in its review to increase power (2630) has considered the following items:

EVALUATION (CONTINUED)

1. Analysis of accidents and transients.
2. Physics tests.
3. Fuel design
4. Ability of the plant structures to accommodate the increase.
5. Modifications of the Technical Specifications.
6. Recent operating history of the plant.

The Staff Evaluation indicates that the results are acceptable.

RADIOLOGICAL CONSEQUENCES

The Staff review of the radiological consequences of an accident is not complete. The Staff indicated that the review and documentation should be completed on or before February 3, 1978.

The problem in the radiological consequences analysis arises with the value of the calculated doses at the exclusion boundary exceeding 10CFR Part 100. The Licensee, based on new x/q values obtained during the last nine months, and using a single compartment model, calculates the dose at the boundary to be within the limits of 10CFR 100. The Staff using a different x/q value, due to different methodology from that of the Licensee and adjusting the FSAR dose calculation exceeds 10CFR 100 limit.

The Staff and the Licensee are now refining the analysis by using a two compartment model analysis which requires additonal information from the Licensee concerning spray characteristics e.g., particle size, distribution, etc. With this information and an agreed upon x/q value, a new dose value at the boundary will be calculated.

The outcome of the refined analysis may have the following consequences.

1. The dose at the site boundary at the present power level of 2440 may exceed the 10CFR 100 limit, hence operation at this level and at the 2560 or 2630 MW cannot be authorized unless modifications are made, either administrative or physical. At this time, this case has a distinct possibility of occurring.

2. The Licensee may seek a relaxation in the containment leak rate defined in the Technical Specification. Leak tests of the containment by the Licensee indicate that the Tech. Spec. value can be reduced by 50%; with this relaxation in the leak rate, the applicant perceives that the dose limits may be within 10CFR 100.

3. Another possibility to reduce to boundary dosage would be to increase the exclusion area.

ACRS GENERIC ISSUES

The SER did not include a discussion of the ACRS generic issues. This matter was discussed with the Staff and they will be prepared to discuss the issues if asked by the Subcommittee.

HARVARD UNIVERSITY
SCHOOL OF PUBLIC HEALTH

HARVARD AIR CLEANING LABORATORY
DEPARTMENT OF ENVIRONMENTAL HEALTH SCIENCES

665 Huntington Avenue
Boston, Massachusetts 02115
617-732-1164

May 29, 1978

Mr. El Igne
Staff Engineer
Advisory Committee on Reactor Safeguards
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attached are my summary comments for the May 25, 1978 subcommittee hearing on the application of Maine Yankee for permission to increase power level to 2630 MW(t). If there are questions or omissions concerning my comments, please call.

Thank you for your kind assistance to me before and at the subcommittee meeting.

Melvin W. First

Melvin W. First, Sc.D.
Professor of Environmental Health
Engineering

MWF:me

Comments of M.W. First, Consultant,

on

The Proceedings of the ACRS Subcommittee Meeting on the Request of Maine Yankee Atomic Power Station to increase Power Level to 2630 MWt. Dr. William Kerr, Chairman.

25 May 1978

In preparation for a request to increase power level, personnel of Maine Yankee conducted a new safety analysis that differed from their pre-operational safety analysis not only because of the higher power level used, but also with respect to methodology to the degree that NRC's guidelines for conducting such studies has altered in the interval of approximately 10 years. The results of the new safety analysis show that in-plant safety considerations and the outside environmental consequences of a DBA will be changed so little at the higher power level as to be negligible, and this conclusion has been confirmed by the appropriate NRC staff members. This unexpected finding (at least to me), is partly the result of the somewhat different analytical methods now advocated by NRC and partly because of operating changes; more specifically, increased coolant temperature, pressure and flow rate.

The newer methods that were used to compute the safety implications associated with a change to an increased power level have made it extremely difficult to judge the significance of the proposed changes by reference to the older calculations. As a consequence, it seems to me that the new safety analysis must stand on its own as a definitive study without reference to the older one. If this is,indeed, the situation, it suggests that this new petition should be reviewed with all of the thoroughness accorded the safety analysis of a new plant. But, if this is not the situation, we must have heard much more about Maine Yankee safety than we ever needed to know. The reason I am including this item in my comments is that it is unclear to me what the basic issues are in this proceedings, nor was I able to clarify this matter to my satisfaction by my questions. Perhaps this stems from my lack of familiarity with established ACRS procedures. Nevertheless, one wonders about the wisdom of asking for the presentation of 15 man-years of wide-ranging technical effort (claimed by Maine Yankee as their preparation time) in a time allotment of two hours.

On the merits of the application, I have a firm impression from the presentations and the documents made available to the subcommittee that (1) Maine Yankee made a careful and thorough review of the safety implications of the requested power increase; (2) that their study results sustain their conclusion that safety will not be adversely affected by the increase; and, (3) that NRC agrees with the major conclusions arrived at by Maine Yankee. Consequently, in the technical areas of special concern to me (occupational and environmental safety), I have no negative comments to communicate to the full committee nor, insofar as I understand the technical issues concerned with reactor physics, do I have any reservations.

It is my understanding that the sufficiency of the meteorological site data are still in question by NRC. It was brought out at the hearing that more refined meteorological data may cause a 10% change in the calculated value of X/Q whereas the best dispersion estimates are no more reliable than a factor of two. As this agrees with my understanding of the matter, I do not understand why NRC wishes to hold back final approval on this basis as it seems to be irrelevant inasmuch as the dispersion study made with the most recent meteorological data shows a satisfactory result. Perhaps these kinds of analytical results would be more meaningful if each value cited could be accompanied by an appropriate confidence interval.

Resolution of the habitability of the control room under LOCA conditions wasn't entirely clear to me. I understood that air in-leakage had been greatly reduced and that this would bring the operators' thyroid dose down to acceptable levels, but I wasn't able to determine if NRC was satisfied with this modification.

Melvin W. First

Appendix VI

MAINE YANKEE PRESENTATION

- Site and Plant Description
- Licensing and Operating History
- Relationship of Various Core Power Levels
- Technical Presentation
  - Incidents Considered
  - Radiological Analysis
  - Safety Analysis
  - Modified TM/LP
  - Fuel Performance
  - Effects on Major Equipment

MAINE YANKEE

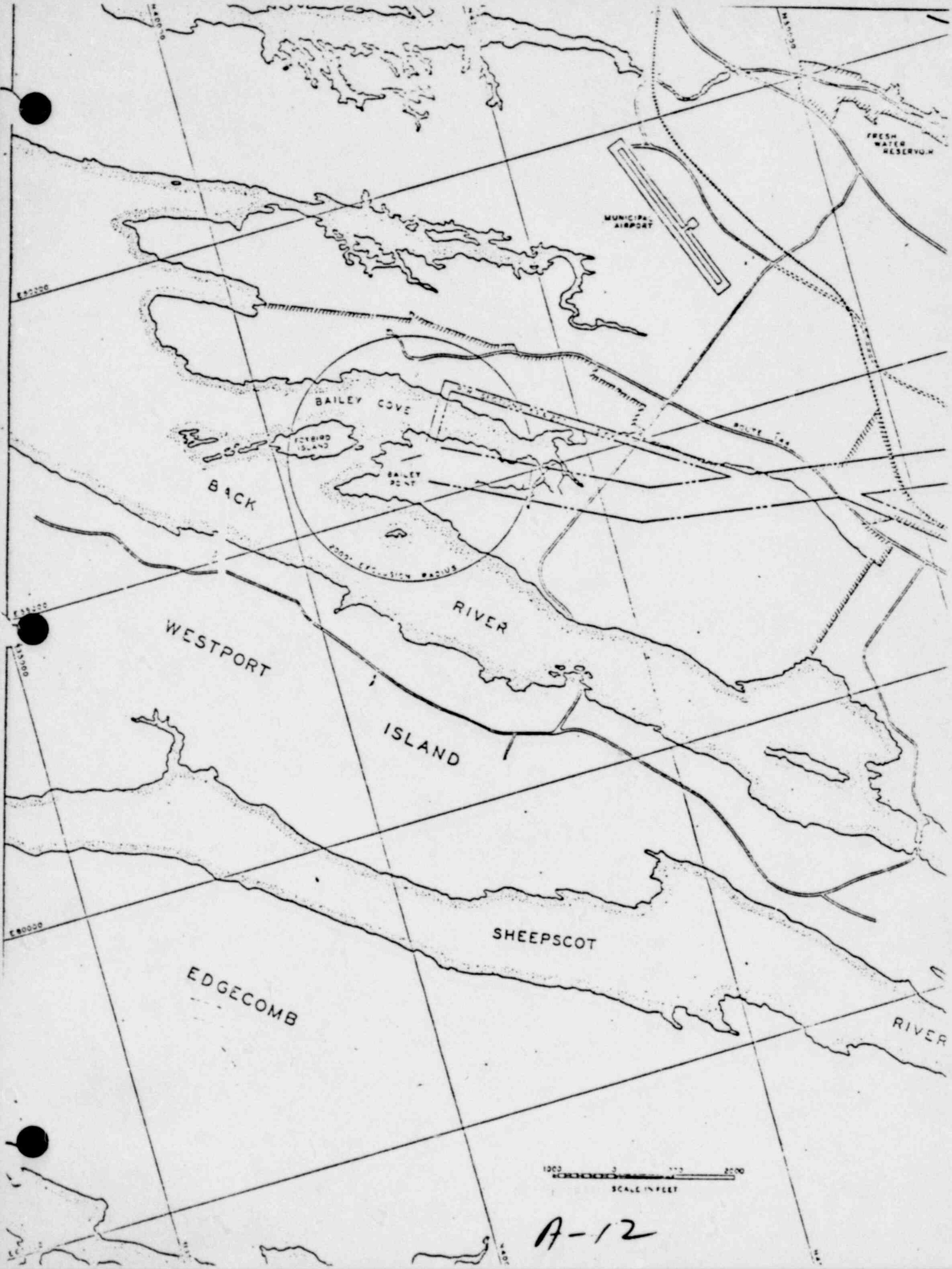
SITE DESCRIPTION

LOCATION

- WISCASSET, LINCOLN COUNTY, MAINE
- 740 ACRES, BAILEY POINT, RIDGE OF BEDROCK
- MINIMUM EXCLUSION RADIUS 2000 FT.
- LPZ 6 MILES

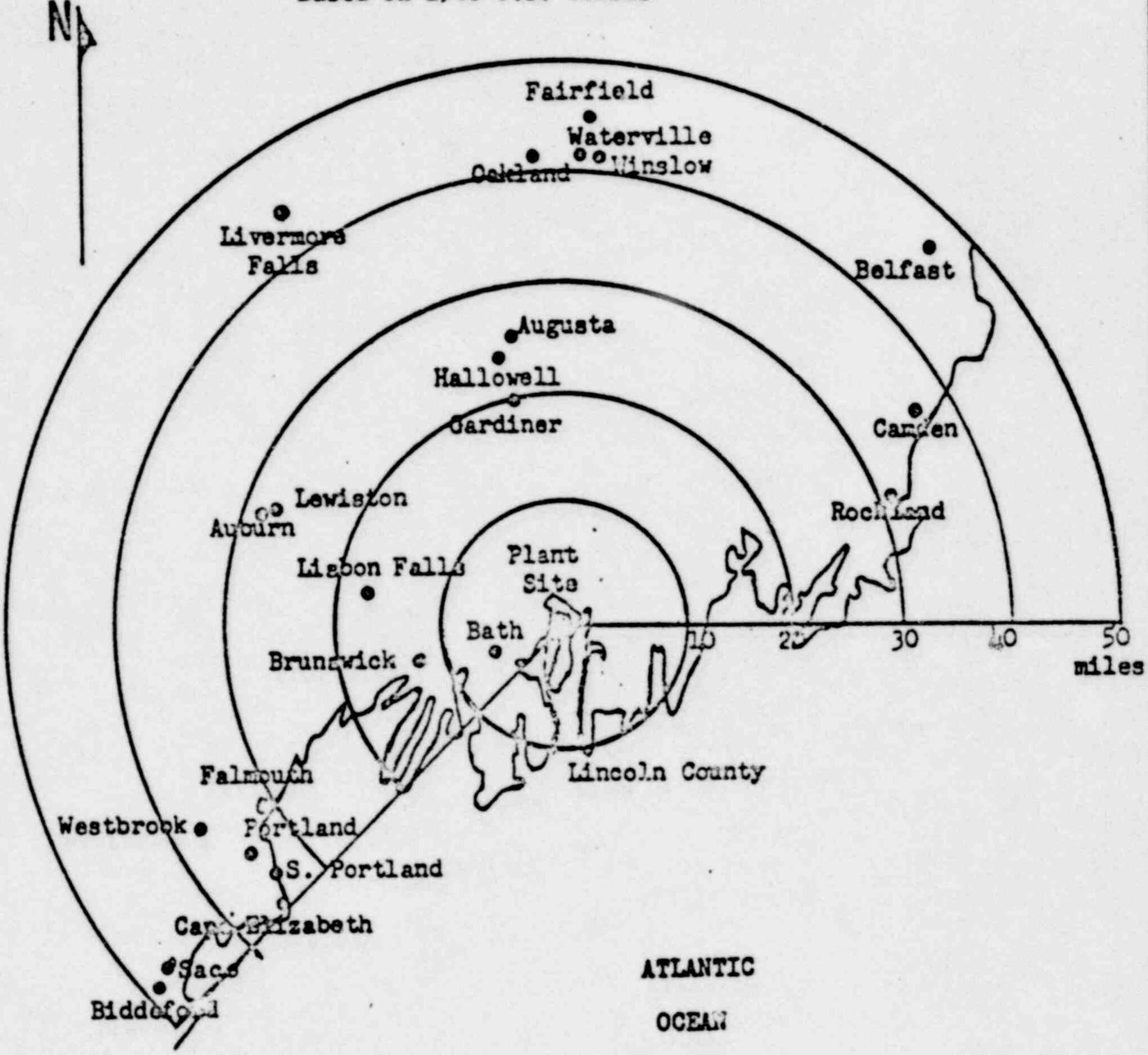
POPULATION

- TOWN OF WISCASSET 4 MILES ~2000
- LEWISTON 26 MILES ~45,000
- BATH 7 MILES ~ 12,000
- LPZ 63 PEOPLE/MI^2 YR. 2000


FRESH WATER RESERVOIR
MUNICIPAL AIRPORT
BAILEY COVE
FOXBIRD ISLAND
BAILEY POINT
BACK
RIVER
WESTPORT
ISLAND
SHEEPSCOT
RIVER
EDGECOMB
SCALE IN FEET

CITIES: POPULATION

Over 2500
0-50 Miles
Based on 1960 U.S. Census



A-13

Fig. 2.1-6

CITIES: POPULATION

Over 25,000
0-100 Miles
Based on 1960 U.S. Census

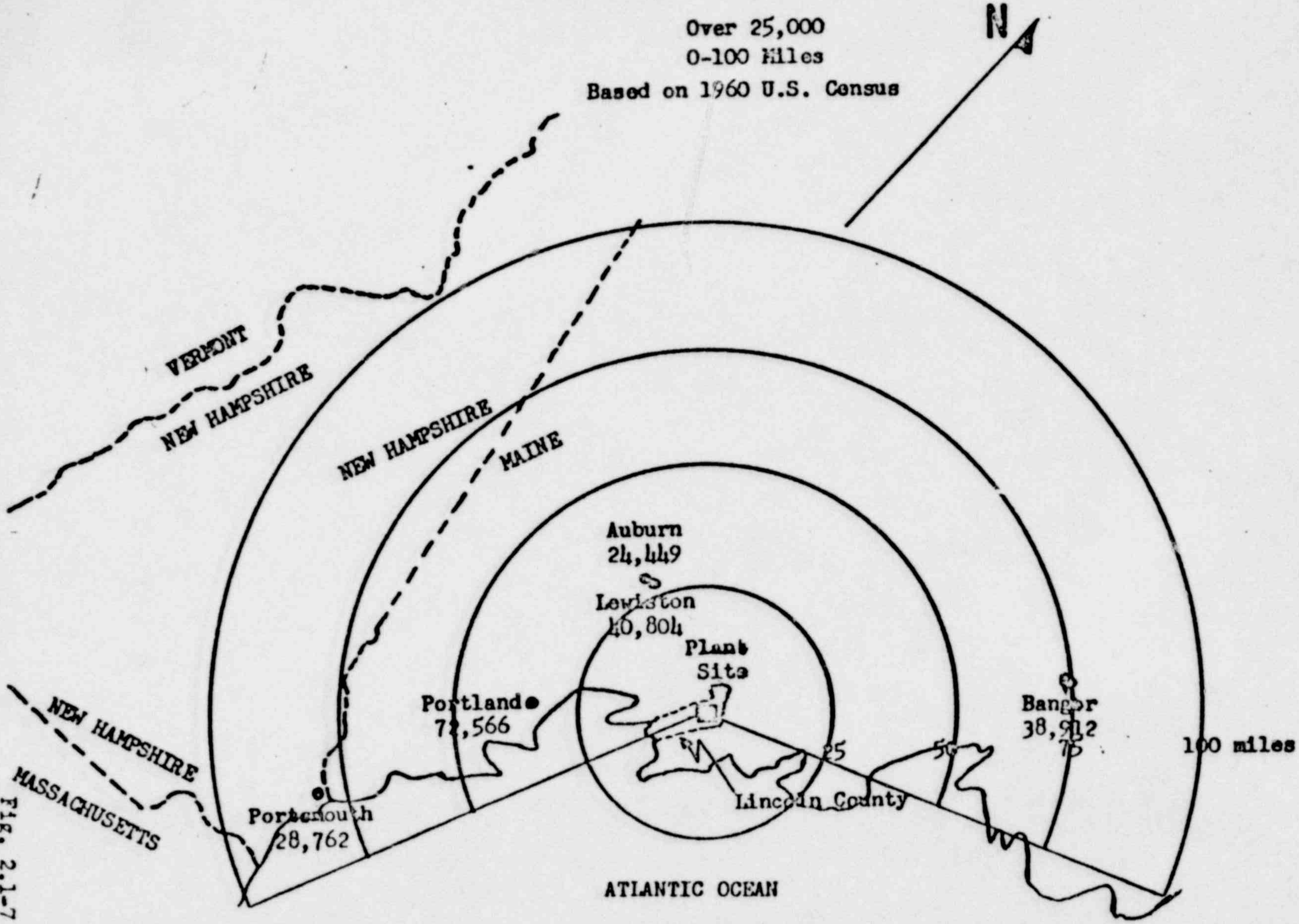
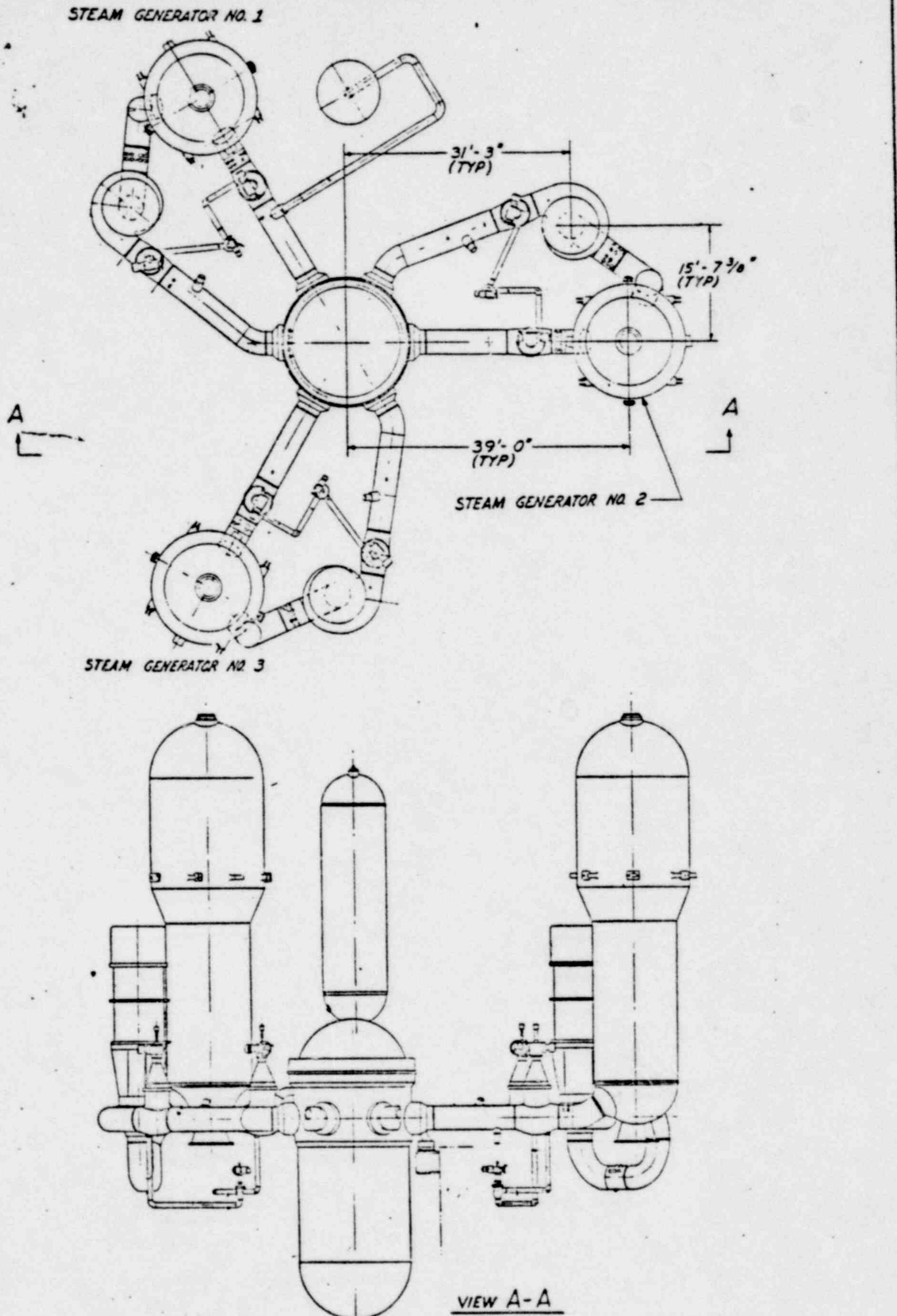
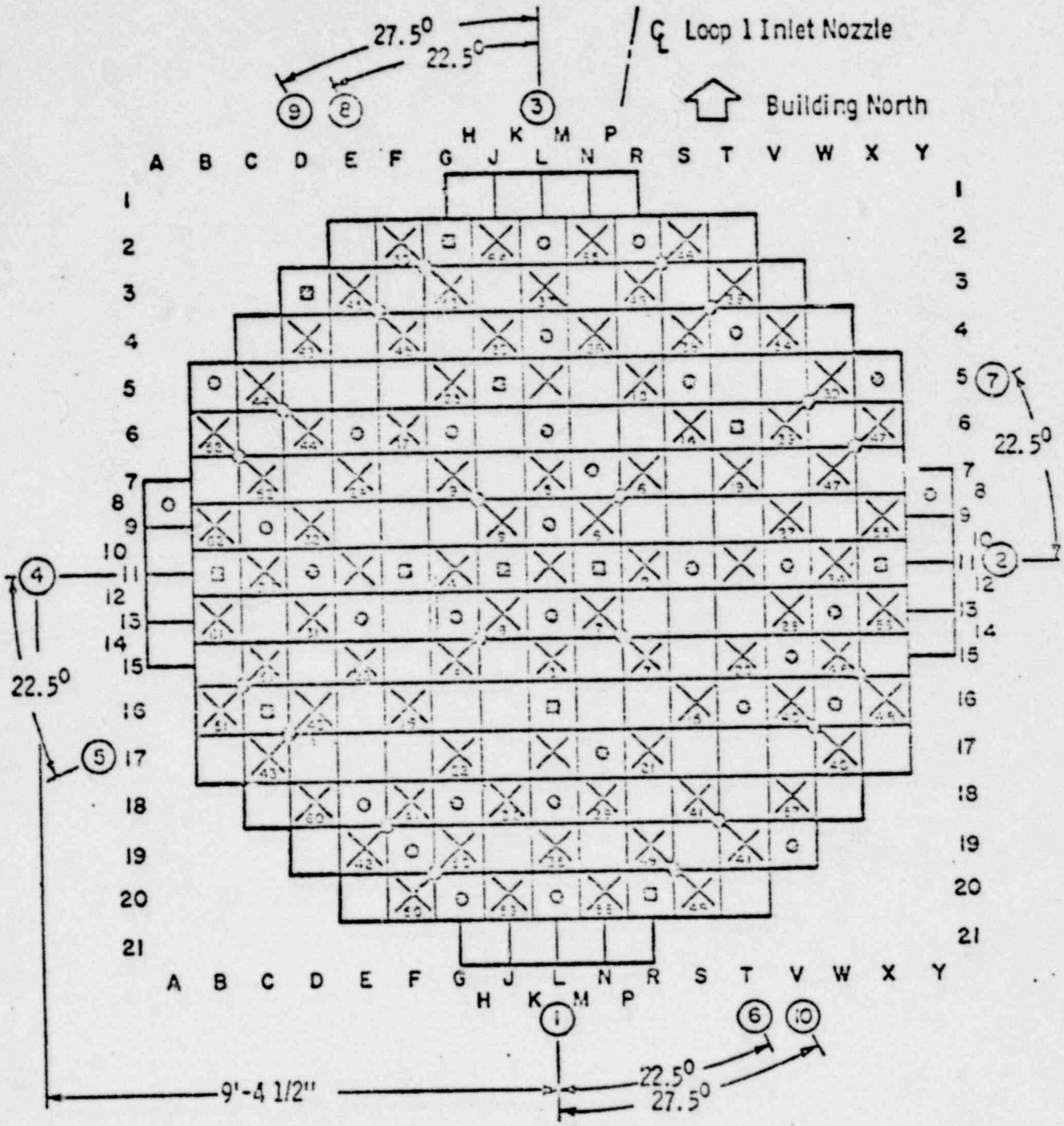


Fig. 2.1-7



MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	Reactor Coolant System Arrangement	Figure A-15



① Thru ④ - Wide Range Logarithmic
⑤ Thru ⑧ - Power Range Safety
⑨ ⑩ - Power Range Control

o - Fixed (33) Detectors
□ - Movable (12) U^{235} Fission Chamber
X - CEA

Maine Yankee reactor core

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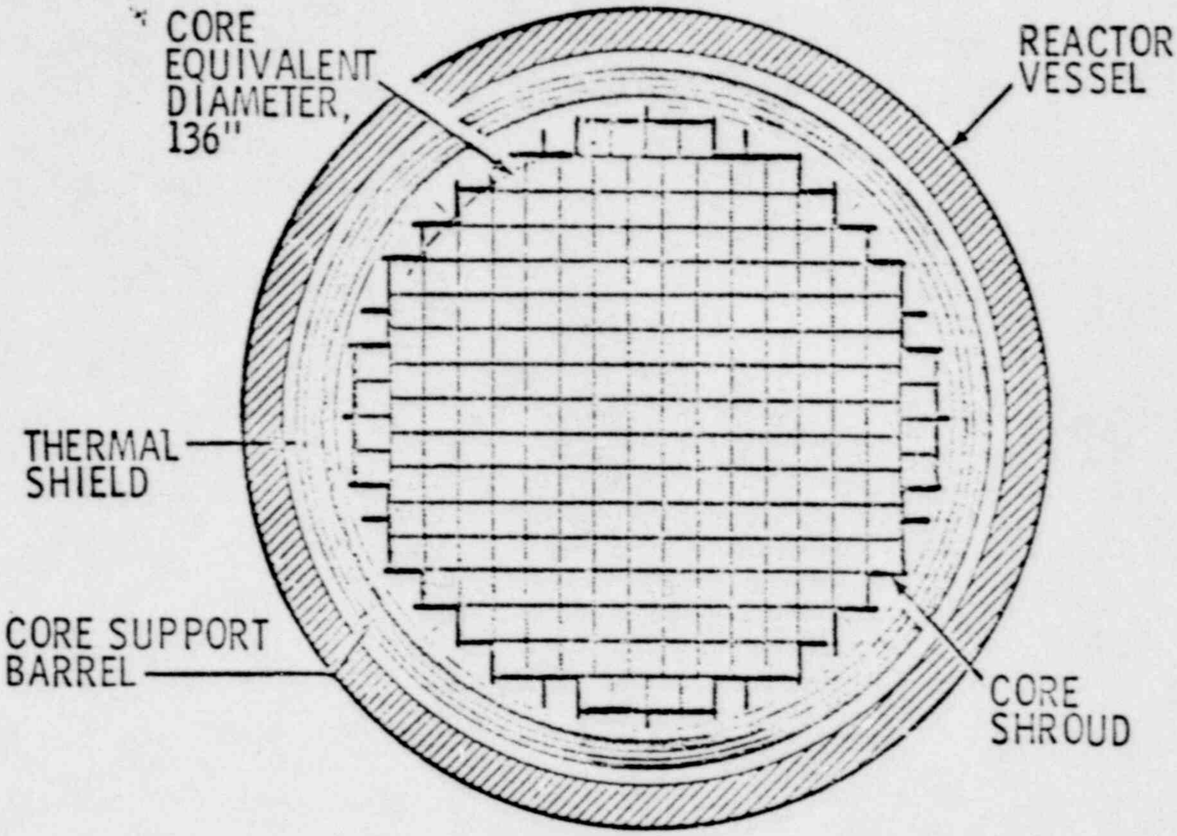

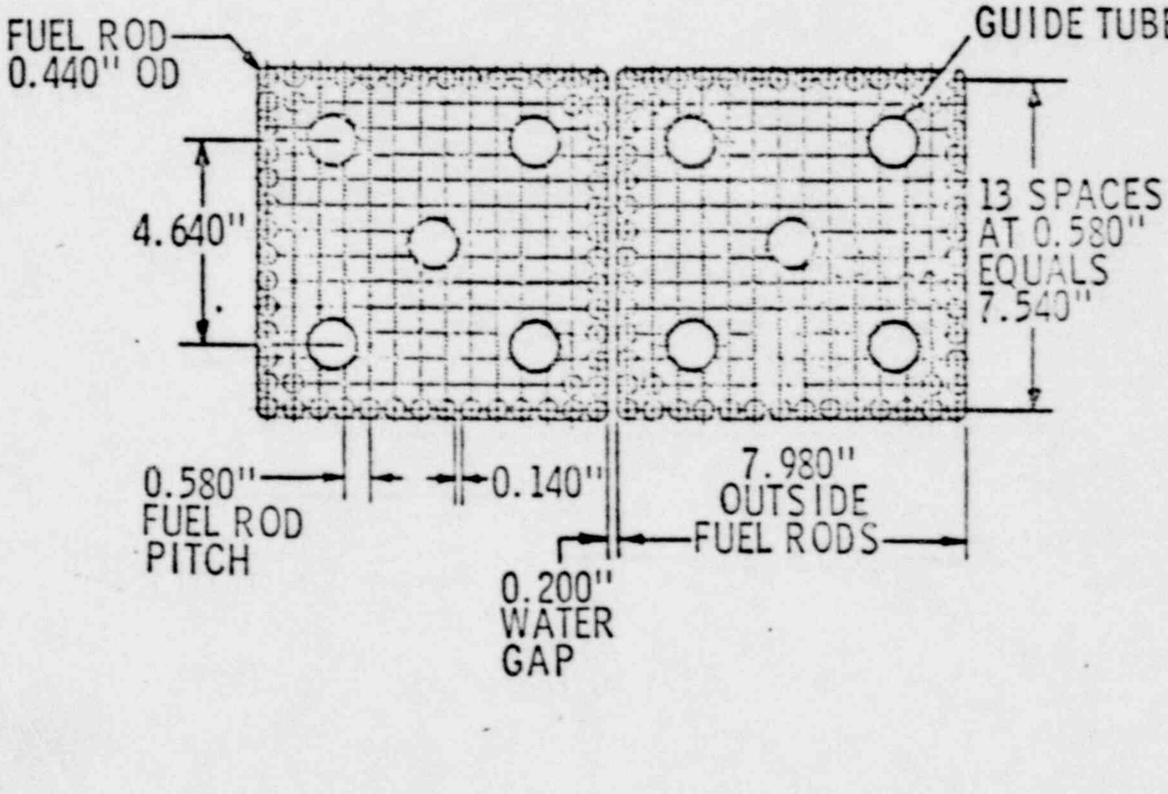

MAINE YANKEE ATOMIC POWER CO. Maine Yankee Atomic Power Station	Reactor Core Cross-Section	Figure 3.6-1

A-17

MAINE YANKEE LICENSING AND OPERATING HISTORY

o Construction Permit	October 1968
o FSAR	August 1970
o ACRS Letter	January 1972
o Staff SER	February 1972
o Operating License (75% of 2440 MWt)	September 1972
o Commercial Operation	December 1972
o Operating License (2440 MWt)	December 1973
o Cycle 1 (10,367 MWD/MT)	Dec. 1972 - July 1974
o Cycle 1A (4500 MWD/MT)	Oct. 1974 - May 1975
o Core 2 (17,100 MWD/MT)	June 1975 - May 1977
o Cycle 3 (9700 MWD/MT)	June 1977 - Present

MAINE YANKEE

Operating History

	Power (MWt)	Burnup	T (°F)	Pressure (psia)	Dates	Fuel
o Cycle 1	1830[1,2]	10,367	538 538	2000 1800	12/72-4/73 4/73-7/74	unpressurized low density
o Cycle 1A	2440[2]	4,500	535	1800	10/74-5/75	72 pressurized remaining Cycle
o Core 2	2440	17,100	537	2100	6/75-5/77	pre-pressurized high density
	2440[3]	-	542	2250	9/76	
o Cycle 3	2440[4]	9,700	542	2100	6/77-present	pre-pressurized high density
	2560[5]	-	550	2250	5/78	65 Cycle 1A

1 OL 75% of 2440 MWt 9/72, OL 2440 MWt 10/73

2 Never achieve 2440 MWt due to 1) bay temp., 2) LHGR limit, 3) RCS Act.

3 Special test

4 During Cycle 3 refueling RPS mods to allow 2630 MWt.

5 5/15/78 operated at 2560 MWt, 2250 psia and 550°F

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MAINE YANKEE

RELATIONSHIP OF VARIOUS CORE POWER LEVELS

	CORE POWER (MWT)	T_C (^oF)	PRESSURE (PSIA)	FLOW (KGPM)	NUCLEAR PEAKING
FSAR	2560[1]	546	2250	324	DESIGN
MAINE YANKEE OPERATION	2440	535-546	1800-2100	360	REDUCED[2]
STRETCH POWER	2630	554	2250	360	REDUCED[2]

1. THERMAL HYDRAULICS 2440 MWT, ACCIDENTS AND TRANSIENTS 2560 MWT
2. IMPOSING SPECIAL RESTRICTIONS ON PLANT OPERATIONS;PDIL AND MONITORING OF CORE POWER DISTRIBUTION
3. 554^{o}F AND 2250 EQUIVALENT TO 546 AND 2100

A-20

TECHNICAL PRESENTATION

- Incidents Considered
- Radiological Analysis
- Safety Analysis
- Modified TM/LP
- Fuel Performance
- Effects on Major Equipment

Incidents Considered

Category 1:

CEA Withdrawal
Boron Dilution
CEA Drop
Malpositioning of Part Length CEA's
Loss of Coolant Flow
Excess Load
Loss of Load
Loss of Feedwater
Steam Line Rupture (SLR)

Category 2:

Steam Generator Tube Rupture
CEA Ejection
Loss of Coolant
Radiological Consequences of SLR Outside Containment
Radiological Consequences of Feedwater Line Breaks Outside Containment

Category 3:

Containment Pressure Analysis
Fuel Handling
Waste Gas System Failure
Spent Fuel Cask Drop
Radioactive Liquid Waste System Leak or Failure

Appendix VII

BASIS FOR RE-EVALUATION OF THE DESIGN BASIS ACCIDENTS FOR RADIOLOGICAL DOSE ASSESSMENT

I - INCREASE IN STRETCH POWER OPERATING CAPACITY FROM 2560 MWT TO 2630 MWT

MAINE YANKEE FSAR ACCIDENTS EVALUATED AT 2611 MWT
STRETCH POWER SUBMITTAL ACCIDENTS EVALUATED AT 2683 MWT
(NET - 3% POWER INCREASE)

II - CURRENT CRITERIA FOR EVALUATING DBA'S AS PRESENTED IN THE STANDARD REVIEW PLANS FOR THE APPLICABLE ACCIDENT.

III - MOST CURRENT METEOROLOGICAL DATA AVAILABLE AT THE TIME OF THE STRETCH POWER SUBMITTAL
(DATA ACCUMULATED FOR THE 1975-1976 OPERATING PERIOD)

DBA'S THAT WERE RE-EVALUATED FOR THE
2630 MWT STRETCH POWER SUBMITTAL

1. STEAM GENERATOR TUBE RUPTURE

2. LOSS OF COOLANT ACCIDENT - DOSE ASSESSMENT
   A. LEAKAGE FROM ENGINEERED SAFETY FEATURES OUTSIDE CONTAINMENT
   B. POST LOCA HYDROGEN PURGE
   C. CONTAINMENT LEAKAGE CONTRIBUTION

3. MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT

4. FUEL HANDLING INCIDENT

STEAM GENERATOR TUBE RUPTURE
FSAR VS. STRETCH POWER SUBMITTAL

MAJOR PARAMETER CHANGES:

1. INCREASE IN PRIMARY AND SECONDARY ACTIVITY LEVELS BASED ON HIGHER POWER OPERATING LEVEL.

2. EVALUATION OF THE RADIOLOGICAL CONSEQUENCES BASED ON PRE-EXISTING AND COINCIDENT PRIMARY COOLANT IODINE SPIKING.

3. RADIOLOGICAL DOSE CONSEQUENCES BASED ON LATEST ATMOSPHERIC DISPERSION FACTORS AT THE TIME OF SUBMITTAL.

TABLE 4.11-2

OFFSITE DOSES FROM
STEAM GENERATOR TUBE RUPTURE

	(0-2 HR) Site Boundary Dose (Rem)		(0-30 DAY) LPZ Dose (Rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
CONSERVATIVE CASE	*6.4 + 0	6.0 - 1	3.0 - 1	3.0 - 2
REALISTIC CASE	1.2 - 4	3.1 - 3	6.1 - 6	1.5 - 4
CONSERVATIVE CASE WITH COINCIDENT IODINE SPIKE	1.1 + 2	1.1 + 0	5.4 + 0	5.0 - 2
CONSERVATIVE CASE WITH PRE-EXISTING IODINE SPIKE	1.1 + 1	7.0 - 1	9.0 - 1	3.0 - 2

*6.4 + 0 = 6.4×10^{0}

MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT
FSAR VS. STRETCH POWER SUBMITTAL

MAJOR PARAMETER CHANGES:

1. INCREASED PRIMARY AND SECONDARY ACTIVITY LEVELS BASED ON HIGHER OPERATING LEVEL.

2. EVALUATION OF THE RADIOLOGICAL CONSEQUENCES BASED ON PRE-EXISTING AND COINCIDENT PRIMARY COOLANT IODINE SPIKING.

3. RADIOLOGICAL DOSE CONSEQUENCES BASED ON LATEST ATMOSPHERIC DISPERSION FACTORS AT THE TIME OF SUBMITTAL.

TABLE 4.14-3

OFFSITE DOSES FROM STEAM LINE BREAK

	(0-2 HR) SITE BOUNDARY DOSE (REM)		(0-30 DAY) LPZ DOSE (REM)	
	THYROID	WHOLE BODY	THYROID	WHOLE BODY
CONSERVATIVE CASE	4.8 - 1*	2.1 - 3	2.4 - 2	1.0 - 4
REALISTIC CASE	4.4 - 6	5.4 - 7	9.7 - 8	1.2 - 8
CONSERVATIVE CASE WITH COINCIDENT IODINE SPIKE	6.5 - 1	2.7 - 3	3.2 - 2	1.3 - 4
CONSERVATIVE CASE WITH PRE-EXISTING IODINE SPIKE	5.3 - 1	3.3 - 3	2.6 - 2	1.6 - 4

*4.8 - 1 = 4.8×10^{-1}

A.28

FUEL HANDLING INCIDENT

FSAR Vs. STRETCH POWER SUBMITTAL

MAJOR PARAMETER CHANGES:

1. INCREASED FUEL ASSEMBLY INVENTORY BASED ON HIGHER OPERATING LEVEL.

2. RADIOLOGICAL DOSE CONSEQUENCES BASED ON LATEST ATMOSPHERIC DISPERSION FACTORS AT THE TIME OF SUBMITTAL.

TABLE 4.17-2

DOSES FROM FUEL HANDLING INCIDENT

DOSE POINT	REALISTIC CASE, REM THYROID	REALISTIC CASE, REM WHOLE BODY	CONSERVATIVE CASE, REM THYROID	CONSERVATIVE CASE, REM WHOLE BODY
SITE BOUNDARY (TWO HOURS)	1.6 - 04	1.4 - 03	2.6 + 1	3.4 + 0
LOW POPULATION ZONE BOUNDARY (DURATION OF ACCIDENT)	3.6 - 06	3.1 - 05	1.3 + 0	1.7 - 1

*1.60 - 04 = 1.60 x 10^{-4}

17-30

LOSS OF COOLANT ACCIDENT
FSAR VS. STRETCH POWER SUBMITTAL

MAJOR PARAMETER CHANGES:

1. INCREASED CORE HALOGEN AND NOBLE GAS INVENTORIES BASED ON HIGHER POWER OPERATING LEVEL.

2. FACTOR OF TWO REDUCTION IN THE ELEMENTAL IODINE REMOVAL RATE CONSTANT (λ_e HR^{-1}) USED FOR THE SODIUM HYDROXIDE SPRAY SYSTEM.

   λ_e USED IN FSAR ANALYSIS = 28.5 HR^{-1}

   λ_e USED IN STRETCH POWER SUBMITTAL = 10.0 HR^{-1}

3. CREDIT FOR A PARTICULATE IODINE REMOVAL RATE CONSTANT OF λ_p = 0.708 HR^{-1} FOR THE STRETCH POWER SUBMITTAL. NO CREDIT FOR PARTICULATE IODINE REMOVAL BY THE SPRAY SYSTEM WAS TAKEN IN THE FSAR.

4. RADIOLOGICAL DOSE CONSEQUENCES BASED ON LATEST ATMOSPHERIC DISPERSION FACTORS AT THE TIME OF SUBMITTAL.

5. POST LOCA HYDROGEN PURGE DOSE CONTRIBUTION WAS CALCULATED BASED ON THE INCREASED POWER OPERATING LEVEL.

6. POST LOCA ENGINEERED SAFEGUARD FEATURES LEAKAGE DOSE CONTRIBUTION.

DOSES FROM LOSS-OF-COOLANT ACCIDENT

TYPE OF ANALYSES	SITE BOUNDARY DOSE (REM) (0-2 HOURS) THYROID	WHOLE BODY	LOW POPULATION ZONE DOSE (REM) (0-30 DAYS) THYROID	WHOLE BODY
CONSERVATIVE CASE	164	9.4	12.4	0.60
REALISTIC CASE	0.86	0.03	0.05	0.001
CONSERVATIVE CASE WITH LIMITED SPRAY REMOVAL CONSTANT	198	9.5	15.9	0.62

DOSES FROM ESF COMPONENT LEAKAGE

	THYROID DOSE (REM)	WHOLE BODY DOSE (REM)
EXCLUSION AREA BOUNDARY (0-2 HR)	1.7 + 0	1.5 - 4
LOW POPULATION ZONE (0 - 30 DAYS)	1.9 - 3	2.4 - 5

DOSE CONTRIBUTION FROM POST LOCA HYDROGEN PURGE

LOCATION	THYROID DOSE (REM)	WHOLE BODY DOSE (REM)
LPZ (0-30 DAYS)	1.4	0.4

MAINE YANKE

ACCIDENT ATMOSPHERIC DILUTION FACTORS (X/Q)

EXCLUSION RADIUS (610 METERS)

DATA PERIOD	DILUTION MODEL	0-1 HOUR X/Q (SEC/M^3)	1-2 HOUR X/Q (SEC/M^3)
VALUES USED FOR FSAR SUBMITTAL	PASQUILL "F" STABILITY CLASS 1 M/SEC INVARIANT WIND	6.48×10^{-4}	6.48×10^{-4}
APRIL 1975 - MARCH 1976	SECTOR INDEPENDENT (5%)	8.07×10^{-4}	5.63×10^{-4}
JANUARY 1977- JULY 1977	SECTOR DEPENDENT (SE - 2.2%) (SSE- 3.2%)	6.24×10^{-4}	5.12×10^{-4}
JANUARY 1977- SEPTEMBER 1977	SECTOR DEPENDENT (SE - 2.4%) (SSE- 3.4%)	6.22×10^{-4}	5.05×10^{-4}
JANUARY 1977- DECEMBER 1977	SECTOR DEPENDENT (N - 2.1%) (SSE-3.4%)	5.93×10^{-4}	5.05×10^{-4}

REVISED LOCA ANALYSIS

PARAMETER CHANGES:

1. REVISED PRIMARY CONTAINMENT SPRAY MODEL.

   A. SPRAYED VOLUME - 47.34% OF TOTAL FREE VOLUME
      ELEMENTAL IODINE SPRAY REMOVAL CONSTANT (λ_E) = 10 HR^{-1}
      ELEMENTAL IODINE DF = 100

   B. UNSPRAYED VOLUME WITH GOOD COMMUNICATION
      VOLUME = 32.27% OF TOTAL FREE VOLUME
      MIXING RATE BETWEEN SPRAYED AND UNSPRAYED = 10 HR^{-1}

   C. UNSPRAYED VOLUME WITH POOR COMMUNICATION
      VOLUME = 20.39% OF TOTAL FREE VOLUME
      MIXING RATE BETWEEN SPRAYED AND UNSPRAYED = 2 HR^{-1}

2. PRIMARY CONTAINMENT LEAK RATE = 0.10% DAY^{-1}

3. RADIOLOGICAL DOSE CONSEQUENCES BASED ON 1977 METEOROLOGICAL DATA (12 MONTHS)

4. REVISED HYDROGEN PURGE DOSE CONTRIBUTION BASED ON ADDITIONAL ZINC AND NRC ZINC CORROSION RATES.

DOSES FROM LOSS OF COOLANT ACCIDENT - REVISED

	TWO HOUR SITE BOUNDARY DOSE		30 DAY LPZ (REM)	
	Thyroid	Whole Body	Thryoid	Whole Body
CONTAINMENT LEAKAGE	176		--	--
ESF COMPONENT LEAKAGE	1.7		--	--
POST LOCA HYDROGEN PURGE	N/A		2.7	1.4

Appendix VIII

TECHNICAL PRESENTATION

- INCIDENTS CONSIDERED
- RADIOLOGICAL ANALYSIS
- SAFETY ANALYSIS
- MODIFIED TM/LP
- FUEL PERFORMANCE
- EFFECTS ON MAJOR EQUIPMENT

DEVIATIONS FROM FSAR

- Core Thermal Power
- RCS Flow
- Core Inlet Temperature
- Turbine Runback
- TM/LP Trip
- Methods

MAINE YANKEE

STEADY-STATE DNBR COMPARISONS

	Core Power MWT	Tc ($^\circ$F)	Pressure (PSIA)	Flow (KGPM)	Nuclear Peaking	MDNBR
FSAR	2560	546	2250	324	Design	2.03[2]
Maine Yankee Operation	2440	535 - 546	1800 - 2100[3]	360	Reduced[1]	2.52 2.32[2]
Stretch Power	2630	554	2250	360	Reduced[1]	2.39 2.01[2]

1 Imposing special restrictions on plant operations: PDIL and monitoring core power distribution

2 Design Power Distribution

3 554°F and 2250 equivalent to 546°F and 2100

FUEL PERFORMANCE

- CLAD COLLAPSE

  RF - 20,000 HOURS, DISCHARGE 12,000 HOURS

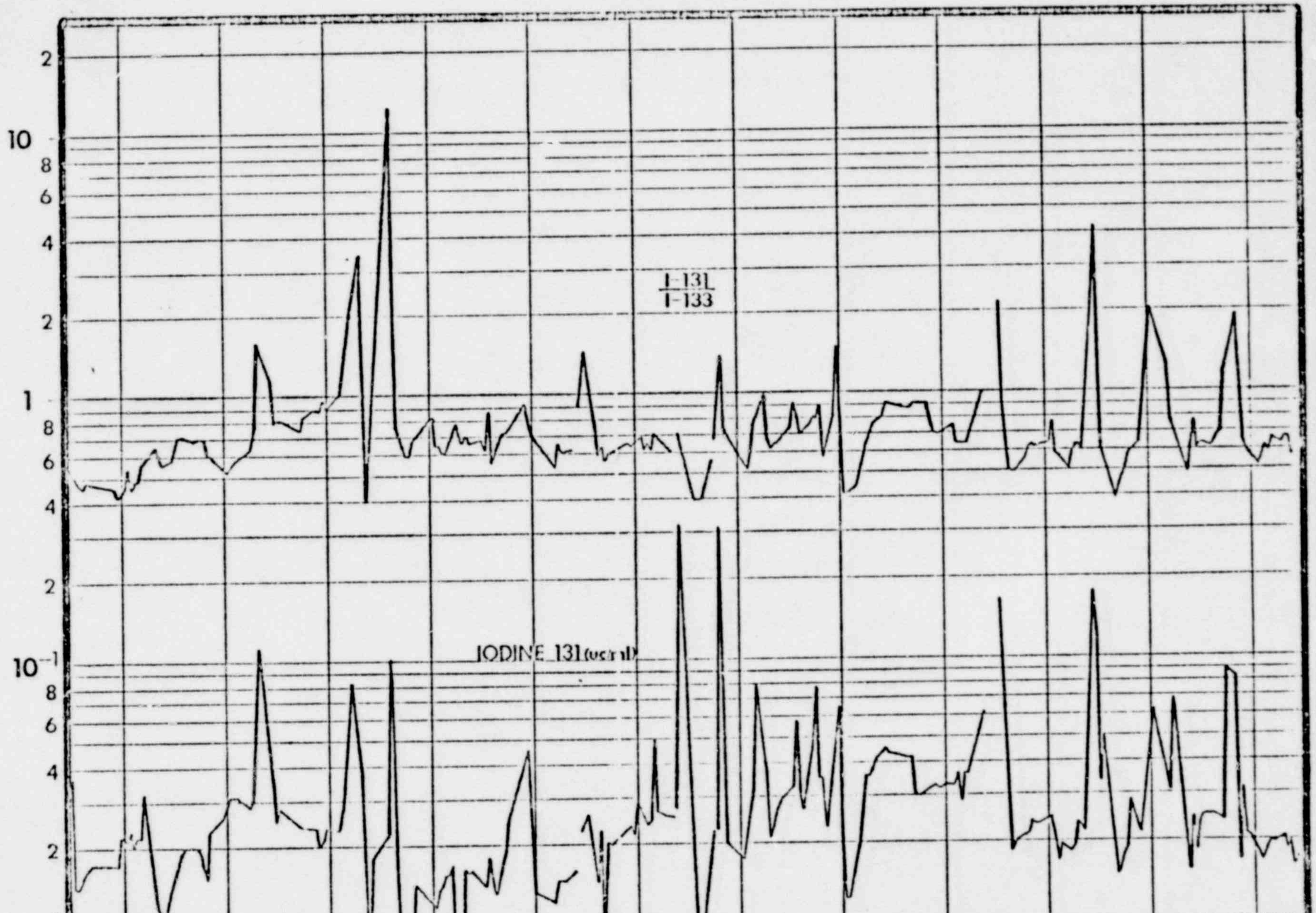
  EFGH ->34,000 HOURS, DISCHARGE 26,400 HOURS

- REACTOR COOLANT SYSTEM ACTIVITY

  CORE 2 - β,γ 2.86×10^{-1} μc/ML
  I^{131} 2.19×10^{-2} μc/ML

  CORE 3 - β,γ 2.87×10^{-1} μc/ML - 3.43×10^{-1} μc/ML
  I^{131} 1.15×10^{-3} μc/ML - 2.24×10^{-3} μc/ML

MAINE YANKEE



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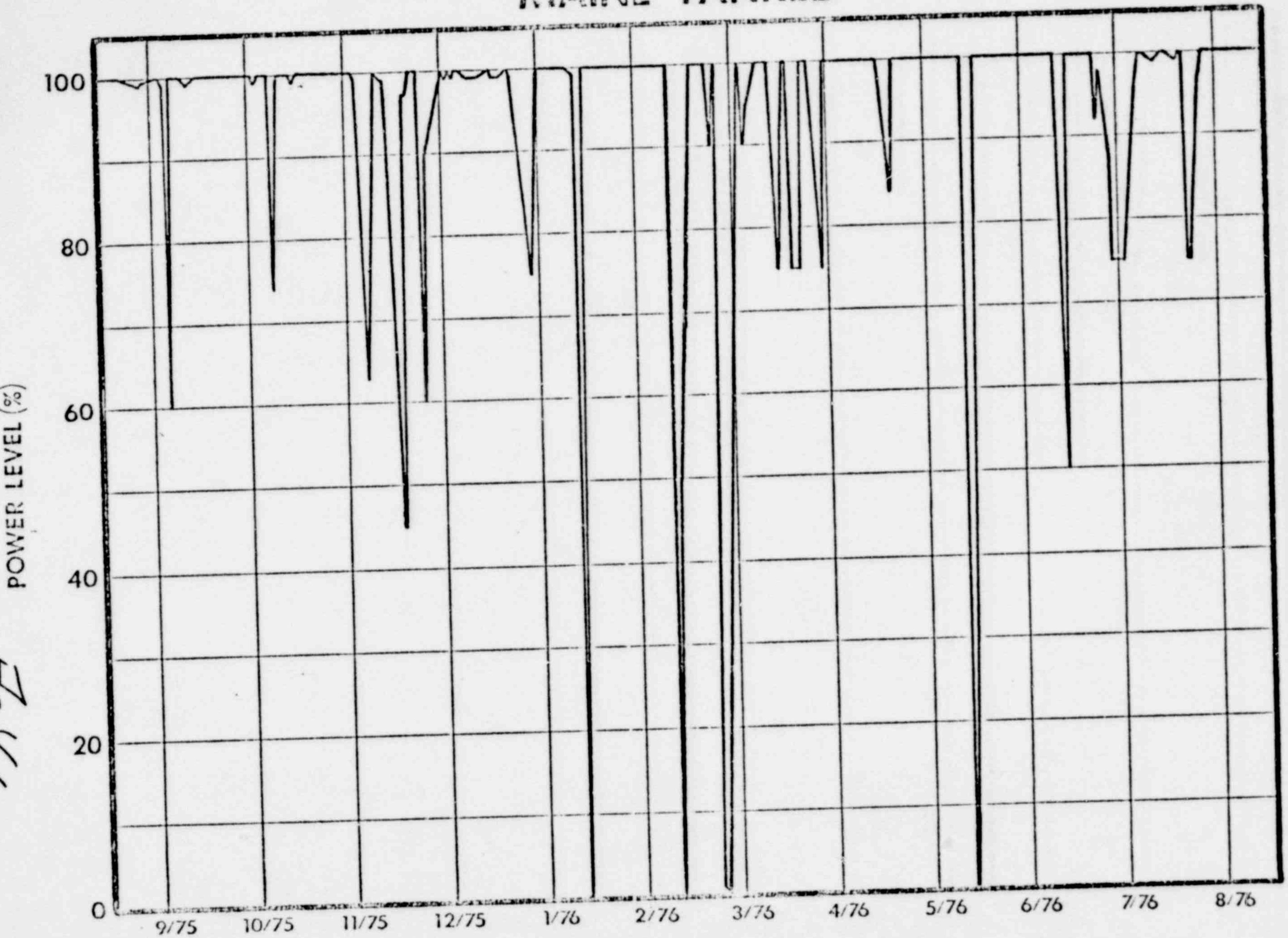
MAINE YANKEE



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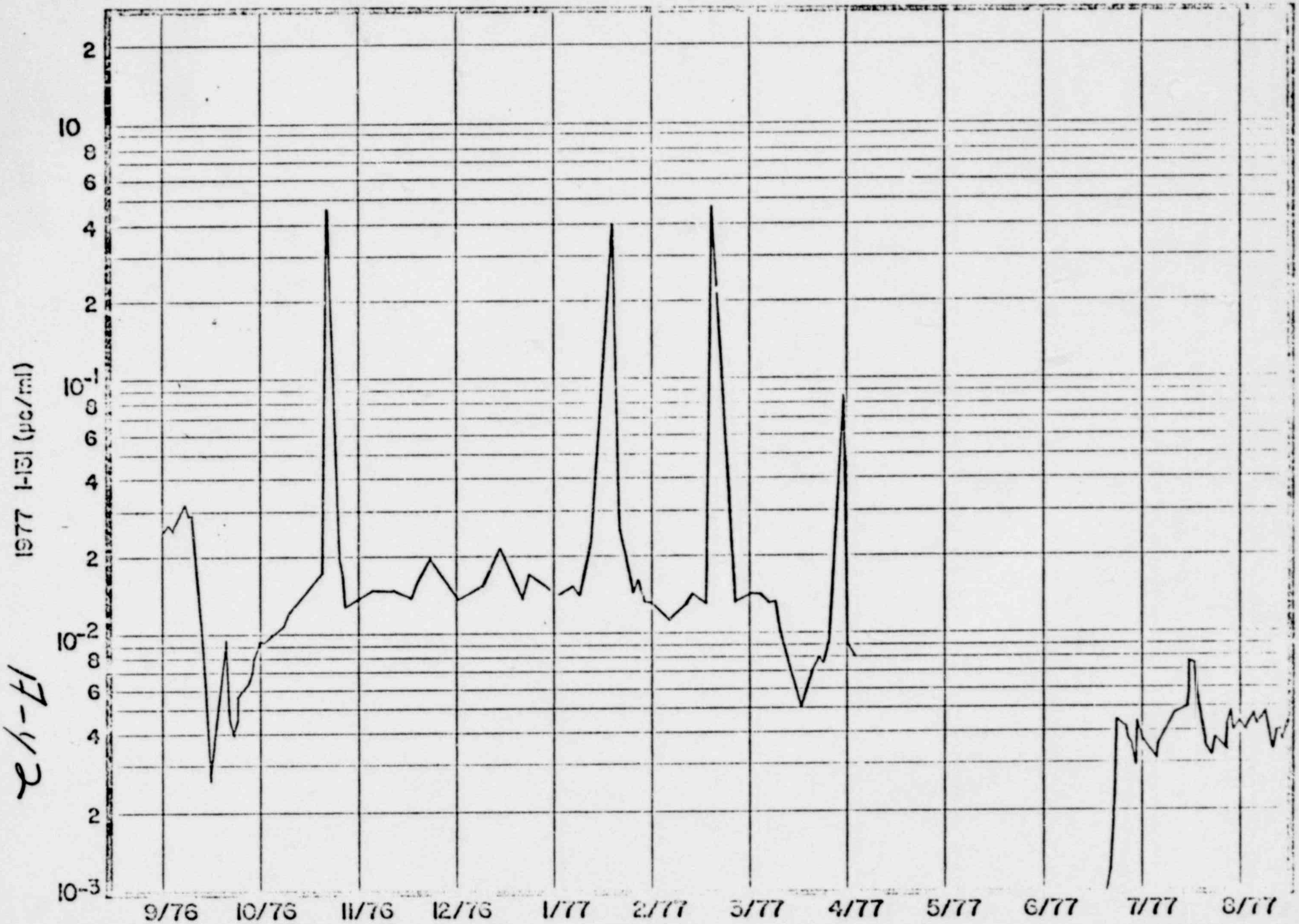
MAINE YANKEE



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MAINE YANKEE

A-43

$P_{VAR} = A\ Q_{DNB} + BT_C + C$

$P_{VAR} = 2016.1\ Q_{DNB} + 17.9\ T_C = 10102$

At 2630 MWt:

Modified System: $P_{VAR} = 1980$ PSIG

$P - P_{VAR} = 230.$ PSID

Previous System: $P_{VAR} = 2162.$ PSIG

$P - P_{VAR} = 48.$ PSID

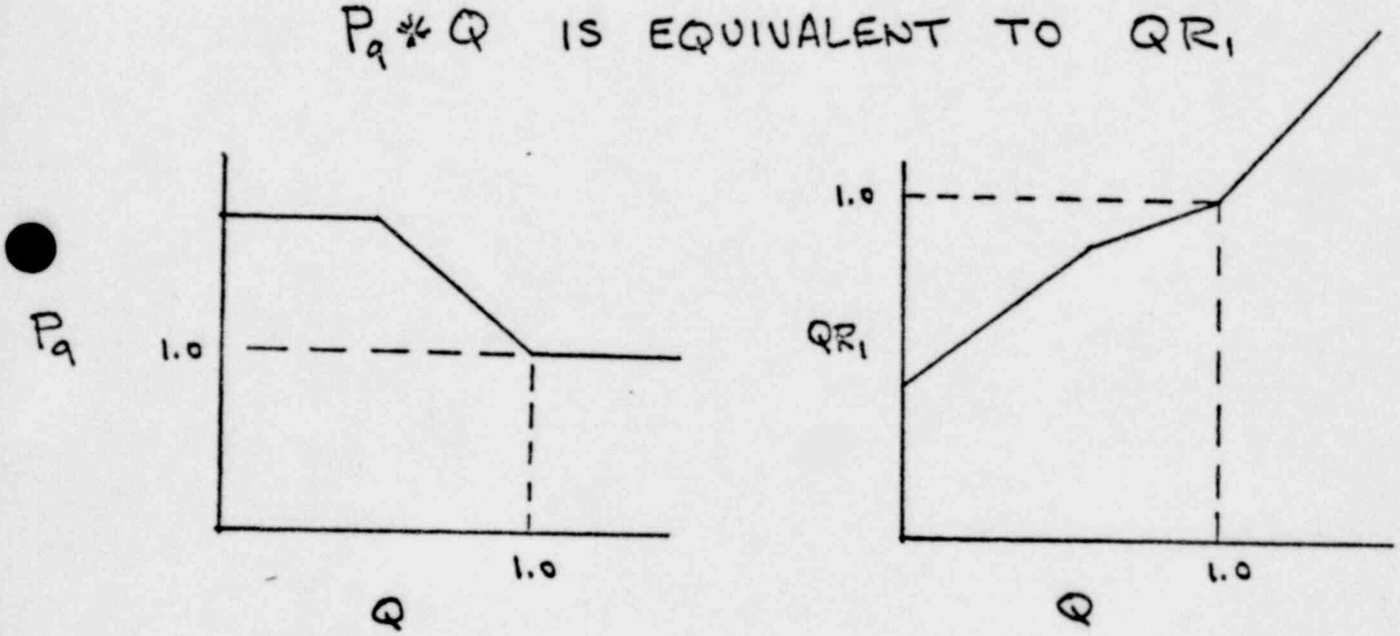
P = operating pressure 2210 psig

MODIFIED TM/LP

$$P_{VAR}^{TRIP} = A(A_1 * Q_{R1}) + BT_c + C \quad \text{(MODIFIED)}$$

$$P_{VAR}^{TRIP} = A\ P_q * Q + BT_c + C$$

$P_q * Q$ IS EQUIVALENT TO QR_1



MODIFIED SYSTEM CONTAINS AN ADDITIONAL FUNCTION THAT RELATES DNB TO SO.

Effects on Major Equipment

RCS

	Maximum Expected	Design Value
Temperature	606°F	650°F (pressurizer 700°F)
Pressure	2300 psia	2500 psia

Secondary

Temperature	530°F	550°F
Pressure	870 psia	1000 psia

Safety Related Equipment

Performance characteristics used to demonstrate that appropriate criteria is met for the various postulated accidents and transients.

Appendix IX

APPENDIX IX
Background Material for Discussion on Use of Class-9 Accidents for Alternate Site Selection

SYNOPSIS OF THE
NUCLEAR REGULATORY COMMISSION MEETING
BRIEFING ON SECY-78-137
MAY 17, 1978

The Nuclear Regulatory Commission met on May 17, 1978, for a briefing by the NRC Staff on assessment of relative differences in Class 9 accident risks in evaluation of alternatives to sites with high population densities.

Mr. Denton reviewed briefly the substance of the existing guidelines to aid in the review of alternative sites from the standpoint of the surrounding population, indicating that if the population density projected at the time of the initial plant operation exceeds 500 persons per square mile averaged over any radial distance out to 30 miles, or the projected population density over the life time of the plant exceeds 1,000 persons per square mile, special attention should be given by the NRC Staff to the consideration of alternative sites with lower population densities.

If the population density at a proposed site exceeds the guideline value specified above, the NRC Staff will institute the special review process for the evaluation of that site with a detailed look at the consequences of Class 9 accidents. However, when a proposed site is in a relatively isolated area, the NRC Staff does not give particular weight to small differences in population density values between the proposed site and the alternate site. Several questions were raised during the course of the meeting. Some of the pre-eminent questions discussed are as follows:

Commissioner Gilinsky asked why the NRC Staff is dealing only with Class 9 accidents in the evaluation of alternatives to sites with high population density.

Mr. Denton responded that the NRC Staff has been routinely dealing with Class 3 through Class 8 (Design Basis Accident) accidents. Class 9 represents all accidents beyond Class 8, including those accidents that lead to core melt.

Mr. Bunch added that in the National Environmental Policy Act (NEPA) reviews, they do discuss all classes of accidents (low probability as well as high probability accidents) in general terms. They discuss the Class 9 accidents in qualitative terms, and make reference to WASH-1400 for more detailed quantitative assessments.

In response to another question from Commissioner Gilinsky regarding the comparison between the overall risks of Class 9 accidents and the other lower Class accidents, Mr. Bunch noted that the results of the analysis in WASH-1400 indicate that the man-rem associated with Class 9 accidents will vary between 50 and a few hundred; considering the probability of occurrence of

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Class 9 accidents the risks associated with these are very low and the exposure appears to be in the same range as the occupational exposure per year basis.

Commissioner Gilinsky asked the reasons for not quantifying the Class 9 risks for a particular site after it has gone through the NEPA site selction process.

The NRC Staff responded that they have been following the guidelines provided in 10 CFR Part 100 for site evaluations. 10 CFR Part 100 provides guidelines and specific methods to calculate risks for accidents between Class 3 and Class 8. There were no special techniques available to determine the risks associated with Class 9 accidents until the development of WASH-1400. However, since WASH-1400 is still being looked at by the Lewis Committee, the NRC Staff has not performed any detailed site specific calculations to determine the risks associated with Class 9 accidents except in certain special case like Perryman. The NRC Staff believes that the techniques available now are good and adequate for assessments of relative differences in Class 9 accident risks in the evaluation of alternatives to sites with high population densities.

In response to a question from Chairman Hendrie regarding the uncertainties associated with this CRAC code model, Mr. Bunch noted that there are some uncertainties associated with this Code, and it still has some limitations. In spite of these deficiencies, he believes that this Code has unique capability of being able to organize information on site characteristics and accident releases, and then generate estimates of the consequences of accidents that reflect an integration of these widely varying but interrelated factors.

Mr. Levine commented that there are large errors and uncertainties associated with the techniques indicated in WASH-1400. In addition, the RSS consequence model was developed to estimate aggregate societal risks for a multitude of sites and not to estimate site specific features. This model may provide reasonable results within a radius of 10 or 15 miles, because the Gaussian meteorological model is accurate within that range. However, the far-out doses are manrem-dependent, and it is not clear how good this model would be in calculating the far-out doses because of lack of sufficient downwind data. He believes that the effects of downwind should be included in this model.

Indicating that this Code was developed to estimate aggregate societal risks for multitude of sites, Chairman Hendrie asked whether it will be appropriate to apply this for site specific calculations, and if so, what would be the magnitude of errors associated with the results of such calculations?

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The NRC Staff responded that the error associated with such calculations would be within a factor of 2, and they believe that a relative factor of 2 is insignificant.

In response to a question from Commissioner Gilinsky regarding the doses in certain distances, Mr. Blond noted that more than 90% of the man-rem would be between 30 and 200 miles radius of the site.

Commissioner Gilinsky asked why they do not consider Class 9 accidents in all cases?

Mr. Denton responded that after developing a competent tool to evaluate Class 9 accident risks, hopefully within a year, they may be able to consider Class 9 accidents consistently in every case.

Mr. Bunch noted that he believes that there is no need to include Class 9 accient risks in each and every case. If the population density of a proposed site is within the guideline values, then there is no need to consider Class 9 accident risks in its evaluation; considering Class 9 accidents in such cases would not provide any additional information, and furthermore, it would not be economically beneficial.

Then Mr. Sege noted that the following issues need more definitive data and clarification:

1. The role of uncertainties on the results of the calculations.

2. The appropriateness of using this Code in site specific calculations and the contribution of such calculations to the evaluation of a specific site.

3. The calculations of this consequence model don't seem to correlate with the intuition that sites with low population densities are better than those with high population densities.

4. Is it really appropriate to confine Class 9 accidents to sites with a population density above 500 persons per square mile?; do we know it for certain that there are no other notable situations where consideration should be given to Class 9 accidents so as to preclude an incomplete and misleading evaluation of that particular site?

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B. 1

HIGHLIGHTS
ACRS SUBCOMMITTEE MEETING ON
SITING EVALUATION
MAY 3, 1978
WASHINGTON, D.C.

The ACRS Siting Evaluation Subcommittee held a meeting on May 3, 1978, at 1717 H Street, N.W., Washington, D.C. The purpose of this meeting was to discuss the CRAC code model and its application in Class 9 accident risk assessments in the evaluation of alternatives to sites with high population densities. CRAC is a consequence model developed for the Reactor Safety Study (WASH-1400) for the calculations of reactor accident consequences.

Dr. Moeller (Subcommittee Chairman), Dr. Kerr, Dr. Siess, Dr. Okrent, Mr. Bender, and the ACRS consultants Dr. Gifford, Dr. Parker, and Dr. Foster were present.

The Subcommittee discussed the following aspects of the CRAC code:

1. Details of the Code.
2. Some of the experiences that have been gained in the application of this Code.
3. Some of its weaknesses and strengths in its current state.
4. The appropriateness of the use of such a Code to estimate and evaluate site specific features.
5. The extent to which the evaluation of low probability Class 9 accidents should be used in site evaluations.

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The Subcommittee recognized the following:

1. The Code provides a means to identify the "important" radionuclides released in an accident.

2. The Code indicates that the acute fatalities would occur predominantly within the 15 and 25 miles radius of the site. The latent cancer fatalities would occur within the 25 to 200 miles radius of the site.

3. The Code further indicates that precipitation can be a very important factor, because it can act as a carrier for the radioactivity or it can bring the radioactivity down to the ground.

OBSERVATION

The Subcommittee identified that the following issues need more definitive data and clarification:

1. The Code needs further refinements and improvements.

2. Since the source term is the predominant contributor to the accuracy of the results of the Code, the NRC Staff need to look more into the source term and more into the influence of various in-containment phenomena on the source term prior to its release to the environment.

3. The Code seems to have certain limitations. According to the meteorological consultants, the Code is principally useful in a dry desert region where meteorology is fairly well know. They cautioned against application of this Code, particularly in Seacoast or Lake Sites due to the difficulties in obtaining sufficient meteorological data in those regions.

4. This Code was developed to estimate aggregate societal risks for a multitude of sites and not to estimate site specific features. Several questions remain to be answered. Is it appropriate to use this Code for site specific calculations? What would be the accuracy of results when this is applied in site specific calculations?

5. What would be the changes on the results of this Code if population evacuation parameter was considered?

6. When considered alone, how important is risk assessment in evaluating the advantages and disadvantages of alternate sites?

7. Can this Code be used in identifying means to reduce risks, in addition to its use in the evaluation of risks?

8. Will the application of this Code shorten the licensing process?

9. Why is this Code used only in the environmental review, but not in the safety evaluation assesment?

The Subcommittee believes that, with further refinements and improvements, this Code would be an useful tool in the evaluation of sites for nuclear power plants.

March 7, 1978 SECY-78-137

COMMISSIONER ACTION

For: The Commissioners

From: Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation

Thru: Lee V. Gossick, Executive Director for Operations

Subject: ASSESSMENTS OF RELATIVE DIFFERENCES IN CLASS 9 ACCIDENT RISKS IN EVALUATIONS OF ALTERNATIVES TO SITES WITH HIGH POPULATION DENSITIES

Purpose: The Staff's criteria call for special consideration of alternative sites when a proposed site has a relatively high population density. The Staff has concluded that, in such instances, assessments of the relative differences in Class 9 accident risks should be included as one element of the site comparisons. This paper provides the basis for the staff's conclusion, and seeks Commission concurrence.

Background: Guidelines Used in the Review of Sites with Relatively Large Surrounding Populations

As noted in the Statement of Considerations to 10 CFR Part 100 it has been the past practice and current policy of the Commission to keep stationary power and test reactors away from densely populated areas (27 FR 3509, April 12, 1962). One basic objective of the criteria in Part 100 is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. As noted in 10 CFR Part 100, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In implementing the provisions of Part 100, we have maintained a conservative approach in evaluating plant safety and in establishing a balance between compensating engineered safety features and population density.

SECY CHAIRMAN

MAR 78 11:33

NOTE: ENCLOSURES A, B, and C MENTIONED HERE ARE NOT INCLUDED.

Contact:
...ch, CSE (49-27323)

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From time to time central station nuclear power reactors have been proposed which would be located in relatively populous areas. One such case was the proposed Newbold Island site.. In 1973, as a result of staff review of Newbold Island, we concluded that there existed an alternative site (adjacent to Salem Units 1 and 2) which was a more desirable alternative from an environmental standpoint and that the "principal factor leading to this conclusion is the fact that the population density at the Newbold site is significantly larger than at the Salem location" (Enclosure A). The proposed facility was subsequently relocated to that alternative site (and is now named Hope Creek).

As a result of the Newbold Island review, guidance was developed to aid in the review of alternative sites from the standpoint of the surrounding population (Enclosures B and C).

The substance of these guidelines is that, if the population density projected at the time of initial plant operation exceeds 500 persons per square mile averaged over any radial distance out to 30 miles, or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile, special attention should be given by the staff to the consideration of alternative sites with lower population densities.

These guidelines do not represent values that determine site suitability. Rather they are a sort of threshold or trigger to indicate the need for additional consideration of population density in the environmental reviews of alternative sites.

Specific guidelines have not been developed that provide the bases for comparing a site whose population exceeds the guideline values to an alternate site with a lower population density. Both sites may be acceptable provided a suitably designed plant is located at each site. Consequently, the balancing between the two sites is necessarily judgmental. For example, it is clear that the consequences of any given release of radioactivity to the environment (routine or accidental) would be proportional to the size and distribution of the surrounding population. However, the relative weight to be given to differences in population densities between alternative sites requires a judgment on the relative weight to be given to risks associated with routine and accidental releases.

Generally, no significant weight has been given to differences in population densities between alternative sites where both are well below the guideline values of Enclosure C. In such instances, the staff has taken that position, based on the experience gained from previous reviews of LWRs at similar sites.

However, for sites where the surrounding population is relatively large, more detailed assessments are called for. A variety of analytical models are available to aid in evaluations of site-to-site differences from the standpoint of consequences of releases of radioactivity (and which account for more factors than population density). One of these is the Reactor Safety Study Consequence Model (CRAC). While the CRAC model has been principally employed in assessments of Class 9 accidents, it has been used to assess the consequences of lesser accidents as well.

Whether any or all of these models should be used to supplement the site comparisons based on population density depend in part on the perceived benefits of siting in relatively low population density areas.

Analysis of the Role of Class 9 Accidents in Environmental Reviews

At the outset of this paper, it was noted that one stated policy objective in keeping reactors from densely populated areas is to minimize total population dose in the event of any accident (large or small). The Statement of Considerations to Part 100 also notes that events more severe than those commonly postulated as representing a reasonable upper limit in consequences are conceivable, although highly improbable. The policy of keeping reactors away from densely populated areas is one step taken to assure that the risks associated with such accidents are extremely low.

Following the enactment of the National Environmental Policy Act (NEPA), the Commission issued guidance on the treatment of accidents in environmental reports of light water reactors in the form of a proposed annex to 10 CFR Part 50, Appendix D. In that guidance (36 FR 2285 , December 1, 1971) it is noted that consequences of accidents beyond the design basis (called Class 9 accidents) could be severe, but that the probability of their occurrence is so small that their environmental risk is extremely low.

The annex stated that the consequences of Class 9 accidents need not be analyzed and, accordingly, until recently the Commission's NEPA environmental reviews have not included calculations of the consequences of Class 9 accidents. Rather, staff environmental impact statements have discussed these accidents only in a qualitative sense by restating the conclusions in the proposed annex and by briefly referencing the existence of a more quantative analysis in The Reactor Safety Study. While it is not entirely clear, the theory of the proposed annex appears to have been that NEPA requires no discussion of events with minimal risk.

While the proposed annex was never formally adopted by the Commission (for the past 6 years it has technically retained its status as a proposed Commission rule), the matter of Class 9 accidents has been discussed extensively in Commission adjudicatory decisions. These decisions [such as Shoreham, ALAB-156, 7 AEC 831, 834-835 (1973) and Zion, ALAB-226, 8 AEC 381, 407-408 (1974)] are generally construed as holding that NEPA does not require that the consequences of Class 9 accidents be considered unless it is established that there is a "reasonable probability" of the accident occurring to warrant consideration of consequences.

These adjudicatory decisions have rested primarily on the absence of significant probability of Class 9 accidents, whereas the rule relies on the absence of significant risk (which takes into account both probability and consequences). The staff's proposal in this instance is not based on a uniquely high probability of accident but rather on unique circumstances which increase the potential consequences and thus the overall risk.

The Commission's practice of not specifically analyzing the consequences associated with a Class 9 accident has received judicial sanction. [See, e.g., Carolina Environmental Study Group v. U.S., 510 F. 2d 796 (D.C. Cir. 1975), Ecology Action v. A.E.C., 492 F. 2d 998, 1002 (D.C. Cir. 1974)] It is unclear whether the basis for these judicial decisions is low risk or low probability.

In sum, it is the present state of law that there need not be any consideration of the consequences of Class 9 accidents in environmental reviews of nuclear license applications. However, this does not preclude the staff from going beyond the strict requirements of the law when it will assist in performing its NEPA review.

Recently the consequences of certain types of Class 9 accidents have been considered by the staff in connection with their reviews of two recent proceedings. In both instances, the justification for doing so was that there were novel aspects of the project such that the consequences (and hence risks) associated with potential accidents appeared to be outside of the parameters considered in the proposed annex. [cf. Citizens for Safe Power v. Nuclear Regulatory Commission, 524 F.2d 129, 1299 (D.C. Cir. 1975)]. In one of those proceedings applicants have taken strong exception to the staff efforts, arguing that the adjudicatory decisions and proposed annex preclude consideration of Class 9 accident consequences absent some showing that such accidents are credible events.

The staff believes that the high population density within the vicinity of the plant may be considered another type of special circumstance warranting a more detailed evaluation of the consequences of Class 9 accidents, especially in view of the policy objectives of Part 100.

Discussion: The staff's bases for recommending that an alternative to the Newbold Island site be considered were general in nature (see Enclosure A). Specific calculations of accident risks were not performed, either on a site-specific basis or on the basis of relative or comparative differences between Newbold Island and alternative sites. Accordingly, the support for the staff's views took the form of qualitative and judgmental arguments.

At about the same period in time, Baltimore Gas and Electric (BG&E submitted for review a proposed application for a reactor at a site in Harford County, Maryland (the Perryman site). This site was, as in the case of Newbold Island, located in a relatively populous area. As a result of the initial staff reviews BG&E was advised that the size of the surrounding population at Perryman needed to receive special consideration. The population density values at Perryman were greater than the guideline values issued after the Newbold Island decision (see Enclosures B and C)

In late 1976, the staff was informally advised by the Baltimore Gas and Electric that they still intended to tender an application for a reactor at Perryman.

In anticipation that a site would be proposed that exceeded the above-mentioned population density guidelines (the Perryman site), NRR staff began exploring various methods to evaluate comparative differences between sites. One of these methods involved the use of the RSS consequence model.* Using the RSS consequence model, the staff performed analyses of the differences between Perryman and other alternative sites from the standpoint of accident risks. Population and other data from the several identified alternate sites in the Perryman application were used for this purpose.

The results of this effort are summarized in Enclosure D which also discusses the current limitations in use of the analyses. The RSS consequence model was developed to estimate aggregate societal risks and not to estimate site specific features. Its applicability to a specific site has not been fully assessed and some specific concerns have been raised as to its applicability for such purposes.** For this reason, it should be emphasized that the results should be viewed cautiously and no significance should be drawn from small calculated differences (e.g., factors of two or so) between sites.

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* The possible uses of the RSS methods to help decision-making in areas such as this was discussed in the memorandum from Lee V. Gossick to Commissioner Kennedy of March 2, 1977.

** The Commission's Risk Assessment Review Group (the "Lewis Committee") has been established for the purpose of reviewing peer-group comments on the final RSS report and the developments in risk assessment methodology that have occurred since the report was published (see SECY-77-350).

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In spite of these limitations, we believe that this type of analysis is useful in the sense of correctly interrelating the important factors. We do believe that the results can be used to assist in the evaluation of relative differences between sites. However, the Commission should be aware that some litigants may argue that such an analysis in these special cases is inconsistent with several Commission adjudicatory decisions. We believe that the Commission should consider the appropriateness of issuing some clarifying statement that consequences of Class 9 accidents can be considered in special cases.

We had intended to include Enclosure D as part of the overall report on the staff's alternative site review portion of the Perryman application (which was issued on December 1, 1977), and to perform similar assessments in any future application where the proposed site has a population density greater than that in the guidelines of Enclosures B and C. This action was precluded by the need to resolve some reservations by the Office of Nuclear Regulatory Research (Enclosure D, if published, would require some modifications to accommodate the RES concerns). Their memorandum on this subject is provided as Enclosure E and a discussion of the memorandum is provided as Enclosure F.

The Office of Nuclear Regulatory Research is organizing a meeting in early 1978 of experts on such consequence modeling in order to develop a greater concensus on the degree of applicability of the RSS consequence model to evaluations of specific sites. We would also note that generic siting studies are part of the development plan for our reassessment of siting policy (see SECY-76-286A). These activities should ultimately provide improved bases for comparing alternative sites. On an interim basis, we recommend that assessments similar to those summarized in Enclosure D be performed in any future application where the proposed site has a population density greater than that in the guidelines of Enclosures B and C.

Recommendation: 1) Pending completion of the Commission's review of its reactor siting policy, that the Staff perform quantitative assessments of the relative differences in Class 9 accident consequences and risks in the review of alternative sites where the proposed site exceeds the general population guidelines of Regulatory Guide 4.7. The results of such assessments of the relative differences between sites, from this standpoint, would be included in any reports on such reviews.

2) That the Commission consider the appropriateness of issuing some clarifying statement to the effect that the proposed Annex to 10 CFR Part 50 Appendix D applies to land-based LWRs of the type licensed during the last decade or so and that more detailed consideration of Class 9 accidents may be warranted for other types of sites or designs. (Note, as stated on page 5, that the staff has performed limited analyses of Class 9 risks in the Clinch River and Floating Nuclear Power Plant reviews; both involve conceptual departures from a typical LWR.) A statement clarifying the annex should also include the Commission's current views on the possible value of such assessments in the evaluation of alternatives to sites with high population densities.

Coordination: CELD has provided the legal analysis for this paper. RES has reviewed the information and concurs. SD concurs. OGC and OPE comments responded to at Enclosure G. Their comment letters are included as Enclosures H and I.

Edson G. Case
Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
See attached

NOTE: Commission comments should be provided directly to the Office of the Secretary by close of business Friday, March 17, 1978.

Commission staff office comments, if any, should be submitted to the Commissioners NLT March 14, 1978, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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APPENDIX C

EVALUATION AND COMPARISON OF RELATIVE RISKS ASSOCIATED WITH LARGE ACCIDENTAL RELEASES AT ALTERNATE SITES

INTRODUCTION

Under the provisions of the Atomic Energy Act of 1954, as amended, the U.S. Nuclear Regulatory Commission regulates nuclear power reactors to minimize their potential danger to life and property. The NRC permits the construction and operation of a power reactor only when it determines that the facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Events which may be anticipated to occur one or more times during the lifetime of a facility are required to be controlled such that no significant radioactivity is released to the environment. Incidents and accidents can be prevented through the proper design, construction, and operation of the facility to assure that this goal is achieved. No design or mode of operation, however, is entirely risk free. Despite the efforts to prevent significant accidental releases from occurring, the possibility exists, however unlikely, that significant accidental releases may occur. NRC requires, therefore, that each application for a construction permit or operating license be accompanied by a detailed assessment of such postulated accidents.

The NRC staff has categorized postulated accidents into four major groups as follows:

1. Anticipated accidents with a moderate probability of occurrence, which lead to no significant radioactive releases.

2. Accidents with a low probability of occurrence, which lead to small radioactive releases.

3. Design basis accidents with a very low probability of occurrence, which lead to large radioactive releases. These accidents are postulated to evaluate the acceptability of the reactor site and to establish performance standards for the reactor's engineered safety features.

4. Accidents with an extremely low probability of occurrence, which involve failures beyond those considered in the design of the plant's engineered safety features. These are typically represented by some combination of failures which leads to core melting and containment vessel failure. These events are accounted for in the regulatory process by assuring that their probability of occurrence is acceptably low. As a result, consequences of events in this group are not specifically analyzed in most applications.

The Commission has a long-standing policy of encouraging the location of reactors in relatively isolated areas, a policy clearly stemming from a consideration of potential consequences of accidental releases. As a result of this policy it is important to review alternative sites with regard to their population differences.

DISCUSSION

There appear to be substantial differences in the number and distribution of people surrounding the applicant's alternative sites. There are also differences in other factors which affect the consequences of accidental releases (e.g., meteorology). Each of these differences was reviewed for the Perryman site and for the applicant's selected alternatives. Some differences were judged significant and these findings were included as part of the overall assessment of the alternative sites discussed in the main body of this report.

However, most of the comparisons of differences and similarities among the alternative sites were qualitative in nature. In an attempt to quantify the comparisons, the staff evaluated the alternative sites using the consequence model developed for the "Reactor Safety Study" (WASH 1400).* This model has the unique capability of being able to organize information on site characteristics and accident releases and then generate estimates of the consequences of accidents that reflect an integration of these widely varying but interrelated factors. While the model contains many simplifying assumptions and limitations the staff believes that its use can provide additional potentially valuable insights to the present alternative site evaluation.

The consequence model used in WASH-1400 (CRAC) considered three general types of effects resulting from large accidental releases. These are (1) acute injuries, such as illness or death, (2) longer term effects, such as increased risks of latent cancers, genetic disorders or thyroid nodules, and (3) economic costs, such as costs of land decontamination or relocation of people from contaminated areas.**

Whether any of these effects will be significant depends on the size of the accidental release and on such factors as speed of evacuation of potentially exposed individuals and meteorological conditions existing at the time of the release. Thus there is no single effect that represents the potential consequences of an

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*For the purpose of this evaluation, only releases to the atmosphere were considered.

**Section 5.5 of WASH-1400, "Risks from Accidental Releases," provides a summary discussion of these factors.

accidental release. One of the key features of the CRAC model is that it combines various related and unrelated situations so as to estimate the probability of a given consequence. The results generally take the form summarized in Section 5.5 of WASH-1400.

The results in Section 5.5 of WASH-1400 are not site specific, they are based on an amalgam or composite of demographic and meteorological conditions at 68 sites.* While this process may have been useful for the purposes of the "Reactor Safety Study," it makes any evaluation of site-to-site variations difficult. While the CRAC code can be used to generate site-specific consequence assessments, its utility for site specific calculations have not been fully assessed. There have been specific concerns expressed regarding its application to site specific assessments, principally arising from some of the simplified assumptions in the consequence model. There is an ongoing review of the final report of the Reactor Safety Study and comments by involved and interested parties on the study. However, as noted above, the CRAC code does permit integrated assessments, which if used judiciously, can provide improved insight as to the significance of variations in site characteristics amongst alternative sites.

For purposes of comparing the candidate sites, a 4100 MWt reactor was assumed (WASH-1400 assumed a 3200 MWt reactor). No variations in design or site characteristics were presumed to affect the probability of an accidental release. Since the principal objective was to examine the <u>relative</u> characteristics of the alternative sites, the accident categories used in WASH-1400 (Table 5-1) were not changed. The key assumptions taken from WASH-1400 in this regard were the PWR release categories and their <u>relative</u> probability. For example, it was assumed for purposes of this review that a release equivalent in magnitude to a PWR-9 in WASH-1400 was 50 times more likely than a release equivalent to a PWR-2. In this way comparisons among the alternative sites could be drawn without regard to the specific value of the probability of a major accident.

Since site specific meteorological information was available for Calvert Cliffs and Perryman, this data was used in the analysis for both sites. The data from these sites were considered to be reasonably representative of the other candidate sites for the purposes of this study. The data for these two sites were modified to reflect estimated differences in directional wind frequenc and then applied to the other sites. Site specific estimates of population distribution and habitable land (land use) were also included as input to the calculations. Some factors that are likely to be site specific were assumed to be constant; for example, a constant set of evacuation speeds was used at all sites (e.g. 1.2 mph).

<u>RESULTS</u>

The results of performing site specific assessments using WASH-1400 consequences model are summarized in Table C.1. As expected the calculations indicate site-to-site variations in the impacts of a major accidental release. For example, the economic costs associated with evacuation were computed to be about 10 times higher at Perryman than at Calvert Cliffs. The calculated mean acute fatalities at Fairhaven were about three times those at Perryman. The differences in both cases can be directly attributed to the number and location of people residing in the vicinity of each site.

TABLE C.1

RATIO OF MEAN VALUES OF CONSEQUENCES AT THE ALTERNATE SITES TO THOSE AT THE PERRYMAN SITE

onsequence*	Perryman	Bainbridge	Carpenter Pt	Calvert Cliffs	Fairhav
	Ratio of alternate sites to Perryman				
. Acute Fatalities	1.0	0.76	0.74	0.45	2.78
. Acute Injuries	1.0	1.5	1.45	0.75	2.33
. Latent Effects from Early and Chronic Exposure	1.0	1.12	1.11	0.55	1.10
Evacuation Cost	1.0	0.30	0.34	0.10	0.80
. Total Cost w/ decontamination	1.0	0.78	0.79	0.38	0.98
Total Man-rem	1.0	1.12	1.12	0.60	0.96

Th sequences do not include the health effects to the transient population o...g in facilities such as offices, institutions, etc., located relatively lose to the reactor but not related to nuclear stations operation, nor do they nclude costs associated with contamination of these facilities as a result of . large accidental release.

TABLE C.2

ACUTE FATALITIES FOR VARIOUS PROBABILITIES FOR ONE REACTOR AT ALTERNATE SITES

Chance per Reactor year	No. of early fatalities Perryman	Bainbridge	Carpenter Pt.	Calvert Cliffs	Fairha
one in 2000	<1	<1	<1	<1	<1
one in 1,000,000	<1	30	10	<1	40
one in 10,000,000	2100	980	1250	600	1800
one in 100,000,000	5700	3200	2900	2800	38,000
one in 1,000,000,000	11,000	7600	21,000	23,000	>100,000

other indices show the same trend, namely that Calvert Cliffs generally ranked lowest in computed consequences, Fairhaven ranked the highest, with Perryman somewhere in between. The total range was generally less than a factor of 5. The distribution of values from the mean was also examined. These results showed similar trends.

In an attempt to gain some additional perspective on the risks (as opposed to expected consequences) associated with large accidental releases, the distribution of a particular risk, namely the acute fatality, for each candidate site is summarized in Table C.2 (using for this purpose the numerical probability estimates of WASH-1400 for various PWR release categories). Other risks calculated by the CRAC Code can be developed as in Table C.2.

Finally, rough estimates were made of risks associated with large accidental releases from a power reactor at the 5 alternate sites, expressed as dollar costs per reactor year. The results are shown in Table C.3. For the purpose of these estimates, the various health effects (excluding acute fatalities) were assumed to have a cost measured by \$1000 per man-rem, after the fashion of Appendix I to 10 CFR Part 50. The "cost" of acute fatalities was taken as \$1,000,000. The results are therefore a measure of site differences in population distribution and to a lesser extent meteorological characteristics. The \$1000/man-rem value is used in Part 50, Appendix I as the cost/benefit index to determine if radwaste treatment augmentation is cost effective. It represents a conservative estimate of dollar costs associated with somatic health effects from low-level radiation arising from normal plant operation (probability of occurrence = 1.). For this evaluation, the cost of a man-rem should be appreciably lower. For example the BEIR Report cites a range of \$12 to \$120 per man-rem for genetically related health effects. It is unlikely that the separate costs of somatic effects would be substantially above this range, although as noted above, a value of \$1000/man-rem has been used for purposes of 10 CFR 50 Appendix I. However, intangibles involved in monetizing health effects warrant the use of a higher value for this analysis.

Quantitatively, the estimated annual public "risks," which might result from these very low probability events, ranged from \$350,000 at Calvert Cliffs to \$700,000 at the more densely populated sites. Perryman was somewhat less than twice that of Calvert Cliffs. These results do not reflect all differences in site characteristics which could have a significant effect on the total risk. For example, the possibility of high evacuation speeds at Calvert Cliffs was noted but has not been considered in the estimates of acute exposure at that site in comparison to the others. Also, the costs for property damage, with and without decontamination, were based on assumptions that land-use characteristics were similar for the 5 sites. By rough estimate, the average cost of land surrounding Perryman and Calvert Cliffs is \$3000 per acre, even though the land at each site is put to different uses (i.e., Perryman with the nearby military complex and Calvert Cliffs with extensive agriculture). A more detailed estimate could well indicate that the costs of interdicting large portions of the Aberdeen Proving Ground and Edgewood Arsenal near Perryman for a period of years (including the possible loss of employment of the 12,000 workers) would be significantly higher than the cost of interdicting the predominantly agricultural lands surrounding Calvert Cliffs.

Comparison of Alternate Sites ... Normalized Annual Dollar Cost

Consequence	Annual Occurrence Rate P	CC	CP	D	FH	$ Cost per Case	$ Annual Risk[3,4] P	CC	CP	D	FH
Acute Fatalities	7×10^{-4}	3×10^{-4}	5×10^{-4}	5×10^{-4}	2×10^{-3}	$1,000,000[1]	700	300	500	500	2,000
Man-rem	565	336	635	635	487	$1000/man-rem[2]	565,000	336,000	635,000	635,000	487,000
Property damage	5×10^{-4}	all cases				as determind by calculation	35,000	10,000	25,000	25,000	30,000
Total							$600,000	346,000	660,000	660,000	519,000

1. This value has been arbitrarily selected. A value of $200,000 per fatality was reported in "Risk Management Guide", ERDA 76-45/11 (June 1977). However, other estimates have been developed which are somewhat higher. This value, as well as other values in this table, should be regarded as illustrative only. A wide range in estimated societal costs of fatalities has been reported. The value used in this table may be on the low side, since it does not include the costs that might be associated with a medical treatment and care of individuals following a major exposure to radiation. However, the results from this table would indicate that the total monetized annual risk is not sensitive to the dollar value assumed for acute fatalities.

2. The $1000 per man-rem is an arbitrary value, selected as illustrative of the societal costs associated with the longer-term health effects that might result from an accidental release. The specific value is that reported in 10 CFR 50 Appendix I, although it is recognized that the considerations that led to the Appendix I value are not directly comparable to this example. As discussed in the text, this estimate may be on the high side.

3. The computed results do not reflect site to site variations in speed or ease of evacuation of the surrounding population. As discussed in the text there is a reason to believe that Calvert Cliffs may be somewhat better than Perryman in this respect. If true (a detailed evaluation would be required to confirm or deny this speculation), the differences between Perryman and Calvert Cliffs would be greater (for all three categories of consequences) than presented.

4. Monetized annual risks associated with low probability, potentially severe consequences events could be estimated in a variety of ways. One alternative would be to estimate costs associated with each of the several types of health effects in Table C.1. The staff is of the opinion that such an approach would not result in estimates significantly above the values estimated here, and could be significantly lower. A different approach would be to adjust these estimates to reflect perceived societal tolerance to (or alternatively, perceived aversion to) very improbable, potentially severe consequence events. Finally, adjustments could be made in the monetized risks to reflect different event probabilities for the various release categories in WASH-1400. Nonetheless, the values cited are regarded as reasonable and are illustrative of the site-to-site variations.

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CONCLUSIONS

The simplifying assumptions and limitations of the present analyses serve to emphasize that results obtained from this use of the CRAC code must be viewed with caution; their principal value in this alternative site review is to indicate trends and to assist in an evaluation of the relative magnitude of site-to-site differences. It should be emphasized that the calculations using the CRAC code would not generally be conducted in the review of alternative sites. As discussed in the main body of this report, the Perryman site has a surrounding population which is, or will be, considerably in excess of the benchmarks of 500 and 1000 people per square mile. Given this circumstance, a special, more detailed assessment was in order.

In applying these results, it is also important to keep in mind that the comparison of health effects from low probability accidents uses site location as the only variable. Health effects from alternative sources of electrical generation at the various sites were not considered.*

Nonetheless, the staff has determined that there are consistent differences among the sites from the standpoint of accident risks, but that in all cases the risks are low. Taking all factors into account, the CRAC analysis supports the conclusion that Calvert Cliffs is superior to Perryman from the standpoint of accidental releases.

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* This topic is addressed in a generic sense in NUREG-0332.

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Appendix X

THE CONSEQUENCE MODEL CRAC PROVIDES A REASONABLE EVALUATION OF THE IMPACT OF POTENTIAL REACTOR ACCIDENTS ON THE ENVIRONMENT

- CALCULATES HEALTH AND PROPERTY RISKS
- REQUIRES SUBSTANTIAL SITE DATA
- LIMITATIONS SET BY RANDOM NATURE OF SITE METEOROLOGY
- PROVIDES SIGNIFICANT INSIGHT INTO POTENTIAL ENVIRONMENTAL EFFECTS
- REQUIRES FURTHER RESEARCH TO IMPROVE LICENSING APPLICABILITY

CRAC EVALUATES TWO TYPES OF ENVIRONMENTAL IMPACTS

- HEALTH EFFECTS (POPULATION)

  IMMEDIATE DEATHS
  IMMEDIATE INJURIES
  LATENT CANCER DEATHS

  GENETIC EFFECTS

- PROPERTY DAMAGE (LAND VALUE)

  INTERDICTION

  DECONTAMINATION COSTS

  CROP LOSS

OBSERVATION: CLASS 9 ACCIDENTS ARE REQUIRED FOR THE PLANT TO SUBSTANTIALLY IMPACT THE SITE

CONSEQUENCE MODEL (CRAC) IS A PROBABILISTIC
TOOL WHICH INTEGRATES PLANT/SITE
CHARACTERISTICS TO PROVIDE
COMPLEMENTARY COMULATIVE DISTRIBUTION FUNCTIONS
(CCDF) FOR HEALTH & PROPERTY CONSEQUENCES

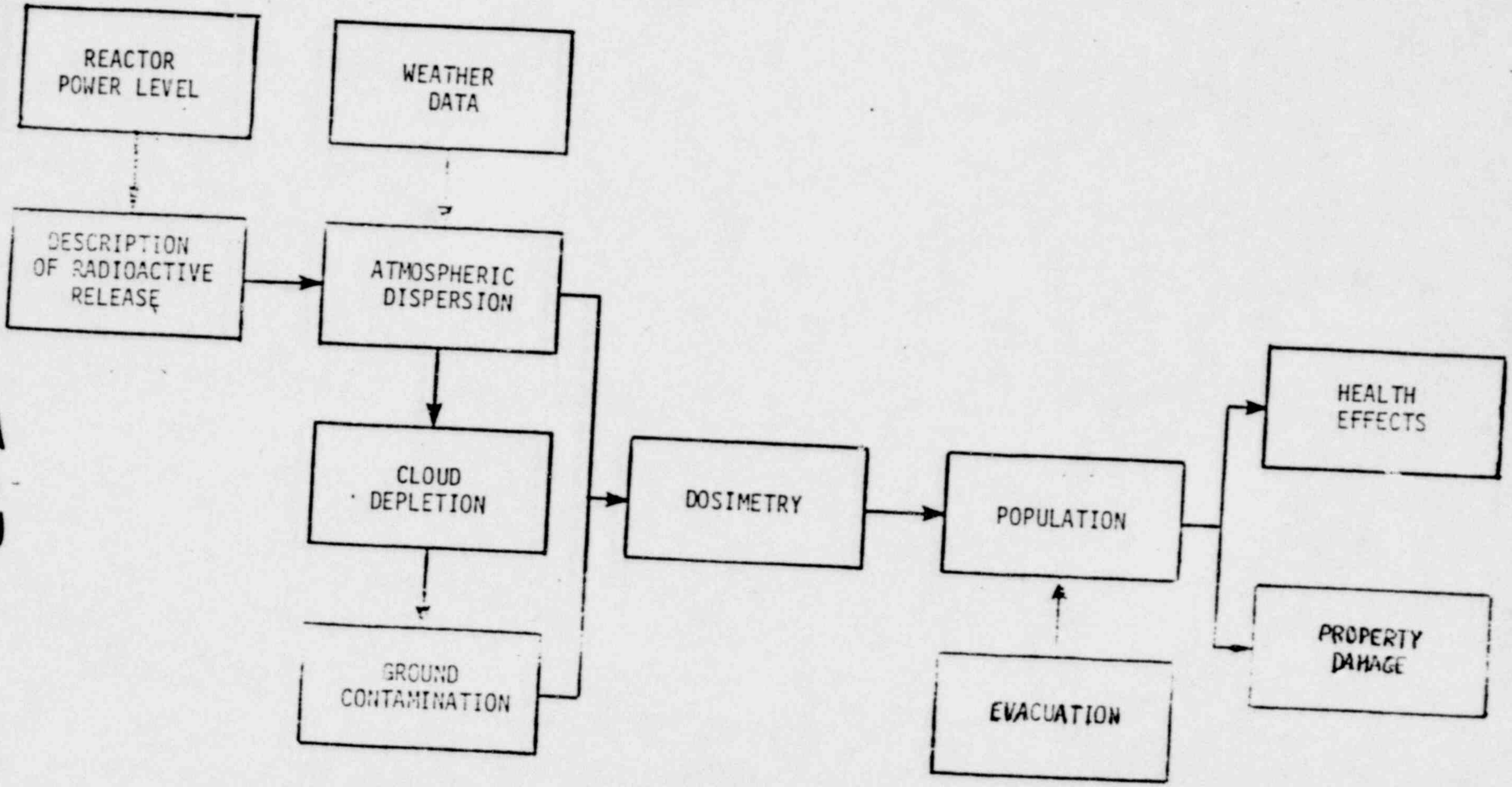
CONVOLUTION OF

1. REACTOR CORE INVENTORY
2. RELEASE CATEGORIES (9-PWR, 5-BWR)
3. WEATHER CONDITIONS (91 START TIMES)
4. POPULATION DISTRIBUTION (16 SECTORS)

ENVIRONMENTAL IMPACT OF PLANT ON SITE IS DETERMINED BY FOUR CHARACTERISTICS

- POWER LEVEL (MAGNITUDE) } RELEASE
- ACCIDENT DESCRIPTION (MECHANISM) } RELEASE
- METEOROLOGY/TOPOGRAPHY ⟶ DISPERSAL
- POPULATION/LAND VALUE ⟶ EXPOSURE

17-77



SCHEMATIC OUTLINE OF CONSEQUENCE MODEL

SITE DATA REQUIREMENTS FOR CRAC ANALYSES

1. HOURLY METEOROLOGICAL DATA FOR ONE YEAR
   A. THERMAL STABILITY
   B. WIND SPEED
   C. PRECIPITATION OCCURRENCE
2. SEASONAL DATA
   A. WIND ROSE
   B. MIXING DEPTH
3. POPULATION DATA
   A. 16 SECTORS
   B. 34 DISCRETE INTERVALS TO 500 MILES
4. LAND USAGE DATA
   A. FRACTION OF HABITABLE LAND
   B. FRACTION OF DAIRY FARMS
   C. FRACTION OF NON-DAIRY FARMS

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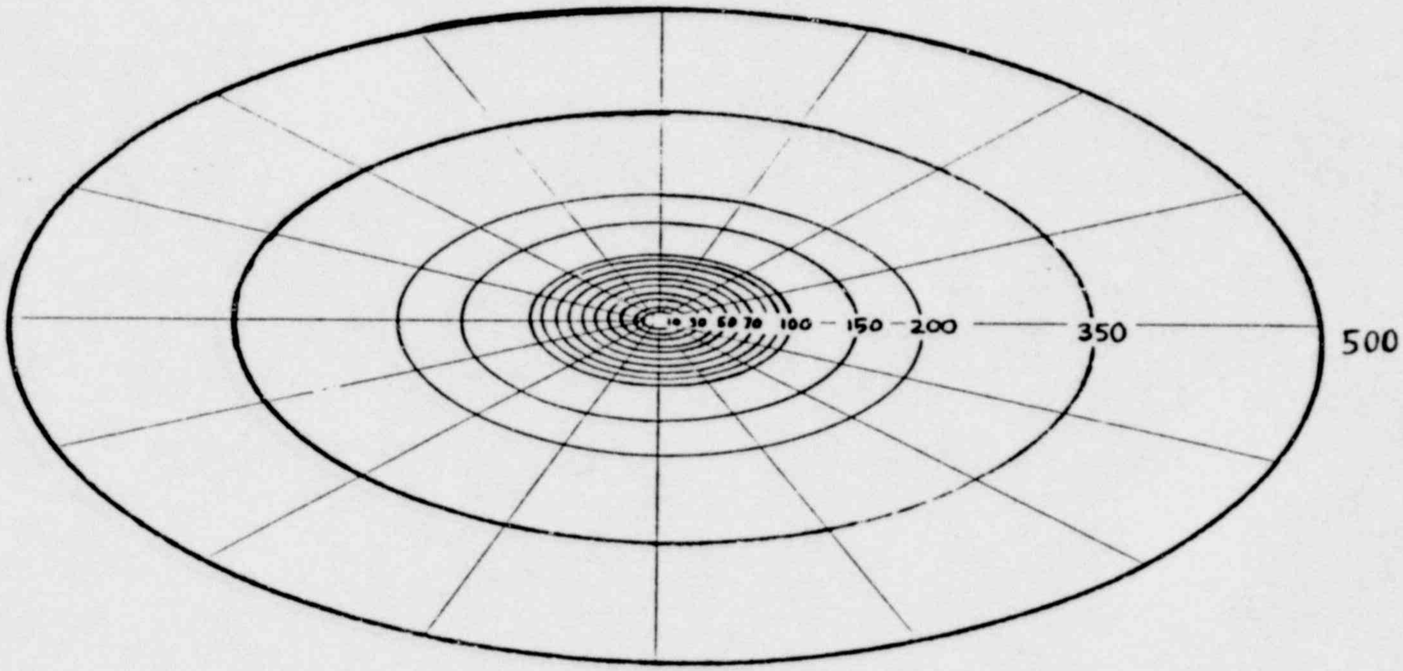
CRAC'S VIEW OF A REACTOR SITE



LIMITATIONS OF CONSEQUENCE MODEL

SITE SPECIFIC CONCERNS

DIRECTION CHANGES CAN BE IMPORTANT

- LOCAL TERRAIN
- MULTI-STATION METEOROLOGICAL DATA

WIND SPEED-STABILITY-PRECIPITATION WIND ROSE SHOULD BE INCORPORATED

LIMITATIONS (CONTINUED)

ERROR SPREADS ON RESULTS DIFFICULT TO ESTABLISH

A. MODEL IMPROVEMENT POSSIBLE IN FOLLOWING AREAS

- PLUME RISE
- RELEASE DURATION
- BUILDING WAKE
- MIXING DEPTH
- ATMOSPHERIC STABILITY CLASSES
- PARTICLE SIZE
- DEPLETION
- PRECIPITATION

B. METEOROLOGICAL DATA UNCERTAINTIES

C. POPULATION LOCATION & VARIATIONS

D. EVACUATION MODEL

E. HEALTH EFFECTS MODEL

CRAC CALCULATIONS PROVIDE SEVERAL VALUABLE INSIGHTS

IN TERMS OF CLASS 9 EFFECTS FOR 3200 MWTH PLANT

A. LATENT CANCER DEATHS (MAN-REM) DOMINATE HEALTH EFFECTS

B. LATENT CANCER DEATHS ARE DOMINATED BY CHRONIC LOW-LEVEL EXPOSURES OCCURRING BETWEEN 25 AND 200 MILES FROM POWER-PLANT

C. IMMEDIATE DEATHS ARE GENERALLY LIMITED TO AREA WITHIN 20 MILES OF POWER-PLANT

D. COSTS ARE DOMINATED BY INTERDICTION AND DECONTAMINATION COSTS WITHIN ABOUT 30 MILES OF THE POWER-PLANT

INSIGHTS

(CONTINUED)

THE IMPACT OF A REACTOR ACCIDENT IS NOT ONE PROBABILITY OR EFFECT BUT, IS A COMPOSITE OF MANY PROBABILITIES AND EFFECTS.

IN CONSIDERING ALTERNATE SITES,
THE BURDEN OF ANY DECISION SHOULD NOT BE LAID ON THE MODEL,
BUT, SHOULD BE PUT ON THE ANALYST,
USING THE INSITES GAINED THROUGH THE MODEL AS A TOOL.

CRAC AND SITING

RESEARCH PROGRAM

1. ADDRESS RECOMMENDATIONS FOR POTENTIAL IMPROVEMENTS IN MODEL

2. EVALUATE SENSITIVITY OF NEW MODELS

REGULATORY PROGRAM

ANALYZE PAST CRITICAL DECISIONS USING THIS TOOL TO GAIN FURTHER INSITE INTO RESULTS.

COMBINED EFFORT

PERFORM PARAMETRIC STUDIES TO DETERMINE RELATIVE INTERACTIONS BETWEEN KEY COMPONENTS OF PROBLEM.

SHOULD USE CRAC ANALYSIS TO FORMULATE AND ESTABLISH A MORE EFFECTIVE SITING CRITERIA.

Appendix XI

D Bunch

USE OF RSS CONSEQUENCE MODEL IN SITE REVIEWS

AGENDA TOPICS

INTRODUCTION

BRIEF REVIEW OF RSS CONSEQUENCE MODEL

NEPA REVIEW OF ALTERNATIVE SITES

POSSIBLE APPLICATIONS OF RSS CONSEQUENCE MODEL

CONCLUSIONS

A-8-1

APPENDIX XI
USE OF RSS CONSEQUENCE MODEL IN SITE REVIEWS

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CURRENT STAFF PRACTICES IN NEPA REVIEWS

- ASSESSMENTS OF ALTERNATIVE SITES INCLUDE A BALANCING OF SIGNIFICANT ENVIRONMENTAL, ECONOMIC AND OTHER ASPECTS, INCLUDING POPULATION DISTRIBUTION.

- IN THE ALTERNATIVE SITE REVIEW NO SIGNIFICANT WEIGHT IS GIVEN TO POPULATION DENSITY IF ALL SITES ARE IN RELATIVELY ISOLATED AREAS.

- IF THE APPLICANT'S SELECTED SITE IS IN AN AREA OF HIGH POPULATION DENSITY (OR INVOLVES OTHER MAJOR ENVIRONMENTAL IMPACTS) A SPECIAL REVIEW IS PERFORMED TO DETERMINE IF THE PROPOSED SITE OFFERS, ON BALANCE, SIGNIFICANT OFFSETTING ADVANTAGES.

- A DETERMINATION IS MADE THAT THERE DOES NOT (OR DOES) EXIST AN OBVIOUSLY SUPERIOR SITE.

SAFETY ASPECTS OF ALTERNATIVE SITE REVIEWS

BACKGROUND:

- ACCIDENT RISKS, OR ANY OTHER ENVIRONMENTAL IMPACT, CAN BE INTERNALIZED OR EXTERNALIZED. THEY ARE INTERNALIZED TO THE EXTENT THAT THE DESIGN INCLUDES FEATURES TO PREVENT OR MITIGATE THE EVENT (THE SOCIETAL COSTS APPEAR AS INCREASED COSTS OF ELECTRICITY). THEY ARE EXTERNALIZED TO THE EXTENT THAT EQUIPMENT MAY NOT WORK AS PLANNED OR MAY NOT COMPLETELY PREVENT A RELEASE (THE SOCIETAL COSTS APPEAR AS A CHANCE THAT THE SURROUNDING POPULATION MAY BE SUBJECTED TO AN EMERGENCY CONDITION).
- CURRENT NRC PRACTICE REQUIRES DESIGN FEATURES TO MITIGATE RISKS OF ALL RELATIVELY LIKELY EVENTS. ONLY CLASS 9 EVENTS NOT EXPLICITLY CONSIDERED IN DESIGN.
- SINCE THE SOCIETAL COSTS OF MORE LIKELY EVENTS ARE INTERNALIZED (ACCIDENT RISKS ARE REQUIRED TO BE ACCEPTABLY LOW), ANY RESIDUAL SAFETY RISKS (EXTERNALIZED COSTS OF ACCIDENTS) ARE DOMINATED BY CLASS 9 EVENTS. THE REACTOR SAFETY STUDY CONFIRMED THIS CONCLUSION.
- SITE VARIATIONS INFLUENCE THE MAGNITUDE OF ANY RESIDUAL RISKS. THERE MAY BE VARIATIONS IN THE PROBABILITY OF ACCIDENTS (DIFFERING SEISMICITY ETC.), OR VARIATIONS IN THE CONSEQUENCES OF AN ACCIDENT (SIZE OF POPULATION, ETC.).

CONCLUSION

SIGNIFICANCE OF SITE VARIATIONS DEPENDS LARGELY ON IMPACT OF THESE DIFFERENCES ON THE MAGNITUDE OF CLASS 9 RISKS. TO DETERMINE THAT A LOWER POPULATION DENSITY SITE OFFERS SIGNIFICANT ADVANTAGES FROM OVERALL ENVIRONMENTAL AND SAFETY POINTS OF VIEW INFERS A SIGNIFICANT DIFFERENCE IN CLASS 9 RISKS.

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MAJOR FACTORS INFLUENCING CONSEQUENCES OF ACCIDENTAL RELEASES

- MAGNITUDE OF RELEASE (INFLUENCES DISTANCE AT WHICH PEOPLE ARE AFFECTED).
- LOCAL AND REGIONAL METEOROLOGY
    - DISPERSION CHARACTERISTICS
    - PREVAILING WINDS
    - PRECIPITATION PATTERNS AND FREQUENCY
- TOPOGRAPHY AND HYDROLOGY
    - UNUSUAL FEATURES AFFECTING DISPERSION OR DEPOSITION
    - UNUSUAL FEATURES EMPHASIZING A SPECIAL PATHWAY
- POPULATION DISTRIBUTION
    - DENSITY, SPATIAL DISTRIBUTION, SPECIAL FEATURES (HOSPITALS, RESORTS)
- FEASIBILITY OF EMERGENCY PROTECTIVE MEASURES
    - LOCATION OF PLANT IN REGARD TO TRANSPORTATION ROUTES
    - FEASIBILITY OF EVACUABILITY, SHELTERING
- LAND USAGE
    - AGRICULTURAL
    - URBAN

EXAMPLES OF METHODS OF COMPARING ALTERNATIVE SITES

RULES OF THUMB

POPULATION VS. DISTANCE

WIND-DIRECTION WEIGHTED POPULATION

RELATIVE HAZARD INDICES

TID-14844, ACRS' SITE POPULATION INDEX

CONSEQUENCE/RISK COMPARISONS

IAEA COST EFFECTIVENESS MODEL

NRR PERRYMAN ANALYSES BASED ON RSS MODEL

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ACUTE FATALITIES FOR VARIOUS PROBABILITIES FOR ONE

REACTOR AT ALTERNATE SITES

CHANCE PER REACTOR YEAR	NO. OF EARLY FATALITIES				
	PERRYMAN	BAINBRIDGE	CARPENTER PT.	CALVERT CLIFFS	FAIRHAVEN
ONE IN 2000	<1	<1	<1	<1	<1
ONE IN 1,000,000	<1	30	10	<1	40
ONE IN 10,000,000	2100	980	1250	600	1800
ONE IN 100,000,000	5700	3200	2900	2800	38,000
ONE IN 1,000,000,000	11,000	7600	21,000	23,000	> 100,000

COMPARISON OF RELATIVE CLASS 9 CONSEQUENCES AT FIVE ALTERNATE SITES

	CONSEQUENCE OF ALTERNATE SITES/CONSEQUENCES AT PERRYMAN				
CONSEQUENCE	PERRYMAN	BAINBRIDGE	CARPENTER PT.	CALVERT CLIFFS	FAIRHAVEN
ACUTE FATALITIES	1.0	0.76	0.74	0.45	2.78
ACUTE INJURIES	1.0	1.5	1.45	0.75	2.33
LATENT EFFECTS FROM EARLY AND CHRONIC EXPOSURE	1.0	1.12	1.11	0.55	1.10
EVACUATION COST	1.0	0.30	0.34	0.10	0.80
TOTAL COST W/ DECONTAMINATION	1.0	0.78	0.79	0.38	0.98
TOTAL MAN-REM	1.0	1.12	1.12	0.60	0.86

MONETIZED COMPARISON OF RESIDUAL ACCIDENT RISKS*

($/REACTOR YEAR)

	0-30 Mile Population	Acute Fatalities	Man-Rem	Evacuation/ Decontamination	TOTAL
Perryman	2.9 x 10	$700	$565,000	$35,000	$600,000
Calvert Cliffs	2.7×10^{5}	300	336,000	10,000	346,000
Carpenter Pt.	1.1×10^{6}	500	635,000	25,000	660,000
Bainbridge	1.2×10^{6}	500	635,000	25,000	660,000
Fairhaven	3.5×10^{6}	2000	487,000	30,000	519,000

* Based on 10^6 $/fatality, $1,000/man-rem, economic costs as calculated from CRAC; no special weighting of any scenarios according to probability or consequences; costs do not include loss of generating capacity or loss of major nearby industrial facilities, if any.

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POSSIBLE APPLICATION OF RSS CONSEQUENCE MODEL

ESTIMATION OF AGGREGATE RISKS

. NEPA REVIEWS OF ALTERNATIVE SITES

EMERGEN PLANNING

SITING CRITERIA

SITE-SPECIFIC ASSESSMENTS

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CONCLUSIONS

- THERE ARE SIGNIFICANT LIMITATIONS IN THE RSS MODEL. HOWEVER, IF CARE IS TAKEN IN SELECTION OF INPUT MODELS/PARAMETERS AND IN INTERPRETATION OF RESULTS, IT CAN BE A USEFUL TOOL IN ASSESSING SITES.

- USE OF THE RSS MODEL DOES NOT APPEAR WARRANTED IN CASE REVIEWS, EXCEPT IN UNUSUAL SITUATIONS.

- EFFORTS UNDERWAY AND PLANNED WILL EXPLORE THE EXTENT TO WHICH USE OF RSS CONSEQUENCE MODEL WILL AID IN THE DEVELOPMENT OF IMPROVED SITING CRITERIA AND BASES FOR EMERGENCY PLANNING. IT DOES APPEAR THAT THE MODEL CAN BE USED TO EVALUATE AND/OR DEVELOP "FIGURES OF MERIT" FOR SUCH ASSESSMENTS.

Appendix XII

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

May 22, 1978

ACRS
ACRS Technical Staff

PROPRIETARY ASPECTS OF DUKE/OCONEE SAFE SHUTDOWN SYSTEM – Fig 1.3-1 only!

In addition to the normal proprietary aspects of the Duke/Oconee Safe Shutdown System, the system description attached hereto gives information sensitive to plant security and should be treated accordingly. Proprietary data is, of course, withheld from public disclosure.

Ragnwald Muller
Senior Staff Engineer

Attachments:

(1) R.K. Major April 6, 1978 Memo "Oconee Nuclear Station - Safe Shutdown System (SSS)"

(2) Duke Power Letter dated February 1, 1978 with Proprietary Attachment -- Control No. 780390049

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APRIL 6, 1978

ACRS
ACRS Technical Staff

OCONEE NUCLEAR STATION - SAFE SHUTDOWN SYSTEM (SSS)

The Oconee Station was designed during the mid-60's. Since that time staff requirements have changed, especially in the areas of fire protection, physical security, and flooding of the turbine building. Oconee is currently being reviewed in each of these areas. In each case the review is concerned with the capability to safely shutdown the plant if the Oconee turbine building were lost or if the systems necessary to shut the plant down were compromised. The proposed installation of the SSS would provide an independent shutdown capability for the Oconee Station and would resolve an area of concern common to the three separate reviews currently being performed.

Envisioned is a separate building containing an independent safe shutdown system. The system would be able to bring all or any combination of the three Oconee units to a shutdown condition in response to specific accident or sabotage scenarios. The system is not designed as a substitute for the current emergency core cooling systems nor does it provide additional redundancy for ECCS equipment. Duke Power makes it clear that the Oconee Nuclear Station is considered a unique situation and may require such a system where other plants do not.

The Oconee Turbine Building contains safety-related systems that provide either power to or cooling water for Class I shutdown systems. Under 10 CFR 73.55 sabotage proteccion would be required for the turbine building. Duke Power feels that adequate protection could not be economically or feasibly provided and if provided it could result in difficulty in performing normal operations. Flooding of the turbine building from external causes or a break in a condenser circulating water system waterbox could disable safety related equipment as well as the normal feedwater system and possibly prevent an orderly cooldown. At one time a turbine building drain system was proposed, but the need for such a system can be eliminated by the proposed safe shutdown system. The SSS can also be used as a redundant shutdown system in the event of a fire and eliminate the need to remove and reroute safety system cables.

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The SSS provides an alternate and independent means to achieve and maintain a hot shutdown condition for all three units. The system is independent of the current shutdown capability, except for the existing remote shutdown panels which would be replaced. The SSS will be able to maintain hot shutdown in all units for a period of 3.5 days without any damage control measures. The system components and the associated structure are designed to Class I seismic requirements.

The system concept is to provide safe shutdown capability by maintaining adequate primary and secondary system inventory. The Oconee reactor coolant system can provide adequate natural circulation flow for decay heat removal in the event of a loss of normal station power. The secondary side steam relief valves will provide an atmospheric heat dump. Sufficient instrumentation will be provided to allow an orderly progression of each unit to hot shutdown conditions. Heating, ventilation, air conditioning, lighting, and communications services will be provided for the safe shutdown facility. An independent diesel electric and battery power system will be provided for the SSS.

Three major subsystems comprise the Safe Shutdown Facility, namely the Emergency Makeup System, the High Head Auxiliary Service Water System, and the Safe Shutdown Facility Power System. The Emergency Makeup System provides borated makeup water to the reactor coolant system from the spent fuel storage pool.

The High Head Auxiliary Service Water System (HHASW) provides feedwater in the event both the normal and auxiliary feedwater systems are unavailable. The suction for the HHASW pump will be taken from the component cooling water system.

The Safe Shutdown Facility includes several and one DC power systems. These systems supply the power necessary for the hot shutdown of the reactor as well as for continuous operation of the security system, in the event of a loss of power from all other power systems. It includes a diesel-electric generator unit, switchboards, a load center, a motor control center, panelboards, battery chargers, an inverter, relays, control devices, and interconnecting cable.

The initial reaction of the staff has been favorable. Duke Power wants approval of the concept before detailed design work is started. It is estimated that design and installation of the SSS would take about 30 months. Duke Power will provide interim protective measures until the SSS is completed.

Richard K. Major
Assistant Engineer

DELETION

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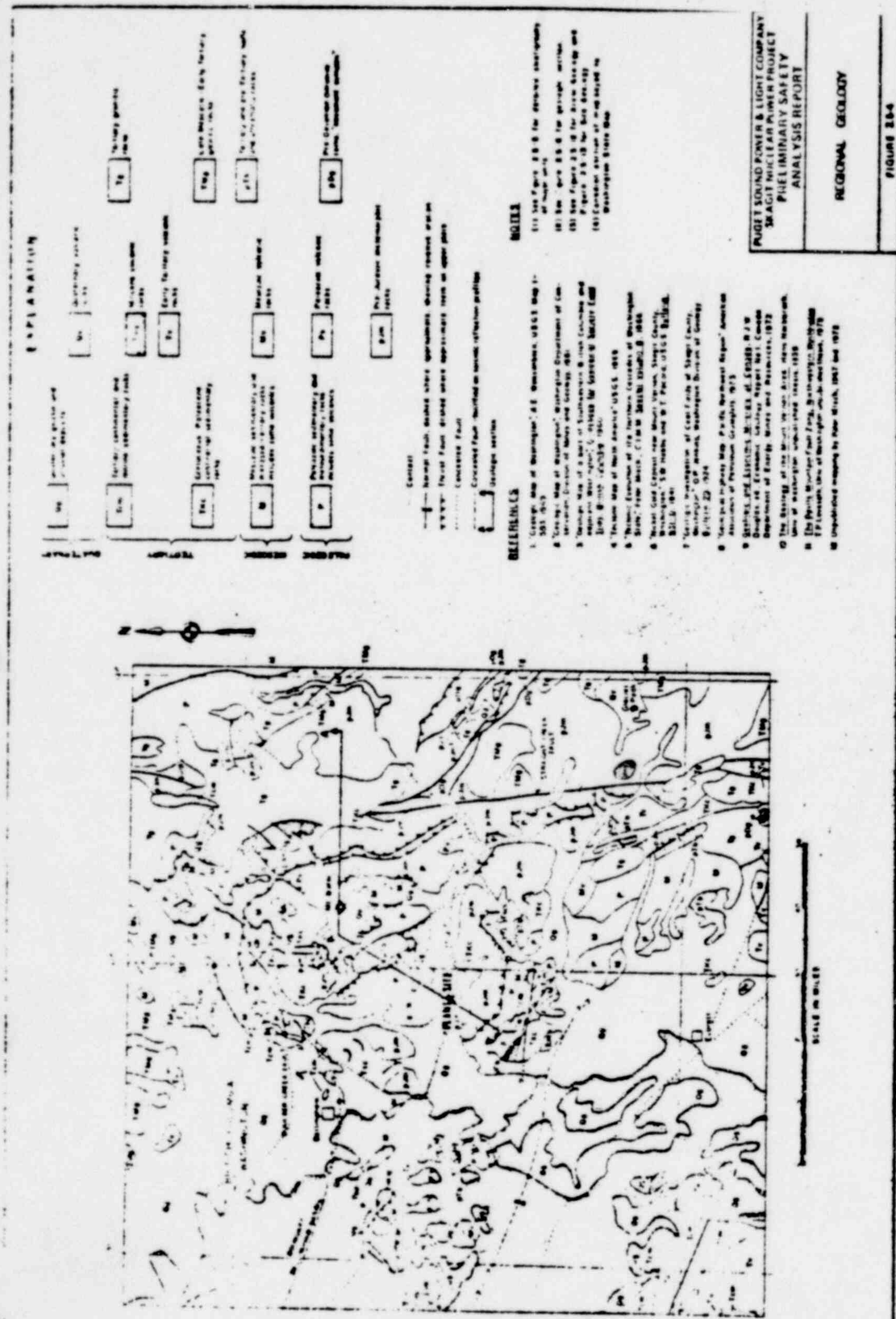
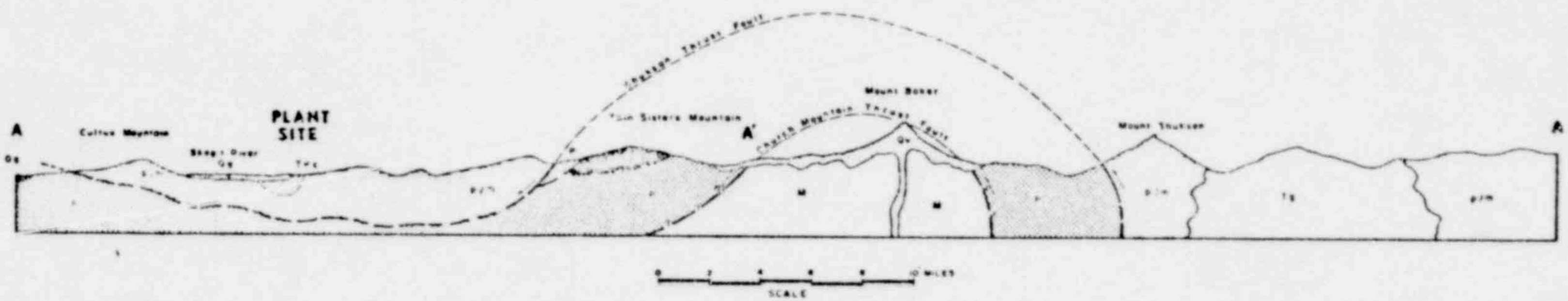
(pgs A-98 thru A-120)


PUGET SOUND POWER & LIGHT COMPANY
SKAGIT NUCLEAR POWER PROJECT
PRELIMINARY SAFETY
ANALYSIS REPORT
REGIONAL GEOLOGY
FIGURE 2.5-4
EXPLANATION
NOTES
REFERENCES



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EXPLANATION

- Qg — Quaternary glacial and alluvial material
- Qv — Quaternary volcanic rocks, Mount Baker
- Tg — Tertiary granitic intrusive rock, Chilliwack Batholith
- Tcc — Cretaceous Paleocene non-marine sedimentary rocks, Chuckanut Formation
- M — Mesozoic undifferentiated sedimentary and meta-sedimentary rocks, autochthonous units
- P — Paleozoic undifferentiated sedimentary, meta-sedimentary and volcanic rocks, Chilliwack Group—Church Mountain Thrust plate
- p-m — Pre-Upper Jurassic metamorphic rocks, Shuksan Metamorphic Suite—Shuksan Thrust plate
- Pre-Tertiary ultramafic rock, Twin Sisters Dunite

REFERENCES

1. Cascadia, The Geologic Evolution of the Pacific Northwest, Bates McKee, McGraw-Hill Book Company, 1972
2. "Fabric, Seismic Anisotropy, and Tectonic History of the Twin Sisters Dunite, Washington", Nikolas I Christensen, GSA Bulletin, Vol 82, p (68), 1971
3. "Tectonic Evolution of the Northern Cascades of Washington State", Peter Misch, CIMM Spec Pat No 8, 1966
4. Additional references listed on Figure 2.5-4, Regional Geology

NOTE

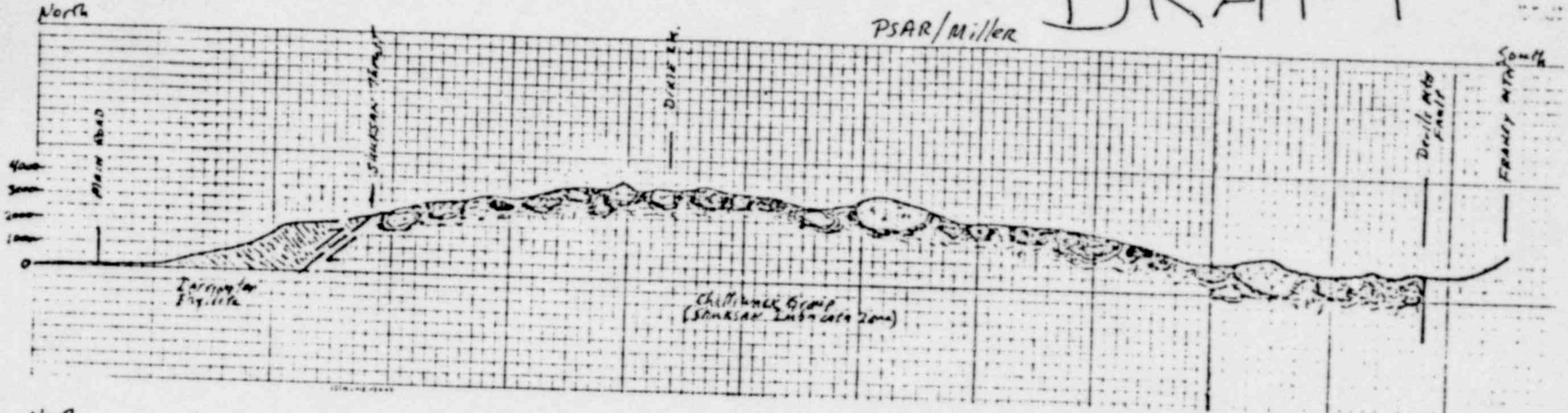
1. See Figure 2.5-4 for location of section.

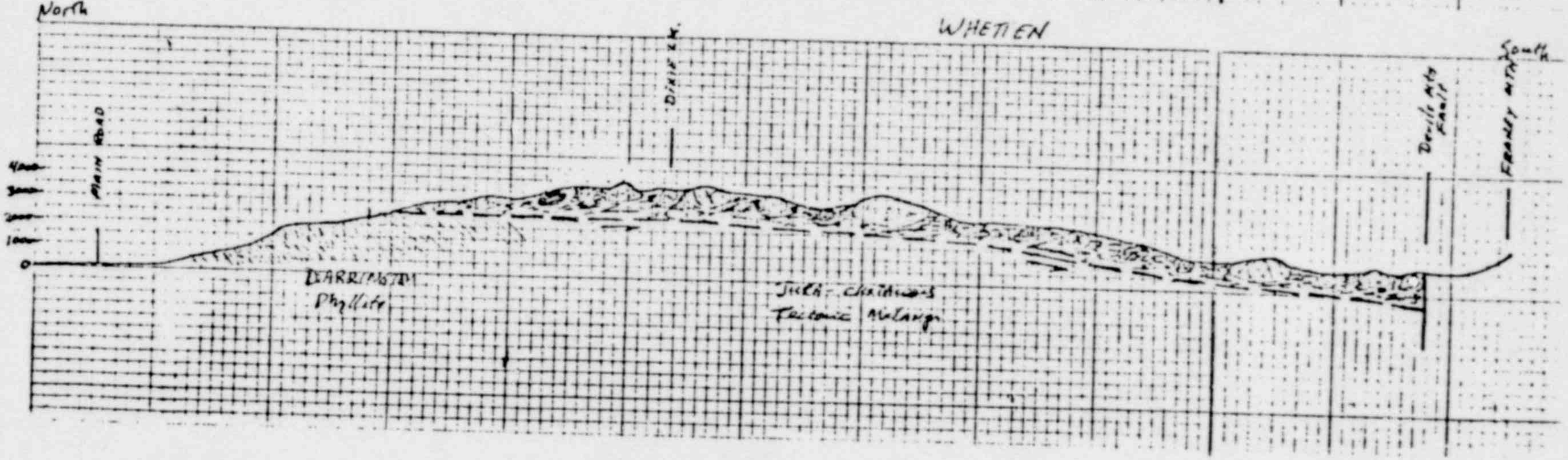
PUGET SOUND POWER & LIGHT COMPANY
SKAGIT NUCLEAR POWER PROJECT
PRELIMINARY SAFETY ANALYSIS REPORT

GENERALIZED REGIONAL GEOLOGIC SECTION

FIGURE 2.5-6

DRAFT 5/5/78


PSAR/Miller
North
South
Main Road
Shuksan Thrust
Dixie Ck.
Devils Mtn Fault
Franey Mtn
Darrington Phyllite
Chilliwack Group
(Shuksan Subduction Zone)


WHETIEN
North
South
Main Road
Dixie Ck.
Devils Mtn Fault
Franey Mtn
Darrington Phyllite
Shuksan-Chilliwack
Tectonic Melange

Appendix XIV

APPENDIX XIV
Sequoyah, Watts Bar, and Bellefonte: Background Material Leading to Seismic Design Reevaluation

UNITED STATE
NUCLEAR REGULATORY
ADVISORY COMMITTEE ON REAC
WASHINGTON, D. C. :

May 25, 1978

ACRS Members

SUBJECT: NRC STAFF EVALUATION OF THE SEISMIC DESIGN BASIS FOR SEQUOYAH, WATTS BAR, AND BELLEFONTE NUCLEAR PLANTS

The NRC Staff has requested that TVA reevaluate the seismic design for the Sequoyah, Watts Bar, and Bellefonte nuclear plants with respect to the current (10 CFR 100, Appendix A and the Standard Review Plan) criteria for seismic design. The controlling earthquake for all three plants is the Giles County, Virginia Earthquake of 1897 (MM VIII). The NRC Staff currently accepts the use of the Trifunac-Brady intensity-acceleration relationship (which associates a mean peak acceleration of 0.25g with an Intensity VIII earthquake) and the use of the Regulatory Guide 1.60 response spectrum. The SSE and OBE values, foundation conditions, and CP dates for these plants are as follows:

	SSE	OBE	Foundation Condition	A/E	CP Date
Sequoyah (W)	.18g	.09g	Bedrock	TVA	5-27-70
Watts Bar (W)	.18g	.09g	Soil	TVA	1-23-73
Bellefonte (B&W)	.18g	.09g	Bedrock	TVA	12-24-74

Similar information for the other nuclear plants located in Tennessee and northern Alabama is as follows:

	SSE	OBE	Foundation Condition	A/E	CP Date
Browns Ferry (CE)	.20g	.10g	Soil	TVA	OL issued
Hartsville (GC)	.20g	.10g	Bedrock	TVA	5-9-77
Phipps Bend (GE)	.25g	.09g	Bedrock	TVA	1-16-78
Yellow Creek (CE)	.30g	.10g	Soil	TVA	9-1-78
	.25g	.08g	Bedrock		

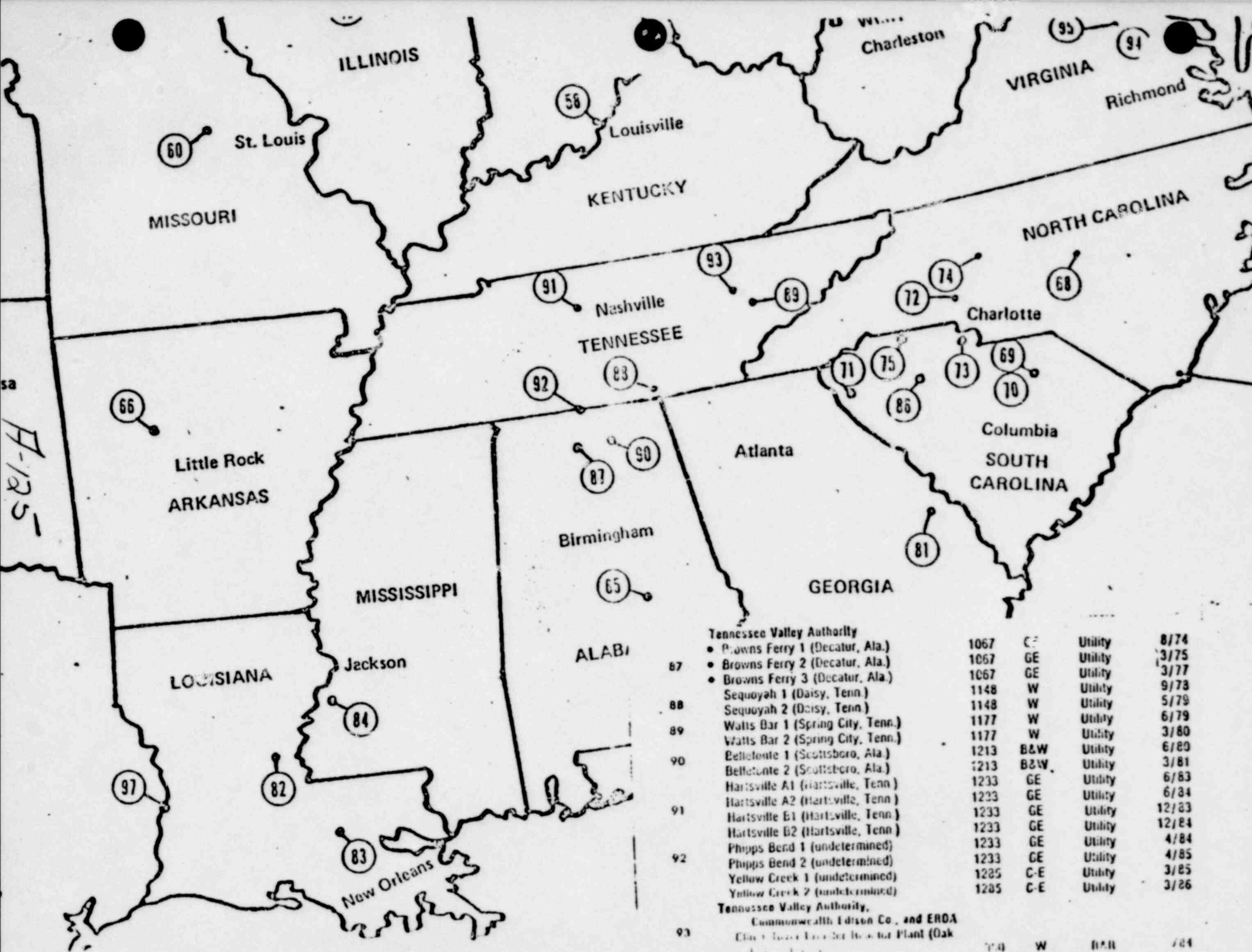
A map showing the location of the nuclear power plants in this part of the eastern United States and three letters, dated January 13, 1978, December 27, 1977, and February 6, 1978 are attached for your information.

A NRC Task Force (H. Rood, Chairman) has been assigned to this review and is expected to issue a report in the near future.

Michele D. Selden
for R. Savio
Staff Engineer

Attachments: As stated

A-125



	Tennessee Valley Authority				
	• Browns Ferry 1 (Decatur, Ala.)	1067	GE	Utility	8/74
87	• Browns Ferry 2 (Decatur, Ala.)	1067	GE	Utility	3/75
	• Browns Ferry 3 (Decatur, Ala.)	1067	GE	Utility	3/77
	Sequoyah 1 (Daisy, Tenn.)	1148	W	Utility	9/78
88	Sequoyah 2 (Daisy, Tenn.)	1148	W	Utility	5/79
89	Watts Bar 1 (Spring City, Tenn.)	1177	W	Utility	6/79
	Watts Bar 2 (Spring City, Tenn.)	1177	W	Utility	3/80
90	Bellefonte 1 (Scottsboro, Ala.)	1213	B&W	Utility	6/80
	Bellefonte 2 (Scottsboro, Ala.)	1213	B&W	Utility	3/81
	Hartsville A1 (Hartsville, Tenn.)	1233	GE	Utility	6/83
	Hartsville A2 (Hartsville, Tenn.)	1233	GE	Utility	6/84
91	Hartsville B1 (Hartsville, Tenn.)	1233	GE	Utility	12/83
	Hartsville B2 (Hartsville, Tenn.)	1233	GE	Utility	12/84
	Phipps Bend 1 (undetermined)	1233	GE	Utility	4/84
92	Phipps Bend 2 (undetermined)	1233	GE	Utility	4/85
	Yellow Creek 1 (undetermined)	1285	C-E	Utility	3/85
	Yellow Creek 2 (undetermined)	1285	C-E	Utility	3/86
	Tennessee Valley Authority, Commonwealth Edison Co., and ERDA				
93	[illegible] Plant (Oak [illegible]	[illegible]	W	[illegible]	[illegible]/84

B43622

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

January 13, 1978

RECEIVED
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS U.S. N.R.C.

JAN 23 1978

AM PM
7,8,9,10,11,12,1,2,3,4,5,6

Docket No. 50-327/328
50-390/391
50-438/439

FACILITY: Sequoyah, Watts Bar, Bellefonte

APPLICANT: Tennessee Valley Authority

SUBJECT: MEETING WITH TVA ON SEISMIC DESIGN BASIS FOR SEQUOYAH, WATTS BAR, AND BELLEFONTE

Representatives of TVA met with members of the staff on December 21, 1977 to discuss verification of the seismic design bases for the subject plants. Transportation problems caused a delay of several hours in the start of the meeting which in turn precluded the attendances of some staff members, attendees were as indicated on the attached list.

Our concerns about the seismic design bases for these plants were discussed along with possible approaches to resolving them. These concerns are documented in the letter of December 27, 1977, attached for reference purposes. TVA suggested a "generic" approach discussing regional seismology which could be applicable to all three plants, but we pointed out difficulties in using only this approach and indicated the need to focus on each plant and site. TVA indicated they would consider a multi-faceted response, including one suggested by us. They stated they would request a meeting to discuss the outline of their proposed response. We urged early action on this matter to preclude any unnecessary licensing delay.

Harley Silver

Harley Silver, Project Manager
Light Water Reactors Branch 4
Division of Project Management

RETENTION PERIOD
☐ Permanent
☐ Temporary for 12 months
☐ Destroy

Enclosures:
As stated

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A-126

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

December 27, 1977

Docket Nos. 50-327/328
50-390/391
50-438/439

Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
830 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

SUBJECT: SEISMIC DESIGN BASIS FOR THE SEQUOYAH, WATTS BAR, AND BELLEFONTE NUCLEAR PLANTS

This letter is to inform you of a question that has arisen concerning the seismic design bases for the Sequoyah, Watts Bar, and Bellefonte plants for which construction permits were issued on May 27, 1970, January 24, 1973, and December 24, 1974, respectively. All three plants lie within a tectonic province where the largest historical earthquake was the 1897 Giles County, Virginia earthquake, an Intensity VIII event. Past and present staff requirements specify that the safe shutdown earthquake (SSE) for plant design be determined assuming that the Intensity VIII event could reoccur near the plant sites. Correlations which were based on distant earthquakes and are now considered inappropriate for converting intensity to ground acceleration for earthquakes assumed to occur near a site, were used in establishing an acceleration of 0.18g as the SSE design basis for each of the three sites. The specific response spectra anchored to the acceleration were selected on the basis of the practice current at the time of reviews for construction permits.

In 1973 Appendix A to 10 CFR Part 100, and in 1975 the staff Standard Review Plan were put into effect. Appendix A lays out the basic approach for determining the SSE while the Standard Review Plan indicates specific Regulatory Guides, procedures, and techniques that may be used for this purpose. Certain aspects of the initial analysis performed for the Sequoyah, Watts Bar, and Bellefonte plants are not affected. We still regard the Giles County Earthquake as being the controlling event for these sites and we still consider that to be an Intensity VIII event. What has changed, however, are the procedures used to convert this intensity to design spectra. We not accept an intensity-acceleration

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relationship based upon a more complete data set (Trifunac and Brady, 1976) which associated a mean peak acceleration of 0.25g with Intensity VIII. We also presently determine response spectra as indicated in Regulatory Guide 1.60 entitled "Design Response Spectra for Seismic Design of Nuclear Power Plants." In general, current practice results in the selection of more conservative response spectra than did our past practice.

Our current approach, as specified in the Standard Review Plan, would require a plant being built in the same region as Watts Bar, Sequoyah, and Bellefonte to be designed to withstand a more conservative design basis earthquake than either plant is currently designed for. Because of the actual procedures utilized for three plants, a detailed analysis of plant response to a larger earthquake than the SSE selected at the construction permit stage of review may show that the plants, as designed, are adequate with respect to the intent of Appendix A and other regulations. This is possible since the procedures generally used, such as the Trifunac and Brady intensity-acceleration correlation and the Regulatory Guide 1.60 procedures for determining response spectra, are general and do not take into account specific site conditions, earthquake magnitude, or distance to the earthquake source.

We will need additional information from you to confirm the adequacy of the seismic design of the Sequoyah, Watts Bar, and Bellefonte plants, and to assess whether the application of current staff practice with regard to selection of seismic response spectra is required for the public health and safety. One approach that might be sufficient is to use existing strong motion records to determine the response spectra predicted for an earthquake of the appropriate magnitude and distance for the site conditions, and then show these spectra to be within the design spectra. In any event, we will need additional analyses from you to conclude that the present plant designs are acceptable, or to determine modifications that may be required.

Please notify us of your schedule for accomplishing this within 60 days of receipt of this letter. We would pleased to meet with you to provide further clarification of this matter.

Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

ccs:
See page 3

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
830 Power Building

FEB 6 1978

U.S. NUCLEAR REGULATORY COMMISSION

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Boyd:

In the Matter of the Application of the Tennessee Valley Authority	))	Docket Nos. 50-327 50-328 50-390 50-391 50-438 50-439

In your letter to Godwin Williams, Jr., dated December 27, 1977, you requested a schedule for the submittal of additional information confirming the adequacy of the seismic design for the Sequoyah, Watts Bar, and Bellefonte Nuclear Plants. This information will be developed and submitted to the NRC in two phases as follows:

1. <u>Phase I</u>

   A report will be developed based on seismic information previously submitted on the Phipps Bend Nuclear Plant docket. This information will be updated and supplemented by additional new information and data. The Phase I report will be submitted on or about May 1, 1978.

2. <u>Phase II</u>

   A report will be developed based on site specific earthquake ground motions. The Phase II report will be submitted to NRC on or about July 3, 1978.

Very truly yours,

J E Gilleland

J. E. Gilleland
Assistant Manager of Power

7804100001

B001/S
1/0

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Appendix XV

OBJECTIVES OF WORKING GROUP:

1. ASSURE TIMELY DECISION ON SEISMIC ADEQUACY OF SEQUOYAH, WATTS BAR, BELLEFONTE

2. ASSURE EFFICIENT USE OF STAFF RESOURCES IN REACHING DECISION

WORKING GROUP PLAN OF ACTION:

1. DEFINE PROBLEM

2. LIST POSSIBLE APPROACHES

3. EVALUATE EACH APPROACH

4. RECOMMEND COURSE OF ACTION

POSSIBLE APPROACHES TO TVA SEISMIC ISSUE:

A. REEVALUATE INTENSITY OF SSE

B. REEVALUATE RESPONSE SPECTRUM ASSOCIATED WITH SSE

C. REEVALUATE DESIGN MARGINS FOR SSE

D. REEVALUATE OBE

E. EVALUATE SEISMIC RISK PROBABILISTICALLY

17-132

A. REEVALUATE INTENSITY OF THE SSE:

1. ASSOCIATE GILES COUNTY EARTHQUAKE WITH TECTONIC STRUCTURE

2. SUBDIVIDE VALLEY AND RIDGE TECTONIC PROVINCE

3. REEVALUATE GILES COUNTY EARTHQUAKE INTENSITY

4. SHOW THAT PLANT SITE AFFECTS INTENSITY

B. REEVALUATE RESPONSE SPECTRA ASSOCIATED WITH THE SSE:

1. DETERMINE RESPONSE SPECTRA FROM STRONG MOTION RECORDS OF APPROPRIATE MAGNITUDE AND DISTANCE
2. DETERMINE RESPONSE SPECTRA FROM STRONG MOTION RECORDS OF APPROPRIATE INTENSITY
3. REVISE INTENSITY-ACCELERATION CORRELATION
4. REVISE SPECTRAL SHAPE
5. REVISE INTENSITY-ACCELERATI[illegible] CORRELATION AND SPECTRAL SHAPE
6. DEVELOP SPECTRA BASED ON PARAMETERS OTHER THAN INTENSITY AND ACCELERATION
7. USE SRP-RECOMMENDED APPROACH

C. REEVALUATE DESIGN MARGINS FOR SSE

1. REEVALUATE ORIGINAL ANALYSIS

2. REANALYZE PLANT STRUCTURES AND FLOOR RESPONSE SPECTRA

3. REANALYZE PLANT COMPONENTS, SYSTEMS, PIPING AND RESTRAINTS

D. REEVALUATE OBE

E. EVALUATE SEISMIC RISK PROBABILISTICALLY

1. DETERMINE PROBABILITY OF EXCEEDING DESIGN ACCELERATION

2. DEVELOP UNIFORM RISK SPECTRA

3. COMPARE SSE PROBABILITY WITH OTHER PLANTS

4. DETERMINE PROBABILITY OF EXCEEDING PART 100 DOSES

RECOMMENDED APPROACHES:

1. DETERMINE SITE-SPECIFIC SSE RESPONSE SPECTRA FROM STRONG MOTION RECORDS OF APPROPRIATE MAGNITUDE AND DISTANCE

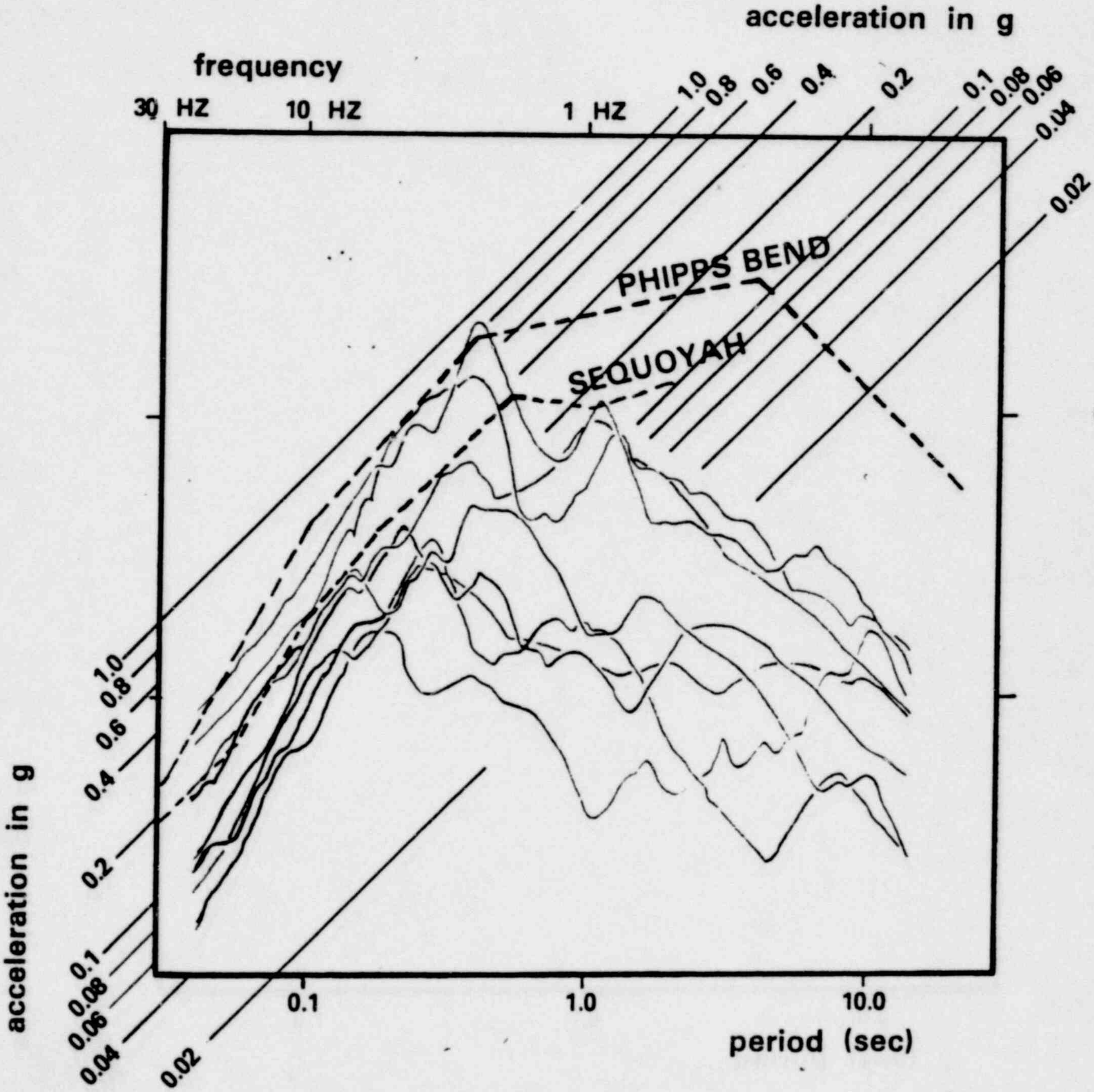
2. DETERMINE SITE-SPECIFIC SSE RESPONSE SPECTRA FROM STRONG MOTION RECORDS OF APPROPRIATE INTENSITY

3. REEVALUATE ORIGINAL SEISMIC STRUCTURAL AND FLOOR RESPONSE SPECTRA ANALYSIS, TAKING INTO ACCOUNT MORE REALISTIC METHODS AND MATERIAL PROPERTIES, AS WELL AS SITE-SPECIFIC SSE RESPONSE SPECTRA

4. REEVALUATE THE OBE TO SEE WHETHER IT MEETS THE RECURRENCE INTERVAL CRITERIA OF APPENDIX A TO PART 100

5. COMPARE THE PROBABILITY OF SSE BEING EXCEEDED AT THE SUBJECT PLANT WITH THAT AT OTHER TVA PLANTS THAT MEET THE SRP CRITERIA

STRONG MOTION RECORDS FROM FOUR EARTHQUAKES



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Appendix XVI

MAY 30 1978

Distribution:

NRC PDR
Local PDR
Docket File
LWR #4 File
R. Boyd
R. DeYoung
D. Vassallo
F. Williams
S. Varga
H. Silver
H. Service
R. Mattson
D. Ross
J. Knight
R. Tedesco
D. Eisenhut
C. Stahle
H. Rushbrook
A. Bournia
F. Schroeder
I. Sihweil
R. Bosnak
H. Denton
D. Muller
V. Moore
R. Vollmer
M. Ernst
W. Gammill
OELD
IE (3)
D. Muller
R. Denise
C. Stepp
R. Rood
L. Reiter
J. Bennett
J. Chan
J. Rajan
✓ ACRS (16)

APPENDIX XVI
Sequoyah, Watts Bar, and Bellefonte:
Reevaluation of Seismic Design

A-139

OFFICE ▶						
SURNAME ▶						
DATE ▶						

MAY 2 0 1978

Docket Nos: 50-327/328
50-390/391
50-438/439

Tennessee Valley Authority
ATTN: Mr. H. G. Hughes
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

[illegible]

In our letter of December 27, 1977, we advised you of our need for additional information to confirm the adequacy of the seismic design of the Sequoyah, Watts Bar, and Bellefonte plants and to assess whether the application of current staff practice for selection of seismic response spectra for these plants is required for the public health and safety.

Since that time, TVA has elected to present information in response to our request in two phases. Phase I consists primarily of attempts to justify a reduced intensity for the Giles County earthquake and for rock sites, and to validate the use of intensity-acceleration relationships other than the Trifunac-Brady relationship. Phase II is to examine site specific response spectra based on existing strong motion records of earthquakes of magnitude similar to the Giles County event. Except for some aspects, this is essentially an approach suggested in our December 27th letter. The Phase I report has been submitted and is under review, while the Phase II report is presently scheduled for August 1978.

To assist in further defining a course of action which would allow a timely finding by the staff on the acceptability of the seismic design of the three plants, we formed an NRC staff Working Group to study and make recommendations on this matter. This Group and other staff representatives have discussed this subject with your representatives on several occasions. The Working Group has completed its assignment, and three copies of its report are enclosed.

The report prepared by the Working Group has been reviewed by NRC management and it has been concluded that three distinct approaches are most likely to lead to favorable resolution of the issue of concern. These approaches are as follows:

OFFICE ▶						
SURNAME ▶			A7-140			
DATE ▶						

1. The first approach is to accept the procedures defined by the Standard Review Plan (SRP), and reanalyze the plant. This is described as Items B.7, C.2 and C.3 in the Working Group Report. Some plant modifications might be needed in order to conform to the SRP requirements. Resolution is assured by this approach. However, we estimate that a period of about two years or more may be required to reach a favorable conclusion for at least the [illegible] plant.

2. The second approach is to accept the recommendations of the Working Group; that is to pursue concurrently Items B.1, B.2, C.1, D, and C.3, as listed in its report. The Working Group, and others within the staff believe there is some likelihood that Items B.1 and C.1.a, of themselves, will lead to a favorable resolution. However, they believe the likelihood of obtaining a favorable resolution through the simultaneous examination of the five items listed above, while not completely assured, is significantly increased over that for Items B.1 and C.1.a alone. We believe the requisite [illegible] and [illegible] analysis will require a period of no less than about six months.

3. The third approach, which is a modification of the second, would concentrate all effort on Items B.1 and C.1.a as described in the Working Group report, with the hope that undivided attention to them will improve the likelihood of success. We believe the requisite analysis will require a period about the same as that described above for the second approach. An integral part of this approach would be to pursue the other items included in the second approach should the concentrated effort on Items B.1 and C.1.a not prove successful. It is possible that because of the greater [illegible] given to Items B.1 and C.1.a in this approach, the additional margins required of the remaining items would be reduced and, therefore, the likelihood of favorable resolution from the combination of all the items would be greater. If the concentrated effort on Items B.1 and C.1.a were not of themselves successful, the subsequent analysis required for the remaining items could extend the overall time for completion for this approach to about nine months to one year.

There are other possible approaches that could be used to attempt resolution of the seismic issue; most of these are discussed in the Working Group report. However, the three approaches discussed above appear to us to be those that warrant your most serious consideration. After careful consideration of the merits and potential drawbacks of each of the three approaches, we find ourselves in essential agreement with the conclusions and recommendations of the Working Group, and strongly suggest that you pursue the second approach described above and discussed in much greater detail in the Working Group Report.

OFFICE ▶						
SURNAME ▶			A-141			
DATE ▶						

MAY 30 1978

You may elect to follow other paths, or to concentrate your efforts on particular aspects of this matter on some priority basis, or possibly to use a different approach for each plant. In any event, we will review your submittals expeditiously. However, even though absolute assurance of a successful outcome cannot be given, we feel that implementation of the approach recommended by the Working Group provides the highest likelihood that the information you supply will permit a timely decision on this issue, and suggest that you follow this approach, particularly for Sequoyah. The results of each item included in the approach should be submitted when available; if after our review of the material submitted at any point a favorable decision can be reached before completion of the total effort, this will of course be done.

If we can provide any other information on this issue, please contact us. We would be pleased to meet with you and your staff on this matter at any time if it would be helpful. In order that we may plan our review efforts and further actions in the licensing process, we would like to know your plans and intentions in this matter and your schedule for accomplishing your efforts.

Sincerely,

/s/

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
Working Group Report

cc:
See Following Pages

17-142

DSS
RMattson
5/2_/78

DPM:DEP:D
RDeYoung
5/24/78

OFFICE	DPM:LWR #4	DPM:LWR #4	DPM:LWR/AD	DSE	OELD	DPM:DIR
SURNAME	HSilver:tlb	SVarga	DVassallo	HDenton		RBoyd
DATE	5/24/78	5/ /78	5/24/78	5/ /78	5/ /78	

cc:

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Mr. Robert B. Borsum
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Bethesda, Maryland 20014

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May 1978

REPORT ON TVA SEISMIC ISSUE

BY

NRC STAFF WORKING GROUP

H. Rood, Chairman
J. Bennett
S. Chan
J. Rajan
L. Reiter

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I. INTRODUCTION

A. Description of the Problem

The Tennessee Valley Authority (TVA or the applicant) has applied to the Nuclear Regulatory Commission (NRC or the Commission) for licenses to operate nuclear power plants at three facilities in the southern Appalachian highlands. These are (1) Sequoyah Nuclear Plant Units 1 and 2, located in Hamilton County, Tennessee; (2) Watts Bar Nuclear Plant Units 1 and 2, located in Rhea County, Tennessee; and (3) Bellefonte Nuclear Plant Units 1 and 2, located in Jackson County, Alabama. The operating license (OL) applications for these plants are currently being reviewed by the NRC staff. As a result of our review, we have concluded that these three facilities are being designed to seismic criteria which deviate from the criteria recommended by NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants" (the SRP).

The seismic criteria used in the design of these three plants were reviewed and found acceptable by the NRC (then the Atomic Energy Commission) during the reviews which preceded issuance of construction permits (CPs) for the plants. However, since the time the CPs were issued for Sequoyah (May 1970) and Watts Bar (January 1973), the Commission's regulations have been modified (Appendix A to 10 CFR Part 100 was adopted in November 1973). Subsequent to that time and following issuance of CPs for Bellefonte (December 1974), the

SRP was issued (September 1975). The applicable sections of the SRP reference Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Revision 1, December 1973), and Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" (October 1973).

Current NRC licensing procedures allow the approval of plants which are designed to criteria other than those recommended by the SRP. However, such deviations must be justified. If in any review, the staff finds that, as a result of such deviations, aspects of the design or the design criteria for the plant are unacceptable, post-CP facility modifications may be proposed by the applicant or be required pursuant to 10 CFR 50.109(a), which states:

> "The Commission may, in accordance with the procedures specified in this chapter, require the backfitting of a facility if it finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security. As used in this section, 'backfitting' of a production or utilization facility means the addition, elimination or modification of structures,

systems or components of the facility after the construction permit has been issued."

All plants currently undergoing OL review by the staff were granted construction permits prior to issuance of the SRP. Hence, the necessity to review and evaluate criteria other than those recommended by the SRP is not unique to the three plants in question. One reason the issue has arisen for these plants is the recent review by the NRC staff of the TVA application for construction permits for the Phipps Bend Nuclear Plant Units 1 and 2. This plant is located in the vicinity (Hawkins County, Tennessee - see Figure 1) of the three TVA plants in question, and is designed to meet the seismic criteria recommended by the SRP. A comparison of the chronology and the seismic design criteria for all four plants is given in Table 1. A comparison of the SSE response spectra for a common damping value is given in Figure 2. As may be seen from Table 1 and Figure 2, the three older plants deviate from the SRP criteria by varying degrees, depending upon their vintage. The plant of greatest concern is Sequoyah, because it deviates from the SRP to the greatest extent, and because its construction will be complete at the earliest date.

B. Working Group Assignment

At the present time the applicant has not adequately justified the seismic criteria used in the design of the three plants. In order to assure that (1) staff decisions on the three plants will be made

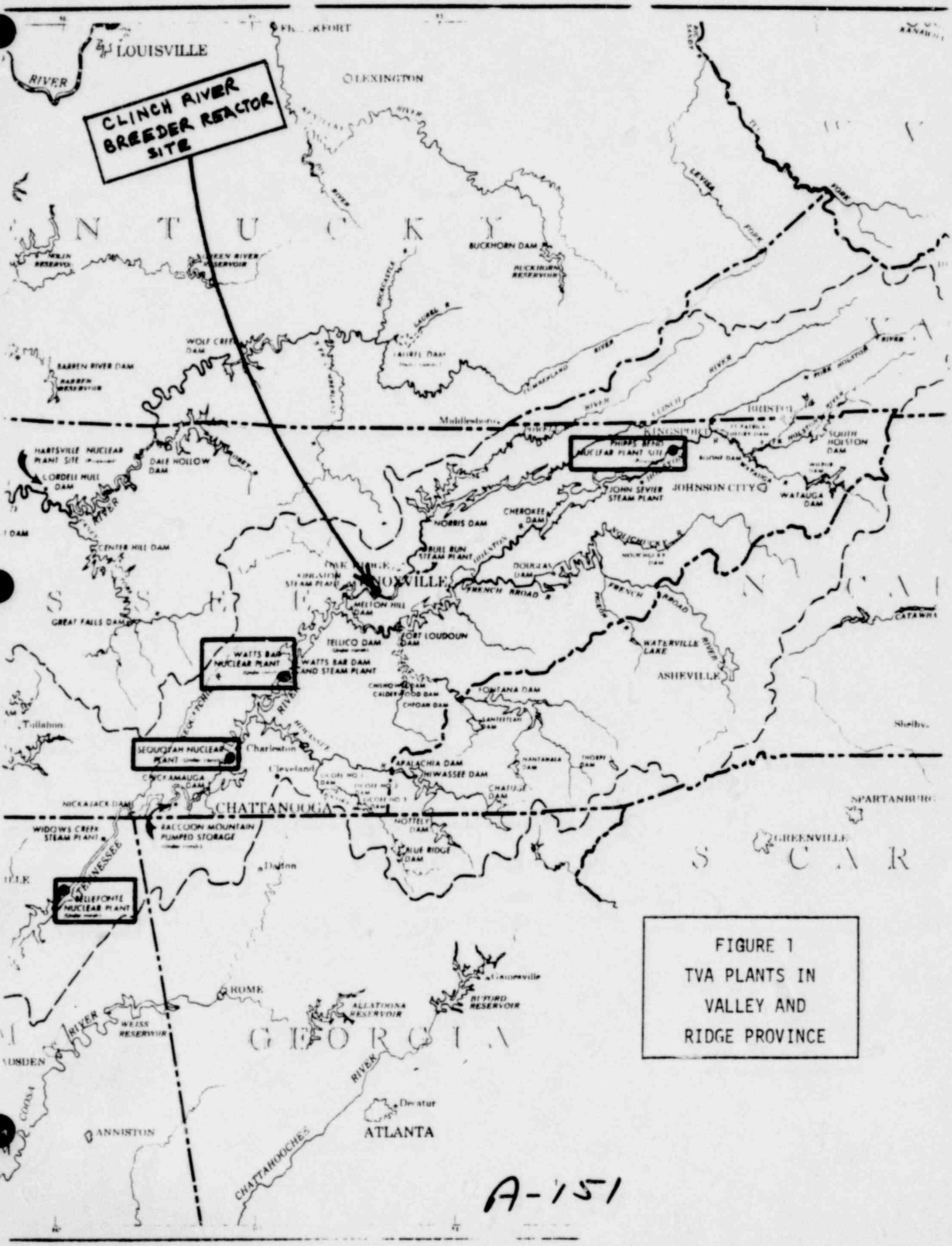

FIGURE 1
TVA PLANTS IN
VALLEY AND
RIDGE PROVINCE

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TABLE 1

COMPARISON OF CHRONOLOGY AND SEISMIC DESIGN CRITERIA

Name of Plant	Sequoyah	Watts Bar	Bellefonte	Phipps Bend
Docket Number	50-327/328	50-390/391	50-438/439	50-553/554
Date CP Application Docketed	10/15/68	5/14/71	6/21/73	11/7/75
Date of CP Issuance	5/27/70	1/23/73	12/24/74	1/16/78
Date OL Application Docketed	1/31/74	10/4/76	Tendered 2/78	1981
Projected Fuel Load Date	1/79	6/79	2/80	5/83
Intensity of SSE (MM)	VIII	VIII	VIII	VIII
Zero-Period Acceleration	0.18g	0.18g	0.18g	0.25g
Type of Response Spectrum	Housner Spectrum anchored at 0.14g but increased to 0.18g at high frequencies	Modified Newmark spectrum anchored at 0.18g	Reg Guide 1.60 spectrum anchored at 0.18g	Reg Guide 1.60 spectrum anchored at 0.25 g
Damping Factors for SSE (%)				
Steel Containment Vessel	1	1	4	4
Other Welded Steel Structures	1	2	4	4
Bolted Steel Structures	2	5	7	7
Reinforced Concrete Structures	5	5	7	7
Vital Piping Systems	0.5	0.5	2-3	2-3

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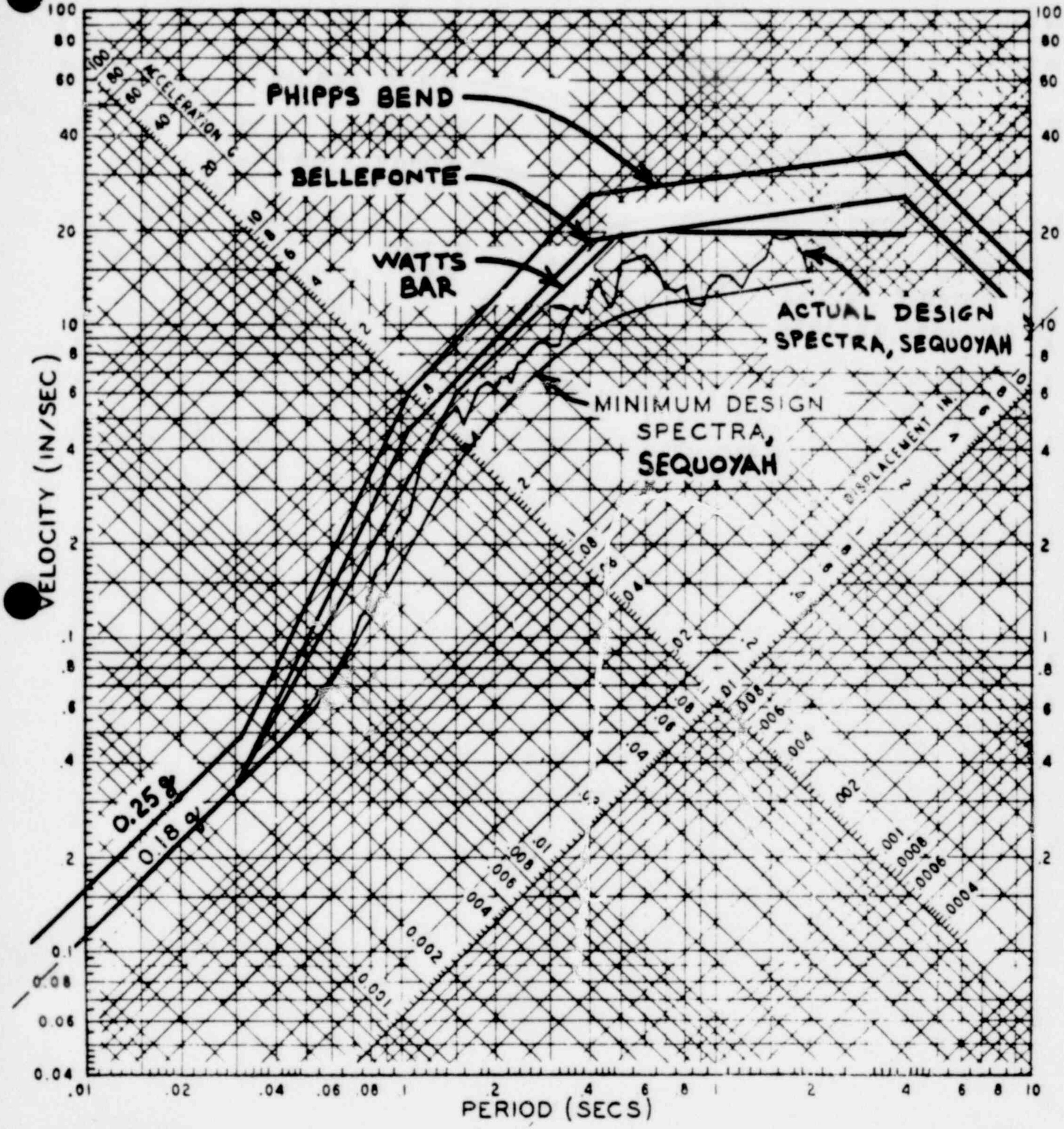

Figure 2 Comparison of Response Spectra for Safe Shutdown Earthquake, 2% Damping

in a timely manner consistent with the construction completion schedules, and (2) staff resources will be used efficiently, a Working Group has been formed. The group consists of five members of the NRC staff who were assigned the task of developing a method of resolving the TVA seismic issue (see Appendix A to this report). The Working Group charter requires that the group evaluate the problem, consider various methods of resolution, and recommend a path of resolution that assures safety while taking into account differences in the time and effort that would be required by applicant and staff, and the extent to which seismic reanalysis of the plant would be required. This report describes the results of the Working Group's efforts.

II. POSSIBLE APPROACHES TO TVA SEISMIC ISSUE

The first task undertaken by the Working Group was to compile a complete list of the possible approaches that the applicant might undertake to evaluate the seismic design criteria used for Sequoyah, Watts Bar, and Bellefonte. The list is given below. Although many of the approaches listed were believed to be impractical or unacceptable at the time the list was compiled, they were nevertheless included for completeness.

A. Reevaluate the Intensity of the SSE

During the CP reviews of Sequoyah, Watts Bar, and Bellefonte, and during the more recent CP reviews of Phipps Bend and the Clinch River Breeder Reactor*, the staff reached several conclusions regarding the factors which define the intensity of the Safe Shutdown Earthquake (SSE). These conclusions are (1) the above plants are located in the Southern Valley and Ridge tectonic province, (2) the largest historical earthquake in that province was the 1897 Giles County, Virginia earthquake, (3) the epicentral intensity of the Giles County earthquake was VIII on the Modified Mercalli scale, and (4) the Giles County earthquake has not been reasonably correlated with any known tectonic structure. These conclusions result in the SSE for any plant in the Southern Valley and Ridge tectonic province being defined by an intensity VIII earthquake that is postulated to occur near the plant. If it could be shown

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* located in Roane County, Tennessee - see Figure 1

that these conclusions should be changed, then the SSE might be revised. Some of the ways that this might be accomplished are:

1. Provide sufficient documentation to permit association of the Giles County earthquake of 1897 with tectonic structure unique to the epicentral area of the earthquake.

2. Provide sufficient documentation to permit subdivision of the Southern Valley and Ridge tectonic province into smaller tectonic provinces. One way this might be accomplished would be to use historic seismicity and/or instrumentally recorded earthquake activity to demonstrate that earthquake activity in the site vicinity is significantly less than that near the Giles County earthquake epicenter.

3. Provide sufficient documentation to show that the Giles County earthquake had an epicentral intensity other than VIII.

4. Provide sufficient documentation to show that if the Giles County earthquake of 1897 occurred adjacent to the site it would have produced intensities at the site different than those which were experienced in the epicentral area of the earthquake.

B. Reevaluate the Response Spectrum Associated with the SSE

If Approach A results in a revision of the SSE intensity downward to VII or less, the use of the SRP seismic design criteria will probably result in response spectra that are lower than the design spectra for the three plants. If the intensity is so revised, this must be verified. If Approach A does not result in a downward revision of the SSE intensity, it may be desirable to reevaluate the response spectra for the three plants, based on the information now available. Ways of generating appropriate spectra include:

1. Determine the response spectra (or suite of time histories) based on existing strong-motion records for earthquakes of appropriate magnitude and distance for the existing site conditions. If necessary, the data base may be supplemented by appropriate scaling of records.

2. Determine the response spectra based on an earthquake of appropriate intensity for the existing site conditions. Use strong-motion records for earthquakes of intensity VIII.

3. Use an intensity-acceleration correlation other than that recommended by the SRP (Trifunac-Brady) to anchor the Regulatory Guide 1.60 spectra recommended by the SRP.

4. Use spectra other than those recommended by Regulatory Guide 1.60 (e.g., generalized rock-site spectra) anchored at the 0.25g value predicted using the Trifunac-Brady correlation.

5. Use an intensity-accéleration correlation other than Trifunac-Brady and spectra other than those recommended by Regulatory Guide 1.60.

6. Develop spectra based entirely, or in part, on parameters other than intensity and acceleration.

7. Use the SRP-recommended approach; Regulatory Guide 1.60 spectra anchored at a peak acceleration value predicted using the Trifunac-Brady relationship (0.25g for an intensity VIII event).

C. Reevaluate Design Margins for the SSE

If Approach A results in an SSE of intensity VIII or greater, and Approach B results in spectra that exceed the design spectrum to a significant degree, a reevaluation of certain key design margins may be required. This may be undertaken in one of two ways. It may be possible to demonstrate adequate margins for safety related structures, systems, and components by a reevaluation of the original seismic analysis, taking into account a few additional effects, such as the use of Regulatory Guide 1.61 damping factors, the use of actual material properties, etc. Alternatively, it may be necessary to undertake a complete reanalysis of safety related structures, systems, and components using the most appropriate spectra or suite of time histories developed during Approach B, above. The reanalysis could be performed using SRP-recommended methods and criteria, or using other methods and criteria. For example, inelastic methods, experimentally determined damping factors, or traveling wave effects could be accounted for.

D. Reevaluate the OBE

After reevaluation of the SSE, the revised SSE may have a zero-period acceleration more than twice that of the OBE used in the design. In this case, two options are possible. The applicant may show that the design OBE is acceptable based on current staff criteria for operating basis earthquakes. Alternatively, the applicant could revise the OBE to be at least one-half the SSE, and reanalyze and, if necessary, modify the plant accordingly.

E. Evaluate the Seismic Risk Probabilistically

As an alternative to the deterministic approaches listed above, the applicant could use probabilistic techniques to assess the risk associated with the plant. Some of the probabilistic options are:

1. Utilizing accepted probability techniques (e.g., McGuire, 1976), determine the recurrence relation for different levels of peak acceleration. Compare the existing design spectra with Regulatory Guide 1.60 spectra scaled to these acceleration levels to estimate the probability of exceeding the design spectra. The effects of various source zones on the probabilities should be considered.

2. Utilizing accepted probability techniques, determine the recurrence relation for individual spectral components of the response spectra. Compare these uniform risk spectra to existing

design spectra to estimate the probability of exceeding the design response spectra (ref. Diablo Canyon studies). The effects of various source zones on the probabilities should be considered.

3. Utilizing accepted probabilistic techniques, compute the relative differences in probabilities of exceedance between the existing design spectra and those used at Phipps Bend and other recently reviewed plants which meet the SRP.

4. Utilizing accepted probabilistic techniques, where possible, determine the probability that the dose guidelines of 10 CFR Part 100 will be exceeded as a result of an earthquake. Compare this probability with the criteria defined in Section 2.2.3 of the SRP for accidents involving hazardous materials or activities.

III. EVALUATION OF POSSIBLE APPROACHES

The various possible approaches listed in Section II, above, were evaluated by the Working Group from the standpoints of the time and effort required to pursue each and the likelihood that undertaking of the approach could contribute to resolution of the issue.

A discussion of the Working Group's evaluation of each approach is given below.

A. Reevaluate the Intensity of the SSE

1. Associate Giles County Earthquake with Tectonic Structure

This approach involves the development of data to show that the Giles County earthquake of 1897 is associated with tectonic structure that does not extend to the vicinity of the plant site. According to the NRC Seismic and Geologic Siting Criteria (Appendix A to 10 CFR Part 100), historical earthquakes which can reasonably be associated with tectonic structure should be assumed to occur no closer to the site than the nearest approach of that structure. Thus, if the applicant were able to identify the structure responsible for the 1897 Giles County earthquake and to map the extent of that structure, the effects of attenuation between the assumed location of the earthquake on the structure and the site would likely result in some reduction in the expected intensity at the site. The Working Group believes that this approach has an extremely limited chance of success. Without some explanation of the causal mechanism of earthquakes in the eastern United States or convincing seismological or geologic evidence of fault activity, it is extremely difficult to conclude that specific historical earthquakes were associated with specific geologic structure. The staff has been reluctant in the past to accept such correlations. Advances in the state of seismic and geologic knowledge in the eastern United States which would permit such conclusions are unlikely, in the short term, except in a few isolated areas.

For the applicant to pursue this approach, the effort required would be comparable to that currently ongoing in the New Madrid, Missouri, and Charleston, South Carolina, areas. These studies entail geological, geophysical, and seismological explorations. Several years of work and several million dollars would doubtlessly be needed to support such an effort.

2. Subdivide Valley and Ridge Tectonic Province

This approach involves the development of information to justify subdivision of the Southern Valley and Ridge tectonic province into smaller provinces, so that the Giles County earthquake would be located in a different province from the plant sites. According to the siting criteria of Appendix A to Part 100, historical earthquakes associated with tectonic provinces other than the one in which the site is located should be assumed to occur at the nearest approach to the site of those tectonic provinces. Thus, if the applicant were able to provide a convincing case for subdividing the Southern Valley and Ridge tectonic province so as to isolate the Giles County earthquake in a different tectonic province, then some reduction in the site intensity would be appropriate. This would result, again, from the effects of distance on attenuation. This approach also has a very limited chance of success in the short term.

Tectonic province is defined in the siting criteria as a region of relative consistency of geologic structural features. Guidance on how this definition is to be implemented is lacking; however, staff practice has been to base conclusions on relatively large-scale provinces such as those identified by Eardley (1951) or Hadley and Devine (1974) which were based strictly on geologic structure. The staff has occasionally accepted lower acceleration levels in certain areas based on seismicity. The effort required to justify subdivision of the Southern Valley and Ridge into smaller tectonic provinces is viewed as major. The NRC staff is currently sponsoring research directed at better defining earthquake sources and their relation to geologic structure in the eastern United States.

These studies involve earthquake monitoring as well as geologic and geophysical investigations. These efforts are mainly concentrated in the northeast and central United States and will involve several years of work. An effort of this scale in the Southern Valley and Ridge province would be required to justify the subdivision.

3. <u>Reevaluate the Giles County Earthquake Intensity</u>

This approach involves providing evidence to show that the Giles County earthquake had an epicentral intensity other than VIII.

If the epicentral intensity of the 1897 Giles County earthquake could be demonstrated to have been less than intensity VIII, the siting cr'.eria of Appendix A to Part 100 would require only that the lesser intensity level be assumed to occur at the site in establi-'ing the safe shutdown earthquake. Detailed reanalysis of historical accounts of the earthquake could be used as a basis for such an assessment.

Approaches of this type have been useful in the past. For example, the 1791 East Haddam, Connecticut earthquake was downgraded after such studies (see the Connecticut Yankee, Montague, and Pilgrim 2 applications). However, in this case the chances of achieving such results appear to be low. The reason for this assessment is that a proposal to reduce the epicentral intensity of the 1897 Giles County earthquake was reviewed only about two years ago by a panel of experts from the USGS and universities. Though several members rated the event as a low or "weak" VIII, the panel decided that the reanalysis of the historical accounts of the earthquake did not warrant a change in the assigned epicentral intensity of VIII.

The cost of this approach is not great in either manpower, equipment, or time. However, in this case it cannot be recommended because of the limited prospects for significant results.

4. <u>Show that Plant Site Affects Intensity</u>
This approach involves the development of evidence to demonstrate that if the Giles County earthquake occurred adjacent to the site, intensities at the site would have been different from those reported for the Giles County event. The siting criteria of Appendix A to Part 100 indicate that in assessing the vibratory ground motion, one should consider the comparative characteristics of the material underlying the epicentral area and the site in transmitting ground motion. Abundant historical data suggest that intensity is consistently greater on soil than on rock; however, accelerograph measurements indicate that ground motion from an earthquake is usually greater on rock than on soil, in the frequency range significant to nuclear power plants. Furthermore, this observation is strongly dependent on soil thickness; thin soil frequently produces very

large amplification at certain frequencies. Thus, though the maximum intensity in the Giles County earthquake was experienced on soil, it is not clear whether or how this observation should be factored into the assignment of ground motion for use in design of the three plants in question. Observations of reduced intensity would only be relevant if they could be demonstrated to occur at frequencies of interest in nuclear power plant design. In the Phipps Bend application, the applicant provided arguments that intensity should be reduced on rock, and suggested that as a result, lower design ground motions were appropriate for the Phipps Bend site. These arguments were not accepted by the NRC staff. Because of the generic nature of this issue and the major impact it could have on licensing policy, an extensive study with peer review by other agencies and consultants is believed to be appropriate to a resolution of this problem. The study would take a considerable amount of time. Therefore, we do not view this approach as viable for resolution of this issue in the short term.

B. Reevaluate the Response Spectra Associated with the SSE

1. Determine Response Spectra from Strong Motion Records of Appropriate Magnitude and Distance

This approach involves determining the magnitude of the 1897 Giles County earthquake from its intensity observations and using the magnitude to identify strong motion records obtained

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on similar foundation conditions to those at the plant site. These strong motion records would be used to compute response spectra with which to check the adequacy of the response spectra used in designing the plant. This approach appears to merit additional work.

One of the critical steps involved here is a determination of the magnitude of the earthquake. Nuttli and Zollweg (1974) and Street and Turcotte (1977) described empirical methods of assigning magnitude based on the area in which certain intensity levels were experienced. Nuttli (1973) and Bollinger (1977) used methods based on the decay of intensity with distance to assign magnitude. Such approaches can and have been applied to assign a magnitude to the Giles County earthquake of 1897 by Bollinger (Private Communication). Using the range of magnitudes identified by Bollinger (about 5.3 to 6.3) as a guide, strong motion records for distances less than about 20 to 25 kilometers and foundation conditions like those at the site (in this case rock for most structures) can be selected. A suite of such records (15 to 20 records) could be used to develop mean and mean-plus-one-standard-deviation spectra. Confidence limits for these spectra should be calculated. To test the sensitivity of these results, similar calculations should be made for other magnitude ranges (4.3 to 5.3 and 6.3 to 7.3) and site conditions. Comparison with the spectra used in designing the plant can then be made to identify significant differences. Scaling of records should be avoided if possible. However, if an adequate data base is not available for the prescribed conditions, scaling of records outside this range of conditions may be necessary. Since such procedures require extrapolation, the sensitivity of the scaling to distance and magnitude should be evaluated. Use of a distance of 15 kilometers for scaling can probably be justified on the basis of the distribution of possible earthquake epicenters in the region surrounding the site. The applicant is currently pursuing this approach, though their current emphasis seems to require scaling of the records. The analysis required to support this approach is relatively minor and probably can be completed in a timely manner. The applicant is attempting to supplement the data base by obtaining overseas strong motion records, the acquisition of which could cause some delays.

2. Determine Response Spectra from Strong Motion Records of Earthquakes of Appropriate Intensity

This procedure involves determination of a set of response spectra directly from intensity. It is unclear whether this approach meets the regulation which requires the determination of acceleration level with subsequent scaling of response spectra corresponding to that level. Instead, this approach involves determination of response spectra directly from time histories corresponding to the appropriate intensity and site conditions. The approach is rather straightforward, following methodology described by Trifunac and Anderson (1978) or Werner and Tsao (1977) and requires no more effort than that involved in Approach B.1, above.

Problems with the approach are that it disregards the distance factor; i.e., earthquakes which produce intensity VII at 60 miles are lumped together with those producing intensity VII in the epicentral area - even though the spectra at the different distances would be expected to be quite different. Werner and Tsao (1977) found that for intensities V, VI and VII the use of the Trifunac and Brady (1975) mean peak acceleration and Regulatory Guide 1.60 spectra resulted in response spectra that fell somewhere between the mean and mean-plus-one-standard deviation of the recorded spectra. Although the scarcity of data precluded them from making a comparison at intensity VIII, the trend observed at the lower intensities suggests that the use of Trifunac and Brady (1975) and Regulatory Guide 1.60 spectra is not overly conservative when compared to actual spectra. It is recommended, therefore, that in addition to Approach B.1, Approach B.2 should also be attempted. Any differences in the results of the two approaches should be explained.

3. Revise Intensity-Acceleration Correlation

Appendix A to 10 CFR Part 100 does not specify a particular intensity-acceleration relationship to use in deriving acceleration from earthquake intensity. Thus, the siting criteria would permit the use of a relationship other than that recommended by the SRP and currently in use. This approach appears to have a limited chance of success, in the short term. The staff adopted the Trifunac-Brady relationship in 1975 after considering all the relationships in the published literature available at the time. The Trifunac-Brady relationship was based on the most complete data set available. Since that time, additional studies with more complete data samples and more correct statistical analysis procedures have been published. The most notable of these is the Computer Sciences Corporation (CSC) study (Murphy and O'Brien, 1978) sponsored by the NRC. While the CSC study has advantages over the Trifunac-Brady study, additional clarification of some of its findings needs to be developed before it can be adopted as a licensing policy. Chief among these problems are the distance dependence of the relationship, the effect of recording site conditions, and the geographic dependence which were identified by CSC.

The staff has supported the Trifunac-Brady relationship in recent licensing actions, in particular in the case of the

Phipps Bend and Clinch River sites. This position has been taken in the face of arguments by the applicant and others that the CSC relationship is preferred. It is the working group's view that adoption of a relationship other than that of Trifunac and Brady should be based not only on completeness and statistical correctness but also on how well response spectra scaled using the new relationship represent the ground motion from earthquakes. The recent study by Werner and Tsao (1977) snow that Regulatory Guide 1.60 spectra anchored at peak accelerations predicted by Trifunac and Brady fall between the mean and mean-plus-one-sigma spectra for measured data at intensity V, VI and VII. The Working Group believes that adoption of a relationship other than that of Trifunac and Brady (1975) should require a major generic study of the entire ground motion problem with external peer review. This effort is not likely to produce results in as timely a manner as some of the other approaches described here.

4. Revise Spectral Shape

This approach involves using spectra developed from strong motion recorded on rock sites for design of structures on rock, and spectra from strong motion on similar soil for design of structures on soil; e.g., spectra similar to those developed by Seed, Ugas and Lysmer (1976). In all cases these generalized spectra would be scaled to the acceleration level specified by standard review plan procedures (0.25g for intensity VIII).

This approach is straightforward and relatively quick but may oversimplify the problem. Werner and Tsao (1977) showed that for rock sites at intensities VI and VII, the Seed, Ugas and Lysmer (1976) spectra anchored at 0.11 and 0.17g (the mean acceleration on rock for these intensities) fall somewhere between the mean and mean-plus-one-sigma spectra. Although these values are somewhat higher than the Trifunac-Brady values, an inspection of their comparative plots indicates that the use of the Seed, Ugas and Lysmer (1976) spectra will not be significantly different.

5. <u>Revise Intensity-Acceleration Correlation and Spectral Shape</u>

   This approach combines the ideas discussed in 3 and 4, above. Advantages and problems with the approach were discussed there.

6. <u>Develop Spectra Based on Parameters Other than Intensity and Acceleration</u>

   Spectra can be developed based on the relationships between intensity and other parameters such as particle velocity, displacement, magnitude, and distance. Studies by several authors (e.g., Nuttli, 1973) suggest that intensity correlates better with particle velocity than with particle acceleration.

It is unclear whether such an approach is permitted in the siting criteria which require that an acceleration level be identified. This approach is not expected to produce results in a timely manner nor are the results expected to be much different. While the concept appears valid, the methodology to pursue this approach is not developed. It is expected that the results would be controversial.

7. Use SRP-Recommended Approach

This meets the staff practice and is acceptable; it would probably result in a detailed reanalysis and possibly in backfitting of the plant, as is discussed in detail under Section III.C.2 and III.C.3 of this report.

C. Reevaluate Design Margins for the SSE

If Approach A results in an SSE of intensity VIII or greater, and Approach B results in spectra that exceed the design spectra to a significant degree, a reevaluation of certain design margins may be required. The margins of interest include the margin to allowable stress or strain for safety related structures, systems, and components.

1. Reevaluate the Original Analysis

It may be possible to show that the plants in question are acceptable, as designed, by reevaluating the original seismic analysis to take into account certain more realistic methods and material properties than were originally used. This approach would also take into account the modified seismic input spectra developed during Approach B, above. The Working Group believes that this approach could be completed in a few months, and recommends that it be undertaken.

The margins to allowable stress or strain for each safety related structure should be reevaluated. In addition, the floor response spectra for floors carrying safety related components should be reevaluated and compared with the design floor response spectra. The following items could be taken into account in the reevaluation of the original seismic structural analysis and floor response spectra determination.

a. Use Regulatory Guide 1.61 Damping Values

The damping values used in the original analysis of the three plants of interest are given in Table 1. As may be seen from this table, the values recommended by Regulatory Guide 1.61 (and used in the Bellefonte and Phipps Bend plants) are larger than those used for the Sequoyah and Watts Bar plants. The SRP and current staff practice support the values of the Guide, and the reevaluation of Sequoyah and Watts Bar using these values is recommended.

As may be seen from Table 1, the most critical structures for Sequoyah are of reinforced concrete. The other structures appear to be less critical, but should also be reevaluated.

b. Use Actual Material Properties

The original analysis of the three TVA plants used concrete and concrete reinforcement strength recommended by the ACI code. Data may be available which would allow a more realistic determination of the actual strength of the concrete and rebar used in these plants. We recommend that the applicant develop the data on actual strength and take this into account in reevaluating the margins of structures. We believe that an increase of 10 to 15% in margin could result from this reevaluation.

c. Consider OBE-Limited Structures

If in the original analysis a structure's design was governed by the OBE, a change in the SSE may not reduce the minimum structural margin. This possibility could be investigated and, where applicable, taken into account in a relatively short time, on the order of a month.

d. Combine Responses to 3D Input Using SRSS

In the design of Sequoyah, the stress in each member of safety related structures was determined by adding the peak value of the stresses due to seismic excitation in the vertical and the major horizontal direction. The SRP recommends the combination using the square root of the sum of the squares (SRSS) of the stresses due to excitation in three orthogonal directions (one vertical and two horizontal). For horizontal members, the method used in the Sequoyah design may be conservative by a factor of 1.0 to 1.4. We recommend that the SRSS combination of three components be undertaken for critical members.

2. <u>Reanalyze Plant Structures and Floor Response Spectra</u>

If Approach C.1, above, does not result in the demonstration that the plants in question are acceptable as designed, a seismic structural reanalysis and possible backfitting of one or more of the plants may be necessary. Such a reanalysis would be a major undertaking, and would require at least a year to complete. As input to the reanalysis, the SSE response spectra or suite of time histories developed during Approach B, above, would be used. Several alternative approaches exist for the reanalysis. These include:

a. <u>Reanalyze Plant Using SRP Methods and Criteria</u>

This approach meets current practice and is acceptable. It would probably require plant modifications, at least for Sequoyah.

b. Reanalyze Plant Using Other Methods and Criteria

Various other methods and criteria have been proposed for evaluation of seismic design. Some of these are discussed below.

(1) Use Inelastic Analysis Techniques

Inelastic design is permitted by Section VI(a)(1) of 10 CFR 100, Appendix A, which states that:

"It is permissible to design for strain limits in excess of yield in some of these safety-related structures, systems and components during the Safe Shutdown Earthquake and under the postulated concurrent conditions, provided that the necessary safety functions are maintained."

One simplified method that has been proposed to account for the inelastic behavior of structures is to modify the design response spectra while using the elastic method for analysis. The result is usually a set of response spectra having lower peak acceleration and higher displacement than the initial spectra.

One difficulty with this approach is that the method is inapplicable unless the strain in all structural members is small; less than about 130% of strain at yield, in our opinion. It is difficult to demonstrate that this limit will not be exceeded, since such a demonstration requires a detailed knowledge of the ductility of structures and structural members when stressed beyond the yield point. Further, unless the structures are designed to be stressed beyond the yield point, the benefits of inelastic analysis may be small.

An additional problem is the difficulty in analyzing the redistribution of stress that results when a member yields. This redistribution may lead to increased stress and consequent yielding in other members, etc.

Another difficulty with this approach is the necessity for demonstrating that structural members will actually yield at the assumed yield stress level. They may, in fact, not yield until much higher stress levels are reached, due to uncertainty in both material properties and the analytical models used. Assuming that yielding occurs at too low a stress level results in non-conservative floor response spectra being applied to plant components and equipment. One way to cope with this difficulty is by performing a second set of calculations, assuming that yielding does not occur, to obtain seismic input motion to components and equipment.

Another difficulty with this approach is the possibility that the damping values recommended by Regulatory Guide 1.61 may already account, to some extent, for inelastic effects.

In one previous review (Diablo Canyon), the staff agreed to accept the simple reduction of ground response spectra to account for inelastic effects, provided that some of the above difficulties were resolved. The Diablo Canyon applicant decided not to use this approach, electing instead to perform plastic analyses of individual members if and when it was found that yield was exceeded.

The Working Group believes that the above-discussed approach to inelastic analysis, if properly justified, as well as more complex plastic analyses, are quite expensive and could take a year or more. For this reason, they are not recommended at this time.

(2) Use Experimentally Determined Damping Values

It has been proposed that the applicant might perform oscillation tests on the as-built structures at the plants in question, in order to determine the appropriate damping factors for use in seismic analysis. Such tests have been performed on the TOKAI-2 1100 MWe BWR in Japan (Private Communication, J. Knight). We see several difficulties with this approach. First, such testing and analysis is expensive and time consuming. Second, since damping increases with stress, to obtain a damping value appropriate for the SSE, the structure would have to be excited to a degree comparable to that produced by the SSE to obtain useful results. This appears to be impractical. Third, it has not been established to the Working Group's satisfaction that the structural response determined from such experiments can be analytically resolved to give unique damping values that are applicable to seismic analysis. This problem results from the complexity of the analytical model and the fact that, in order to get significant response, the oscillator must be placed at the top of the structure, whereas the actual seismic excitation would be propagated upward through the foundation of the structure, rather than downward from the top.

(3) Account for the Traveling Wave Effect

It has been proposed that since the seismic wave length at high frequencies is less than the dimension of the plant structures, that the amplitude of the translational seismic input to the structures at such frequencies is, in effect, reduced. The reduced translational input due to the traveling wave effect is partially compensated by an increased rotational input that also results.

The SRP-recommended and previous methods of seismic analysis assume that the seismic input is uniform across the base of the structure, which eliminates consideration of both the above translational and rotational effects.

In one previous case (Diablo Canyon), the staff has accepted the translational reduction that results from accounting for this effect. The attendant increase in torsion was approximated by simplified assumptions about structural eccentricity. This approach has been, and remains, controversial. Some experts in the field maintain that a detailed, three-dimensional, finite element soil-structure-interaction analysis should be used rather than the simplified model.

As a result of the above considerations, the benefits of pursuing this approach are questionable.

3. <u>Reanalyze Plant Components, Systems, Piping and Restraints</u>
In addition to the reanalysis of structures and floor response spectra discussed in Approach C.2, above, it may be necessary to determine the margin to code-allowable stresses for components, etc. This would be the case if the new floor response spectra exceed the design response spectra. Where no margin to code-allowable stresses exists, the lack of margin must be justified, if possible, by showing that the component's functional capability will not be impaired. Alternatively, corrective modifications to the plant should be defined. The Working Group has compiled a set of criteria for reanalysis of components, systems, piping and restraints, should such reanalysis be required. The criteria are given below.

If it is found to be required, the reanalysis of the mechanical components, piping systems and equipment will be based on input loadings defined by response spectra. These spectra will include the amplification of ground motion by the structure supporting the equipment; i.e., the revised floor response spectra determined in C.2, above. This approach would require at least a year to complete and would probably result in plant modifications, at least for Sequoyah.

a. Reanalysis of Piping Systems

In the original design of the piping system, loads due to (1) thermal expansion and (2) dead load normal operational stresses due to system pressurization, were analyzed per ANSI Code for power piping and the ASME Boiler and Pressure Vessel Code requirements in effect at that time. The loading combinations to be considered and the allowable stress limits for the purpose of the reanalysis should meet the current requirements of Section III of the ASME Code and be consistent with the current regulatory staff positions. All piping will be classified into two categories, rigid or flexible. Rigid piping is that which has a period of less than 0.03 second. All piping with periods greater than 0.03 second is classified as flexible. A dynamic analysis will be performed of all flexible piping systems having piping that is six inches and greater in diameter. A dynamic analysis will also be performed of the more critical smaller lines. The mathematical models for these analyses will be the same as those used in the original design. An approximate dynamic analysis will be performed on the balance of the critical systems. The maximum acceleration for rigid piping or equipment will be considered to be the same as the structure (or ground for piping or equipment located on the ground) at the point of the piping support.

A dynamic seismic analysis will be performed on applicable flexible piping systems by the response spectrum method. The piping system will be mathematically modeled to represent the dynamic and elastic characteristics of the pipe system. The flexibility calculations will include the effects of torsional, bending, shear, and axial deformations. The frequencies and mode shapes for all significant modes of vibration of the piping system will be determined from the flexibility and mass matrices of the mathematical model.

Standard structural analysis methods will be used to determine the contribution of each mode to the total displacements, inertial forces, moments, and stresses. For piping systems which span more than one floor, the most severe floor response spectra, to which any portion of the pipe is subjected, will be used to represent the input motion.

The movements of the piping supports and restraints will be based on the maximum of the floor movements adjacent to the support location. The stresses induced in the piping due to restraint movements will be considered as expansion stresses and will be assumed to act concurrently with the thermal stresses. Seismic Category I piping systems will be evaluated for excitation in each of two orthogonal horizontal directions and will be individually combined with the excitation in the vertical direction. The stresses, moments, etc., at any point in the piping system will be taken to be the largest value resulting from either of these combinations.

The analysis of piping systems (less than 6-inch diameter piping), which are not considered critical for safe shutdown of the plant following a seismic event, may be done by simplified dynamic analysis methods. This analysis should include the pipe deadweight as well as horizontal and vertical seismic loadings.

The seismic-induced effects of non-Category I piping systems on Category I piping will be accounted for by including in the analysis of the Category I piping a length of the non-Category I system equal to at least the first seismic restraint or anchor beyond the point of change in classification.

The modal responses from earthquake responses should be combined in accordance with Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

b. Reanalysis of Mechanical Equipment

Design Class I mechanical equipment will be reanalyzed by the response spectrum modal superposition method. In those instances where the components, such as tanks, heat exchangers, valves, and pumps, are shown to be rigid (all natural frequencies greater than 20 Hz), the equipment will be checked for the maximum acceleration of the supporting structure. For Code Class II and III pumps and valves which were designed to the Codes and standards that were in effect when the items were purchased, it must be demonstrated that the stress limits were sufficiently low to provide assurance that no gross deformation would occur in active components. For pumps which are part of the nuclear steam supply system, the reanalysis will include forces resulting from seismic accelerations in the horizontal and vertical directions.

These forces will be applied simultaneously at the center of gravity of the pump. The pump support design should be checked to ensure that the natural frequencies (usually in excess of 30 Hz in the original design) do not result in any amplification of the seismic floor accelerations in the pump-support structures.

Pumps which were designed to standards other than the ASME Code requirements will have to be seismically qualified for the service conditions. This qualification may be done by analysis, testing, or a comparative review of the pump design. The support design of Code Class II and III pumps will be evaluated to determine capability to withstand the effects of the OBE and SSE. Supports of the active safety related pumps will be shown not to deflect and impair the operability of the pump.

The Code Class II and III valves which are part of the nuclear steam supply system were generally designed to the pressure and temperature requirements of the American Standard Association (ASA). The testing requirements of these valves should be reviewed to ensure that they include hydrostatic shell and seat leakage tests. The Code Class II and III valves which fall within the scope of supply of the balance of plant were designed, manufactured, and tested in accordance with the ASME Code for Pumps and Valves for Nuclear Power. These designs should be reviewed to ensure that the valve operators and yokes have a natural frequency greater than 20 Hz and that these will maintain operability when subjected to a 6g load across the yoke support.

Primary system equipment (steam generators, reactor coolant pumps, pressurizer, reactor vessel, vessel internals, fuel assembly, and control rod drive mechanisms) will be re-analyzed with revised input loads from the dynamic loop piping analysis..

The design basis analysis for the steam generator was by a lumped mass/beam model of the structure and by a response spectrum modal analysis technique. This model included the piping and support stiffnesses. It was used to evaluate the shell, tube bundles, and other pressure boundary components. The nozzles and support feet will have to be reanalyzed by the static stress analysis or similar methods with revised loads from the dynamic loop analysis.

The reactor coolant pump will have to be reanalyzed using a modal analysis method based on revised response spectra to qualify the internal component, flange bolts, and other pressure boundary components. Nozzles and support feet will be reevaluated by performing static stress analysis based on revised input loads from the dynamic loop piping analysis.

The shell and heater rods of the pressurizer will be reanalyzed using a lumped mass, modal analysis technique and based on revised response spectra. The most highly stressed components of the reactor vessel are inlet and outlet nozzles. The nozzles will be reanalyzed by the static stress analysis techniques with revised loadings from the dynamic loop analysis.

The dynamic response of the reactor internals will be based on a mathematical model which includes the containment building with the reactor vessel supports, the reactor vessel, and the reactor internals. For the vertical earthquake analysis, a single-degree-of-freedom system model may be used for the internals. The mathematical model for the horizontal earthquake analysis will consist of beams, concentrated masses, and linear springs.

The reevaluation will be made for the simultaneous occurrence of horizontal and vertical seismic input motions. The total seismic response will be obtained by adding the responses for vertical excitation absolutely to the separate results for the N-S and E-W directions. The larger of the two values so determined at each point in the model will be considered as the earthquake response. The response spectrum method of analysis will be used. For the normal-plus-SSE and the normal-plus-SSE-plus-LOCA loading conditions, acceptance criteria ahould assure adequate core cooling and core shutdown. The core geometry should not deform beyond acceptable limits. The maximum allowable deflections should not impair the structural and mechanical integrity as well as the functional capability of the internals.

The effects of the SSE on the fuel assembly will be evaluated by performing a non-linear time history analysis. This analysis will use a time-history input to a reactor vessel and internals model and evaluate the core plate motions and integrity of the fuel. The control rod drive mechanisms will be evaluated by means of a dynamic time-history analysis. This analysis will be performed by applying revised floor acceleration time-histories to a linear elastic model at the operating peak acceleration and at reactor vessel support elevations.

c. Reanalysis of Electrical Equipment

The reanalysis of Design Class I instrumentation and electrical equipment will be based on the same earthquake design bases as those for the structures and other mechanical equipment. The amplification of ground accelerations due to the response of the structures at the location of the equipment will be considered in the reevaluation. Revised acceleration response spectra for horizontal free field ground motion, resulting from postulated earthquakes at the plant site should be examined closely for amplification of the ground motion in the range of frequencies above 20 Hz.

Equipment with resonant frequencies above 20 Hz will be considered rigid. The acceleration in every part of the equipment can, therefore, be assumed to be the same as that of its supports. For both rigid and flexible equipment (frequencies above and below 20 Hz), it must be demonstrated by either tests or analysis that accelerations obtained from the response spectra for its location in the building will not damage the device to the extent that it will fail to initiate and maintain its safety function, nor prevent other devices from performing their safety functions. In addition, (1) the Design Class I and IE electrical equipment must be able to perform its required functions of providing electrical power, control instrumentation, and protection for the engineered safety features; and (2) the reactor protection systems must be able to shut down the unit and maintain it in a safe shutdown condition. The effects of seismic accelerations will be determined by either physical tests or analysis for all Design Class I and major equipment. Most physical tests were conducted by single axis, sine beat methods. These should be supplemented with multi-axis, multi-frequency testing to demonstrate compliance with IEEE-344-1975 requirements.

d. Definition of the Systems to be Reanalyzed

Several options exist with respect to the systems and components to be reevaluated. In two recent OL reviews, the North Anna Power Station Unit 1 (Docket No. 50-338) and

(4) Service water system,

(5) Containment spray system equipment,

(6) Auxiliary feedwater system, including pumps, water supplies, piping, valves,

(7) Pressurizer and main steam safety valves,

(8) Circuits and/or equipment required to trip the main feedwater pumps,

(9) Main feedwater isolation valves,

(10) Main steam line stop valves,

(11) Main steam line stop valve bypass valves,

(12) Steam generator blowdown isolation valves,

(13) Batteries (Class IE),

(14) Control room ventilation,

(15) Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the above tested equipment cannot be operated,

(16) Emergency lighting,

(17) Post-accident monitoring system.

The systems and equipment listed above are required to mitigate the short-term effects following a rupture of a main steam line. In the event it is necessary to maintain hot standby following this event, additional systems, including reactor containment ventilation cooling units and systems required for obtaining reactor coolant samples will have to be analyzed. For achieving cooldown, the steam generator power-operated relief valves (which can be operated manually), controls for defeating automatic safety injection actuation during a cooldown and depressurization and the residual heat removal system, including pumps, heat exchangers, and systems valves and piping necessary to cool and maintain the reactor coolant system in a cold shutdown condition will have to be reevaluated.

If it were necessary to evaluate systems required to cope with a loss-of-coolant accident, the following systems would need reevaluation.

(1) Containment Spray System
(2) Containment Purge System
(3) Containment Isolation System
(4) ECCS Systems - Safety Injection, RHR Systems
(5) ECCS Pump Room Emergency Filter Systems
(6) ECCS Pump Room Coolers
(7) Containment Fan Coolers
(8) Auxiliary Building Normal HVAC System Isolation Dampers
(9) Emergency Lighting
(10) Post-Accident Monitoring Systems

As a final alternative, it might be required that all safety related systems be reevaluated.

D. Reevaluate the OBE

Appendix A to Part 100 states that the operating basis earthquake (OBE) is the earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. It also indicates that the maximum vibratory ground acceleration of the OBE should be at least one-half the maximum vibratory ground acceleration of the SSE.

Further, Appendix A states that if an applicant believes that the particular seismology and geology of a site indicate that some of the criteria of Appendix A need not be satisfied, the specific criteria should be identified and supporting data should be presented to clearly justify such departures.

The current NRC licensing practice requires that the applicant provide information about earthquake recurrence intervals which demonstrate that the OBE is an earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. The staff has recently found acceptable and issued licenses for a number of plants for which the OBE acceleration is less than one-half the SSE acceleration. These include the Byron-Braidwood, Clinton, Koshkonong, Marble Hill, and Phipps Bend plants. The Working Group believes that the OBE for the three TVA plants should be reevaluated to determine whether the design OBE meets the criterion of being an earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. If the design OBE meets this criterion, an acceleration level less than half the SSE would be acceptable and consistent with current licensing practice.

E. Evaluate the Seismic Risk Probabilistically

To supplement the above-described deterministic approaches, the seismic risk can be, at least in part, evaluated using probabilistic methods. Several such approaches are evaluated below.

1. Determine the Probability of Exceeding the Design Acceleration

This approach involves identification of earthquake source zones in the site region. Within each source zone the earthquakes of some selected size (epicentral intensity) are assumed to have an equal probability of occurring. Based on this assumption, recurrence relations (earthquake probability per year versus intensity) are calculated for each source zone; and considering effects of attenuation the probability of exceeding some acceleration level at the site can be calculated. Such procedures have been developed by McGuire (1976) and Cornell (1968). Comparison of the plant design response spectra with Regulatory Guide 1.60 spectra scaled by means of the Trifunac and Brady relationship to the acceleration levels of computed probabilistic intensities provides an estimate of the probability of exceeding the design spectra. Critical elements in this approach are the configuration of the source zones, upper bound on intensity for use in the analysis, and the selection of an attenuation relationship. These problems are resolvable as technical decisions. The sensitivity of the conclusions to these parameters and other assumptions should be tested. It should be pointed out, however, that if the overall procedure is to provide a resolution to the problem, a higher level decision must be made on the acceptable level of risk. Such a decision would need to factor in the ideas described in E.4, below.

This approach is rather straightforward and can be performed quickly at little cost. The main problem with the approach is that it only resolves the issue to the point of assessing the earthquake risk. In a complete assessment the probability that the earthquake will result in unacceptable consequences also needs to be considered; such an approach is described in E.4.

2. <u>Develop Uniform Risk Spectra</u>

The initial elements of this procedure are like those described in E.1, above. Earthquake source zones are identified and the probability of earthquake occurrence anywhere within the zone is developed. These probability calculations are combined with a regression analysis relating response spectral amplitude at some frequency to magnitude or epicentral intensity, distance, and generalized site conditions. The probability that some spectral component will be exceeded is calculated by summing or integrating the probabilities over all source zones. The results are response spectra for which the response at each frequency has some fixed probability per year of being exceeded; the resulting spectra are called uniform risk spectra. Procedures to calculate the uniform risk spectra have been described by Anderson and Trifunac (1978) and in the San Joaquin Early Site Review (Project No. 499). Again, studies should be performed to test the sensitivity of the results to various assumptions. This approach is not difficult or particularly expensive.

The objections to this approach are the same as those identified for E.1. Specifically, additional calculations are needed to determine the probability of unacceptable consequences resulting from occurrence of the earthquake.

3. Compare SSE Probability with Other Plants

The procedures described in E.1 or E.2 can be used to compare the probability of exceeding the SSE at the subject plant to those at other TVA plants which meet the Standard Review Plan criteria (Phipps Bend, Yellow Creek, and Hartsville).

This approach appears to be a valid way to estimate the significance of differences in the ground motion assumed in the design. The procedures are relatively straightforward and the calculations can be made quickly. The relative risk calculations described tend to be more stable and less sensitive to the assumptions. This approach also avoids problems identified in connection with items E.1 and E.2 in that it minimizes the dependence on the additional assessment of the probability that the earthquake will result in unacceptable consequences. The determination that the seismic design of the subject plant is acceptable depends on the acceptability of the plants to which it is compared; i.e., adequate safety margins are assumed to be present in the comparison plants.

The Working Group recommends that the applicant be requested to undertake this approach for Sequoyah. This would involve a comparison of the probability of occurrence of the SSE at Sequoyah with that at other TVA plants which meet the SRP. This information may be of use to the staff in determining whether a reanalysis and possible backfitting of Sequoyah is necessary to provide substantial, additional protection required for public health and safety, pursuant to 10 CFR 50.109(a).

4. <u>Determine the Probability of Exceeding Part 100 Doses</u>
In this approach the probability that an earthquake will result in the dose criteria of Part 100 being exceeded is determined. As with Approaches E.1, E.2, and E.3, above, this approach involves determining the likelihood of occurrence of earthquakes

that have a variety of acceleration levels, from smaller, relatively likely earthquakes to larger, less probable ones. The range would include the OBE on the low end and would extend beyond the SSE on the high end. Then, for each level of earthquake, the effects of the earthquake on safety related components would be evaluated and a probability estimated that, given the earthquake, the damage to the plant would be so severe that the dose guidelines of 10 CFR Part 100 would be exceeded. By combining the probabilities of earthquake occurrence, structural damage, and radiological release, one could determine the probability that an earthquake would result in the Part 100 dose guidelines being exceeded. Such an approach is philosophically satisfying to many because it results in a quantitative estimate of a parameter of basic interest, the radiological risk to the public due to earthquakes.

In the past, estimates have been made of the seismic risk for nuclear plants. The Reactor Safety Study (WASH-1400) for example, estimated that the probability of a core melt accident being caused by an earthquake is 5×10^{-7} per reactor year for nuclear plants on soil of average properties and 6×10^{-8} for nuclear plants on firm sites. The uncertainty in these values was estimated to be plus or minus an order of magnitude. The study concluded that "at this level of probability, earthquake-induced accidents should not contribute significantly to reactor accident risks."

Unfortunately, the time and effort involved in taking this approach for the TVA plants is very large, and the uncertainty of the results would also be very large. The large uncertainty results, at least in part, from the difficulty in determining the response of structures, systems, and components to loads that are significantly in excess of the design loads. For these reasons, this approach is not recommended.

IV. CONCLUSIONS AND RECOMMENDATIONS

The objective of the Working Group's effort was to recommend to NRC management a course of action that will allow a timely staff finding on whether or not the present seismic designs of the three plants in question are acceptable. If not, a detailed reanalysis and possible backfitting of the three plants could be required. After evaluating the possible approaches that might be undertaken by the applicant, we have concluded that several of the approaches can be completed in a timely manner, and will probably provide sufficient information to allow the staff to make a decision regarding the acceptability of Sequoyah, the plant of most concern. However, the Working Group does not wish to foreclose the possibility of pursuit of any approach, if the applicant believes that the results would contribute to a decision regarding the safety of the plants. The recommendations contained herein are intended to assure that the applicant undertake the approaches that we believe to be necessary for the staff to reach a decision.

We conclude that the approaches listed below can be completed in a few months, and will result in information that will contribute to a decision on the acceptability of Sequoyah as designed and constructed.

1. Approach B.1: Determine site-specific SSE response spectra from strong motion records of appropriate magnitude and distance.

2. Approach B.2: Determine site-specific SSE response spectra from strong motion records of appropriate intensity.

3. Approach C.1: Reevaluate the original seismic structural and floor response spectra analysis, taking into account more realistic methods and material properties, as well as site-specific SSE response spectra.

4. Approach D: Reevaluate the OBE to see whether it meets the recurrence interval criteria of Appendix A to Part 100.

5. Approach E.3: Compare the probability of the SSE being exceeded at the subject plant with that at other TVA plants that meet the SRP criteria.

It is possible that Sequoyah can be shown to be acceptable by completing some, but not all of the above-listed approaches. For example, Approach B.1 (site-specific response spectra based on records of appropriate magnitude and distance), in combination with Approach C.1.a (Regulatory Guide 1.61 damping factors), might be sufficient to establish the seismic acceptability of Sequoyah. However, it is desirable to reach a decision on the acceptability of Sequoyah as soon as possible, since a detailed seismic reanalysis would probably result in a delay in fuel loading until the analysis had been completed and any necessary plant modifications had been made. The Working Group believes, therefore, that it would be prudent for the applicant to undertake all the above approaches simultaneously, rather than sequentially. This will reduce the likelihood that a decision on the issue will be delayed, thus minimizing the delay in fuel loading that might result if a detailed reanalysis is found to be necessary.

Therefore, we recommend that the applicant be requested to undertake all of the above-listed approaches immediately, and to submit the results of each approach as soon as they become available. If a decision can be reached before all the approaches are completed, the remaining effort could be curtailed. If, on the other hand, the results of all of the above approaches are required to reach a decision, their simultaneous, immediate pursuit will permit a decision to be made at the earliest possible time. The Working Group believes that all the above approaches can be completed by September of 1978. Approach B.1 is currently under way, with the results scheduled for submittal by July 1, 1978.

We recommend that following submittal of the above information, the staff be directed to evaluate it and reach a finding shortly thereafter regarding the acceptability of Sequoyah as designed and constructed. If it is not found to be acceptable, and a detailed reanalysis is required, such an effort could involve the most appropriate spectra or suite of time histories developed in Approach B (reevaluate the response spectra), Approach C.2 (reanalyze plant structures and floor response spectra), and Approach

C.3 (reanalyze plant components, systems, piping and restraints). We recommend that detailed reanalysis not be undertaken until a decision has been made regarding the acceptability of Sequoyah as designed and constructed.

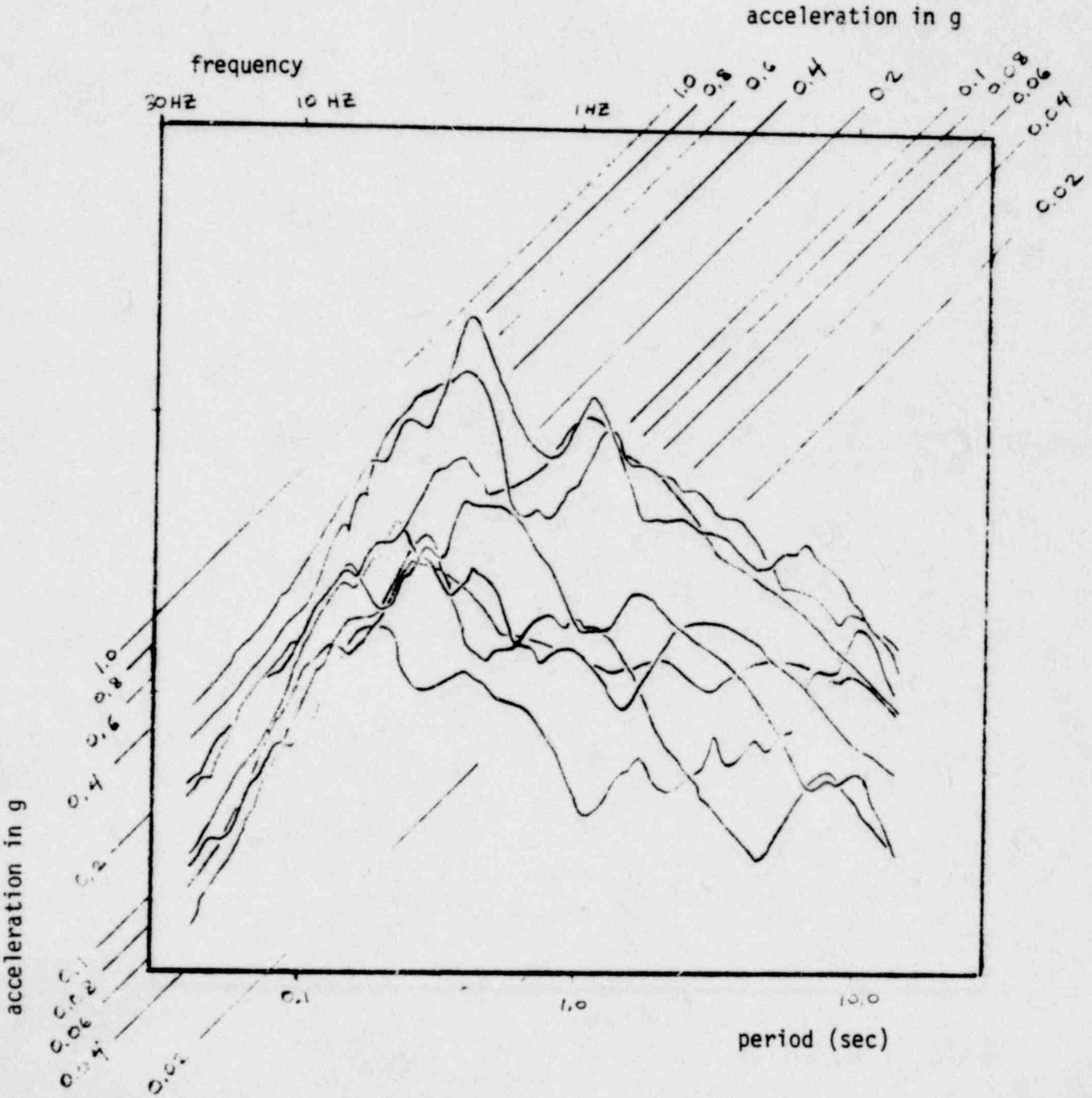
To scope the potential results from Approach B.1 in combination with the major part of C.1, we have plotted readily available measured response spectra from four earthquakes and compared them with the design response spectra for concrete structures for Sequoyah and Phipps Bend. The Phipps Bend plant, as discussed earlier, is in the Southern Valley and Ridge tectonic province and was designed to meet the recommendations of the SRP. The comparisons shown account for the use of Regulatory Guide 1.61 damping factors with Sequoyah. Figure 3 shows the perpendicular horizontal components of ground motion recorded during four earthquakes that fall within the range of investigation discussed above in Section III.B.1. Figures 4 through 7 show the spectra for each of four earthquakes, compared with the design response spectra for concrete structures. In all of these figures, 7% damping is used with the measured spectra and the Phipps Bend design spectrum; 5% damping is used with Sequoyah. This procedure thereby accounts for the change to 7% damping for concrete structures at Sequoyah recommended in Approach C.1.a. As may be seen, the design spectrum for Sequoyah falls below the spectra for the Parkfield earthquake, is comparable to the spectra for the Helena earthquake, and is above the spectra for the Lytle Creek and San Francisco earthquakes. These figures are preliminary in nature. However, we believe that comparisons of this kind, if extended to include comparisons with other rock-site records from a sufficient number of earthquakes of appropriate magnitude and distance, and extended to include the effects of other factors discussed in Section III.C.1, can provide a basis for a timely decision on the acceptability of Sequoyah. A similar approach may be taken to reach a decision for Watts Bar and Bellefonte, but on a more extended schedule.

In summary, the Working Group recommends that the applicant be requested to immediately undertake the above-described program to provide the staff with sufficient information to decide whether or not a detailed reanalysis of Sequoyah is required. We believe that the necessary information can be developed in a few months, and that a staff decision on the issue can be reached shortly thereafter.

FIGURE 3

STRONG MOTION RECORDS FROM FOUR EARTHQUAKES

All records adjusted to 7% damping
Two horizontal records from each earthquake
All records are from rock sites



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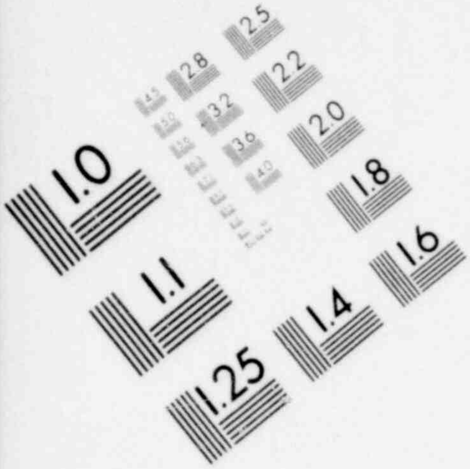


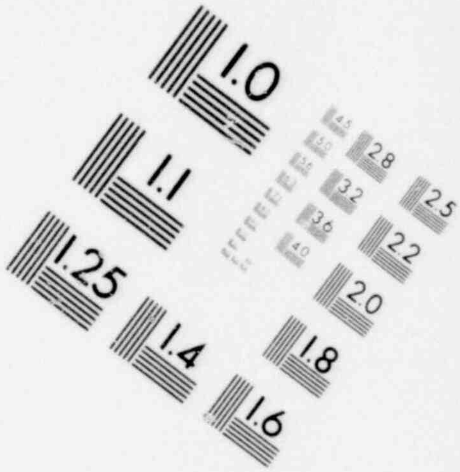
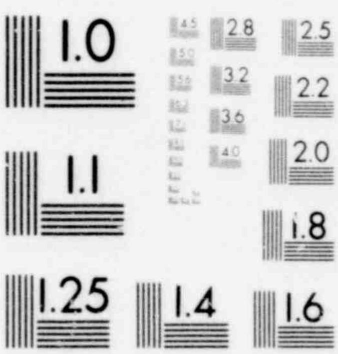


IMAGE EVALUATION
TEST TARGET (MT-3)



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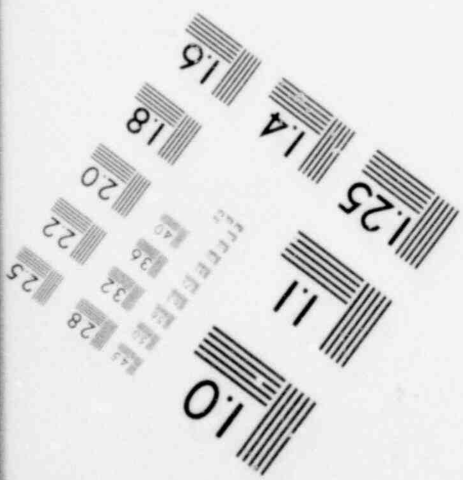
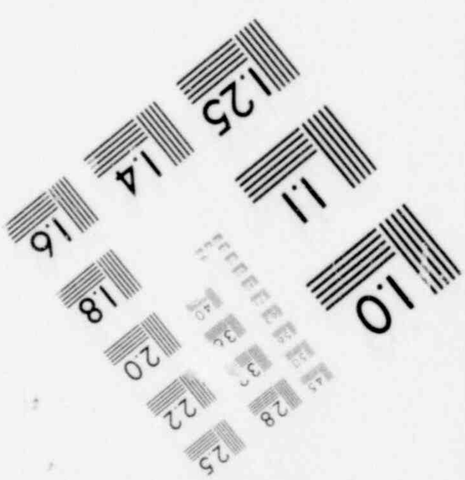
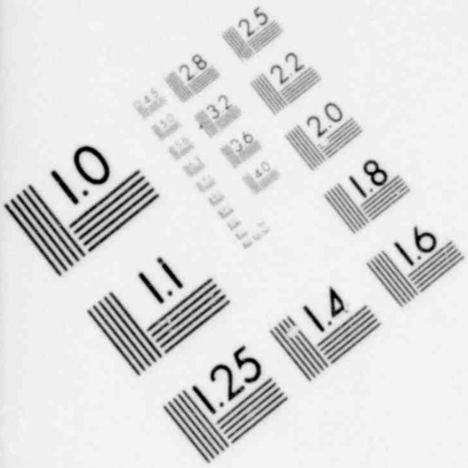






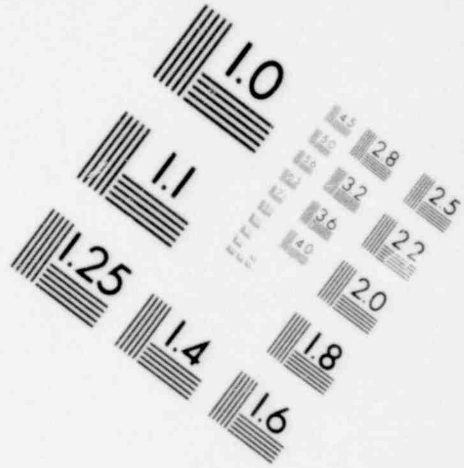
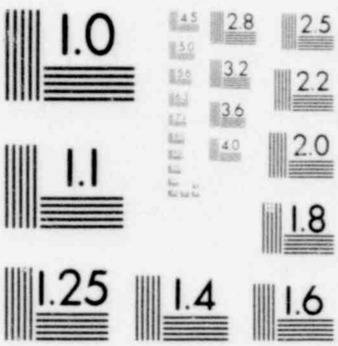


IMAGE EVALUATION TEST TARGET (MT-3)



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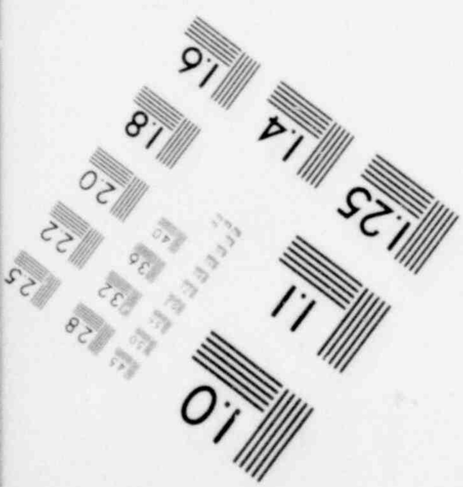


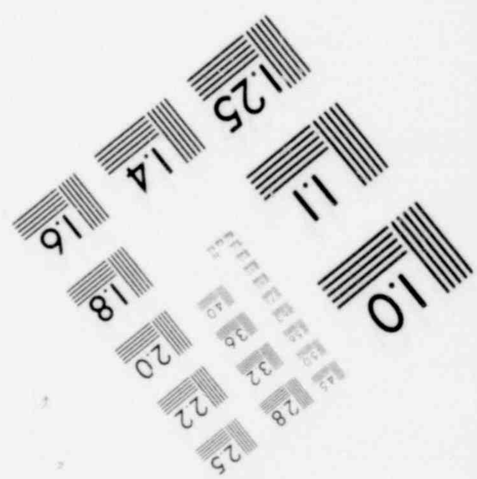
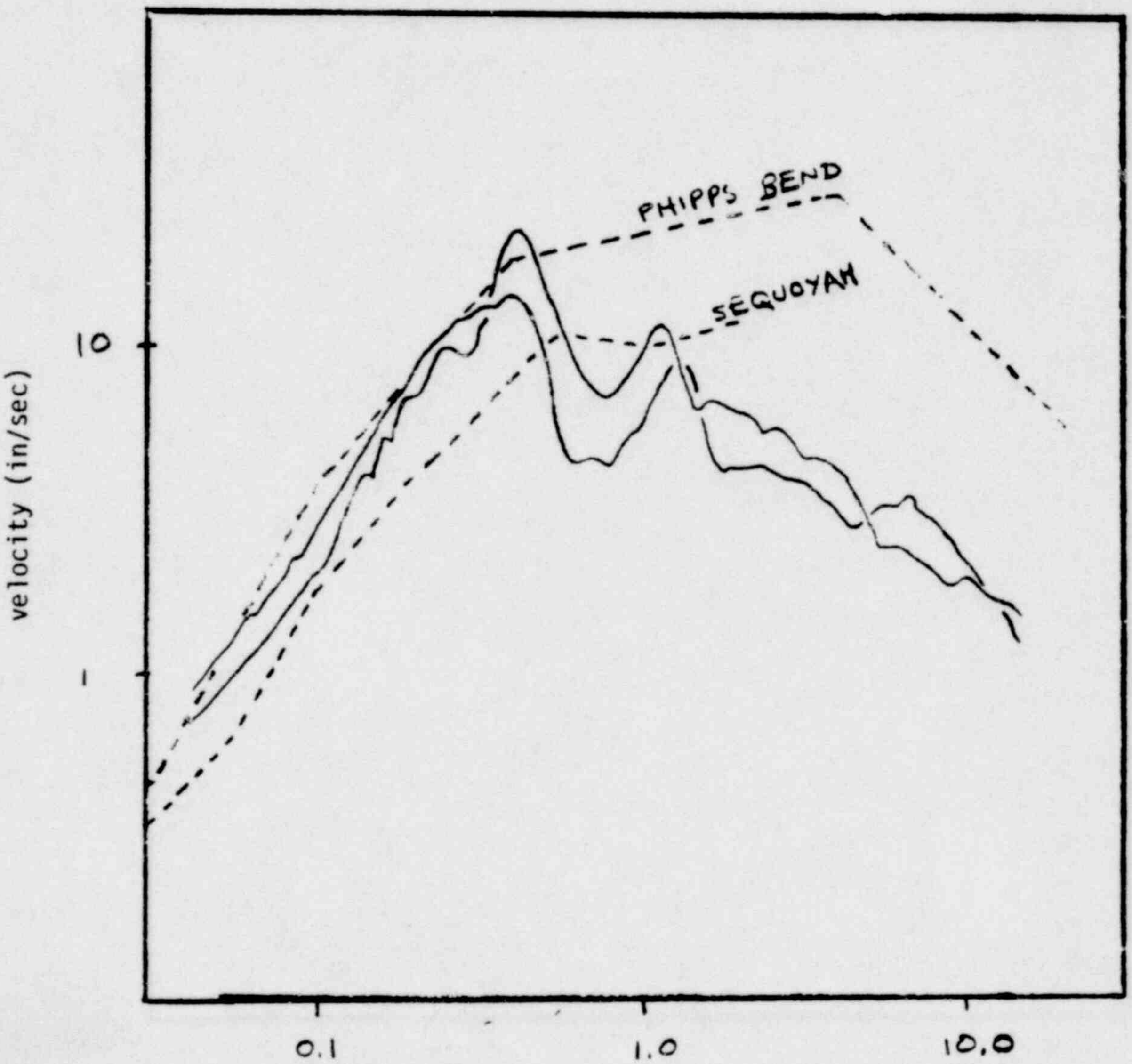


FIGURE 4

PARKFIELD RECORDS VS. SEQUOYAH AND PHIPPS BEND DESIGN SPECTRA

Recorded at Temblor 6/27/66
Distance: 10 to 35 km*
Site Intensity: VI
Range of published body wave and local magnitude: 5.3 - 5.9



*Range of published distances. Variation due to differences in way distances are measured.

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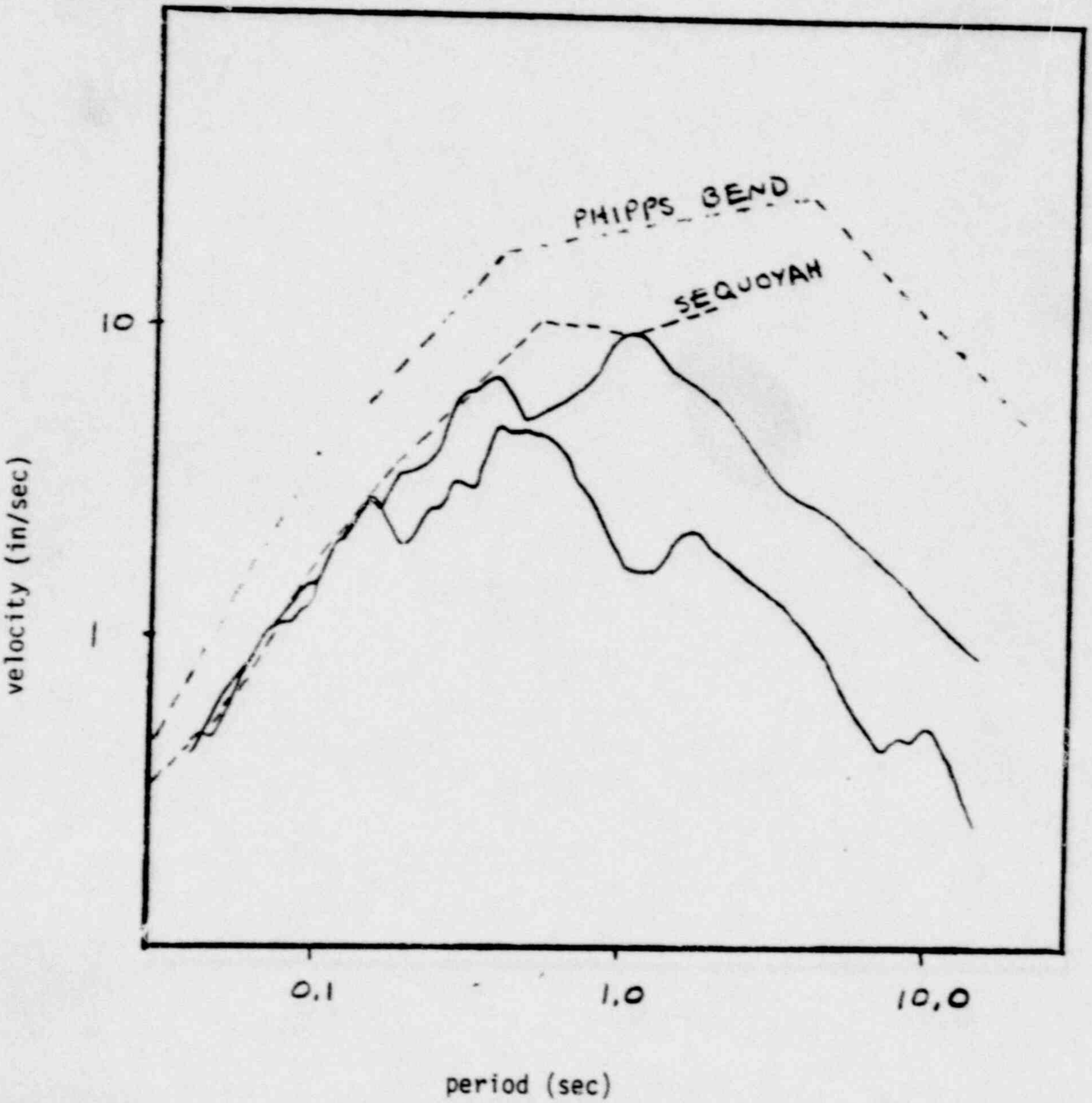
FIGURE 5

HELENA RECORDS VS. SEQUOYAH AND PHIPPS BEND DESIGN SPECTRA

Recorded at Federal Building 10/31/35
Distance: 7 km
Site Intensity: VII
Range of published body wave and local magnitudes: 5.5 - 6.0



A-009

FIGURE 6

LYTLE CREEK RECORDS VS. SEQUOYAH AND PHIPPS BEND DESIGN SPECTRA

Recorded at Allen Ranch 9/12/70
Distance: 19 km
Site Intensity: V-VI
Range of published body wave and local magnitudes: 5.4 - 5.7

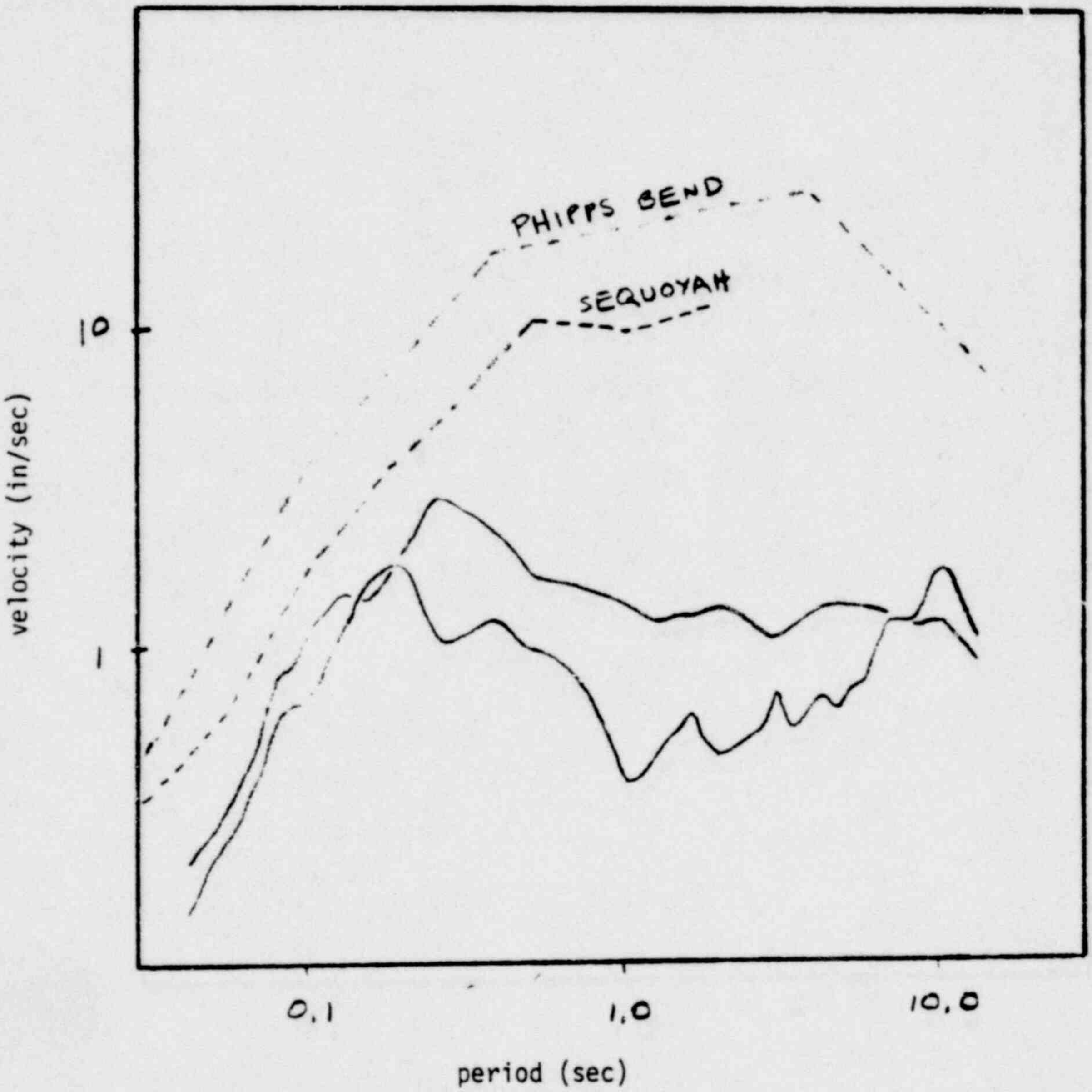
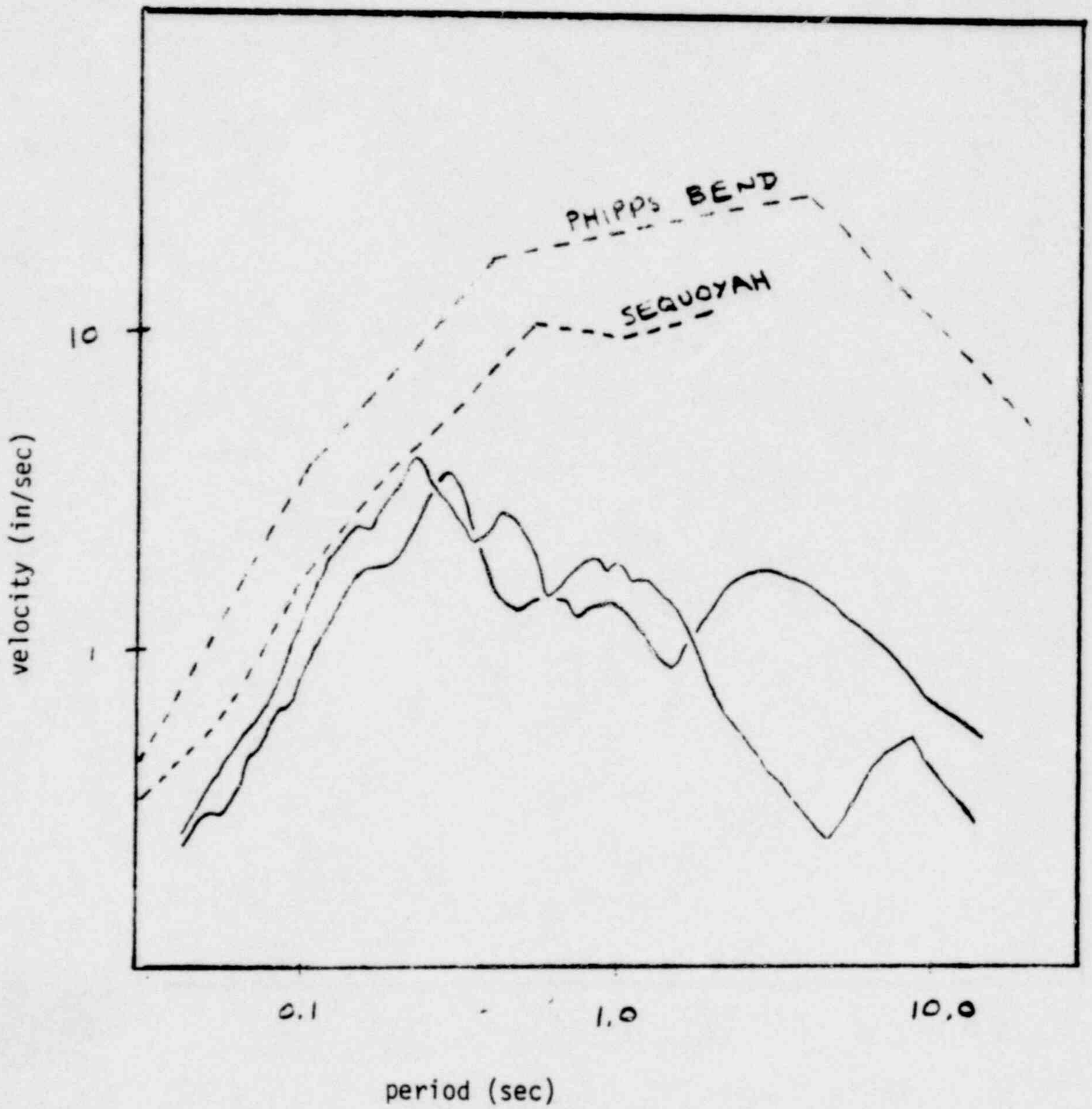


FIGURE 7

SAN FRANCISCO RECORDS VS. SEQUOYAH AND PHIPPS BEND DESIGN SPECTRA

Recorded at Golden Gate Park 3/22/57
Distance: 13 km
Site Intensity: VI
Range of published body wave and local magnitudes: 5.3



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

MAR 30 1978

APPENDIX A

MEMORANDUM FOR: J. Bennett, Division of Site Safety and Environmental Analysis
S. Chan, Division of Systems Safety
J. Rajan, Division of Systems Safety
L. Reiter, Division of Site Safety and Environmental Analysis
H. Rood, Division of Project Management

FROM: Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation

SUBJECT: WORKING GROUP ASSIGNMENT

You have been assigned to form a Working Group to develop a method for resolving the TVA seismic issue in order to assure that (1) Staff resources will be used efficiently, and (2) Staff decisions on the three facilities involved will be made in a timely manner consistent with construction completion schedules.

You are to form a dedicated team that will work essentially full-time on this task until its completion. At its initial meeting set for 9:00 a.m. on April 3, 1978, the Group will select one of its members to act as the Chairman.

The Charter for the Working Group is attached.

Harold R. Denton for

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosure:
Charter

cc w/enclosure:

R. Boyd	J. Knight
H. Denton	K. Kniel
R. Mattson	C. Stepp
D. Vassallo	R. Bosnak
W. Gammill	I. Sihweil

CHARTER FOR

WORKING GROUP ON TVA SEISMIC ISSUE

DEFINITION OF PROBLEM: Three TVA plants, SEQUOYAH, WATTS BAR, and BELLEFONTE, are being designed and constructed to seismic criteria that predate the Standard Review Plan. Using the current SRP criteria, and the intensity value for the Giles County earthquake recently assigned by the USGS, the design basis ground motion would be defined by Regulatory Guide 1.60 response spectra scaled to 0.25g. The seismic design basis motion actually being used varies among the three sites but all spectra are scaled to 0.18g. The SRP criteria use the Trifunac and Brady correlation of intensity vs acceleration. This correlation is more recent and based on more data than previous correlations. Also, SEQUOYAH and WATTS BAR have been designed using spectra that are less conservative than the Regulatory Guide 1.60 spectra. In order to reach an affirmative licensing decision, it must be shown that all safety related structures, systems, and components in these plants can withstand the effects of the SSE without loss of capability to perform their safety functions. The essential point is that it must be shown that each plant conforms to the Commission's regulations or that there are sound technical bases for deviations. Because of their vintage, it is not necessary that the plants conform to the Standard Review Plan. It is NRR policy (see NRR Office Letter #9) that deviations from the Standard Review Plan are permitted provided they are identified and justified to the staff's satisfaction.

OBJECTIVE: The objective of the Working Group is to evaluate the problem, consider various methods of resolution, and recommend a path of resolution that assures safety taking into account differences in the time and effort that would be required by applicant and staff, and the extent to which seismic reanalysis of the plant would be required. The methods of resolution for the three facilities may differ in detail because of differences in the extent of deviation from Standard Review Plan requirements as a result of different staff approval dates for the designs.

SCHEDULE: The following tasks will be conducted on the schedule indicated.

TASK 1 - Problem definition and identification of possible approaches to solving the problem. The problem will be defined by the group, starting with the general and proceeding to the specific aspects, in as much detail as necessary. The group will generate a list of possible approaches to solve the problem. The group should take into account the significant efforts expended by DSE Geosciences Branch to date toward resolution of the problem. The attachment to this charter provides a synopsis of those efforts. This task should take about a week. NRR management will be briefed on the results of this task upon its completion.

TASK 2 - Evaluation of each approach. A scope of evaluation will be determined for each alternative approach, and the group will organize itself so as to evaluate each alternative approach. The evaluation will include a comparison of the time and effort required to pursue the approach, the development of an outline of the activities involved, and an estimate of the relative likelihood that the approach would be successful. This task should take about a week. NRR management will be briefed on the results of this task upon its completion.

TASK 3 - The various alternatives will be compared and a determination made as to whether any are sufficiently promising to warrant action. This task will take about two weeks, including the writing of a summary report to NRR management describing the alternatives considered and the group recommendation.

TASK 4 - TVA will be called in and the results of the Working Group efforts will be discussed. TVA will be told of the NRR decision on the recommendations of the Working Group and encouraged to pursue that course or, if warranted, to explore alternative approaches on their own.

APPENDIX B

BIBLIOGRAPHY

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Appendix XVII

APPENDIX XVII
Seismicity of Southeastern New York
and Northern New Jersey

Reports

Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey

Abstract. *Seismic activity in the greater New York City area is concentrated along several northeast-trending faults of which the Ramapo fault appears to be the most active. Three nuclear power plants at Indian Point, New York, are situated close to the Ramapo fault. For a reactor site in use for 40 years, the probability that the site will experience an intensity equal to or in excess of the design (safe shutdown) earthquake is estimated to be about 5 to 11 percent.*

The Ramapo fault system, which bounds the Triassic-Jurassic Newark graben on its northwest side, has been known for about 100 years but has been commonly presumed to be an inactive fault. Prior to the advent of plate tectonic concepts in the late 1960's, Triassic deformation was generally thought to be "the last dying gasp of Paleozoic orogeny." The separation of North America from Africa in the Triassic-Jurassic is generally recognized as the last great tectonic event in the area, which greatly influenced the subsequent geologic history. The hypothesis that the fault is dead now appears to have been tenable only in the near absence of local instrumental earthquake data. Although a number of workers since 1964 (*1*) have suggested correlation of earthquakes with this and other nearby faults, the data were insufficient to definitely establish such correlations. The recent improvement in the seismographic coverage for this area enabled us to determine precise locations for 33 earthquakes and many focal mechanism solutions. The results clearly indicate that seismic activity is related to faults that trend northeast to north-northeast.

More people live within 80 km of the Indian Point reactors than within the same distance from any other nuclear power plant in the United States. The reactors are situated within 1 km of a major branch of the Ramapo fault system. As late as 1972, however, the Final Environmental Statement (*2*) for Indian Point reactor unit 2 stated, "There are no truly major faults in or near the site." This was disputed by the State of New York, and that concern led to hearings on seismic safety held before the Atomic Safety and Licensing Appeal Board (ASLAB) of the Nuclear Regulatory Commission (NRC) in 1976 and 1977. In 1975 Ratcliffe (*3*) recognized an individual fault, possibly of the Ramapo system, that passes beneath reactor unit 3. Since then, considerable effort has been devoted to geologic mapping and studies of local earthquakes near the reactors (*2, 4–6*). Since late 1976, several shocks have occurred on the Ramapo fault both to the southwest and northeast of the plant as well as almost directly beneath it.

Scientific information and judgment are intimately involved in several of the questions litigated in the NRC hearings on Indian Point. Since we participated as scientific experts in these hearings, we also briefly summarize our views about some of the difficulties encountered in applying the existing federal regulations known as Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" (*7*), to sites in the East.

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Figure 1 shows earthquakes in the northeastern United States and adjacent parts of Canada from 1970 to 1977 as detected by networks in the area. Since 1970 the number of seismic stations in this region has steadily increased. For the period covered in Fig. 1 the station coverage is more complete for New York State and adjacent areas and poorer for New England. For New York and adjacent areas the detection is probably complete for events larger than magnitude (m_b) 2. Since 1974 the detection is complete for $m_b > 1.8$ for the area near the Ramapo fault. We determined the magnitudes (m_b) of these and other events used in this report, using Nuttli's scale (*8*).

The overall spatial distribution of these events is remarkably similar to that of historical events for the period 1534 through 1959 (*9*). Both the record in Fig. 1 and the historic shocks show concentrations of seismic activity in the northern, western, and southeastern parts of New York State; the central part of the state is essentially aseismic.

Earthquake locations, faults, and focal mechanism solutions for southeastern New York and northern New Jersey are shown in Fig. 2. A more detailed description of the seismic data is given else-

Fig. 1. Epicenters of earthquakes (1970 through 1977) in northeastern North America located by various networks in the area. Note the northeast alignment of earthquakes in northern New Jersey and southern New York. Stars denote events of unknown origin.

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where (6, 10). Figure 2 shows 33 events ($\leq m_b \leq 3.3$) for the period 1962 to 1977 located with an accuracy of 5 km or better. Instrumental data for events prior to 1962 were generally found to be insufficient to allow us to meaningfully investigate their possible correlation with faults.

Figure 2 shows a strong spatial correlation of epicentral locations with surface traces of faults in this area. A large majority of the events lie on or very close (within 1 to 2 km) to the faults. Furthermore, an examination of the focal mechanism solutions shows that for each solution one of the nodal planes trends north to northeast, which is also the predominant trend of the faults in this area. This remarkable spatial correlation and the consistency of the nodal planes with the trend of the mapped faults leave little doubt that earthquakes in this area occur along preexisting faults.

About half of the events plotted in Fig. 2 are almost colinear and lie along or close to the Ramapo fault system. The Ramapo fault system can be traced as a single continuous fault between point A and event 26; near event 26 it splays into a number of branches (5, 11). One of these branches (the Thiells fault) passes within 1 km of Indian Point (triangle in Fig. 2). The association of seismic activity with this major fault system is particularly clear in Fig. 3, where the hypocenters of events with reliable focal depths occurring within 10 km of the fault trace are projected onto a vertical cross section perpendicular to the trend of the fault. The southeasterly dip of the hypocenters in Fig. 3 agrees with the dip of the fault determined from focal mechanisms and geologic evidence (4, 5).

Relatively little activity is found within the Triassic Newark basin, the area between the Ramapo fault and the Hudson River (Fig. 2). Similarly, very little activity (Figs. 1 and 2) is found to the northwest of the line connecting events 4 and 7. Some activity is found to the southeast of the Ramapo fault in the area east of the Hudson River. Hence, most of the activity in Fig. 2 is located within and bounding the Precambrian Hudson Highlands.

The Ramapo fault system has experienced at least four periods of movement from Precambrian to Jurassic time (11). Although Triassic-Jurassic movement has not been demonstrated along the Ramapo fault on the east side of the Hudson River, seismic activity is nearly continuous along the entire zone A–A'. Figure 3 indicates that seismic slip on the Ramapo fault extends to a depth of about 10 km. In contrast, much of the seismic activity northwest of the Ramapo fault occurs at shallow depths (1 to 2 km) and is of swarm type. This evidence suggests that the Ramapo fault may have a greater seismic potential than adjacent faults northwest of it.

Focal mechanism solutions indicate that high-angle reverse faulting is the predominant mode of contemporary fault movement in this area; this differs from the sense of movement during the Triassic-Jurassic (4, 5, 11). Thus, the state of stress in this area has changed with time. The present maximum compressive stress direction is nearly uniform and trends west-northwest and indicates reactivation of southeast- or northwest-dipping faults.

In a plate tectonic framework, the east coast of North America was located along a plate boundary during the Triassic but is presently a region interior to a lithospheric plate. In a worldwide study of intraplate phenomena, Sykes (12) found that intraplate earthquakes, such as those in eastern North America, tend to occur along major preexisting faults that were reactivated by continental fragmentation in the Mesozoic or Cenozoic eras. Many of these reactivated faults are still seismically active today but, of course, not to the extent that they were during the initial stages of continental rifting.

On the basis of focal mechanism solutions, Aggarwal (10) postulated that the activity in the New York City area may belong to a larger seismotectonic province extending southwesterly to Virginia approximately along the Fall Line. Cretaceous and Cenozoic deformation is found along that zone in Delaware, Maryland, and Virginia (13). The abrupt changes in the courses of several major rivers near the Fall Line have been at

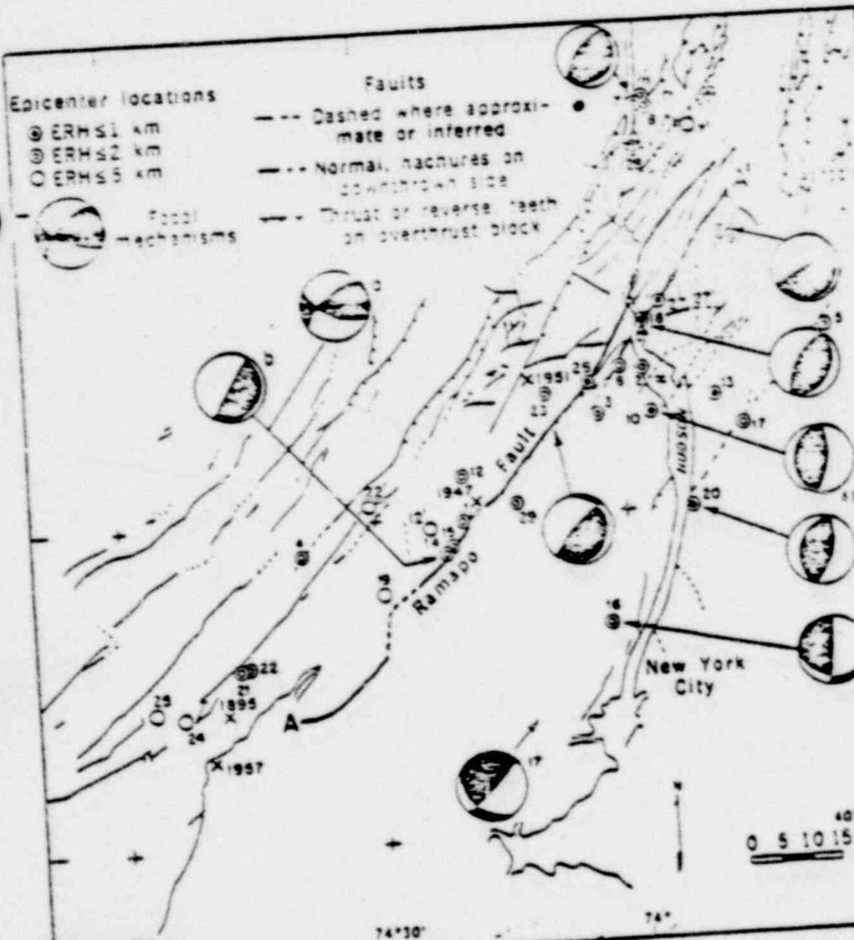

Fig. 2. Fault map (4, 5, 29) of southeastern New York and northern New Jersey showing epicenters (circles) of instrumentally located earthquakes from 1962 through 1977. Indicated uncertainties (*ERH*) in epicentral locations represent approximately two standard deviations. Focal mechanism solutions are upper-hemisphere plots; the dark area represents the compressional quadrant. For event 14 there are two possible focal mechanism solutions; the data, however, are more consistent with solution *b* than *a*. The Ramapo fault and two of its major branches (*A–A'*) are shown by the heavy lines; x's denote locations for other events discussed in the text. The solid triangle shows the location of the Indian Point nuclear power reactors.

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tributed to Cenozoic deformation (14), possibly along reactivated northeast-trending faults.

[illegible]ough the instrumentally located [illegible]s in Fig. 2 are small in magnitude and cover only a 15-year time span, the historic record of felt shocks shows that much larger earthquakes have occurred in the greater New York City area. Among the larger events known to have occurred during the last 250 years are three shocks (1737, 1884, and 1927) of intensity VII on the modified Mercalli (MM) scale and three (1783, 1895, and 1957) of intensity VI. Since precisely located shocks of the last 15 years show such a close relationship to northeast-trending faults, the larger felt shocks, for most of which precise locations are not available, are most reasonably interpreted as occurring along the same faults. In other areas where a longer record of instrumental locations is available, larger shocks show an even greater tendency than smaller shocks to be localized on major throughgoing faults (15).

Large uncertainties are inherent in efforts to locate earthquakes solely from felt reports (9); consequently, the larger events cannot be unequivocally associated with a specific fault. Within the uncer[illegible]y in the [illegible], however, some of [illegible] events may have occurred on the [illegible]mapo fault. The 1884 shock was felt from Maryland to New Hampshire; [illegible]en bricks and cracked plaster were reported at 30 sites from eastern Pennsylvania to central Connecticut. Although two recent catalogs (9, 16) place the epicenter in Brooklyn, New York, both list Rockwood (17) as their original source of data. He placed the center of the zone of maximum shaking in northeastern New Jersey. An epicentral location in that area is supported by newspaper reports of foreshocks that were felt in Paterson, New Jersey (18). Felt reports for the 1737 shock are much more limited. The smaller felt area of the 1927 event places it somewhere along the north shore of New Jersey near Asbury Park, well off the Ramapo fault.

Felt reports for shocks in 1895 and 1957 and limited instrumental data for 1957 (m_b = 4.4) indicate that they occurred near the southwest end of the Ramapo fault near point A in Fig. 2. The 1783 earthquake was located by Smith (9) in New Jersey near the Ramapo fault. In addition, felt reports and limited in[illegible]rumental data indicate that an earth[illegible]ke (m_b = 3.9) in 1951 occurred about [illegible]km northwest of the Ramapo fault and a shock of m_b = 3 in 1947 occurred on or close to the fault (Fig. 2).

Thus, on the basis of the precise locations in Fig. 2, the 120-km length of the Ramapo fault, and the history of felt shocks in the area, we conclude that the

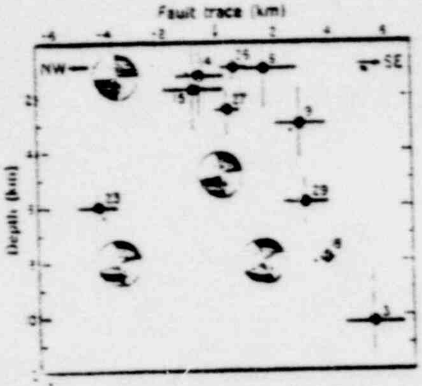

Fig. 3. Composite (stacked) vertical cross section showing focal depths and focal mechanism solutions (the dark area is the compressional quadrant) for events within 10 km of the Ramapo fault trace. The event number is keyed to the epicenter number in Fig. 2; only those events are plotted for which reliable focal depths could be determined. Bars represent one standard deviation. Northeast of epicenter 26 (Fig. 2) horizontal distance is measured from one of two major [illegible]ches of the fault on the basis of fo[illegible]chanism solutions.

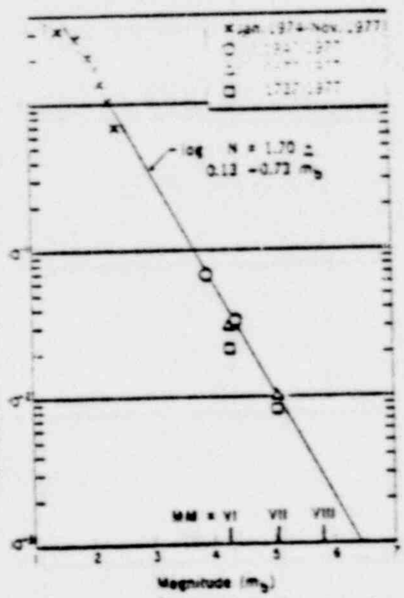

Fig. 4. Cumulative number (N) of earthquakes of magnitude m_b or greater per year as a function of magnitude. Data sets are each for the 120-km-long segment of the Ramapo fault and for shocks located within 10 km of the fault. The question mark denotes the minimum value, that is, the incomplete detectability of events of that magnitude. The slope of the curve, 0.73, was determined independently for recent shocks in New York and adjacent areas. The intensity-magnitude relationship is from (19). The uncertainty, ± 0.13, in the value of a (log N = a − bm_b) represents the 95 percent confidence interval.

Ramapo fault is capable of generating an earthquake of at least intensity VII, the nominal design earthquake of the Indian Point reactors. The relatively short period of historical data (about 250 years) is not sufficient to establish an upper bound to the size of shocks for a particular region unless some geologic or tectonic criteria are invoked.

Perhaps the most important question involving earthquakes and the seismic safety of Indian Point is: How active is the Ramapo fault? We calculate the probability of occurrence (Fig. 4) of earthquakes of intensity VII and VIII within 10 km of the fault by extrapolating the occurrence of smaller shocks to larger magnitudes using the well-known relationship log N = a − bM, where N is the cumulative frequency of shocks, M is the magnitude, and a and b are constants. The b value, 0.73, was obtained for shocks recorded throughout New York and New Jersey (10) and is assumed to be applicable to this subregion. The x's in Fig. 4 represent the rate of occurrence of shocks within 10 km of the Ramapo fault between points A and A' that occurred from 1974 to 1977. We fitted the solid line through the x's, using the slope 0.73.

To check the predictability of the frequency-magnitude relationship thus determined, the rates of occurrence of historical events discussed [illegible] are shown in Fig. 2 for three different periods. We estimated the magnitudes of the historical events from the following relation between intensity (I) and magnitude, m_b = −0.20 ± 0.05 + (0.75 ± 0.03)I, for the East Coast (19). The rates of occurrence of events for the periods 1947 to 1977 (1951 and 1957 earthquakes) and 1887 to 1977 (1957, 1895, and 1884 earthquakes) are in excellent agreement with those predicted by the solid line (Fig. 4). Squares (Fig. 4) indicate the rate of occurrence of events up to intensity VII for the period 1737 through 1977 if both the 1737 and 1884 (MM VII) earthquakes occurred on or near the fault.

For the entire fault, the relationship between N and m_b predicts shocks of MM ≥ VII about once per 97 years, if no upper bound is placed on the maximum size of possible earthquakes. If, however, we assume that shocks of MM ≥ IX or MM ≥ VIII cannot occur, then the corresponding recurrence times for MM ≥ VII are about 105 and 137 years. These estimates are subject to possible systematic errors in determining magnitude, to uncertainties in the relation between m_b and I and the b value, and to possible errors in extrapolating

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Table 1. Probability of equaling or exceeding intensities VII and VIII at Indian Point.

Method	Estimated recurrence time (years)		Probability for exposure interval of 40 years (%)	
	VII	VIII	VII	VIII
1. Earthquake frequency-magnitude relationship (Fig. 4)				
a. Events within 10 km of site only:				
No upper bound on size of events	580	2050	6.7	2.0
Excluding events of MM ≥ IX	630	2870	6.1	1.4
Excluding events of MM ≥ VIII	810		4.8	
b. Events along entire Ramapo fault:				
No upper bound on size of events	300	1050	12.5	3.7
Excluding events of MM ≥ IX	340	1830	11.1	2.1
Excluding events of MM ≥ VIII	530		7.3	
2. MM VII shocks occur at random once per 100 years along faults of total length 252 km	1800		2.2	
3. Probabilistic calculation by McGuire (20) based on historic events				
a. No upper bound on size of events	1000	3160	3.9	1.3
b. Excluding events of MM ≥ IX	2240	7080	1.8	0.6

the data to larger magnitudes. We estimate that they may be uncertain by a factor of 2 to 3.

Using this log N–m_b relationship, we derive in Table 1 (method 1) the recurrence times for MM intensities at the reactor site to equal or exceed intensities VII and VIII for three different upper bounds on the size of possible earthquakes. The [illegible] probabilities of equaling or exceeding intensities VII and VIII for an exposure interval of 40 years, the presumed lifetime of the nuclear power plants, are also tabulated.

First (method 1a) we calculate the contributions to site intensities only from earthquakes within 10 km of the site. The intensity at a distance of up to 10 km, for earthquakes of moderate size, is expected to be nearly the same as that at the epicenter (*20*). Thus, the probability that site intensity will equal or exceed, say, VII once in 40 years from earthquakes within 10 km of the site is equivalent to the probability of occurrence of earthquakes of MM ≥ VII. For an earthquake more distant than 10 km the probability that its intensity at the site will equal or exceed a given intensity is a function of the size of the earthquake and the decay of intensity with distance. Approximating the fault zone as a line source, and assuming the intensity-distance relationship of McGuire (*20*) for the East Coast, we used the procedure developed by Cornell (*21*) in method 1b (Table 1) to integrate over the entire fault length. The probabilities of equaling or exceeding intensities VII and VIII thus calculated are not greatly affected by the use of a line instead of an areal source but are sensitive to the intensity-distance relationship. Other attenuation curves (*22*), also considered appropriate for the eastern United States, give higher estimates than those in Table 1 (method 1b).

Table 1 shows that the calculated probabilities are not greatly dependent on the maximum size of earthquakes. Estimates obtained by excluding events of MM ≥ IX, however, are probably more realistic. Thus, the probability that the MM intensity at the reactor site will equal or exceed VII, the design (safe shutdown) earthquake, once in 40 years is about 5 to 11 percent. For MM ≥ VIII, the probability is about 2 percent.

Method 2 (Table 1) is a more approximate calculation based on the historic rate of occurrence of shocks of intensity VII in the greater New York City area. We take the rate as 2.5 shocks per 250 years since the 1927 event is assigned MM VII in one catalog (*16*) and VI in another (*9*). We assume that these shocks, like those in Fig. 2, occur along major northeast-trending faults, which we estimate have a total length about three times that of the Ramapo fault. Assuming a rupture length of about 5 km for MM VII (*23*), we obtain a total of 72 rupture segments, four of which we take as being within 10 km of Indian Point. This gives a recurrence time of 1800 years for MM VII within 10 km of the plants. This calculation suffers from our poor knowledge of the lengths of rupture zones for eastern earthquakes and of their extent in depth. Since precisely located shocks in the area have computed depths that are less than 11 km, the calculated recurrence time is not greatly affected by the depth of seismic faulting in individual shocks.

McGuire (*20*) calculated probabilities for exceeding given intensities for a number of sites near the East Coast by randomly varying the locations of historic shocks within individual seismic provinces. He showed that his method is stable to uncertainties in the designation of seismic provinces and the size of specific shocks. The approach used in the federal siting appendix (*7*), however, is highly sensitive to those parameters. Differences of up to two MM intensity units can be obtained with the existing procedure, depending on how the seismic provinces are drawn. For a 10,000-year return period, McGuire calculated a shock of intensity 8.3 for New York City under the assumption that shocks larger than MM IX cannot occur. If his results are applied to Indian Point, we obtain return periods of 2240 and 7080 years (method 3, Table 1) for intensities VII and VIII, respectively. Some of his other calculations, which probably are not as realistic as the above, yield shorter return periods for the same intensities.

We think that method 1 provides the most realistic estimate since it is based on data from the area of the Ramapo fault, whereas in method 3 a random distribution of activity in space is assumed. [illegible] factor of 10 than th[illegible] seismologists (*24*) for the same intensity; their estimate suffers from an assumed random distribution of activity in space and much more limited data than that used in this study. The 5 to 11 percent probabilities we obtained, of course, should not be equated with the probability of significant damage or accidental radioactive release.

Indian Point reactors 2 and 3 are designed for an input acceleration of 15 percent of the earth's gravitational acceleration, g (*25*), for very high frequencies. The power plants, however, are situated within a few kilometers of branches of the Ramapo fault system, where earthquakes as shallow as 1 to 2 km occur. Now that we have demonstrated that the Ramapo fault is active, it is not clear whether nearfield accelerations, which can be as high as $0.5g$ at high frequencies for moderate-size earthquakes (*26*), have been adequately considered in the design of the reactors. The Advisory Committee on Reactor Safety of NRC recently recommended a minimum design of 25 percent of g for new reactors in the East (*27*).

We believe that our calculations provide the public and policy-makers with

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quantitative numbers against which to judge whether the risk is acceptable or not. It is clear, however, that not a great [many] of the approximately 70 nuclear power plants now in operation in the United States can be allowed to operate at a risk of 5 to 11 percent without the probability becoming high that shaking will exceed that of the design earthquake for at least one of them over a 40-year period.

The Indian Point seismic hearings before NRC brought out a number of problems about the applicability of the existing federal regulations (7) to sites in the East. By these regulations a capable fault is defined on the basis of either (i) demonstrated fault movement younger than 500,000 years or (ii) macroseismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault. There is no evidence for surface breakage in any earthquake in the central or eastern United States, with the possible exception of questionable ground breakage during the New Madrid, Missouri, earthquakes of 1811–1812. Yet we know that a number of large and damaging shocks have occurred in these areas. The Ramapo fault is typical of many eastern [fault]s in that almost all of the rocks in the [regi]on, with the exception of scattered postglacial deposits less than 15,000 years old, are older than 150 × 10^6 years. Hence, it is very difficult to tell if earth movements are as old as 150 × 10^6 years or if they happened in the past 0.5 × 10^6 years. Thus, surface breakage is not a good indicator of either "capability" or seismic risk for many eastern sites.

The hearings demonstrated that the word "macroseismicity," which is not defined in the regulations, is rarely used or defined by seismologists. Various scientific witnesses differed to a large extent in their concept of macroseismicity (28). For much of the East, instrumental data of sufficient precision to demonstrate a relation to specific faults are very limited in time. Hence, it is not surprising that no fault in the central or eastern United States has as yet been declared legally capable.

In the absence of capable faults, the concept of "tectonic provinces" is used in deriving the intensity of the design earthquake from the historic record of shocks. The intensity at the site is calculated by moving historic shocks in the [sam]e province to the site and shocks in adjacent provinces to the closest point within those provinces (if the shocks cannot reasonably be correlated with a tectonic structure). Although this procedure may appear conservative in terms of design safety, it is so only if reasonably large tectonic provinces are used. At the Indian Point hearings it was clear that the scientific witnesses had greatly varying opinions about the size, designation, and concept of tectonic provinces (28). These ambiguities can result in a number of small provinces being invoked to keep critical historic shocks at a distance such that their intensities at the site are much lower than those near the epicenter. In the case of Indian Point, this leads to a design earthquake of intensity VII or VIII depending on the designation of tectonic provinces.

The rate of seismic activity along the Ramapo fault and in the East in general is clearly less than that for major faults in, say, California or Japan. Although the federal siting regulations put the question of the capability of a fault as a yes-no decision, the present rate of movement along faults obviously varies by many orders of magnitude. We believe recognition must be given to the fact that some faults are more "capable" than others. Until this is done, the public may well equate the designation of capability with size and rate of occurrence of earthquakes like those along, say, the San Andreas fault in California. In the context of siting nuclear power plants and other critical facilities, we believe that the rate of activity must be judged in comparison to the design earthquake of the plant. The rate of activity along the Ramapo fault is such that it probably only warrants concern for critical facilities such as nuclear power plants and hospitals for which integrity must be ensured at a high level of confidence.

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Comments on Seismicity in Southern New York -
Northern New Jersey

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Abstract. A cumulative frequency curve constructed for nine earthquakes over an eight year period of magnitude less than or equal to m_b 2.4 has been proposed as the basis for determining the activity of the Ramapo fault and for assessing the seismic safety of the Indian Point nuclear power plants. To support the proposed cumulative frequency curve, all earthquakes of Modified Mercalli intensity greater than or equal to MM V within 75 to 100 kilometers of the Ramapo must be assumed to have occurred on the Ramapo fault. The tacit assumption to a Ramapo origin for almost all of these larger earthquakes is unrealistic and definitely conflicts with the available historic reports of those events.

INTRODUCTION

In a recent report to Science entitled "Earthquakes, Faults, and Nuclear Power Plants in Southern New York and Northern New Jersey", Aggarwal and Sykes (1) developed a position that the Ramapo fault, a strand of which passes near the site of the Indian Point nuclear power plants, is an active fault. From this position, they reached the conclusion that the probability is estimated to be about 5 to 11 percent that the site will, in the next 40 years, experience an intensity equal to or in excess of the design (safe shutdown) earthquake (i.e., a Modified Mercalli intensity of VII).

This conclusion was reached from an analysis of the historic record of earthquakes in the general area over the past 250 years, and upon recordings of microearthquakes by local networks over a 4 year to 8 year span. Clearly, the best test of a predictor developed statistically is to examine how well the predictor performs in predicting what actually happened in the most recent past. The conclusion of Aggarwal and Sykes is derived from their Figure 4, which shows the cumulative number of earthquakes of magnitude m_b or greater per year as a function of magnitude. The figure purports to utilize data sets valid for shocks located within 10 km of a 120 km long segment of the Ramapo fault.

In this paper, we show that the conclusions derived from Figure 4 of Aggarwal and Sykes lead to the prediction that 32 earthquakes of Modified Mercalli intensity V or greater should have occurred within the last 250 years. The historic record reveals that for a much larger area, namely 30,000 square kilometers, only 23 such events have occurred within

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he last 250 years. Further, no such event has occurred within the limited area centered along the Ramapo fault.

Aggarwal and Sykes devoted a considerable portion of their paper to establishing that a number of the larger historic earthquakes reported to have occurred over a wide area of southern New York and northern New Jersey actually originated on the Ramapo fault; their later development of earthquake recurrence statistics is based upon that premise. As will be shown in this paper, the tacit assumption of a Ramapo origin for some of the larger earthquakes definitely conflicts with the available historic reports of those events.

The methods employed by Aggarwal and Sykes, although generally accepted as a tool useful in characterizing the seismicity of a given region, are not ordinarily employed to extrapolate from a set of data on instrumentally recorded microearthquakes of very limited duration upward to damaging earthquakes with recurrence intervals measured in centuries or even millenia. The procedures they have followed underscore the generally well-recognized problems inherent in extrapolating to a time period which is many times longer than the duration of the available instrumental data base and in neglecting the physical limitations on earthquake size that exist for every earthquake producing structure.

<u>HISTORIC SEISMICITY</u>

Critical review of the data prepared by Aggarwal and Sykes for the Ramapo fault indicates a somewhat less than compelling data base supporting the reported seismic activity rate along the Ramapo. Indeed, the very conclusion that the Ramapo fault is "active" is in itself somewhat unique. Elsewhere in the world, active faults are typically well defined by their seismicity and geologic data is used to define more precisely the surface traces of active faults, and often to determine recurrence rates of fault movement st 250 years. Further, no such event has occurred within the limited area centered along the Ramapo fault.

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Within the region surrounding the Ramapo fault, seismic history is reasonably complete for large intensity events for a time period of between 200 and 250 years. For this period of record, historic seismicity within the region does not define any specific fault structures; see Figure 1. At best, the historic seismicity can be relegated to broad seismic source areas which encompass subregions of highest earthquake density with reasonable uniformity.

Aggarwal and Sykes have shown only a selected portion of the seismicity in the region about the Ramapo fault based on a location precision criterion (in part). It is not indicated whether all epicenters have been screened by their criterion or only those adjacent to the Ramapo. Regardless of screening, there are abundant epicenters which clearly are not associated with the Ramapo fault; many or all of these earthquakes have been used to derive the regional b-value of 0.73. There is no indication of the area considered to be the region.

For purposes of establishing our regional data base, we have selected a 150 km by 200 km region of somewhat uniform earthquake density; the selected region (encompassing all of Figure 1) has had more than 100 reported earthquakes. Since no regional data were presented by Aggarwal and Sykes we cannot compare regional data assumptions directly.

However, the data shown on Figure 1 are essentially those data used by Aggarwal and Sykes. These earthquake locations from various data sources (2,3,4,5) are presented in Table 1 excluding obvious aftershock sequences and earthquakes swarms. Aggarwal and Sykes suggest a "remarkably similar" distribution exists between historical events for the period 1534 to 1959 and the events detected by the current networks from 1970 to 1977. When the historical record and the urrent seismic activity in southeastern New York are compared (3), however, they do not display a remarkable similarity for earthquake activity along the Ramapo fault. In fact, the seismic record for southeastern New York shows a broad distribution of epicenters rather than a concentration along the Ramapo fault.

Seismic source areas within the region have been selected to conform to geologic structural trends and to contain areas of uniform earthquake density. Assuming the conclusion reached by Aggarwal and Sykes is correct (that earthquakes are related to northeast trending faults exposed at the surface) we have selected seismic source areas which trend northeast, including the proposed source area about the Ramapo fault. The primary purpose in selecting these source areas is for comparison of the seismicity in various portions of the region. Source areas are shown on Figure 1, and include the Ramapo fault source area, the Highlands source area adjacent to and west of the Ramapo, and the Lower Hudson source area which encompasses the rather dense cluster of historic events adjacent to and east of the Ramapo area. For the Ramapo fault, we have accepted essentially the same Ramapo source area as that suggested by Aggarwal and Sykes, but have shifted the area somewhat eastward to comply with known fault geometry. A tabulation of earthquakes by source area is presented in Table 1 and 2.

Locations of the larger historic events are indicated on Figure 1. Evaluation of this data indicates that no historical earthquakes of intensity greater than MM IV have occurred within the Ramapo seismic source area (or along the Ramapo fault).

We have examined the merit of relocating the larger historic events along the Ramapo fault as was suggested by Aggarwal and Sykes. In general, we found no basis in historic reports to justify relocation of events to the Ramapo fault.

We specifically examined in detail the suggested relocation of the 1884 earthquake to some point along the Ramapo fault. The highest reported intensities for the 1884 event were along the southwestern end of Long Island (2,3,6). Rockwood, a noted authority on earthquakes in late 1800's, stated that he had devoted much time to the study of the material gathered in regard to this earthquake. He found the highest intensities to be located at Jamaica and Amityville in the western part of Long Island.

In their discussion of the 1884 earthquake, Aggarwal and Sykes acknowledge the Brooklyn epicenter location reported in Earthquake History of the United States (2) and the Smith catalog (3), but note that Rockwood (6) is cited in each as the original source of data. Further discussing Rockwood's report they state that "He placed the center of the zone of maximum shaking in northeastern New Jersey". This is not the case. Rockwood clearly identifies the area of maximum shaking as "The area of Intensity IV is nearly elliptical, its longer axis extending from Hartford, Conn., to West Chester, Pa., and having its center near New York, being about 200 miles long by 70 miles wide". Rockwood further states that "The only places where the reported intensity reached V were Jamaica and Amityville in the western part of Long Island". Rockwood states later in his report that "An examination of the map at once indicates that the cause of this earthquake is to be sought in the vicinity of New York City".

Rockwood continues in his report to describe examples of the intensity V shaking. The case reported by Aggarwal and Sykes is actually a comment on the intensity IV isoseismal rating utilized by Rockwood is the scale of I to VI where IV = Strong and V= Severe.

In addition to the reports of mainshock data, the New York Times (7) cites the occurrence of aftershocks which were felt along the southwestern end of Long Island as well. The Paterson Daily (8) Press reports foreshocks and aftershocks purportedly felt in Paterson, New Jersey, but foreshock reports came only after the mainshock. Rockwood states "there were sundry reports of light succeeding shocks at various hours on the 11th, but none were confirmed by two observers, and all were apparently due to the excited imagination of the public".

Based on evaluation of Rockwood's reports, the suggestion that the epicenter of the 1884 event was along the Ramapo fault is clearly not justified by the existing data.

SPATIAL DISTRIBUTION OF EARTHQUAKES AND GEOLOGIC AND TECTONIC STRUCTURES

The geologic significance of the suite of Precambrian rock comprising the New York-New Jersey Highlands has been summarized by Aggarwal and Sykes. The prominence of this zone geologically and geomorphically as well as its tectonic history of recurrent deformations through geologic time, suggests that a more than coincidental importance may be attributed to the spatial relation of the seismicity within the Highlands. The faulting system within the Highlands exposed at the surface includes faults within the Highlands source area and eastward up to and including the Ramapo fault. Considering the distribution of epicenters within the Highlands faulting system (see ure 1), it is concluded that earthquake activity can be spatially related to surface traces of faults in the Highlands system. The linear trend of faulting from southwest of the Schooley's Mountain area (point X on Figure 1) through Lake Hopatcong (location A) to the Wappingers Falls area (point X') and beyond, may represent the most significant and continuous faulting system in the region.

The Highlands linear is quite prominent over much of its length as observed on various forms of high altitude photography and satellite imagery. Northeast of Wappingers Falls area, the marked linear continues, but is no longer within the Precambrian rocks. Studies are presently underway to define the nature of the linear as expressed by geology. A portion of the linear has been previously termed the Beacon-Copake anomaly(9), where the linear extends into Paleozoic rocks of New York and is marked by apparently isolated knobs of outcropping Precambrian rock. The geologic importance of the linear has not been established to date.

Clearly, the bulk of historic epicenters located within the Highlands are readily associated with the Highlands faulting system. Also, the three localities mentioned above (points X and X' and Location A) are sites of earthquakes swarms within the New York-New Jersey region shown on Figure 1. The Schooley's Mountain area (point X) had more than 125 small events reported in December of 1977(4); both Lake Hopatcong (location A) and Wappingers Falls (point X') have had well recorded earthquake sequences during the last decade.

Simple correlation of the number of events for the period since 1970 within the seismic source areas as outlined on Figure 1 shows 13 main events along the Highlands linear and events along the Ramapo; for the complete period of historical record, there have been 21 events along the Highlands linear and 15 events along the Ramapo. The larger intensity historical events and hundreds of swarm events have also occurred along the Highlands linear; this includes a magnitude $m_b = 3.3$ event at Wappingers Falls, and older events of up to intensity MM VI and perhaps MM VII. The magnitude of the largest instrumentally recorded event that has occurred along the Ramapo source area is m_b 2.5; the largest historically reported intensity is MM IV.

Based on the distribution of epicenters, the actual number of events association with the Ramapo source area, and the location of the larger historic events along the Highlands linear, it is clear that the Ramapo fault is not the geologic structure controlling the seismicity of the region. In particular, there is no justification for considering all of the seismic activity in the region to be concentrated on the Ramapo fault, as Aggarwal and Sykes have implicitly done by applying regional data to their recurrence curve for the fault.

RECURRENCE INTERVALS FOR EARTHQUAKES

A common method of evaluating the seismicity of any given area is by determining the cumulative frequency of various size earthquakes within the area. Earthquake size may be expressed by either an intensity rating or a magnitude scale. Cumulative frequency curves are constructed by plotting the cumulative number of events greater than or equal to a particular size per unit time. For such curves, it is customary to determine the time spans for which various size earthquakes would be completely reported and to normalize these data to the number of events per year. For example, based on the population distribution specifically within the Ramapo source area, and the generally large felt area for ensity MM V events in the northeastern United States, it is reasonable to assume that most, if not all, events of intensity MM V (and greater) should have been felt since some time prior to the Revolutionary War, or approximately the last 230 years.

Assumptions of the period of complete or near complete record for various size events within the region are shown on Figure 2. Thus, for MM≥V, despite the cluster of events in more recent years, the population density controlling felt reports within the Ramapo source area suggests that MM V events should not have gone unnoticed for a much larger period of time than suggested by the intensity distribution. As suggested by Aggarwal and Sykes, detection of events larger than magnitude (m_b) 2.0 has been complete, or nearly complete, since about 1970, or for approximately 8 years.

No satisfactory magnitude-intensity relationship exists. Since it is necessary to utilize both intensity and magnitude data to develop cumulative frequency curves over the entire range of available data, we have used Aggarwal's relationship (10) for ease of comparing results:

$$m_b = -0.20 + 0.75I$$

Cumulative frequency curves are typically normalized with regard to area as well as time in order that seismicity for differant areas or different structures may be meaningfully compared. The unit area most commonly used in evaluation of seismicity is 1000 square kilometers. Thus, the standard recurrence curve for an area such as the Ramapo source area would relate the cumulative number of events per year per 1000 square kilometers to various size events.

Although this approach was not taken by Aggarwal and Sykes, we have normalized all data within the region to determine a recurrence per 1000 square kilometers area. Both the data presented for the Ramapo source area (by Aggarwal and Sykes), and the historical seismicity data for the region are normalized 1000 square kilometers and are plotted on Figure 3.

As shown on Figure 3, there is as much as an order of magnitude difference between the cumulative frequency curve based on the regional historical seismicity and the Ramapo source area data. Data points for the curve for the Ramapo source area are limited because no historical events larger than intensity MM IV have occurred. One of the difficulties in constructing a reasonable recurrence curve for the Ramapo fault is the limited data base. Only a few small earthquakes ($m_b \leq 2.5$) have occurred within the source area since 1974 when a number of sensitive seismic stations were established in the area. Aggarwal and Sykes have used the cumulative level of activity implied by these events to predict a level of activity for larger earthquakes by extrapolation at the regional b-slope of 0.73 to the higher magnitude events. The estimate of the cumulative level of activity is subject to ge sampling errors because of a small sample size. Extrapolation on this basis typically leads to extremely unreliable return periods for higher magnitude earthquakes. We can find no basis for Aggarwal and Sykes extrapolation of the Ramapo microearthquake data to larger size events based on a regional b-value 0.73. Since the figure presented by Aggarwal and Sykes makes no effort to normalize the two different data sets by area, their curve has meaning only if the two data sets are drawn from the identical area. Moreover, with no events greater than MM IV occurring along the Ramapo in reported history, and with approximately 230 years or more of significant population of the area along the fault, we find little reliability in extrapolating recurrence intervals for intensity MM VII and MM VIII earthquakes.

In addition we can find no justification for the plotting of large intensity regional earthquakes along the curve that

-ports to represent the cumulative frequency of events ior the Ramapo fault as was done in Figure 4 of Aggarwal and Sykes. This assumes that essentially all earthquakes which have occurred within the region shown on Figure 1 actually occur along the Ramapo fault. This is clearly an incorrect assumption.

For example, the recurrence curve presented by Aggarwal and Sykes predicts that approximately 32 earthquakes of intensity MM$\geq$ V should have occurred along the Ramapo fault in the last 250 years. For the entire region shown on Figure 1, there have been only 23 events of intensity MM$\geq$ V during the period of record, 1698 to present; and none of these have been located along the Ramapo fault. To satisfy their recurrence curve, however, all of these events and an additional 9 events would have to be located along the fault.

In areas of high seismicity with several geologically active faults, such as the San Andreas fault system, earthquake data sets have been subdivided into small subsets of the system to evaluate activity of the individual segments (10). Such analyses revealed that while the recurrence rates for various segments are similar, the levels of activity for these segments vary by more than an order of magnitude. This demonstrates that event for a fault system with a relatively high level of activity, no single subset is necessarily representative of the whole system. Further, it is shown that only the fault system as a whole can be used to identify the maximum earthquake associated with the system.

The variation in both a-values and b-values and data subsets is perhaps more clearly demonstrated by Wyss and Lee (12). For individual segments along approximately 50 km of the San Andreas fault near Hollister, California, the return period for a magnitude 3 event varies by almost an order of magnitude. Further, this variation can be seen along the same fault.

segment with discrete samples in time. The b-values for the same fault are shown to vary from 0.8 to 1.4 for six-months data samples during a one-year total sample(12).

Based on these and other data, we believe that the approach taken by Aggarwal and Sykes for defining the seismicity along the Ramapo fault is invalid. The a-value for the fault cannot be well defined by 9 to 11 events collected over a period of about four years. It should be noted that less than half the data used to determine the a-value can actually be associated with the Ramapo fault when the spatial relations between the hypocenters and the fault plane are considered. We conclude the a-value selected for the data subset by Aggarwal and Sykes has no validity with respect to either the true a-value for the Ramapo fault source area or the a-value of the Highlands linear faulting system as a whole.

To further identify the difficulties inherent in the approach of Aggarwal and Sykes, we have utilized the 1970 through 1977 period of record to construct cumulative frequency curves for various seismic source areas; see Figure 4. It should be clearly noted, however, that the number of events for each source area (individually) is statistically insufficient to determine slope and intercepts for the recurrence curves with any real reliability. Data used by Aggarwal and Sykes for the Ramapo have been normalized and presented on the cumulative frequency curve on Figure 4 together with small magnitude data for the same time period for the Highlands linear source area. Both data sets include a four year sample for magnitudes less than m_b 2.0, and an eight year sample for magnitudes greater than m_b 2.0. The Highlands data plot essentially along the same curve developed for the Ramapo data. We conclude that there is nothing unique about the seismicity of the Ramapo.

To examine the implications of the Aggarwal and Sykes recurrence curve, we have calculated the probability of not observing a single event of a particular intensity during a reasonable detection period assuming that the activity rates of Aggarwal and Sykes are correct. In calculating these probabilities, we have made the usual assumption that earthquake occurrences follow a Poisson distribution. Thus, the probability of not observing a single event of intensity greater than or equal to I in t years is given by:

$$P_I = P \text{ (0 events with intensity} > \text{I in t years)}$$

$$= \frac{e^{-\lambda t}\ (\lambda t)^0}{0!} = e^{-\lambda t} \qquad (1)$$

where λ is the average rate of occurrence (i.e., mean number of events per year greater than I).

The values of P_I for different intensities and the associated detection periods are shown in Table 3. For the stated assumptions, the chance of not observing an earthquake greater than or equal to intensity VI in 230 years is 1 in 4,000. The chance of not observing a single earthquake greater than intensity V is 1 in 10^{12}.

The short return periods indicated by Aggarwal and Sykes (Figure 4) do not seem credible in view of their incompatibility with the historical seismicity data.

Another way of interpreting this result is that if in fact earthquakes of intensity greater than or equal to MM V occur on the average 7.8 years, as suggested, one should have seen almost with certainty several events of intensity MM$\geq$ V specifically along the Ramapo fault in the last 100 years. Since not one earthquake of intensity V or greater has been located on the fault (or within the source area) it is unreasonable to assume these earthquakes occur with a return period of 7.8

earl. It is concluded that the short return periods predicted by the Aggarwal and Sykes curve, therefore, are not statistically valid. Based on the microearthquake data presented in Figure 4, we conclude that there is nothing to suggest that the Ramapo fault (or the Ramapo source area) differs from the adjacent Highlands linear with regard to activity levels of earthquakes. Indeed, as only low intensity events (MM$\leq$IV) have ever been reported along the Ramapo, it is clear that the Ramapo fault is not the geologic structure controlling the seismicity in the region or the combined Highlands-Ramapo source area.

There remains a marked discrepancy between the shorter return periods of recent instrumental data (low magnitude) and the inconsistently longer return periods of regional historic data (larger intensity). An inconsistency between microearthquakes and larger magnitude data is not expected unless the microearthquakes result from different causative mechanisms.

A portion of the discrepancy is clearly the result of the error in the data plot of Aggarwal and Sykes. They state that the record of events for $m_b \geq 2.0$ has been complete since 1970. For this data, however, they have used only the 1974 to 1977 time period in normalizing the data with respect to time. This oversight results in an error of a factor of two, which is to say that all plotted events between m_b 2.0 and m_b 2.5 appear to be twice as frequent as they indeed are, based on the actual data.

Other possible contributions to this discrepancy may include an incorrect relationship between magnitude and intensity. The cumulative frequency data are based on instrumental magnitudes determined by Lamont Doherty from the local telemetered seismic network. The macroseismic recurrence curves are derived from felt report intensity evaluations. The two data sets are related through an empirical magnitude-intensity relationship. Again, for purposes of this discussion, we have assumed the same relationship as outlined by Aggarwal and Sykes for convenience in making data comparisons. The data supporting this relationship, however, are yet unpublished.

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Another possible explanation could be that magnitudes determined over the past few years for instrumentally recorded microearthquakes are incorrect. If the magnitude determinations are too high as has been previously suggested (13), this could partially explain the discrepancy dependent upon the degree of error made in the magnitude assignments. Street and others (14) have pointed out that errors in m_b estimations of 0.5 magnitude units can occur when using Lg periods that are not close to one second (which is frequently the case with microearthquake recordings).

CONCLUSIONS

Based on the data shown in Figures 1, 3, and 4, we conclude that the Ramapo fault is not unique in terms of its seismicity. The bulk of the historic seismicity within the region, however, does appear to be included within the three identified source areas. The only differences between the Ramapo fault data and the rest of the Highlands fault system is the fact that for both short-term (4 to 8 years) and the historical record, the larger events do not appear to be associated with the Ramapo. Indeed, in approximately 250 years of record, no earthquake of intensity greater than MM IV has ever occurred along the Ramapo. Further, based on statistical analysis of the return periods proposed by Aggarwal and Sykes, and the review of the data presented by others (11,12) we conclude that return periods of larger events cannot be accurately determined by a four-year sample of nine events of less than m_b 2.5. This appears to be the case even in areas of active seismicity and must certainly apply to areas of low to moderate seismicity.

In summary, based on the historical seismicity, the spatial distribution of events and the apparent lack of large intensity events, we conclude that the Ramapo fault is not the geologic structure controlling the seismicity of the region or the Highlands faulting system. Further, we believe that the conclusions reached by Aggarwal and Sykes regarding return periods of large intensity events along the Ramapo or for the region are unjustified and incorrect. Return periods for larger intensity earthquakes based on the actual historic record indicate that the frequency of occurrence predicted by Aggarwal and Sykes are more than an order of magnitude too short. We suggest that the data can be more correctly analyzed by assuming a more realistic source area, and that resulting recurrence intervals for large events then correspond with the observed historical record.

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ble 1. List of all epicenters within the limits of Figure 1 excluding aftershock sequences or swarms. For swarms the largest earthquake is taken as representative of the swarm. Source areas are those outlined on Figure 1 and described in Table 2. Earthquakes located within the source areas are identified by the following designators: LH, Lower Hudson Source Area; HL, Highlands Source Area; R, Ramapo Source Area. Earthquakes are listed in the order of decreasing size with all m_{bLg} magnitudes converted to intensity according to the Aggarwal relationship, $m_{bLg} = 0.2 + 0.75I$, (10).

Table 2. Summary of characteristics of the region of study and the three subregional source areas.

Table 3. Probabilities of not observing earthquakes of various intensities based on the recurrence rates from Aggarwal and Sykes and this paper. Two estimates of complete record of intensity levels, one based on an assumed population distribution and another taken from the record intervals indicated on Figure 2, are used with the recurrence rates to determine four sets of probabilities. These data sets are presented in the form of chance estimates which are simply the inverse of the probabilities.

Date	Lat	Long	MMI	Source Area
12-18-1737	40.[illegible]	74.0	VII	R
08-10-1884	40.6	74.0	VII	HL
06-01-1927	40.3	74.0	VI-VII	HL
11-30-1783	41.0	74.5	VI	HL
09-01-1895	40.7	74.8	VI	LH
03-23-1957	40.63	74.83	VI	LH
12-10-1874	40.9	73.8	V-VI	LH
08-23-1938	40.13	74.53	V-VI	
09-09-1848	40.4	74.0	V	R
02-05-1878	40.8	73.8	V	R
10-04-1878	41.5	74.0	V	
03-09-1893	40.6	74.0	V	LH
05-12-1926	40.9	73.9	V	LH
01-25-1933	40.2	74.7	V	LH
01-04-1947	41.03	73.58	V	LH
09-03-1951	41.25	74.25	V	LF
10-08-1952	41.7	74.0	V	LH
03-27-1953	41.1	73.5	V	R
09-14-1961	40.75	75.75	V	R
12-27-1961	40.50	74.75	V	HL
11-17-1964	41.2	73.7	V	HL
07-11-1872	40.9	73.8	IV-V	R
09-10-1877	40.1	74.8	IV-V	HL
06-07-1974	41.57	73.94	IV-V	LH
[illegible]4-13-1976	40.84	74.05	IV-V	LH
06-08-1916	41.0	73.8	IV-V	HL
1698	41.38	73.47	IV	R
1702	41.4	73.5	IV	HL
1711	41.4	73.5	IV	LH
08-06-1729	41.5	73.5	IV	R
02-05-1908	41.4	73.2	IV	HL
07-19-1937	40.72	73.71	IV	HL
10-24-1942	40.97	75.25	IV	HL
03-29-1950	41.05	73.60	IV	LH
08-17-1953	41.0	74.0	IV	
03-31-1954	40.25	74.00	IV	HL
12-20-1962	40.99	74.33	IV	LH
05-11-1976	40.49	73.80	IV	LH
01-21-1977	39.98	74.33	IV	LH
08-02-1938	41.08	73.70	III-IV	R
02-15-1972	41.29	73.61	III-IV	HL
03-11-1976	40.95	74.35	III-IV	HL
08-20-1976	41.12	73.76	III-IV	R
09-02-1977	41.31	73.92	III-IV	LH
04-29-1975	41.59	73.88	III-IV	R
07-19-1975	41.43	73.79	III-IV	

Date	Lat	Long	MMI
08-22-1975	41.14	73.95	III-IV
07-02-1977	40.70	74.935	III-IV
12-09-1977	41.56	73.88	III-IV
12-23-1977	40.77	74.76	III-IV
05-18-1804	40.75	74.0	III
01-25-1841	40.75	74.0	III
03-05-1861	40.7	74.2	III
04-21-1881	40.95	73.10	III
01-04-1885	41.3	73.9	III
01-31-1885	41.3	73.8	III
10-24-1925	41.4	73.3	III
06-26-1933	41.0	73.8	III
09-03-1937	40.83	74.25	III
07-29-1938	41.0	73.7	III
08-23-1938	41.2	73.7	III
12-06-1938	40.8	74.3	III
07-28-1941	41.13	73.75	III
04-01-1947	41.01	74.30	III
03-10-1977	41.18	74.15	III
10-14-1977	41.56	73.95	III
10-24-1975	41.60	73.98	III
04-08-1974	41.22	73.99	III
06-15-1975	41.58	73.94	III
11-22-1976	41.00	73.86	III
05-16-1938	40.8	74.3	II-III
12-08-1951	41.7	73.9	II-III
09-22-1976	41.29	73.95	II-III
12-05-1976	40.77	74.76	II-III
12-11-1976	40.72	74.01	II-III
11-27-1977	41.02	74.22	II-III
01-10-1973	41.39	73.98	II-III
11-10-1975	41.06	74.32	II-III
10-27-1977	41.07	74.59	II-III
12-25-1878	40.8	73.8	II
05-01-1910	40.7	73.5	II
05-22-1926	41.7	73.9	II
10-12-1937	41.2	73.8	II
10-21-1938	41.17	73.67	II
09-13-1939	40.8	74.0	II
05-21-1966	41.14	74.03	II
10-06-1969	40.96	74.64	II
06-10-1977	40.70	74.89	I-II
11-30-1964	41.21	73.95	I-II
03-06-1976	41.17	73.81	I-II
10-28-1976	40.89	74.49	I-II

Source Area	Source Area Dimension, km Length	Width	Tota rea, km^2	Number of Earthquakes	Earthquake Densi Events/1000 km^2
Region	200	150	30,000	91	3.0
Highlands Linear	150	25	3,750	21	5.6
Lower Hudson	135	40	5,400	38	7.0
Ramapo	125	18	2,250	15	6.7
Sum of three source areas	——	——	11,400	74	6.5

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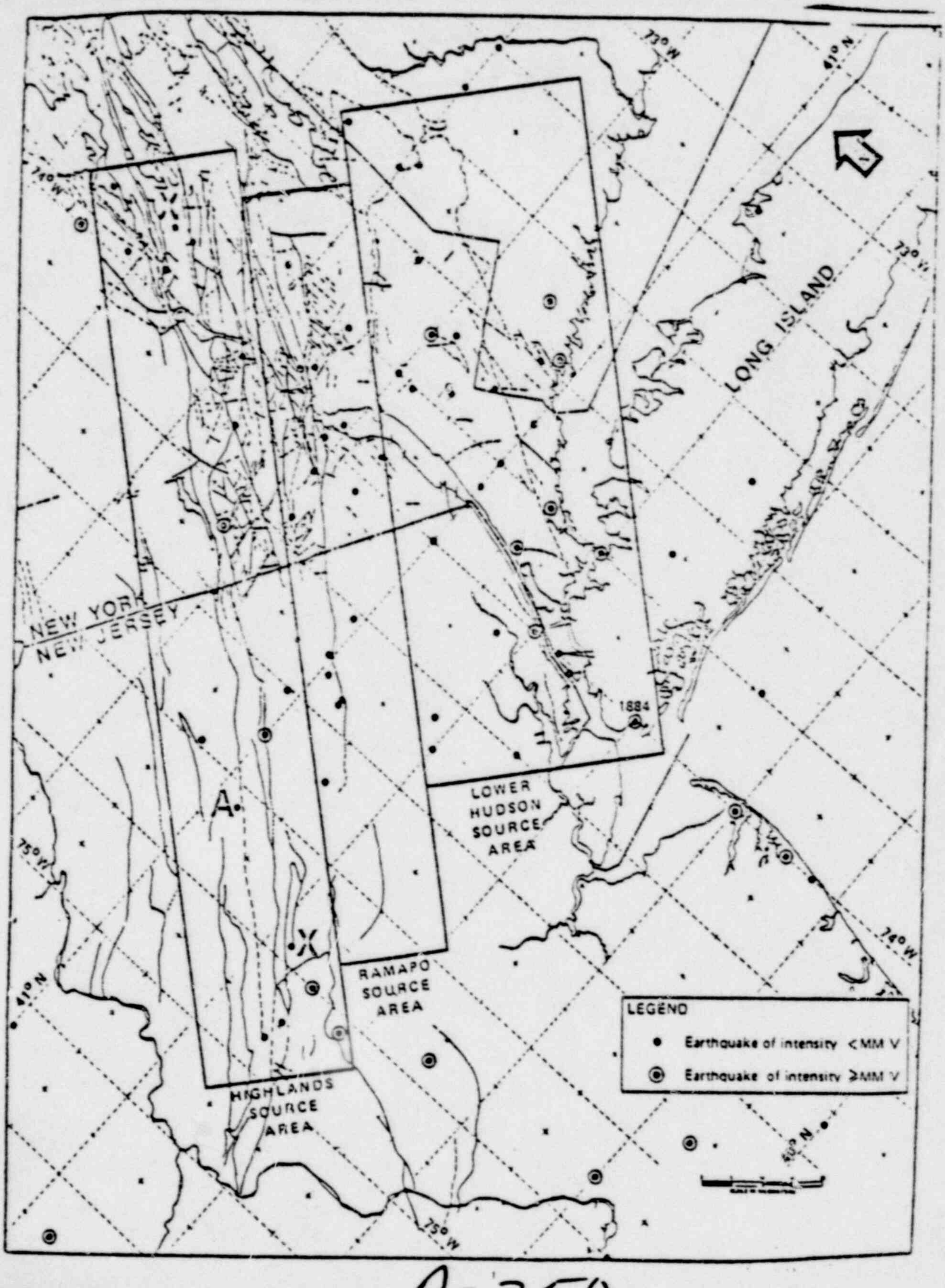
Intensity	Assumed Period of Complete Record in Years, t	Mean Rate of Occurrence Per Year, λ	Probability of Not Observing a Single Event of Intensity > I	Chance of Not Observing a Single Event of Intensity > I
Recurrence rates from Aggarwal and Sykes and complete record estimates based on assumed population distribution				
VIII	250	2.92×10^{-3}	4.81×10^{-1}	1 in 2
VII	250	1.03×10^{-2}	7.59×10^{-2}	1 in 13
VI	230	3.64×10^{-2}	2.32×10^{-4}	1 in 4300
V	230	1.28×10^{-1}	1.50×10^{-13}	1 in 6.66×10^{12}
Recurrence rates from Aggarwal and Sykes and complete record estimates based on data presented in this report				
VIII	250	2.92×10^{-3}	4.81×10^{-1}	1 in 2
VII	250	1.03×10^{-2}	7.59×10^{-2}	1 in 13
VI	200	3.64×10^{-2}	6.90×10^{-4}	1 in 1450
V	130	1.28×10^{-1}	5.65×10^{-8}	1 in 1.77×10^{7}
Recurrence rates from this paper and complete record estimates based on assumed population distribution				
VIII	250	9.33×10^{-5}	9.77×10^{-1}	1 in in 1.0
VII	250	3.41×10^{-4}	9.18×10^{-1}	1 in 1.1
VI	230	1.24×10^{-3}	7.51×10^{-1}	1 in 1.3
V	230	4.54×10^{-3}	3.52×10^{-1}	1 in 2.8
Recurrence rates and complete record estimates based on data presented in this report				
VIII	250	9.33×10^{-5}	9.77×10^{-1}	1 in 1.0
VII	250	3.41×10^{-4}	9.18×10^{-1}	1 in 1.1
VI	200	1.24×10^{-3}	7.80×10^{-1}	1 in 1.3
V	130	4.54×10^{-3}	5.54×10^{-1}	1 in 1.8

Figure 1. Epicenters of earthquakes (1698 through 1977) primarily in southeastern New York and northern New Jersey. Portions of Pennsylvania and Connecticut are also included in the limits of the figure. Geographic coordinates of the corners of the figure are: 41.96°N, 73.77°W; 40.93°N, 72.62°W; 40.79°N, 75.59°W; and 39.77°N, 74.42°W. Three subregional source areas, discussed in the text, are identified on the figure. X, A, and X', mark the locations of the Schooley's Mountain, Lake Hopatcong, and Wappingers Falls earthquake swarm sites respectively. Note the number of epicenter locations in the figure are less than the number of epicenters listed in Table 1 because of the co-location of several epicenters; epicenter symbols are based on the largest event reported for a location. Faults and linears shown are taken from the brittle structures map of New York , the geologic map of New Jersey, and unpublished notes of the New Jersey State Geological Survey (15).

Figure 2. Number of earthquakes per decade for each MM intensity category listed in Table 1. Dashed vertical lines indicate the authors' estimate of complete record for the intensity categories based on frequency of reported occurrence.

Figure 3. Cumulative number (N) of earthquakes of magnitude m_b or greater per year per 1000 km^2 as a function of magnitude. Circles, ⊙, are data points based on all data presented in Table 1, taking into account the estimates of complete record presented on Figure 2; X's are data points presented by Aggarwal and Sykes (1) normalized with respect to area for comparison. The regional recurrence curve is based on values determined for the four Modified Mercalli intensity intervals: IV, V, VI, and VII; a somewhat steeper slope can be obtained using only the intensity intervals; V, VI, and VII.

Figure 4. Cumulative number of earthquakes of magnitude m_b or greater per year per 1000 km^2 as a function of magnitude, derived from Table 1, for the subregional source areas. Subset of Highlands and Ramapo source areas data for the time period 1970 to 1977, evaluated in the same manner, are also presented. The regional recurrence curve, presented in Figure 3 is repeated for reference.



73° W
41° N
LONG ISLAND
74° W
73° W
NEW YORK
NEW JERSEY
1884
LOWER
HUDSON
SOURCE
AREA
A
75° W
RAMAPO
SOURCE
AREA
41° N
HIGHLANDS
SOURCE
AREA
74° W
LEGEND
Earthquake of intensity <MM V
Earthquake of intensity ≥MM V
40° N
75° W

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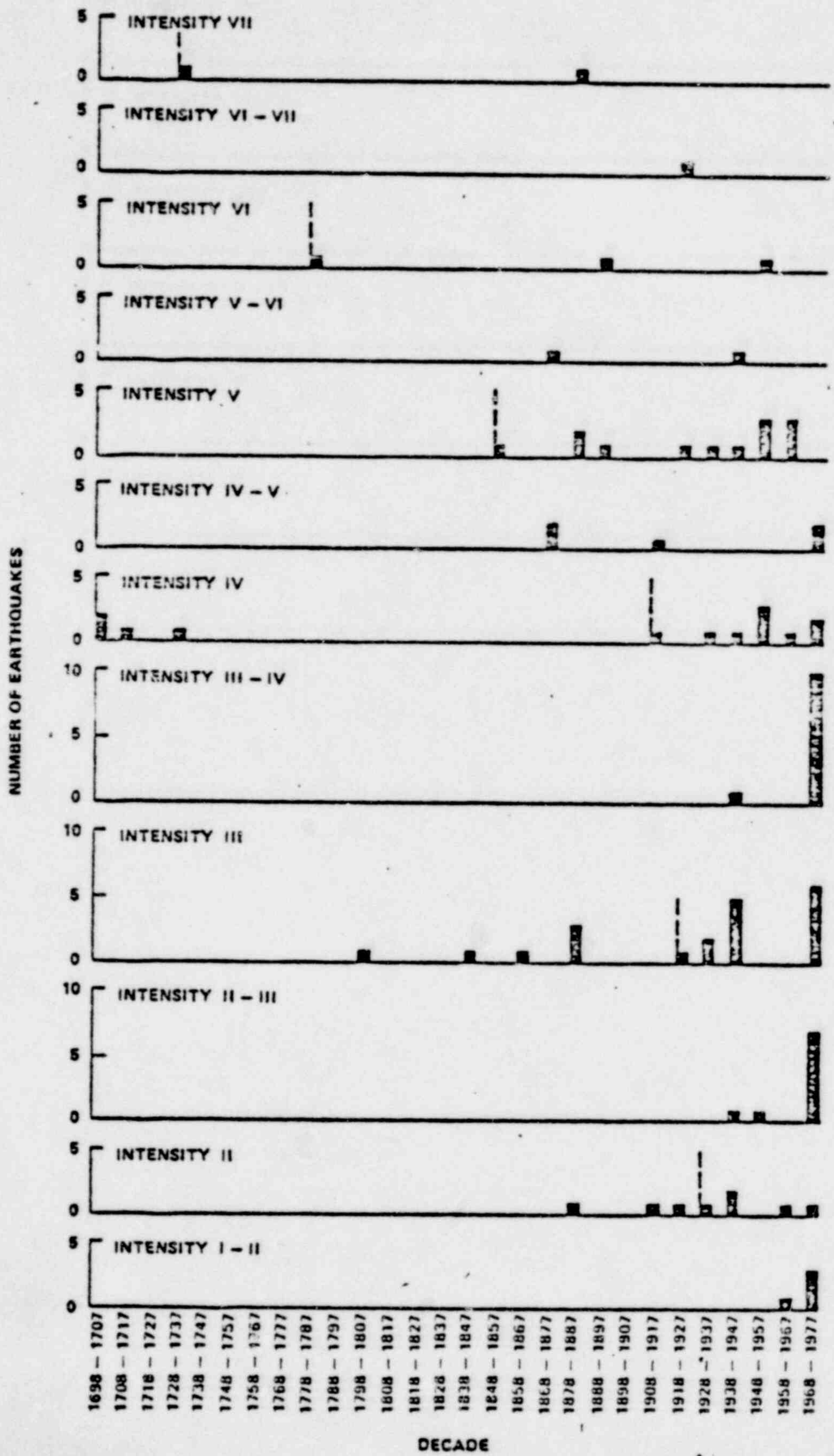
INTENSITY VII
INTENSITY VI – VII
INTENSITY VI
INTENSITY V – VI
INTENSITY V
INTENSITY IV – V
INTENSITY IV
INTENSITY III – IV
INTENSITY III
INTENSITY II – III
INTENSITY II
INTENSITY I – II
NUMBER OF EARTHQUAKES
DECADE
1698 – 1707
1708 – 1717
1718 – 1727
1728 – 1737
1738 – 1747
1748 – 1757
1758 – 1767
1768 – 1777
1778 – 1787
1788 – 1797
1798 – 1807
1808 – 1817
1818 – 1827
1828 – 1837
1838 – 1847
1848 – 1857
1858 – 1867
1868 – 1877
1878 – 1887
1888 – 1897
1898 – 1907
1908 – 1917
1918 – 1927
1928 – 1937
1938 – 1947
1948 – 1957
1958 – 1967
1968 – 1977

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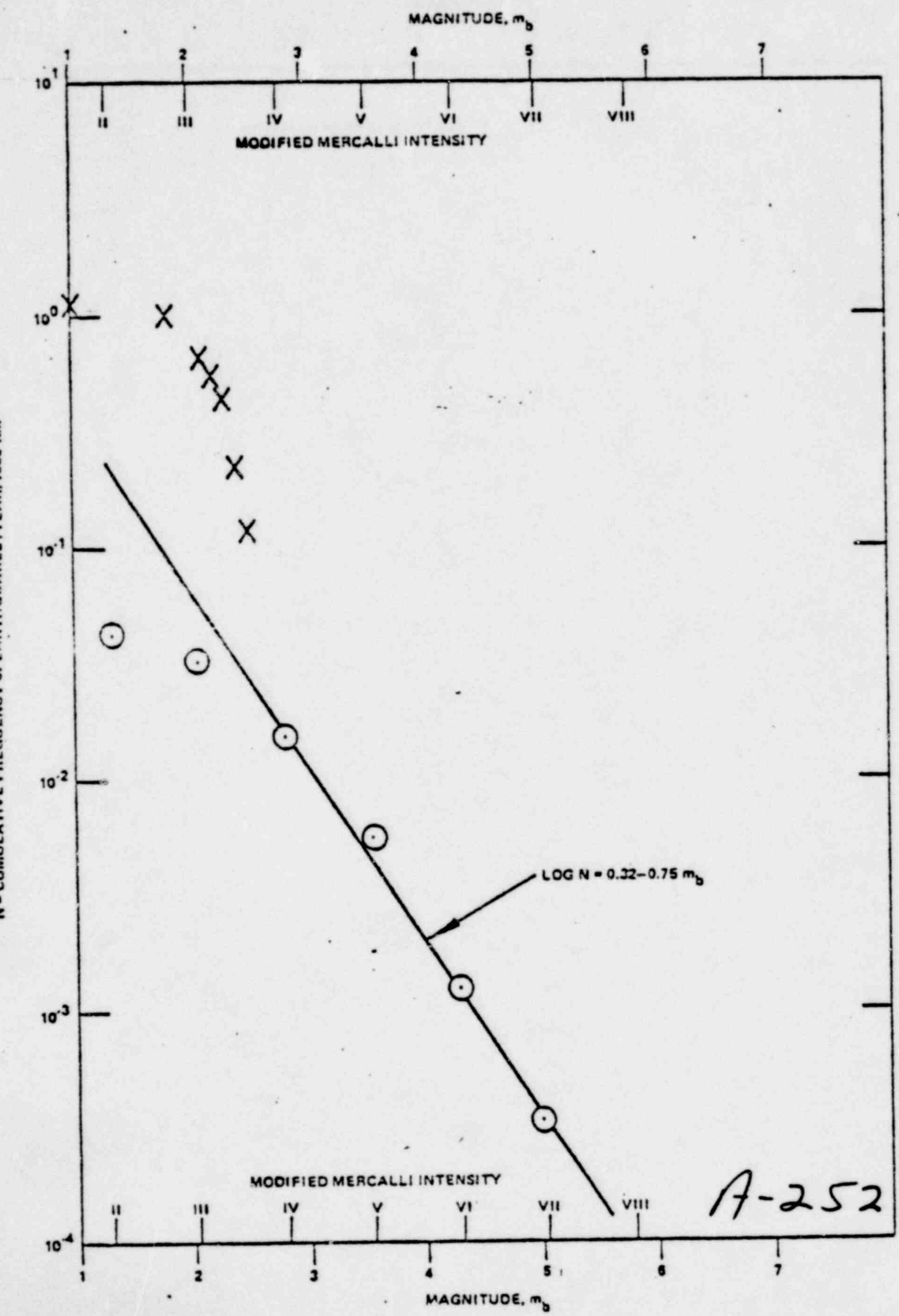
MAGNITUDE, m_b
1
2
3
4
5
6
7
10^1
II
III
IV
V
VI
VII
VIII
MODIFIED MERCALLI INTENSITY
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
N = CUMULATIVE FREQUENCY of EARTHQUAKES/YEAR/1000 KM^2
LOG N = 0.32–0.75 m_b
MODIFIED MERCALLI INTENSITY
II
III
IV
V
VI
VII
VIII
1
2
3
4
5
6
7
MAGNITUDE, m_b

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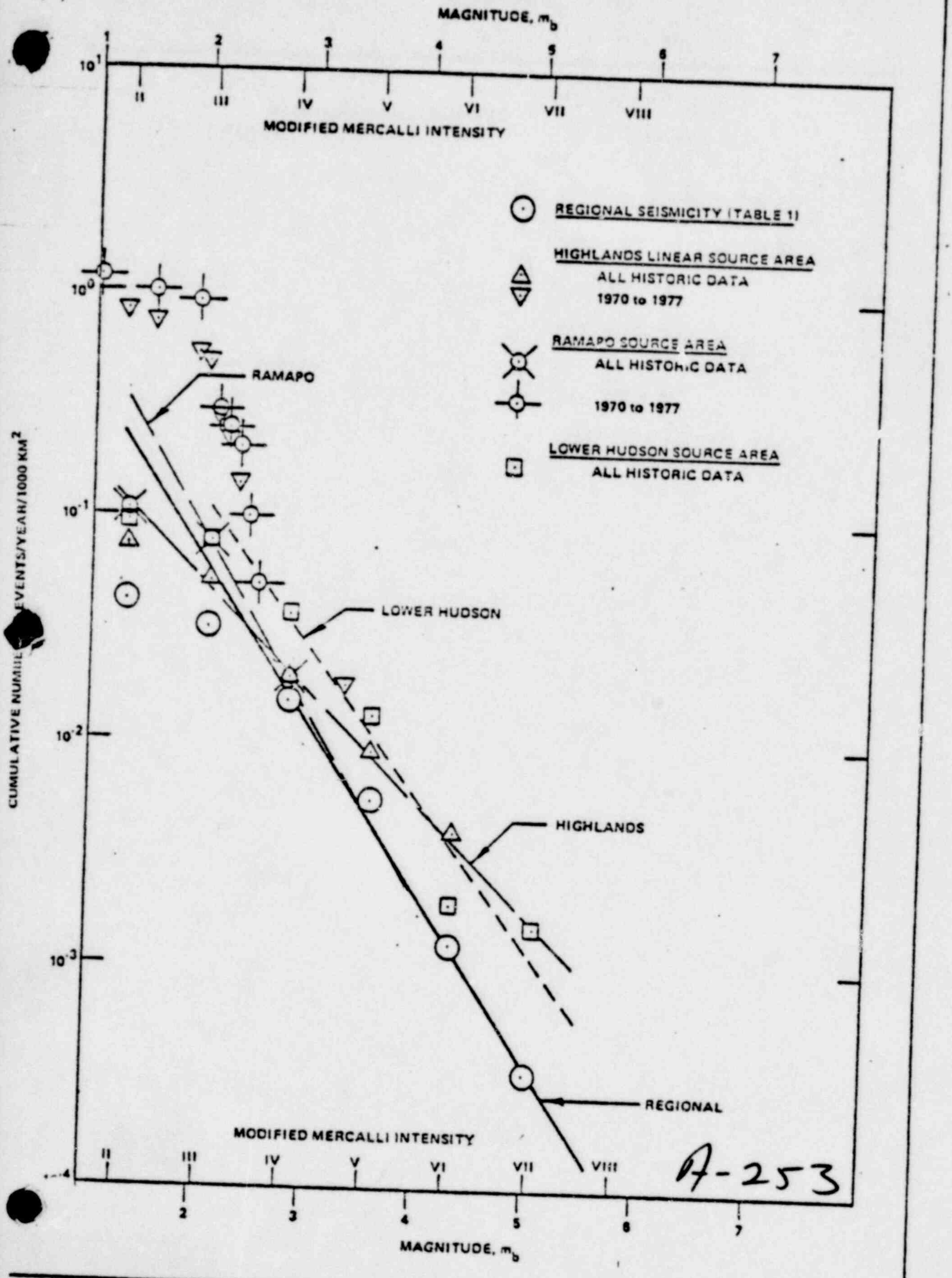

MAGNITUDE, m_b
MODIFIED MERCALLI INTENSITY
REGIONAL SEISMICITY (TABLE 1)
HIGHLANDS LINEAR SOURCE AREA
ALL HISTORIC DATA
1970 to 1977
RAMAPO SOURCE AREA
ALL HISTORIC DATA
1970 to 1977
LOWER HUDSON SOURCE AREA
ALL HISTORIC DATA
RAMAPO
LOWER HUDSON
HIGHLANDS
REGIONAL
CUMULATIVE NUMBER EVENTS/YEAR/1000 KM^2
MODIFIED MERCALLI INTENSITY
MAGNITUDE, m_b

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Appendix XVIII

JOHN C. MAXWELL
GEOLOGIST
5322 WESTERN HILLS DRIVE
AUSTIN, TEXAS 78731
512-454-1931

ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS U.S. N.R.C.
MAY 22 1978
AM PM
7,8,9,10,11,12,1,2,3,4,5,6

May 19, 1978

CT-0988

Mr. H. Etherington
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Washington, D. C. 20555

Comment on: Earthquakes, Faults and Nuclear Power Plants in southern New York and northern New Jersey, by Y. P. Aggarwal and Lynn R. Sykes, Science, 28 April, 1978.

On first reading this is a lucid and compelling analysis of the Ramapo fault system, establishing it as seismically active and suggesting high seismic risks for a nuclear plant built near the fault. On analysis, however, it seems to me the arguments advanced largely fall apart. The probabilities arrived at are derived from statistical analysis and the validity of such an analysis is directly proportional to both the amount of data available and to the unquestioned applicability of the data to the problem being analyzed. The Aggarwal-Sykes analysis is weak in both respects, as will be discussed below.

Since the early 60's at least, the Lamont Geological Observatory has maintained a seismic network in northern New Jersey and southern New York State. Apparently, significant additions to the network were made in the Ramapo area about 1974, and the data demonstrating seismic activity along the Ramapo fault zone date from then. The Lamont data and those from other sources appear to indicate a tendency for seismic energy to be released along old, northeasterly trending zones of faulting, of which the Ramapo fault is one. Aggarwal and Sykes' Fig. 1 shows widespread seismic activity throughout New England and adjacent parts of Quebec during the 1970-1977 period. Zones of seismic activity appear to have northerly or northwesterly strikes, particularly a broad belt extending north-northwesterly from the Mohawk valley for about 500 kilometers. Other centers of activity are present in western New York State, perhaps with a generally northwesterly trend, and in northern New Jersey and southern New York with a more northerly trend. A wide

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scatter of earthquakes is also present from Connecticut to Massachusetts and northward into Quebec.

Focal mechanism solutions for nine recent small earthquakes in northern New Jersey and southern New York, four of which are associated with the Ramapo fault zone, indicate high angle reverse faulting, with an apparent small component of strike-slip movement, suggesting regional compression, acting in a northerly direction. According to Aggarwal and Sykes the earthquakes are shallow, many within 1 or 2 kilometers of the surface. Many are concentrated along exposed zones of ancient brittle crystalline rocks. It is tempting to suggest that the region is undergoing a shearing motion, with a counterclockwise rotation sense. The local coincidence of seismic activity with ancient fault systems is related to an accidental orientation favoring relief of north-south compressive stress.

The Aggarwal-Sykes frequency analysis involves a few small earthquakes detected in the general vicinity of the Ramapo fault in the 1974-1977 period, plus six earthquakes of intensity VI and VII, felt in the greater New York City area over the past 250 years. One of the latter was located in northern New Jersey near Asbury Park, well away from the Ramapo fault zone. Despite some evidence to the contrary, Aggarwal and Sykes have assumed that the other five occurred on the Ramapo fault. They then note that an emperical relationship determined for the southern New York - northern New Jersey area seems to rationalize the small seismic events on the Ramapo fault with the larger historical events on a frequency-magnitude plot. It is the interpretation of this plot which led Aggarwal and Sykes to their prediction of a rather high probability for an intensity VII or larger earthquake on the Ramapo fault near the Indian Point site within the 40-year life of the plant.

The acceptance of the above conclusion depends entirely on the judgment that the Ramapo fault is the dominant fault for seismic strain release in the southern New York - northern New Jersey area. This conclusion seems unacceptable because of the improbability that all or most of the larger seismic events cited should be attributed to activity

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along the Ramapo fault. Fig. 1 in the "Comments on the Seismicity in southern New York - northern New Jersey," issued by PASNY's Seismic Consultants shows quite clearly that seismic activity has been broadly distributed over known faults in the New Jersey Highlands and in the New York City-Dutchess County area, as well as in the Ramapo fault zone. In fact, intensity V and larger earthquakes seem to characterize the Highlands and New York-Dutchess County faults, while avoiding the Ramapo zone! It, therefore, seems much more probable that the five intensity VI and VII earthquakes used in the Aggarwal and Sykes analysis should be assigned to fault zones other than the one associated with the Ramapo fault.

In their Table 1, Aggarwal and Sykes indicate three methods for calculating the probability of equalling or exceeding intensities VII and VIII at Indian Point. Of these the third, probabalistic calculation by McGuire based on historical events throughout the region, would seem to be the best that can be done with the historic data for analyzing the seismic threat to the Indian Point site. That analysis gives exceedingly low probabilities of VII or VIII events affecting a particular spot within the New England seismic zone.

As a general comment, I would like to suggest that a predictive analysis for frequency versus magnitude of earthquakes probably is only useful at a regional scale. The large amount of data gathered for the San Andreas fault system, for example, gives a reasonable basis for rough predictions of probability of occurrence of events of certain intensity along the entire San Andreas system. However, it is easy to see that these same probabilities do not apply to the various parts of the system, and in fact could give very misleading results. Compare, for example the northernmost part, which appears to be essentially locked, with parts near Los Angeles which are deforming by creep at the present time. It seems to me that the determination of the probability of occurrence of an event of a certain intensity along the northeastern end of the Ramapo fault must be essentially meaningless unless that conclusion applies equally to a large region surrounding this particular site.

John C. Maxwell

John C. Maxwell
Consultant

c.c. Mr. E. Igne, Staff Engineer

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Appendix XIX

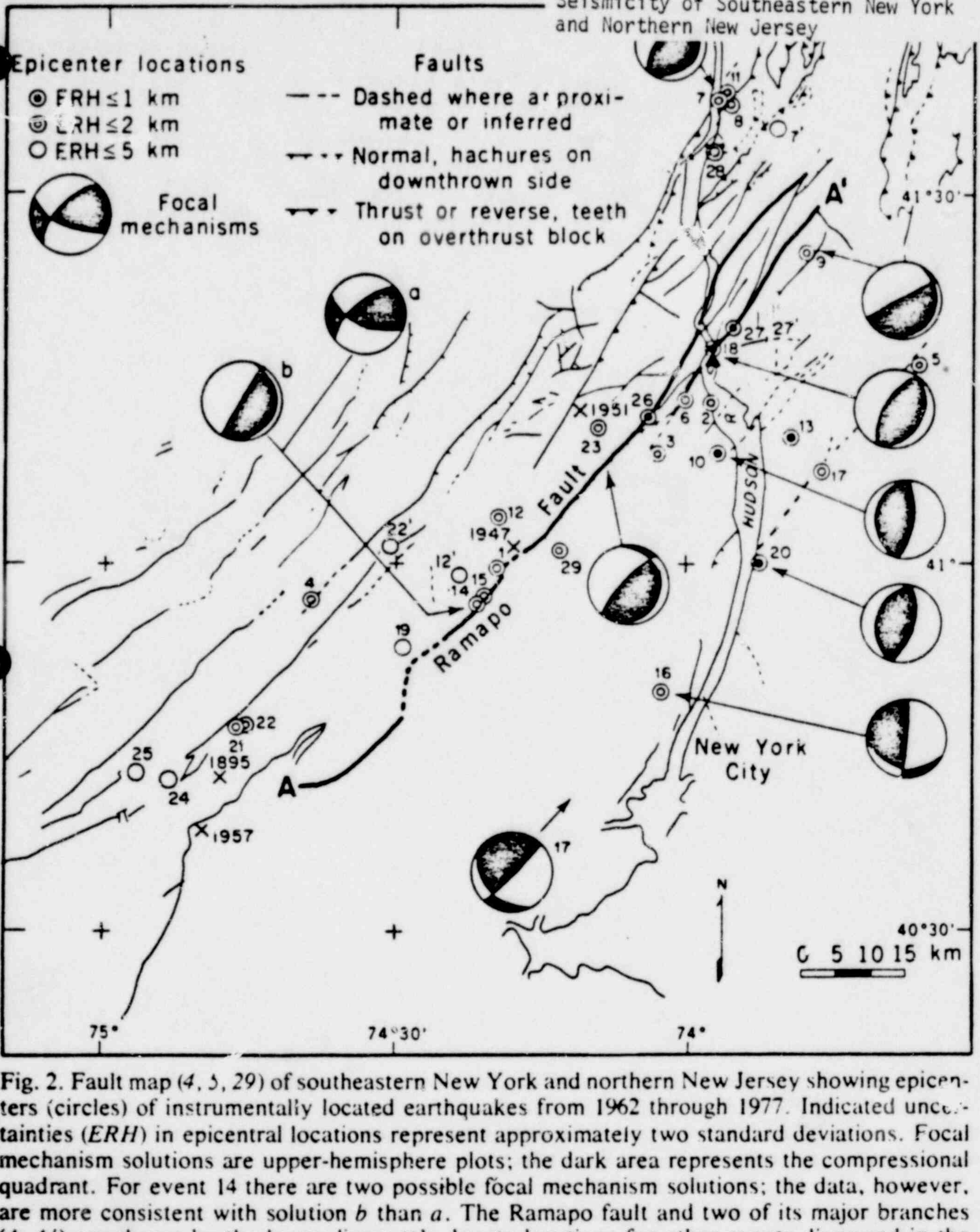


Fig. 2. Fault map (*4, 5, 29*) of southeastern New York and northern New Jersey showing epicenters (circles) of instrumentally located earthquakes from 1962 through 1977. Indicated uncertainties (*ERH*) in epicentral locations represent approximately two standard deviations. Focal mechanism solutions are upper-hemisphere plots; the dark area represents the compressional quadrant. For event 14 there are two possible focal mechanism solutions; the data, however, are more consistent with solution *b* than *a*. The Ramapo fault and two of its major branches (*A–A'*) are shown by the heavy lines; ×'s denote locations for other events discussed in the text. The solid triangle shows the location of the Indian Point nuclear power reactors.

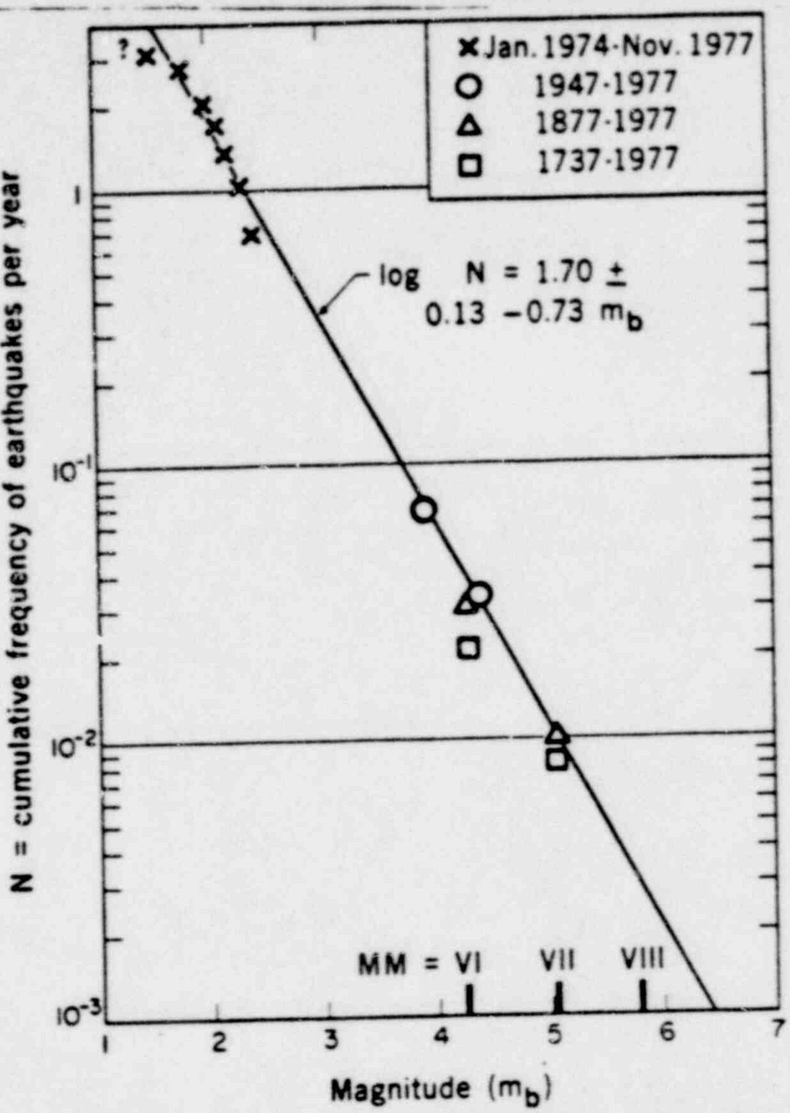
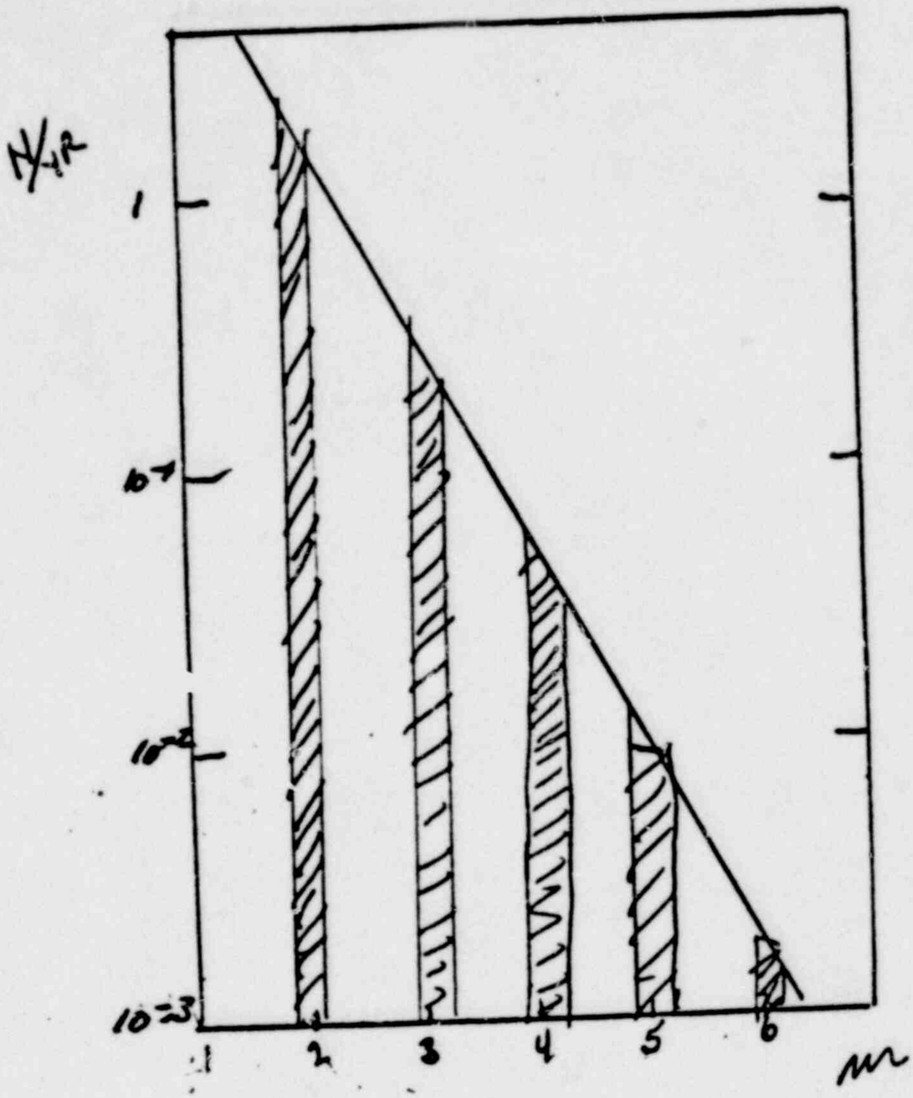


Fig. 4. Cumulative number (N) of earthquakes of magnitude m_b or greater per year as a function of magnitude. Data sets are each for the 120-km-long segment of the Ramapo fault and for shocks located within 10 km of the fault. The question mark denotes the minimum value, that is, the incomplete detectability of events of that magnitude. The slope of the curve, 0.73, was determined independently for recent shocks in New York and adjacent areas. The intensity-magnitude relationship is from (*19*). The uncertainty, ± 0.13, in the value of a ($\log N = a - bm_b$) represents the 95 percent confidence interval.

$\log N = 2 - .8m$			$\log N = 2.0 - 1.2\,m$	
m	N/YR	YR	N/YR	YR
3	.398	2.5	.025	39.8
5	.01	100	.0001	10,000
6	.00158	630	6.3×10^{-6}	1.58×10^{5}

$$\log N = a - b\,m$$



RECURRENCE CURVE

— $\log N = a - b\,m$

— # OF EVENTS/YEAR PLOTTED AGAINST MAG.

— LOGARITHMIC LINEAR - SMALL CHANGES IN SLOPE MAKE VERY LARGE CHANGES IN RECURRANCE TIMES

Appendix XX

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

May 26, 1978

APPENDIX XX
Davis-Besse: Background Material on Orifice Rod Assembly Failures

H. Etherington, Chairman
B&W Water Reactors Subcommittee

B&W ORIFICE ROD ASSEMBLY (ORA) FAILURES

On Friday, May 19, Davis-Besse 1 reported failures of several ORAs. The latching mechanism on the ORAs is similar to that on the Burnable Poison Rod Assemblies (BPRAs) which caused problems at Crystal River 3. (The Committee was briefed on this at the April meeting) The ORAs weigh only about 1/3 as much as the BPRAs and, whereas the BPRA's were believed to "chatter" at 4 pump operation (and were therefore restricted to reduced operation), the ORA's being lighter and held up by flow are believed to encounter "chatter" problems at 2 pump (reduced flow) operation. (Davis-Besse 1 had considerable time on 2 pumps which is thought to be a reason the ORA failures occurred there first). (The CR-3 ORA's were inspected and were not badly worn)

The Staff called for a meeting with B&W and affected B&W 177FA licensees on Wednesday, May 24. B&W explained what inspections had been made and what additional information would be gathered by June 15th. (Complete inspection of all ORAs at Davis Besse 1, Metallurgical exam of B-4 (new) end fitting, work in hot cell on B-3 end fitting, pressure pulse effects analysis, inspection of Oconee-3 ORAs after June 6 shutdown, flow tests at Alliance Research Center, TMI-2 inspection when BPRA retainers installed) B&W felt the problem probably was not generic but was flow related and requested approval to continue operating on the basis that failure of an ORA would not be a safety problem and that, especially with increased attention to the Loose Parts Monitors, failure of an ORA would not escape undetected. Davis Besse planned removal of some ORA's because, even with the resultant increased by-pass flow, DNBR was not expected to be a problem.

At this point the Staff has not taken any limiting action, but has asked B&W to submit an analysis of why dislodging of an ORA would not be a safety problem. This will be discussed at the NRC Staff session Friday morning.

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The cure to the problem is probably a question of material selection for the lands of the holddown latch assembly (see attached slide). A harder, stronger material would prevent the wear which permits the ORA's to become dislodged.

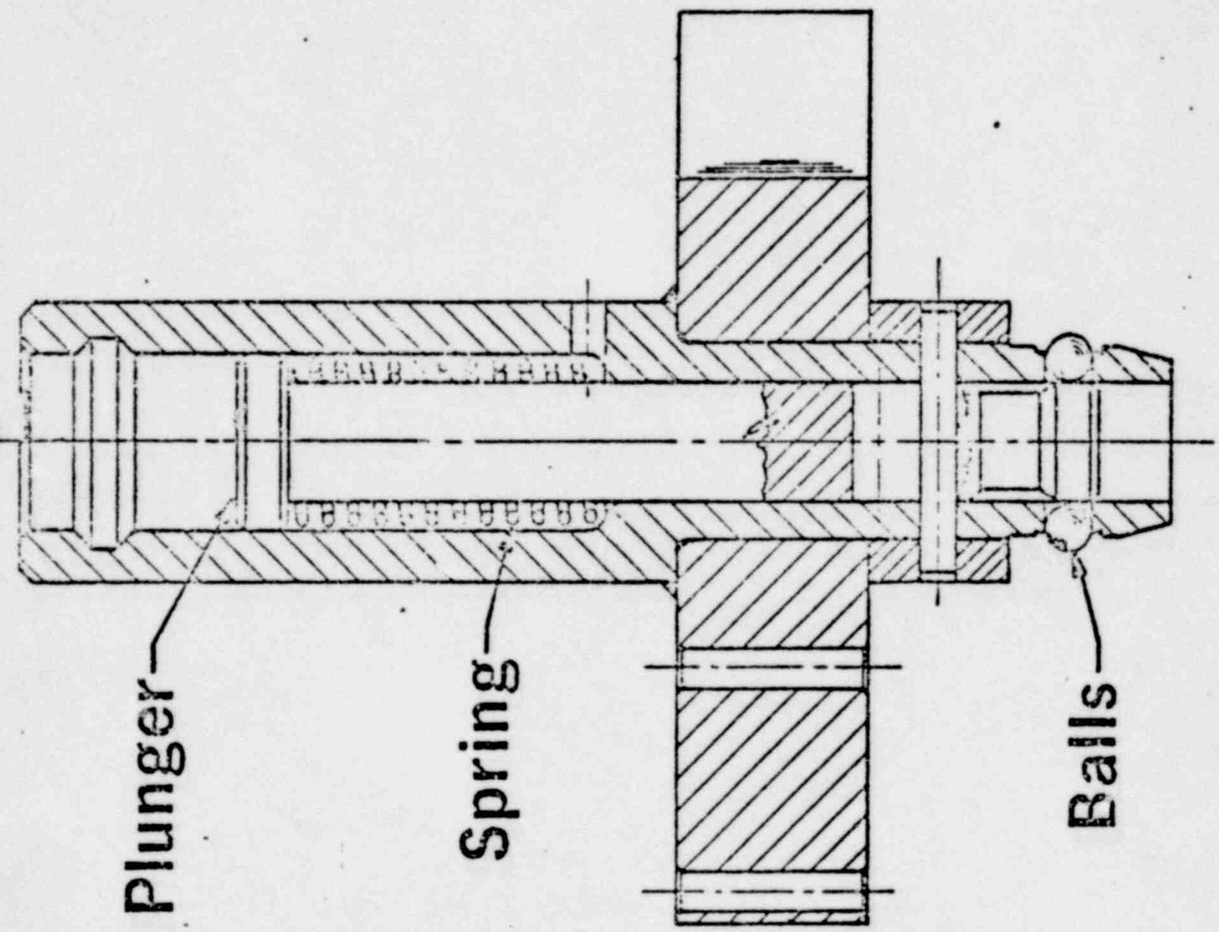
(The B&W Mark C fuel to be used in Bellefonte and newer plants, has a positive locking mechanism which would not have this problem)

An update on the CR-3 problem is attached.

Ragnwald Muller
Senior Staff Engineer

Attachments:

(1) Slides used by B&W
(2) FPC Letter dated May 16


Plunger
Spring
Balls

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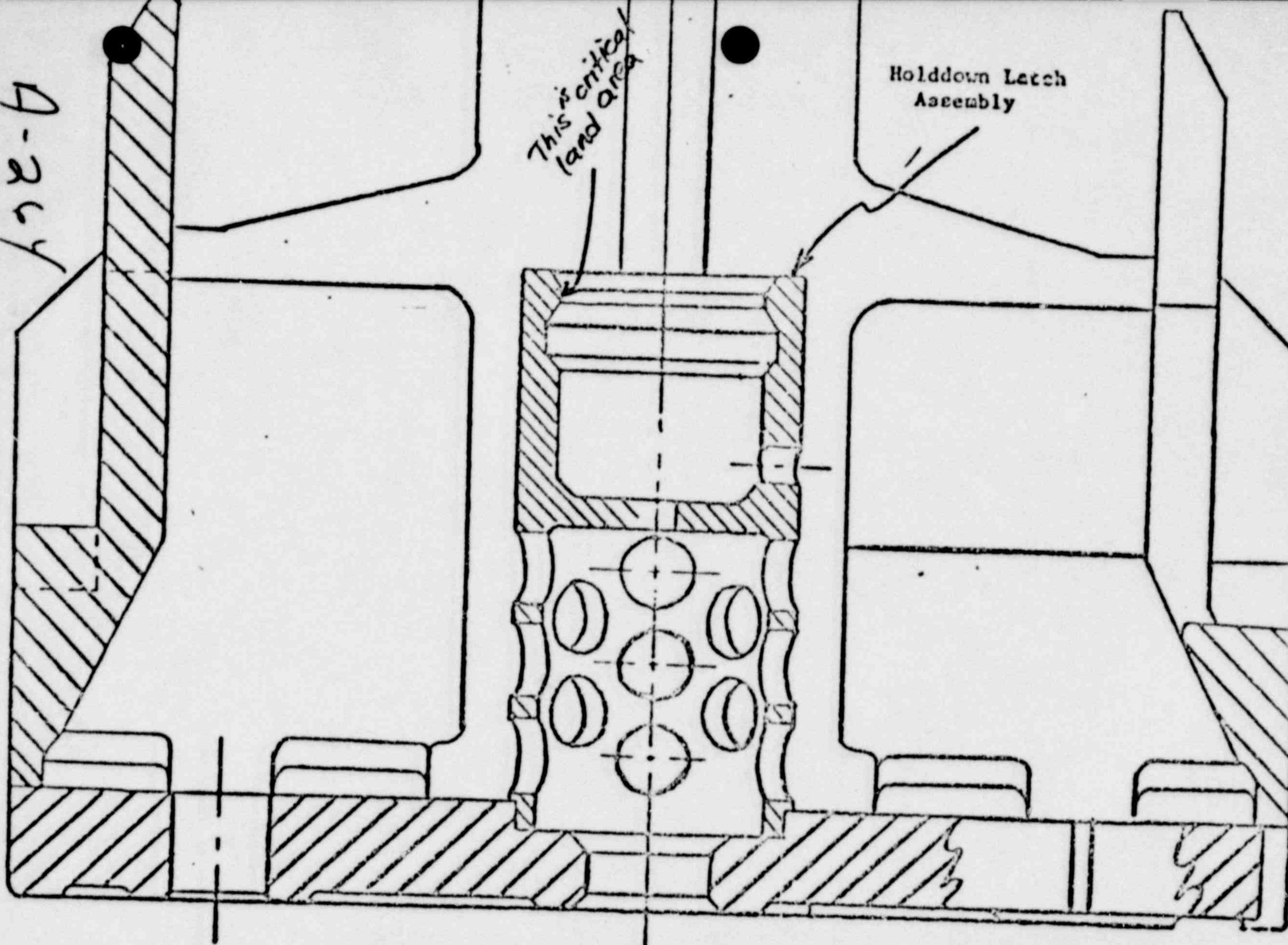
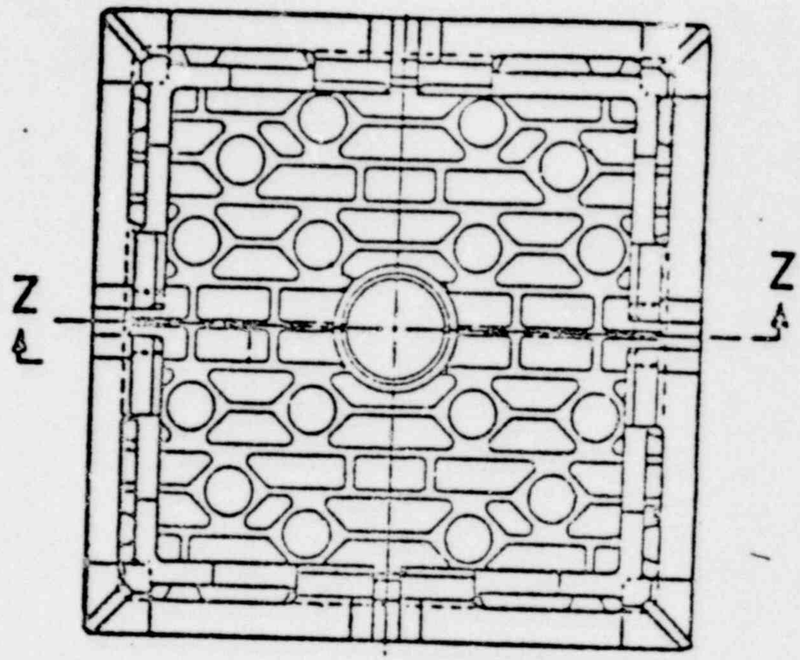
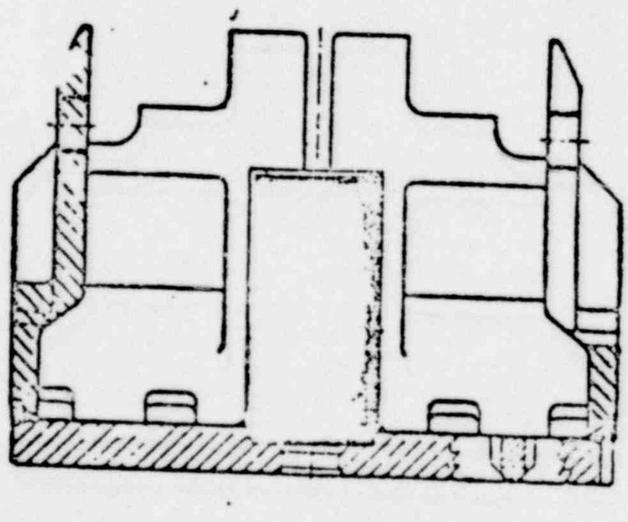


UPPER END FITTING AND HOLDDOWN LATCH ASSEMBLY

WILCOX
DATE
BY
REVISION
SUBJECT
NO

UPPER END FITTING ASSEMBLY





SECTION Z-Z

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MATERIAL PROPERTY COMPARISON
HOLDDOWN LATCH ASSEMBLIES

CORE	MATERIAL	YIELD STRENGTH PSI	HARDNESS BHN	HARDNESS R_B
CR-III	BAR	36,000	149	
		63,000	217	
ANO-I	BAR	37,000	156	
		63,000	217	
OCONEE-II	BAR	33,500	~145	77.1
		69,000	197	
DAVIS BESSE-I	CAST	40,700	~153	80
		48,000	~170	85
SMUD	BAR	63,000	207	
		75,000	217	
GENERIC RELOADS	CAST	~38,000		~78
		~47,000		~78-80

POSSIBLE WEAR CONTRIBUTORS

1. REDUCED FLOW; I.E. 2 AND 3 PUMP OPERATION

2. MARK-B4 END FITTING

   A. CAST HOLDDOWN LATCH

   B. REDUCED PRESSURE DROP

3. OTHER

   A. PUMP PRESSURE PULSATIONS

   B. MINOR MANUFACTURING VARIABLES

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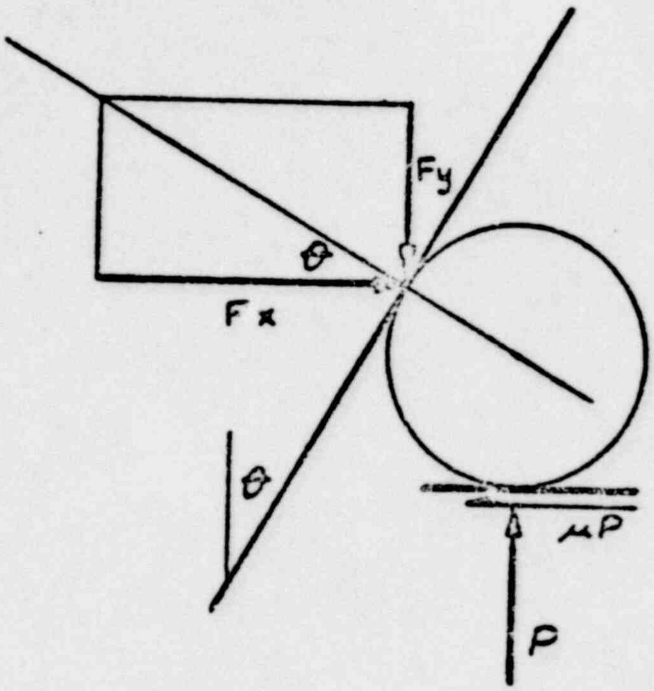
PRELIMINARY PUMP OPERATING HISTORY

MAY 23, 1978

REACTOR	CYCLE	NUMBER OF OPERATING DAYS 1 PUMP	2 PUMPS	3 PUMPS	4 PUMPS
DAVIS-BESSE-1	1	1	35	41	194
TMI-1	1				
	2	6	4.5	2	277
	3	1.5	1.5	2	305
	4	-	>1	>1	27
TMI-II	1	-	16	28	2
ANO-I	1				
	2	>1	3	1	
	3	>1	1.5	2	
OCONEE-I	°	19	5.5	9	
OCONEE-II	°	27	12	13	
OCONEE-III	°	5	3	18	
CR-III	1		7.5	24	
RANCHO SECO	1	6	8	6	608
	2	>1	>1	14	

° OPERATING SINCE HK-B4 END FITTING

BPRA UNLATCHING CONDTION



FOR BALLS TO RETRACT
$F_x \gg \mu P$

FOR $\mu = .4$
$\mu P = .4P$

	30°	45°	60°	75°
Fx =	1.72P	1.0P	.58P	.27P

OPERATING PLANT COMPARISON

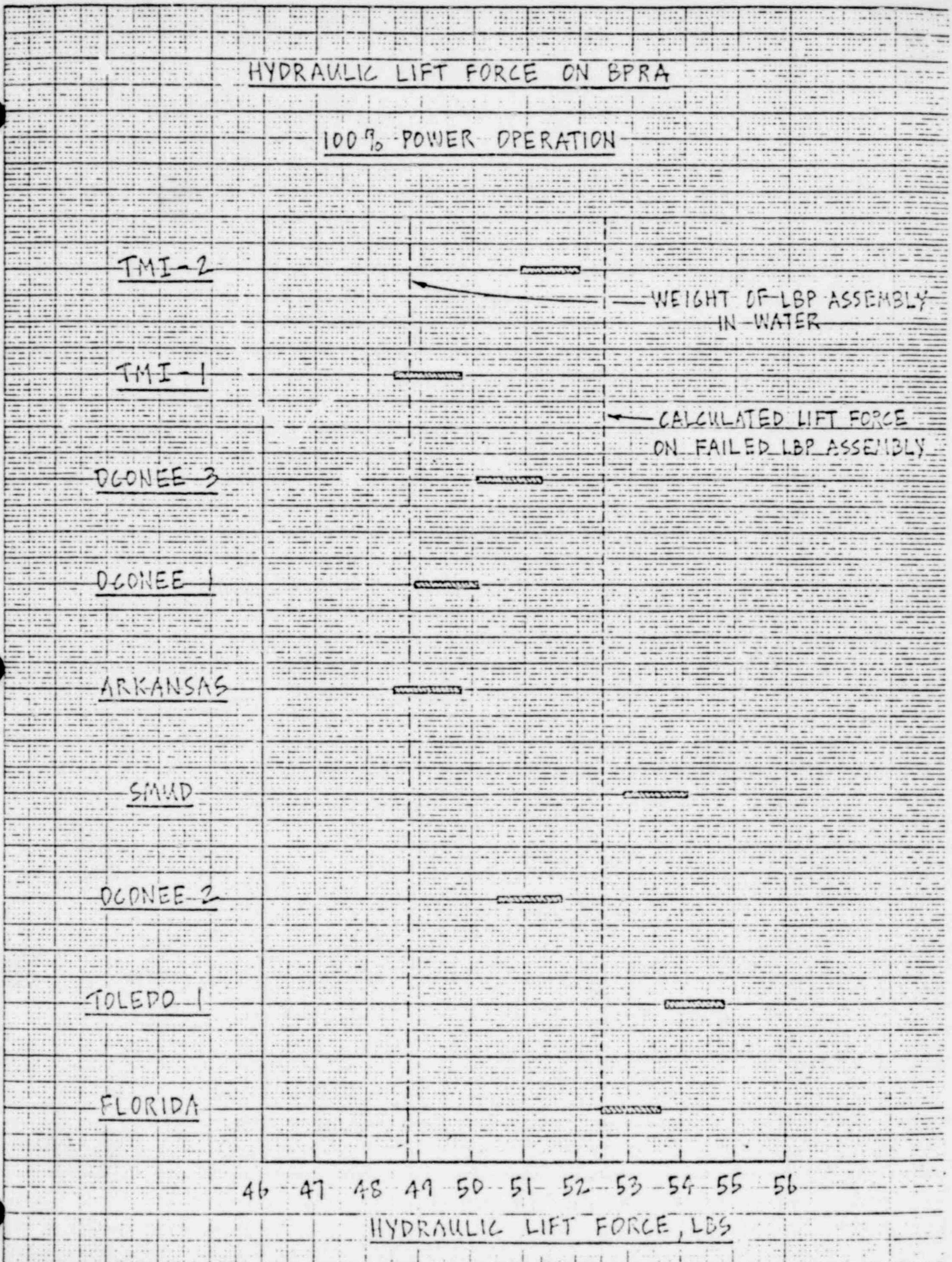
NSS	3	4	5	6	7	8	9	11	14
PLANT	DUKE I	DUKE 2	TMI-1	TMI-2	CR-3	ANO-1	DUKE-3	SMUD	DB-1
PUMP TYPE	W	BINGHAM	W	BINGHAM	B-J	B-J	BINGHAM	BINGHAM	B-J
RATED POWER (MWT)	2568	2568	2535	2772	2452	2568	2568	2772	2772
SYSTEM FLOW (+ 2%) (% OF 352000 GPM)	109	111	109	110	112	109	110	114	113.4
LICENSED (MIN) FLOW	106.5	106.5	106.5	105	105	106.5	106.5	105	105

A-270

THERMAL-HYDRAULIC DESIGN
177 FA PLANTS

	MAXIMUM CORE BYPASS FLOW (% OF SYSTEM FLOW)	
	TOTAL	GUIDE TUBE
INITIAL BASE - ALL FA'S CONTAIN CRA's, BPRA's, OR ORA's	6.0	1.7
TYPICAL PRACTICE - 36 TO 44 ORA's REMOVED	8.3 (7.8)	4.0 (3.6)
ALL BPRA's/ORA's REMOVED	10.4	6.4

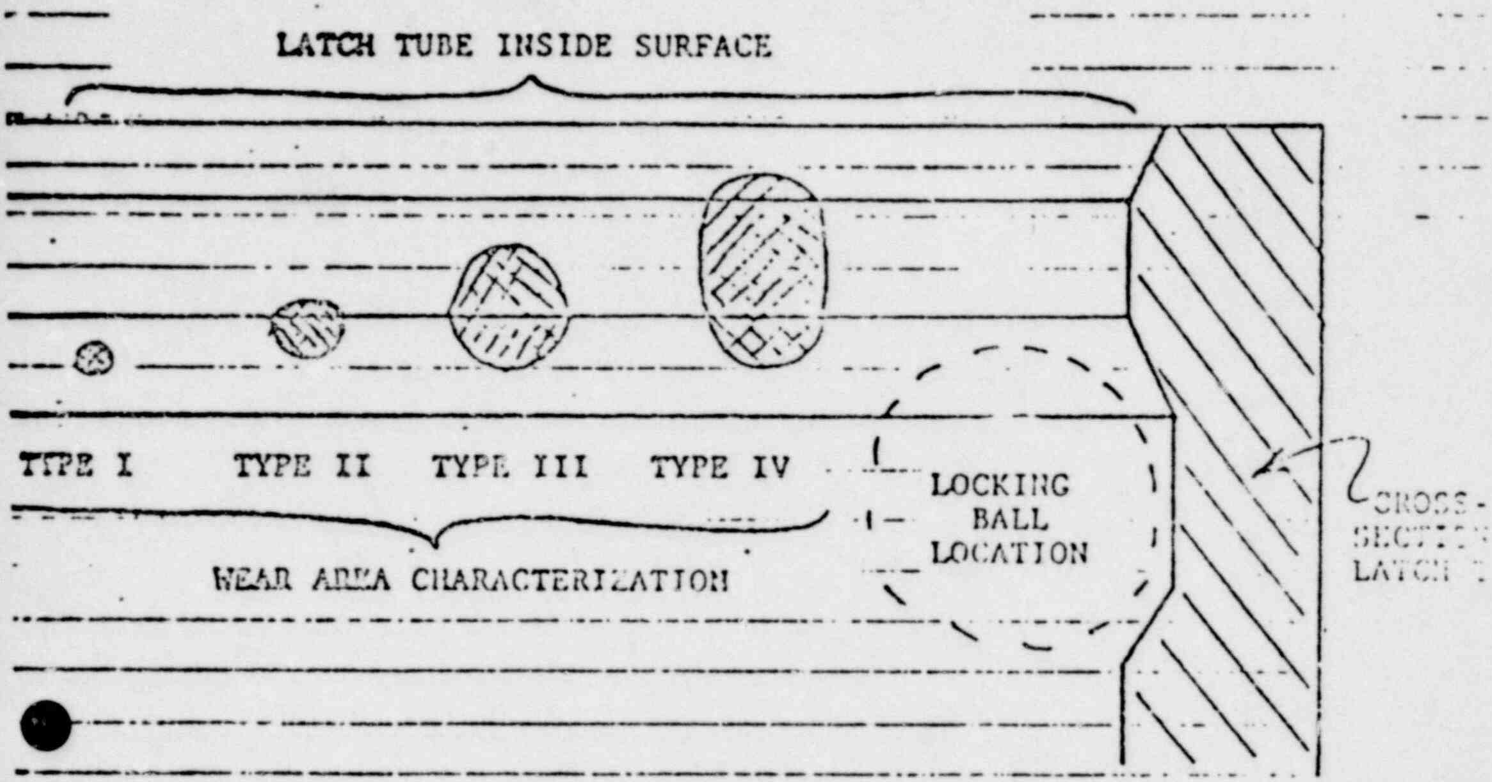
NOTE: REMOVAL OF ONE ORA INCREASES CORE BYPASS BY ~ 0.04%


HYDRAULIC LIFT FORCE ON BPRA
100% POWER OPERATION
TMI-2
WEIGHT OF LBP ASSEMBLY IN WATER
TMI-1
CALCULATED LIFT FORCE ON FAILED LBP ASSEMBLY
OCONEE 3
OCONEE 1
ARKANSAS
SMUD
OCONEE 2
TOLEDO 1
FLORIDA
46 47 48 49 50 51 52 53 54 55 56
HYDRAULIC LIFT FORCE, LBS
RLH 5-18-78

A-272

FIGURE 1

QUALITATIVE DESCRIPTION OF LATCH TUBE WEAR



TYPE I - SMALL WEAR AREA ON LOWER CHAMFER ONLY.

TYPE II - WEAR INTO VERTICAL LAND, BUT LESS THAN HALF THE HEIGHT.

TYPE III - WEAR INTO VERTICAL LAND, GREATER THAN HALF THE HEIGHT.

TYPE IV - WEAR INTO UPPER CHAMFER.

BPRA BPRA

PLANT	NO. OF LATCH TUBES EXAMINED	PERCENTAGE OF TUBES SHOWING WEAR OF: TYPE I	TYPE II	TYPE III	TYPE IV
CR-3	68	7	49	34	10
OCONEE-2	25	52	44	4	0
ANO-1	36	47	53	0	0
D-B-I	24	54	13	20	13

Florida Power
CORPORATION

May 16, 1978
File: 3-0-3-a-3

Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit No. 3
Docket No. 50-302
Operating License No. DPR-72
Florida Power Corporation

Dear Sir:

Attached are responses to Items 1. through 9. of Part I, Enclosure 1 to your letter of May 2, 1978.

This information is being submitted in accordance with the schedule of activities outlined in our letter to you of May 15, 1978.

Please advise if further discussion on the attached is desired.

Sincerely,

Q. B. DuBois /for
W. P. Stewart
Director, Power Production

luv

A-274

1. Q: Describe cleanup operations for removing debris from the primary coolant system. This should include a description of any grappling, flushing, filtration, and vacuum cleaning techniques to be used. You should also describe which method(s) will be used for each component (e.g. fuel assemblies, reactor internals, steam generators, piping, valves, etc.).

A: Debris removal from the RCS has/will be accomplished by a variety of means. A summary, by component, follows:

Fuel Assemblies: BPRA pins were removed from the guide tubes by mechanical grabbers. Upper and lower end fittings were cleaned with a combination of mechanical grabbers, picks and vacuums. Cleaning was preceeded and followed by a detailed video inspection of all upper and lower end fittings as well as a detailed side view inspection of selected fuel assemblies.

Plenum: Video inspections were conducted and debris removed with a mechanical grabber. A free path check of all control rod guides is planned.

The Reactor Vessel: . Vacuumed to remove all debris and video inspected. This inspection included the inlet and outlet piping.

Core Support Assembly: Again video inspections were accompanied with mechanical grabbers, picks and vacuuming. Some debris was simply knocked out through the bottom and will be vacuumed up later.

B-OTSG: Debris was manually removed from the OTSG upper tubesheet and lower head. A visual inspection of the J leg piping showed no debris. All tubes found with debris will be cleaned with a stiff rod and cable. A 100% free path test of all tubes will be followed by eddy current inspections.

A-OTSG: A visual inspection of the upper head revealed no foreign material. Following a free path inspection of 100% of the tubes, the lower head and J leg piping will be visually inspected. Any foreign material will be removed.

2. Q: Describe the cleanup inspection procedures and techniques which will be used. This description should include any methods used to identify the absence of residual debris and the capabilities of the inspection techniques to identify the debris.

A: Cleanup inspection procedures and techniques consist of CR#3 instructions based on approved plant operations and maintenance procedures. They basically give detailed instructions for the removal/disassembly of components within the pressure boundary and debris collection. Inspections for debris have been primarily video using assorted underwater television equipment. Documentation of inspection and cleanup operations is by video tape and independent observation by at least two observers.

All debris observed using video equipment will be removed.* Manual grabbers have removed pieces of debris from 12 feet long down to less than one inch. Vacuuming has removed debris from several inches in length down to debris that appears as specs on the video screen.

3. Q: Describe the damage inspection procedures and techniques used. Identify which components will be inspected for damage, and what criteria will be used to determine the acceptability of any components found damaged.

A: Damage inspection procedures consist of a combination of station approved operation and maintenance procedures and procedures developed by Babcock & Wilcox Company. Components to be inspected include the Reactor Vessel, Core Support Assembly, Plenum, Fuel Assemblies, Control Rod Drive Mechanisms, and Once Through Steam Generators.

Criteria for acceptability is based on the application of each component examined, detailed Engineering evaluation of any damage observed, and inspection of like components in other Babcock - Wilcox NSSS. Documentation of examinations is by video tape and independent observation by at least two observers.

4. Q: Provide the results of the cleanup and inspections discussed above. Itemize the total debris recovered and any debris that is not recovered.

A: Results of the cleanup are analyzed and documented as each operation is completed; including size estimates of the debris recovered. Documentation is by video tape and procedural sign-off. Debris has been removed from the fuel assemblies (upper & lower end fittings and BPRA guide tubes), reactor vessel, core support assembly, plenum and Once Through Steam Generators. Both BPRA spiders and couplings were recovered, one intact in the plenum, one in pieces in OTSG. Of the total

* See attached supplemental Information page.

4. A: Cont'd.

of 403'8" of BPR rod in the two BPRA, total inventory to date is 397'8". Further searches for debris will be free path checks of the reactor plenum, the OTSG A&B tubes, and final cleanup of the Core Support Assembly.*

5. Q: Determine the potential effect(s) that residual poison and metallic fragments will have on plant operations. As a minimum address the following areas:

   a. Flow blockage of fuel assemblies. This should include a conservative estimate of channel blockage at the end fittings, grids, and in between grids. You should address the potential for DNB and local cladding hot spots which may cause cladding perforations. The potential for propagation of fuel failures and the means of monitoring and/or mitigating such conditions should also be discussed.
   b. The potential for blockage and/or binding of the control rod drive systems due to residual coolant debris. Any procedures planned to mitigate and/or monitor these conditions should be provided.
   c. Blockage of the guide tubes which would prevent control rod insertion and safe shutdown operations.
   d. Mechanical damage to primary internals due to impacting.
   e. Blockage and/or binding of any orifices, valve seats, and vent valves in the primary coolant system.
   f. Blockage and/or erosion of steam generator tubes.
   g. The effects that the residual debris will have on pumps and any other components with moving parts.
   h. Effects on coolant chemistry and crud levels.

   A: B&W has evaluated the potential effects of residual poison and metallic fragments and has determined that none of the effects will be detrimental to safe operation. As discussed in question 4 the vast majority of the debris has been removed and any small fragments remaining in the system will soon be flushed to the bottom of the reactor vessel where their effect will be minimal. This position is supported by the previous operation of Arkansas Nuclear Unit One and Oconee Unit Two for several

* See attached supplemental Information page.

5. A: Cont'd.

months with similar size debris in the system with no adverse effects. Considerations was given to the following areas:

a. Flow blockage of fuel assemblies -

The potential effects of residual poison and metallic fragments on DNB are minimal. See response to Question 1 & 2 for cleanup procedure. Any debris left in the system will eventually be carried to the core inlet and become trapped in the lower end fitting or lower end spacer grid. The effect of debris trapped in the lower end fitting has been evaluated using the crossflow codes LYNX1/LYNX2. The results demonstrate that blockage of 20% of the fuel assembly inlet flow area decreases the DNBR by less than 0.1%. A blockage this large is extremely unlikely since it would require several large pieces of debris to be lodged in the same fuel assembly.

It is highly unlikely for debris to work its way into the active fuel region of the core. The largest strip that can fit through a spacer grid would be approximately 0.140" wide. However, if one assumes that a blockage does occur at the spacer grid just below the point of minimum DNBR and that 75% of the flow area in two adjacent channels is blocked, the resulting reduction in DNBR is approximately 5%. This calculation does not consider that turbulence intensities are very high behind the blockage. A study of pressure and flow in a fuel bundle containing blockages conducted by Battelle Pacific Northest Laboratories measured turbulence intensities five times greater than normal for the area just behind the blockage. This increase in turbulence should offset the loss of flow due to the blockage.

The potential for propagation of fuel failures due to a blockage is extremely remote. The means that is used to postulate the first failure (forcing the coolant from one channel) protects the adjacent channels because more coolant is forced into these channels; thereby, increasing the margin and reducing the possibility of further failures.

The water chemistry is monitored daily and any fuel failure would be detected by this routine inspection.

b. Control rod drive system

B&W inspection of upper plenum cover showed no debris. If coolant debris were ever to reach the Control Rod Drive Mechanisms (CRDM) internals, it would necessarily have to exist in the area between the upper plenum cover and the reactor vessel nozzles. Since no evidence of debris was found on the plenum cover, the possibility for debris in the mechanism is essentially precluded.

In addition, a Diamond Power Supply Company (DPSC) representative was called to the Crystal River site to inspect the control rod drive leadscrews and closure insert components. The results of this inspection, conducted under the reactor vessel head are: there is no aluminum oxide debris in the CRDM internals.

Further, it was pointed out by DPSC that -

1. Inspection of CRDM components after design life testing have shown that a considerable amount of metallic debris could be present with no detrimental affect on mechanism operation.
2. Inspection of drives which have been ratchet tripped have shown that chips from the leadscrew can be present in the rotor assembly area of the mechanism. Presence of these chips has never prevented a control rod from being tripped or driven into the core.

In summary, based on the above information B&W and DPSC concur that further CRDM inspection is not justifiable and that the CRDM's may continue in normal operation.

c. Possible blockage of control rod guide tubes -

B&W has made an extensive effort to identify and retrieve all the loose pieces from the primary system. Detail description of the efforts made is given in the responses to the question 1 thru 4.

In summary, debris from all the guide tubes of fuel assemblies 3C35 and 3C37 were removed and cleanliness of the guide tubes of these two assemblies were verified by a special probe.

There is a possibility that the small pieces might get into guide tubes and cause some interaction with moving components. However, based on ANO experience, the probability of this occurrence is very small.

d. Possible damage to primary internals -

A detailed inspection of the reactor internals has been carried out and no structural damage detrimental to the function of the reactor internals has been found. In fact the only damage attributable to the loose debris is some minor dings near a large flow hole in the plenum cylinder. This is believed to have been caused by impacting of the LBP spider coupling before the assembly escaped entirely from the fuel assembly. The fact that no other structural damage was found in the internals, although a significant amount of debris was found on the fuel assembly lower end fittings, the lower internals and the lower head of the reactor vessel, suggests that the parts that are able to pass through the system are too small to cause structural damage.

In conclusion, based on the fact that (1) a detailed inspection of the reactor internals was performed and no detrimental structural damage found, (2) all debris that is found will be removed and, (3) possible remaining debris would be small, no detrimental effects on the function of the reactor internals either present or future are expected from the LBP failures and resulting debris.

e. Effects on RCS valve seats or vent valves -

All vent valves, including the seating surfaces, were visually inspected with a TV camera. This inspection revealed no detrimental structural damage. The only indication of any type was a minor impact mark on one vent valve jack screw, believed to have occurred during removal of the plenum assembly (Plenum assembly was removed from the vessel without the aid of the indexing fixture to facilitate removal of the LBP assembly lodged in the plenum region).

In addition to the detail inspection, the vent valves were exercised and found to operate freely.

In conclusion, based on the results of the visual inspection which revealed no detrimental damage and the fact that the valves moved freely when exercised provides sufficient evidence that the function of the valves have not been impaired.

The possible effects of residual debris on the Pressurizer Safety valves was also considered. Our findings show that any debris particles, which could be drawn from the pressurizer into the safety valves by the suction created when those valves lift, would pass through the valves and into the discharge system without obstructing flow. While marring of the valve seating surfaces could occur and result in leakage after the valve closes, this in no way compromises the safety function of these valves.

f. Effects of Steam Generators -

Video inspections of the OTSG B upper tubesheet have revealed damaged tube ends and tube to tubesheet welds. This damage however is not extensive enough to effect the safe operation of the steam generator.

Erosion of steam generator tubes is not anticipated since all partially attached chips and internally lodged debris will be removed; if the debris cannot be removed the tubes will be plugged. Calculations of the effect of the damage shows insignificant changes to the generator pressure drop and reactor coolant flow characteristics. In order to confirm these conclusions the reactor coolant loop flow signal will be monitored at 40, 75, and 100% power.

As any debris remaining in the system will be in the form of small fragments of little mass, additional damage is not anticipated.

g. Possible effects on Reactor Coolant Pumps -

B&W has reviewed the videotapes of burnable poison rod pieces and spring pieces assumed to have passed through the reactor coolant pumps. None of the pieces shown in these tapes are believed to have had sufficient mass or density to significantly damage the pump impeller on impact. Operational data surrounding the incident is limited. The only data available is verbal, and this data indicates that the pump vibration levels following the incident were comparable to the normal pump vibration levels prior to the incident. In addition, the fact that seal injection was maintained makes it unlikely that any foreign material could have entered the seal areas.

Based on the above, disassembly or inspection of the reactor coolant pump is not warranted. B&W recommends continued operation. Due to the lack of data surrounding the incident, additional conservatism will be added by the following action:

> "Startup and escalation data pertaining to the RCP seals and pump vibration data should be obtained and compared with baseline data for these pumps. This data should be forwarded to B&W for final recommendation and confirmation of our assessment".

h. Effects on coolant chemistry and crud levels -

The effects on coolant chemistry and crud levels are expected to be minimal. The increased boron in solution will be insignificant next to normal soluble boron levels used for plant control (1-2 ppm if all the boron in both BPRA's were dissolved in the coolant).

Suspended debris, including Aluminum Oxide, may have an initial abrasive effect on any crud buildings, but this debris will be removed by the makeup and purification filters subsequent to plant startup.

6. Identify the cause of the BPRA failure addressing possible manufacturing, design, or installation errors. Please include:

   a. A description of the "as found" condition of all BPRA in the reactor. Address any indications of improper seating or wear.

   b. Details of nondestructive inspections of the BPRAs, both damaged and undamaged, and orifice rod assemblies. Address any anomalies found with the holddown latch assemblies.

   c. A description of any destructive examinations that have been performed. Address any metallography that has been completed in the areas of wear.

Response

The cause of the two BPRA separating from their fuel assemblies is still under investigation.

a. Coupling spider assembly of BPRA B-47 was discovered in the steam generator B. The assembly was badly beaten up and was broken up in many pieces. These pieces were collected and sent to B&W's Lynchburg Research Center (LRC) Hot Cell Facilities for visual and dimensional inspection.

   Coupling spider assembly of BPRA B-52 was found in the plenum cylinder with several full and partial length burnable poison rods attached and one locking ball present.

   Many full and part length individual burnable poison rod pieces were found in the guide tubes or upper end fittings of the fuel assemblies from which they came out. A long length of the burnable poison rod was also found wedged into the upper end fitting of an adjacent fuel assembly and a small segment was found lying across the upper end fittings of a nearby fuel assembly.

b. Following defueling at Crystal River 3 (CR-3), all 66 remaining BPRAs were subjected to a lock test, and all were found to be locked in their respective fuel assemblies. During removal of the 66 BPRAs, all ball-lock couplings were visually examined; nothing unusual was seen. Nine (9) of the BPRAs were visually examined full-length and 360° around. Nothing unusual was seen. All fuel assembly holddown latch assemblies (68) containing

BPRAs were visually examined 360° around on the inside. Two wear areas were seen on each latch assembly, oriented at 180° to each other.

Three fuel assemblies had wear in the holddown latches which approximated that observed in the holddown latches of fuel assemblies 3C35 and 3C37. While the results of the holddown latch inspections are still being evaluated, preliminary results indicate the wear in the latch assemblies at CR-3 is much higher than the wear observed at Oconee or ANO.

Orifice Rod Assemblies (ORA) at Crystal River 3 were examined as well as the corresponding holddown latches in each fuel assembly. No evidence of wear or any abnormal condition was seen.

Each of the forty ORAs were identified and checked for orientation with respect to the fuel assemblies. The ball latching mechanisms were examined for ball orientation and condition. The holddown latches were examined for evidence of wear, and general visual appearance. Each ORA was reinserted into its corresponding fuel assembly and was verified to be locked in place.

None of the holddown latch assemblies had wear marks, or any features except for two tiny spherical dimples corresponding to the location of the latching balls.

Inspection of ORAs at Oconee, and holddown latches at Oconee and ANO-1 provide additional verification of the observations at CR-3. No evidence of wear or abnormal operation has been seen for any of the ORAs and ORA holddown latches. These results show that ORAs have been used with no failures and no degradation of any kind.

The results of the recent ORA latch mechanism examination firmly supports the current plans of reusing present ORAs. This same ORA design will also be used, as required, to replace BPRAs which are removed. Administrative steps will be taken at Crystal River to assure that the ORAs locking balls are oriented in a direction different than that in which the BPRA locking balls were oriented.

c. Destructive Examination

B&W has not performed any destructive examination on the recovered coupling spider assemblies to date. However, radiographic examination of the coupling spider assemblies was made. Nothing unusual was found which could indicate functional loss of any internal component during the operation.

9. Describe the remedies planned to prevent future occurrence of similar failures.

Response

To avoid future occurrence of similar failure at Crystal River site, BPRAs are replaced by ORAs as stated in response to question 6.b.

Question 7. In your presentation on April 6, 1978, you indicated that the poison rod assembly was lifted out by action of the hydraulic forces within the core. Provide your analysis of this phenomena. The complete analysis should include any simplifying assumptions, conservatisms, and test results used in your evaluation of this phenomena. Describe what provisions are being considered to preclude this condition and how these provisions will effect other plant operations.

Response The lift force on the BPRA was calculated in the following manner: a quarter core LYNX1 model was used to calculate the axial flow and pressure distribution within each fuel assembly containing a BPRA. The formloss coefficients used in this analysis were developed from test data and have been used in all previous Mark-B4 fuel assembly analyses. The lift force was calculated by multiplying the unrecoverable pressure drop times the effective area. The initial calculations predicted a best estimate net lift force of zero to two pounds. A BPRA lift test has recently been completed at Alliance Research Center and the calculational model was adjusted slightly to benchmark the test results. A reanalysis was then performed for the Crystal River 3 BPRA's using the benchmarked model and, as a result, the net predicted uplift force has been revised to three to five pounds.

The lift force on the BPRA is no longer a concern for this plant since all BPRA are being removed from the core.

Question 8 Also during your April 6, 1978 presentation, you indicated that the orifice rod assemblies were lifted by the action of the hydraulic forces. Your basic assumption as to why these assemblies did not experience failure was that they are considerably lighter than the hydraulic forces and therefore are in, essentially, continuous contact with their restraints. This condition was assumed to eliminate, or minimize, the impact (fatique) damage that resulted in failure of the poison rod assemblies. If this is true, provide an analysis on the effects that low flow operations will have on the orifice rod assemblies.

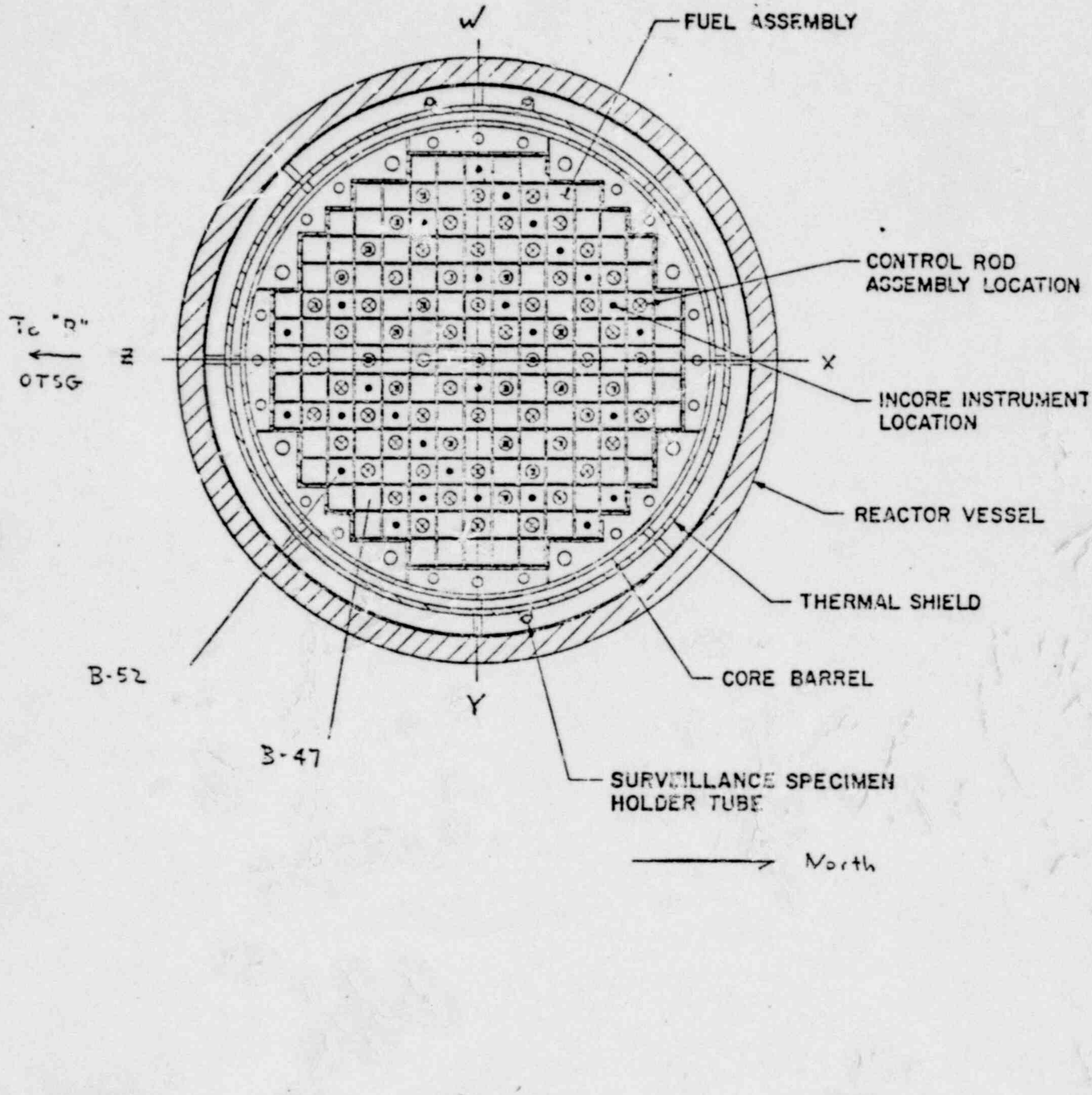
Response The lift force on the ORA was calculated in the same manner as that for the BPRA. The calculated lift force on the ORA is approximately sixty pounds during four pump operation and 35 pounds for three pump operation. The weight of the ORA in water is sixteen pounds. Therefore, for three pump operation the minimum positive lift on the ORA is nineteen pounds. This lift force is fifteen pound higher than the net force on the failed BPRA (under four pump operation) and ten pounds higher than the highest lift force experienced by any BPRA. This margin is sufficient to insure that the ORA's will always be exposed to a positive uplift force during four pump and three pump operation. Furthermore, three pump operation is not the usual operating mode and is used only for limited periods of time. Two pump operation has not been considered because of its limited use.

Supplemental Information

At the present time, all observed debris has been removed from the CSA except for 4 small pieces of BPRA pin. These pieces are located in the lower grid support posts and are estimated to range in length from 4 to 8 inches. If efforts to remove these pieces fail, justification for not removing them will be provided.

Appendix XXI

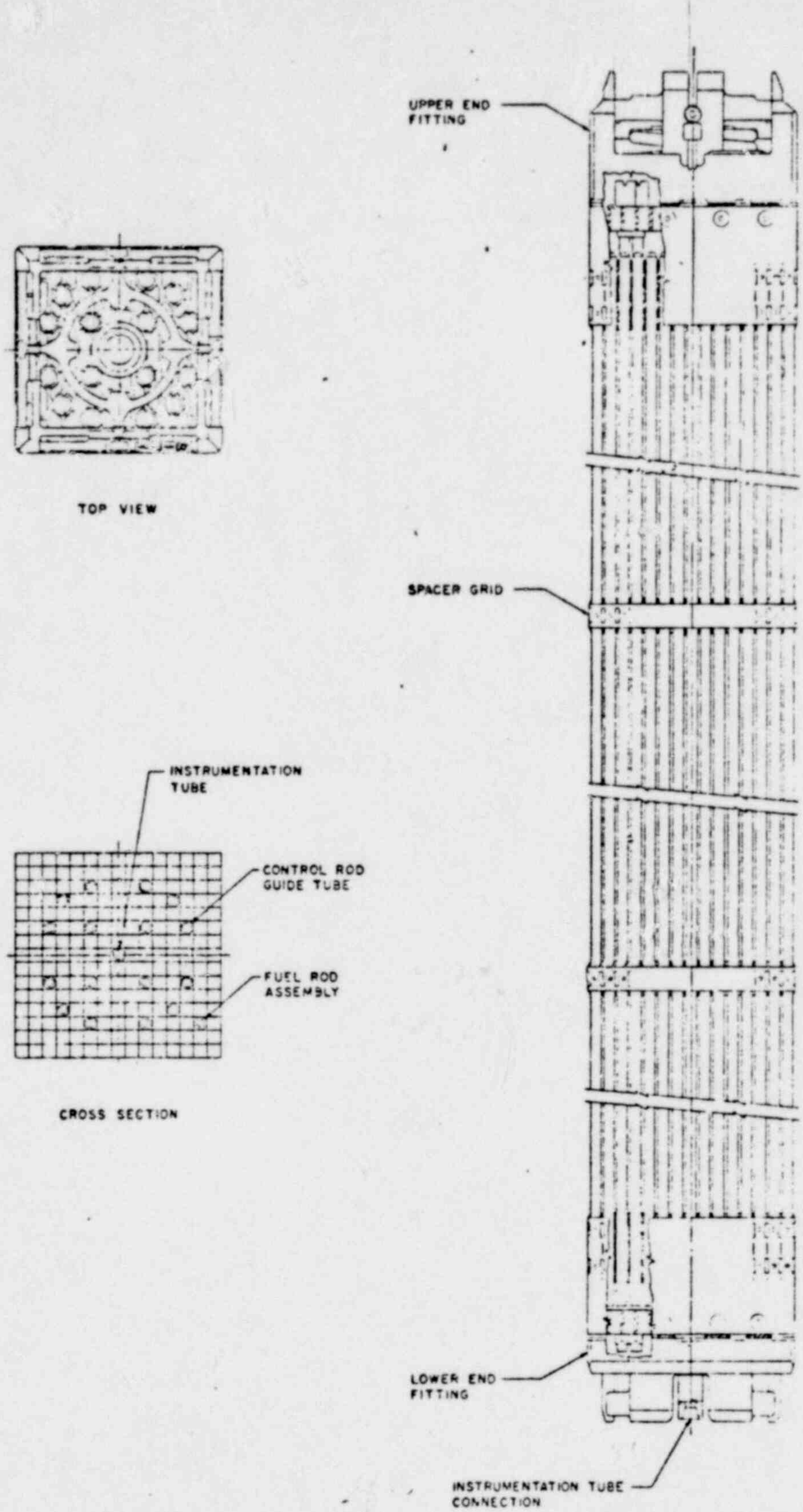
APPENDIX XXI
Davis-Besse: Orifice Rod and Burnable Poison Rod Assembly Failures



REACTOR VESSEL AND INTERNALS
CROSS SECTION
CRYSTAL RIVER UNIT 3

FIGURE 3-56

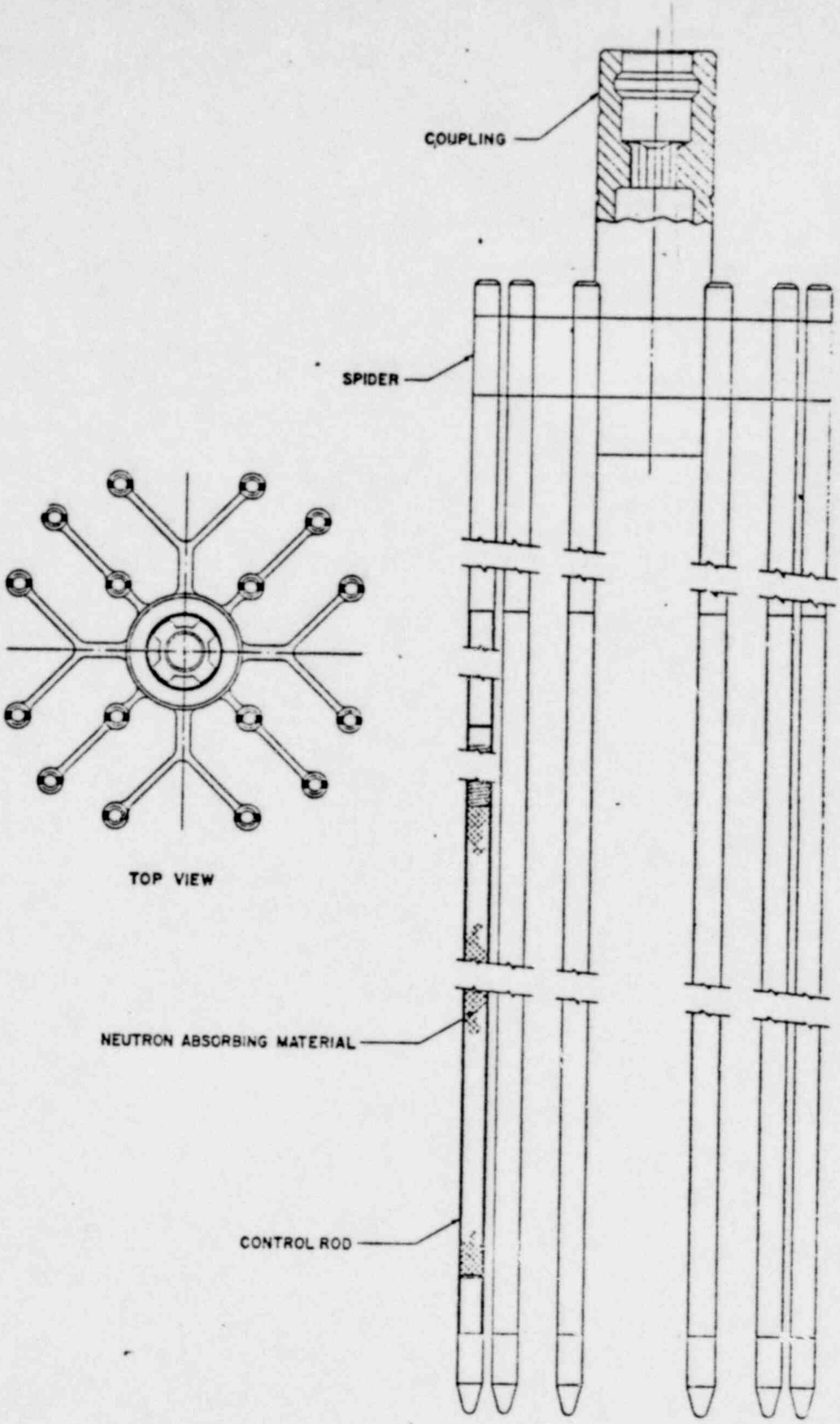
A-286



FUEL ASSEMBLY
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 3-61

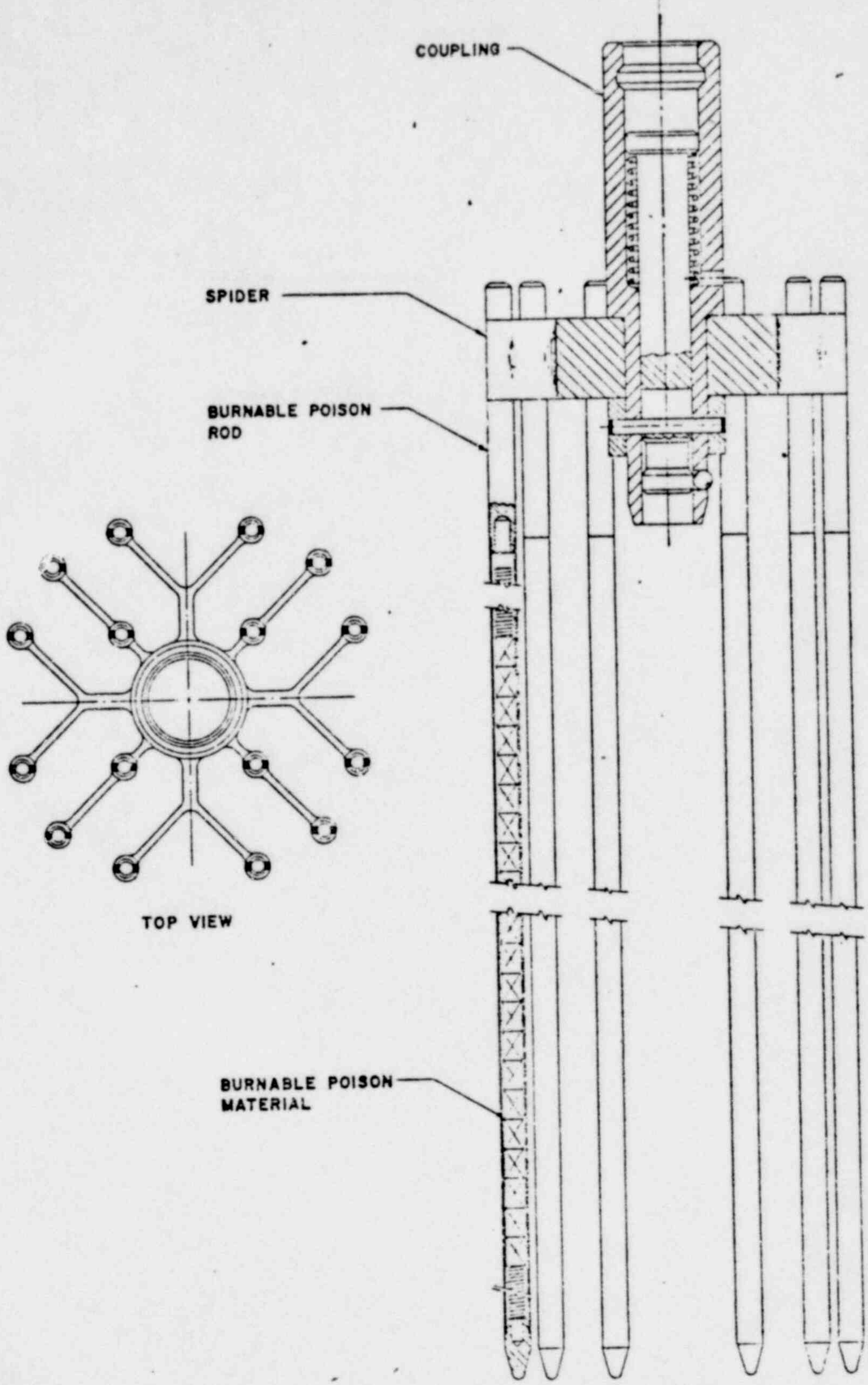


CONTROL ROD ASSEMBLY
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 3-62

A-288

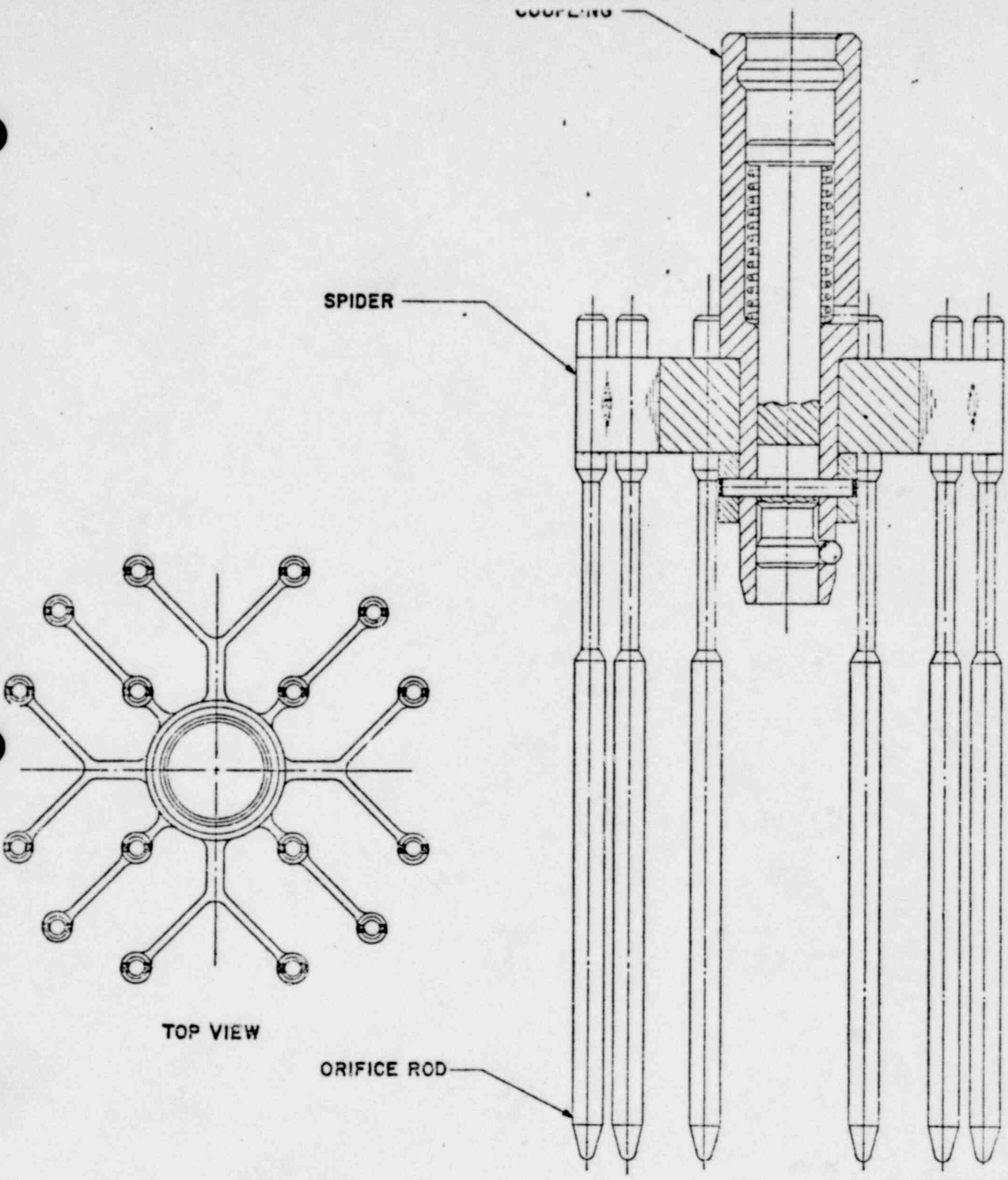


BURNABLE POISON ROD ASSEMBLY
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

FIGURE 3-60

A-285

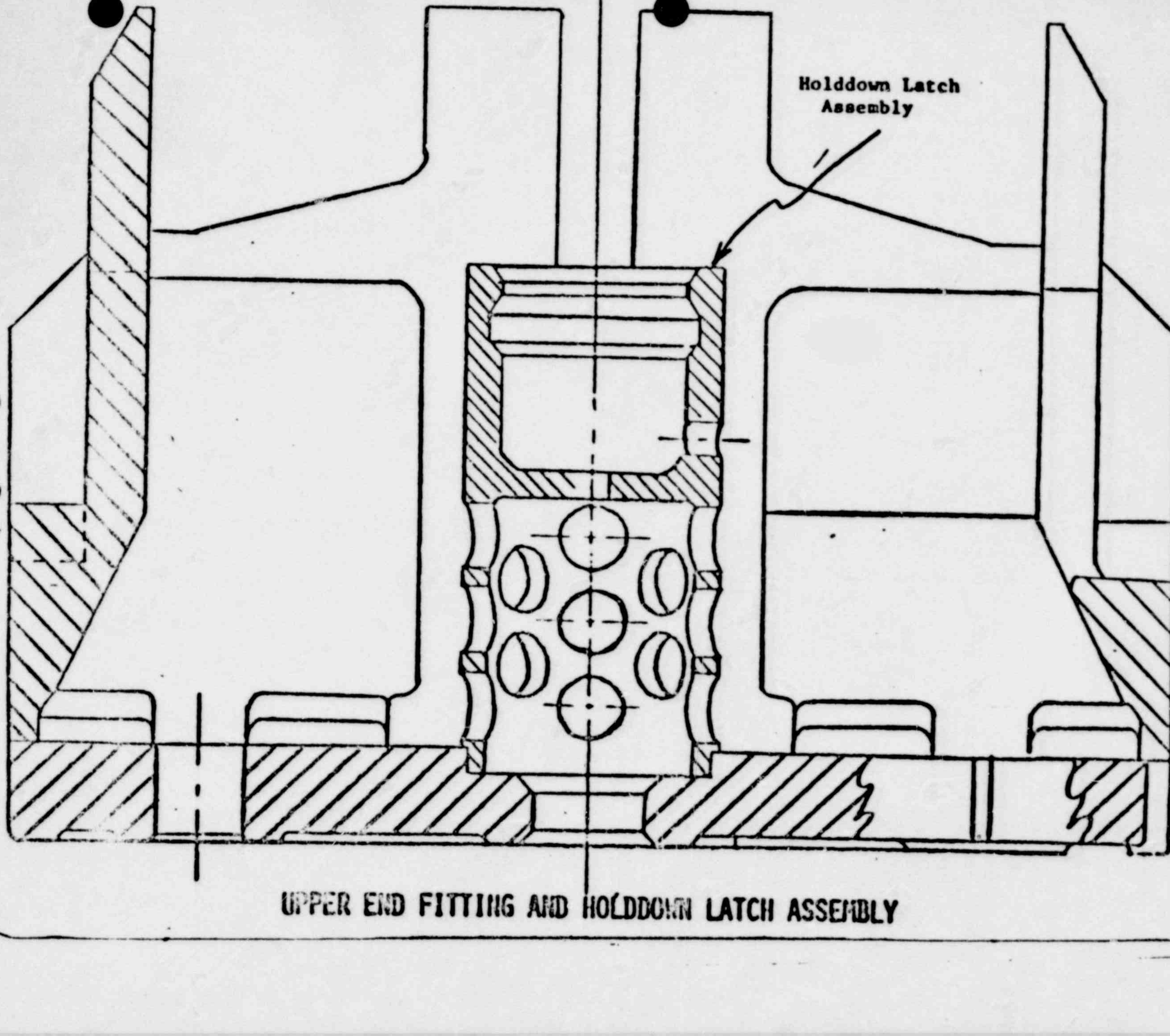


ORIFICE ROD ASSEMBL
CRYSTAL RIVER UNIT

FLORIDA POWER CORPORATION

FIGURE 3-

A-290



UPPER END FITTING AND HOLDDOWN LATCH ASSEMBLY

A-291

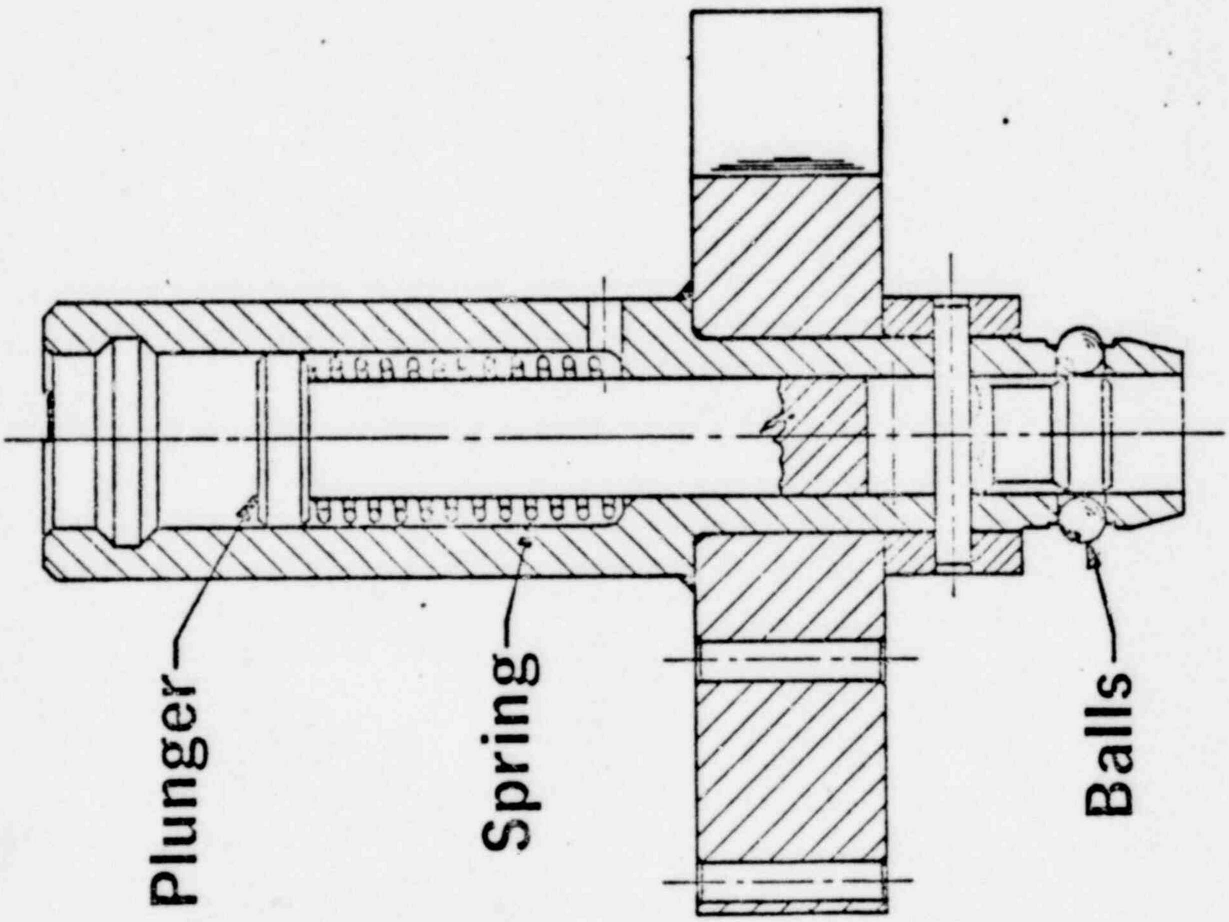
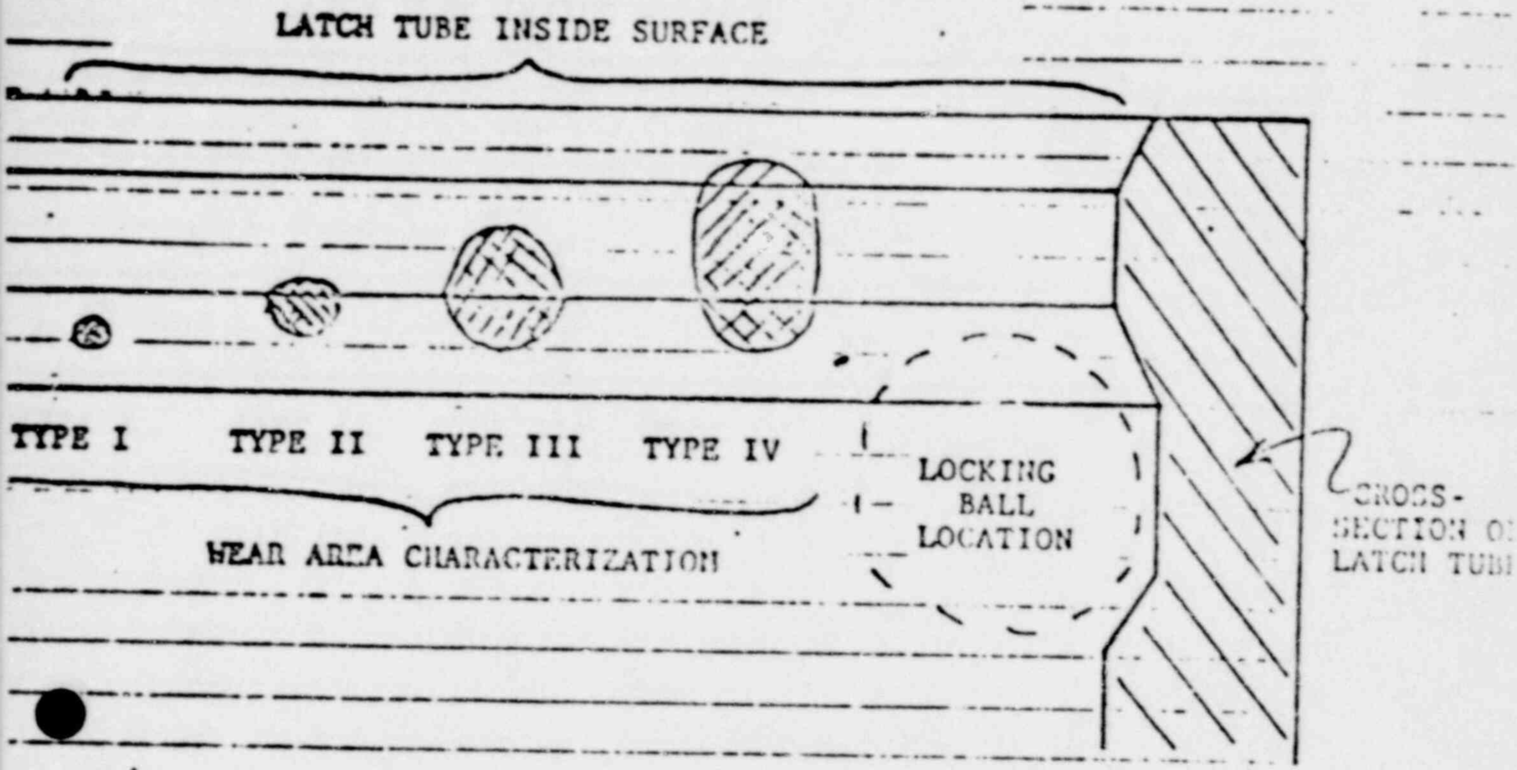

Plunger
Spring
Balls

FIGURE 1

QUALITATIVE DESCRIPTION OF LATCH TUBE WEAR



TYPE I - SMALL WEAR AREA ON LOWER CHAMFER ONLY.
TYPE II - WEAR INTO VERTICAL LAND, BUT LESS THAN HALF THE HEIGHT.
TYPE III - WEAR INTO VERTICAL LAND, GREATER THAN HALF THE HEIGHT.
TYPE IV - WEAR INTO UPPER CHAMFER.

PLANT	BPRA NO. OF LATCH TUBES EXAMINED	BPRA PERCENTAGE OF TUBES SHOWING WEAR OF TYPE I	TYPE II	TYPE III	TYPE IV
CR-3	68	7	49	34	10
OCONEE-2	25	52	44	4	0
ANO-1	36	47	53	0	0
D-B-I	26	54	13	20	13

A-293

3M

POSSIBLE WEAR CONTRIBUTORS

1. Reduced Flow; i.e. 2 and 3 pump operation
2. Mark-B4 End Fitting
   - A. Cast Holddown Latch
   - B. Reduced Pressure Drop
3. Other
   - A. Pump Pressure Pulsations
   - B. Minor Manufacturing Variables

MATERIAL PROPERTY COMPARISON
HOLDDOWN LATCH ASSEMBLIES

CORE	MATERIAL	YIELD STRENGTH PSI	HARDNESS BHN	R_B
CR-III	Bar	36,000	149	
		63,000	217	
ANO-I	Bar	37,000	156	
		63,000	217	
Oconee-II	Bar	33,500	~145	77.1
		69,000	197	
Davis Besse-I	Cast	40,700	~153	80
		48,000	~170	85
SMUD	Bar	63,000	207	
		75,000	217	
Generic Reloads	Cast	~38,000		~78
		~47,000		~78-80

A-29C

SUMMARY OF ORA LATCH TUBE INSPECTIONS

PLANT	LATCH TUBE CHARACTERISTICS BHN	YIELD, KSI	TUBES EXAMINED	REMARKS
OCONEE-1	-	73.8	7	ORA FOR 2 OR 3 CYCLES
OCONEE-1	170	34.8	1	ORA FOR 3 CYCLES
OCONEE-2	170	43.0	7	ORA FOR ONE CYCLE
ANO-1	217	63.8	8	ORA FOR ONE CYCLE
ANO-1	156	37.0	12	ORA FOR ONE CYCLE
CR-3	149-159	42-36	19	ORA FOR 270 EFPD
CR-3	170	43	2	ORA FOR 270 EFPD
CR-3	217	63.8	19	ORA FOR 270 EFPD
PB-1	153-168		9	ORA FOR 85 EFPD

Type 1 - 11%
2 - 11%
3 - 67%

SUBJECT

A-297

PRELIMINARY PUMP OPERATING HISTORY

MAY 23, 1978

REACTOR	CYCLE	NUMBER OF OPERATING DAYS 1 PUMP	2 PUMPS	3 PUMPS	4 PUMPS
DAVIS-BESSE-1	1	1	35	41	194
TMI-1	1				
	2	6	4.5	2	277
	3	1.5	1.5	2	305
	4	-	>1	>1	27
TMI-II	1	-	16	28	2
ANO-I	1				
	2	>1	3	1	
	3	>1	1.5	2	
OCONEE-I	*	19	5.5	9	
OCONEE-II	*	27	12	13	
OCONEE-III	*	5	3	18	
CR-III	1		7.5	24	
RANCHO SECO	1	6	8	6	608
	2	>1	>1	14	

* OPERATING SINCE MK-B4 END FITTING

CATALOG NO

PLANT STATUS SUMMARY

PLANT	POWER LEVEL	STATUS	NEXT REFUELING
OCONEE 1	2568	100%	SEPT-OCT
OCONEE 2	2568	100%	OCT-NOV
OCONEE 3	2568	100%	JUNE
TMI-1	2535	100%	1979
TMI-2	2772	IN STARTUP TESTING	1979
CR-3	2452	SHUTDOWN	1979
DAVIS BESSE	2772	SHUTDOWN	1979
RANCHO SECO	2772	~70%	LATE 1978
ANO-1	2568	100%	1979

OCONEE I, CYCLE 4 CORE LOADING DIAGRAM
REV. 1

FUEL TRANSFER CANAL →

X

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
					FA	FD	G5	G9	F7					
			FG OXX	GO	GH C17	1D06	1D48 C56	1D08	G4 C48	GE	FT R00010			
		GP	FY OXX	1D26 C18	6P OXX	6C C29	1D51 OXX	60 C32	6H OXX	1D21 C52	GL OXX	FS		
	G2	FM OXX	1B15 C49	72 OXX	5Q A04	1D46 OXX	5X C16	1D12 OXX	[illegible]	71 OXX	1B03 C42	FK OXX	FX	
	G1	1D45 C19	70 OXX	5Y C05	1D27 OXX	5F C23	1D03 OXX	5J C27	1D29 OXX	5W C07	5U OXX	1D20 C57	FZ	
F5	FJ C04	6Y OXX	65 A026	1D28 OXX	1D17 C01	5C OXX	1D59 C21	6K OXX	1D18 C30	1D25 OXX	6Q A06	6L OXX	GN C24	F8
G8	1D30	6A C55	1D11 OXX	5E C22	5R OXX	1D07 C26	6T OXX	1D35 C61	61 OXX	6S C25	1D44 OXX	67 C46	1D52	FC
G7	1D56 C41	1D09 OXX	5B C14	1D01 OXX	1D61 C02	6X OXX	1D05 C12	5Z OXX	1D58 C38	1D02 OXX	6U C10	1D50 OXX	1D60 C50	GQ
[illegible]	1D22	5G C51	1D15 OXX	6M C53	66 OXX	1D19 C13	5H OXX	1D32 C15	5T OXX	6G C06	1D14 OXX	69 C54	1D42	GB
F9	FP C44	6R OXX	5K A07	1D16 OXX	1D36 C03	5S OXX	1D57 C47	62 OXX	1D37 C39	1D23 OXX	64 A02	6V OXX	GJ C20	F3
	GF	1D43 C60	5V OXX	6F C43	1D53 OXX	63 C58	1D04 OXX	6N C59	1D24 OXX	68 C28	5N OXX	1D13 C40	GG	
	G3	GK OXX	1B01 C36	6E OXX	5D A08	1D10 OXX	6W C06	1D49 OXX	6B A01	5P OXX	1B07 C37	FN OXX	FW	
		GC	FL OXX	1D54 C45	6Z OXX	6D C35	1D38 OXX	5L C34	6J OXX	1D55 C11	GM OXX	FU		
			FF R00009	FV	FQ C33	1D41	1D47 C31	1D31	FR C09	GD	FH OXX			
					F6	GA	G6	FE	F4					

Z

Fuel I.D. (1DXX - Batch 4; 5X, 6X or 7X - Batch 5; FX or GX - Batch 6; 1BXX - Bat[illegible]
Orifice (OXX) or Control Rod ID (CXX - CRA; AXX - APSRA)
Source I.D. (R000XX - Secondary)

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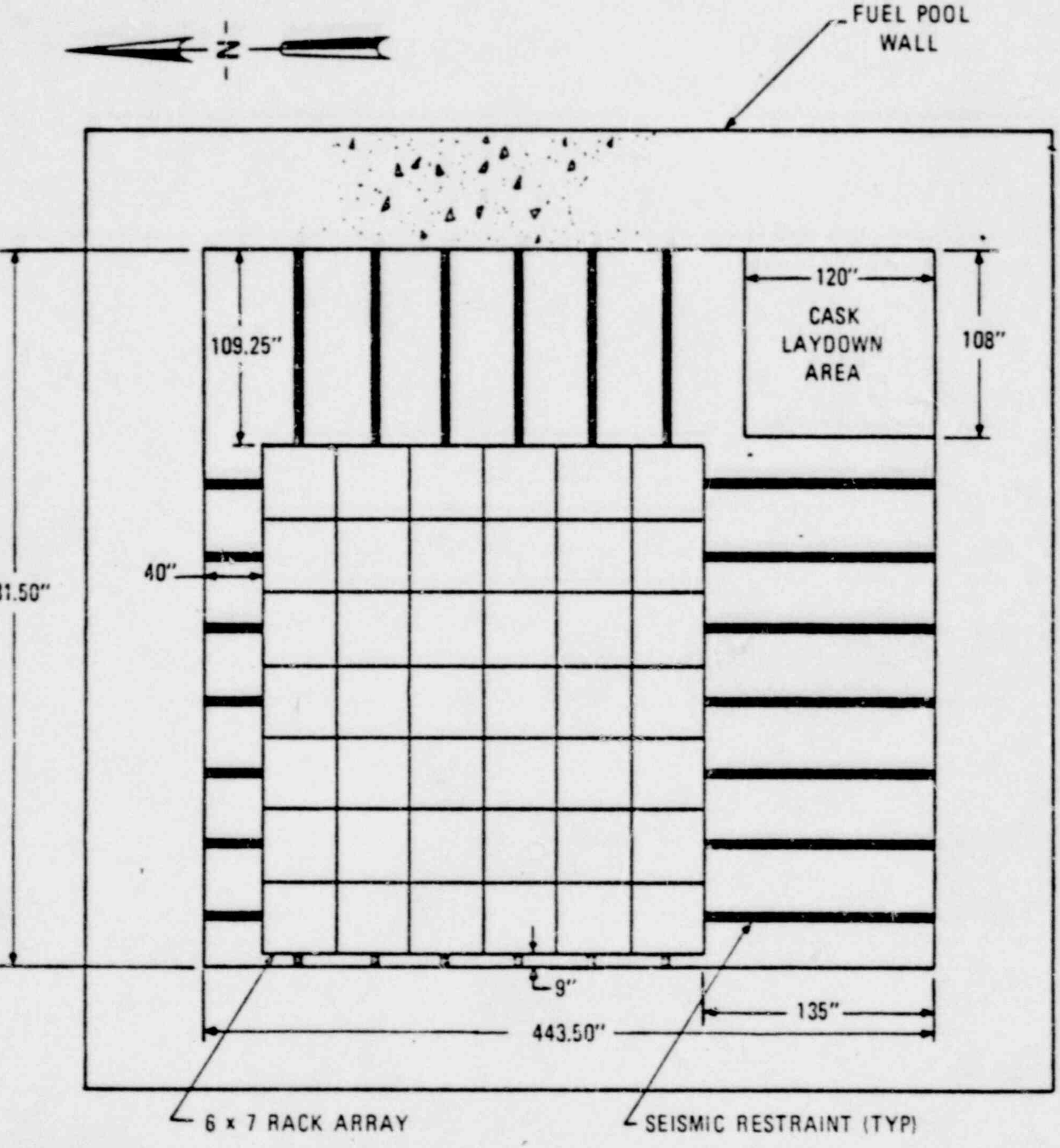
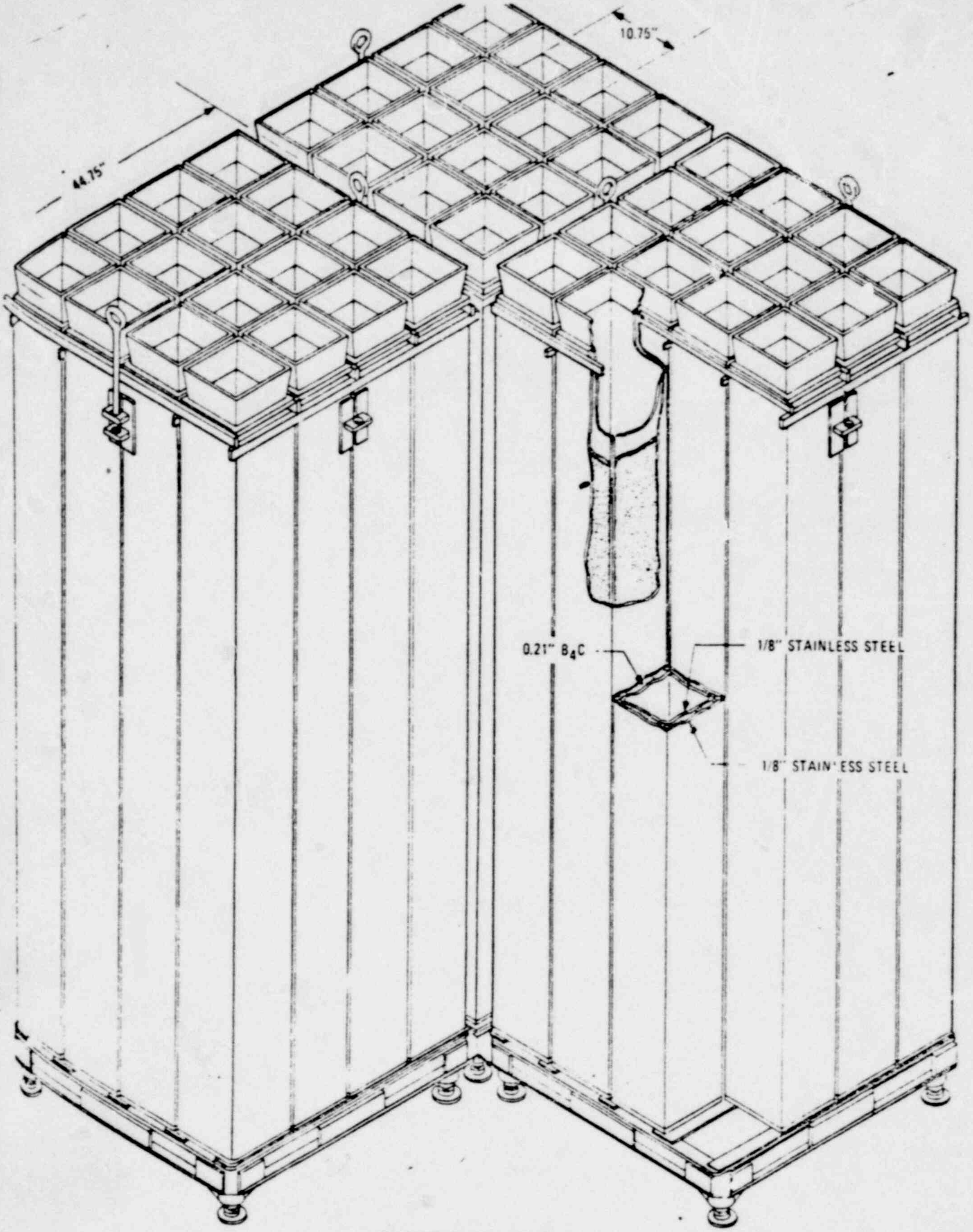
NOTE: NJ00 prefixes Batches 5 and 6 F.A. I.D.'s

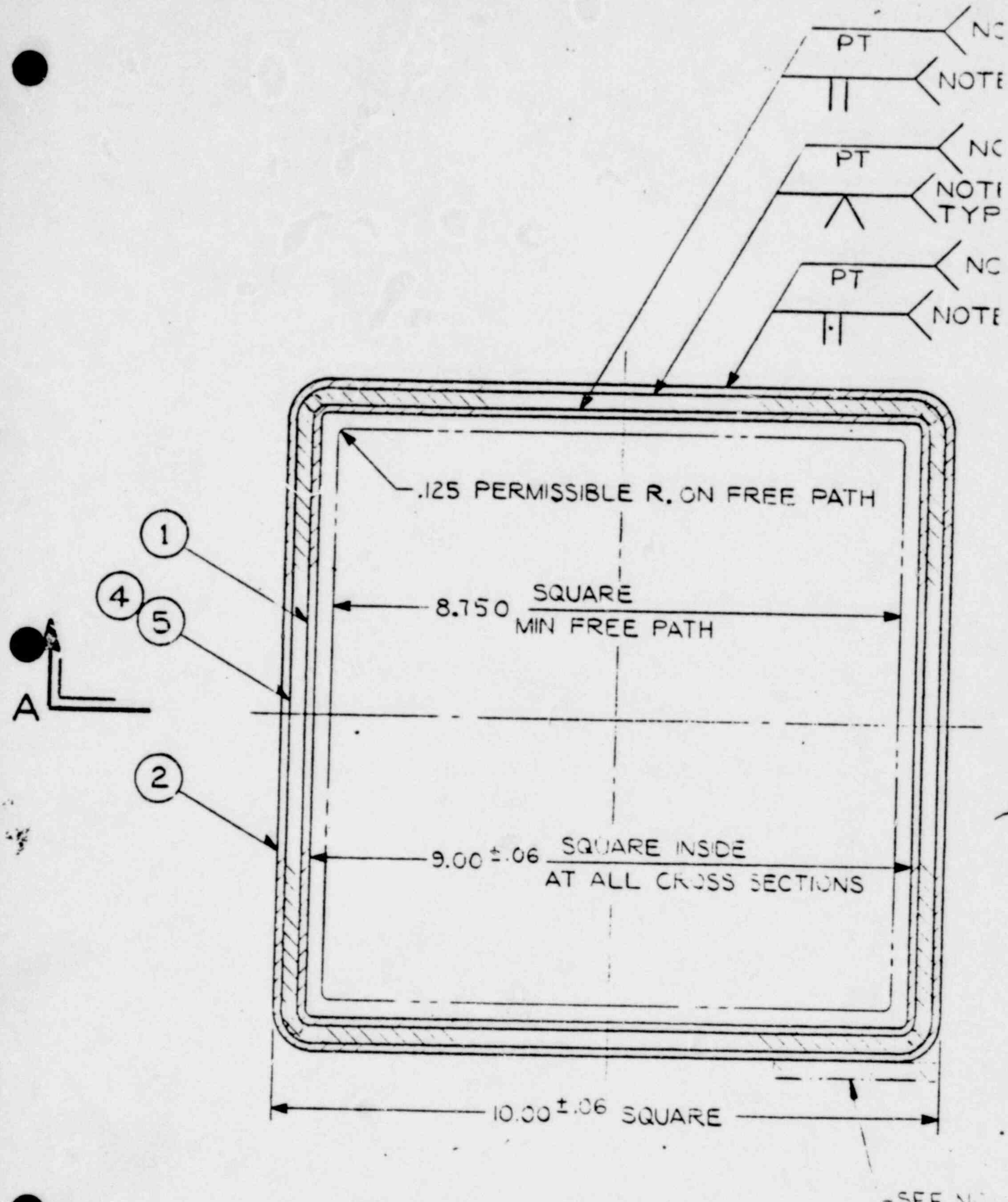
Appendix XXII



CONN. YANKEE SPENT FUEL POOL
INTERMEDIATE MODIFICATION ARRANGEMENT
FIGURE 3-2



SPENT FUEL RACKS
FIGURE 3-1



SECTION B-B

SCALE: 3" = 1'-0"

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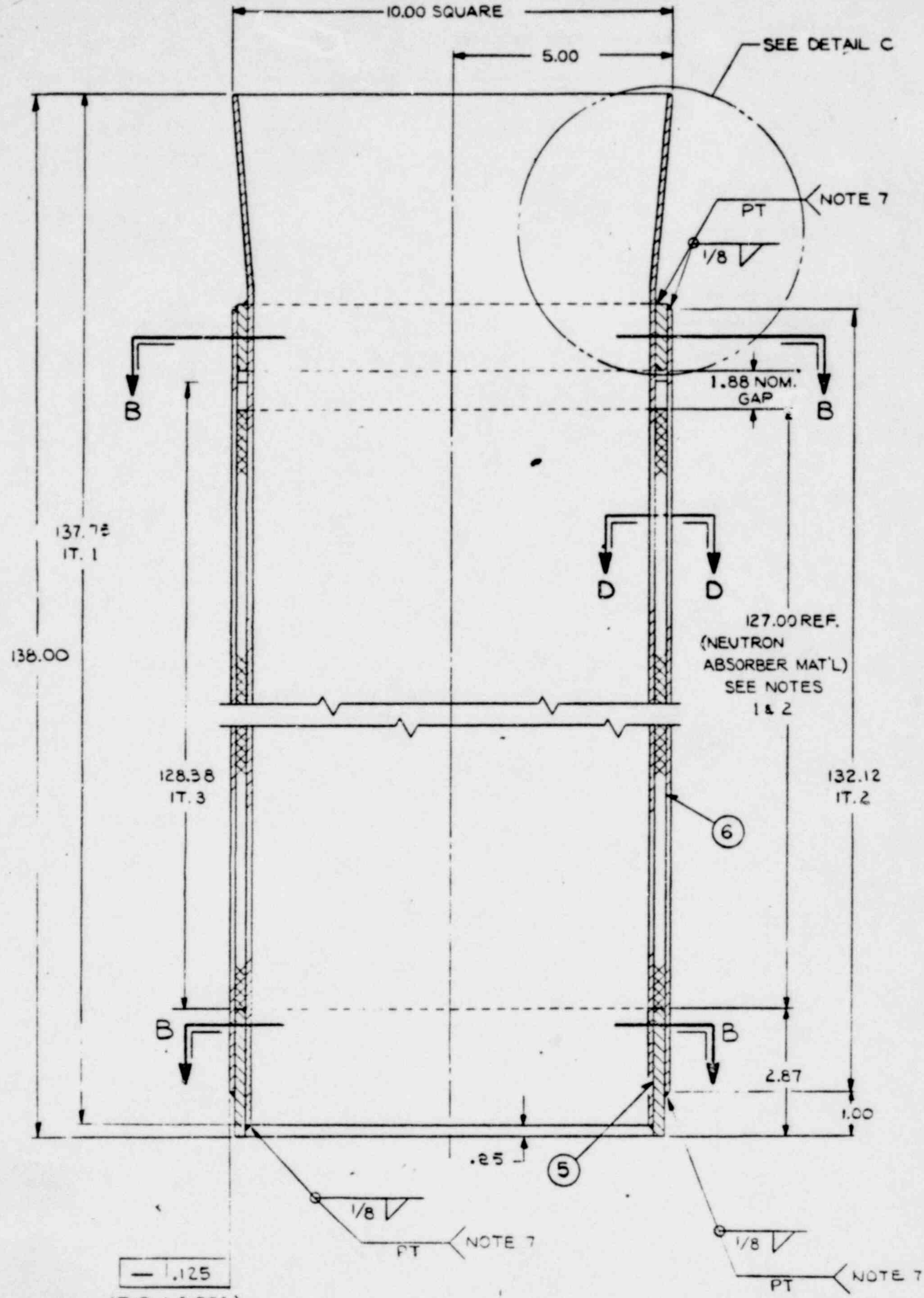

SECTION A--A

SCALE: 6" = 1'-0"

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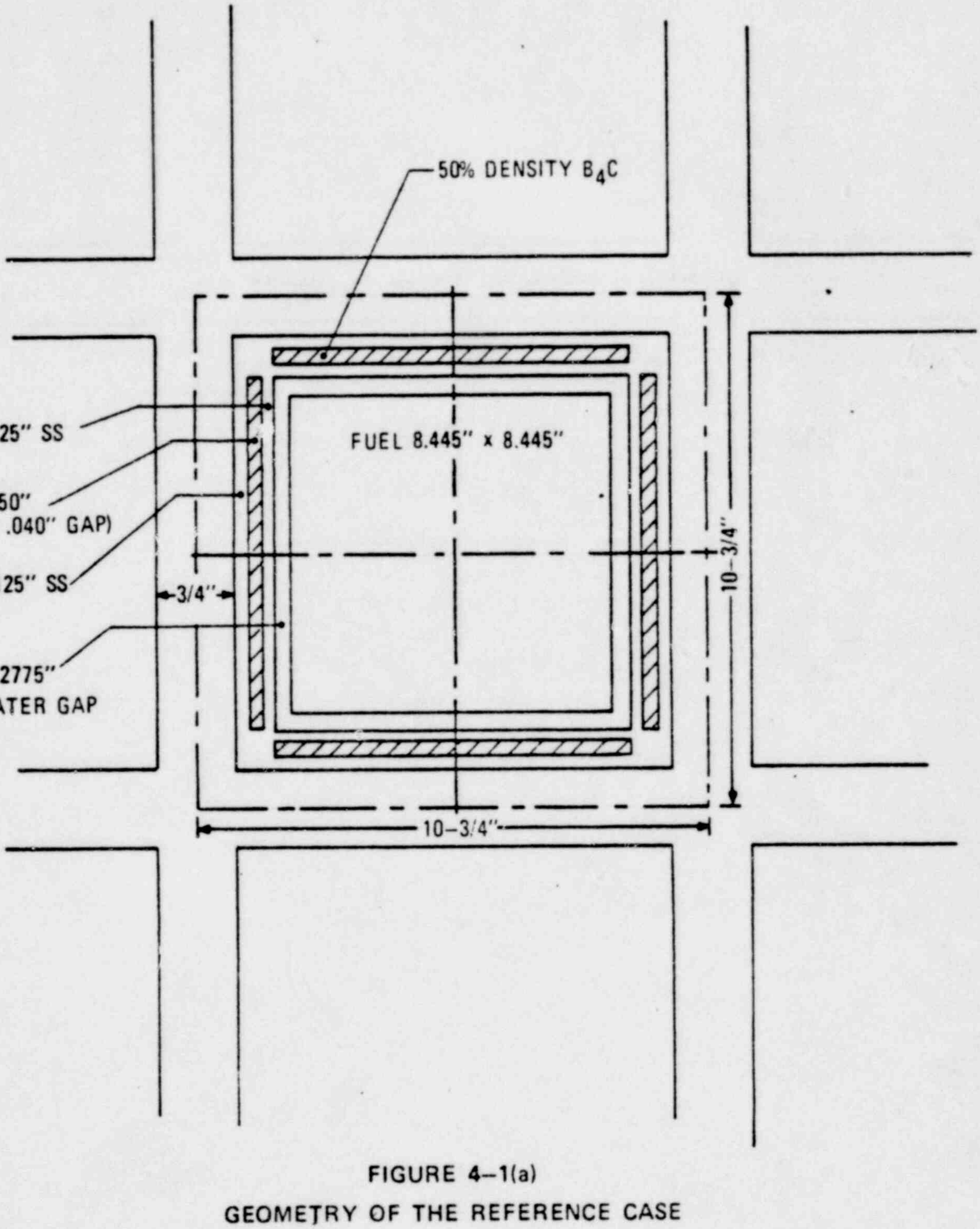

FIGURE 4–1(a)
GEOMETRY OF THE REFERENCE CASE

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Appendix XXIII

ATWS WORKING GROUP MEETING
MAY 26, 1978
WASHINGTON, D.C

APPENDIX XXIII
Highlights of ATWS Subcommittee Meeting,
May 26, 1978

- MEETING HIGHLIGHTS -

The ATWS Working Group met on May 26, 1978, to continue discussion with the NRC Staff on their position on ATWS as stated in NUREG-0460: "Anticipated Transients Without Scram for Light Water Reactors." ACRS members present included Messrs. Bender, Etherington, Kerr, and Mark. Consultants in attendance included Messrs. Bennett, Ditto, Epler, Lee, Lipinski and Saunders. The Meeting discussion involved NRC response to the attached list of 20 questions. Highlights of the meeting included:

1. In response to the question of how the ATWS fix would be changed if the ATWS safety goal were varied from 10^{-5} to 10^{-7}, NRC noted the following: (1) if the safety goal is 10^{-5} per reactor year, the Westinghouse fix would be identical to the one now proposed. CE and B&W would probably need only 1-3 safety relief valves instead of 3-5 the relief valves now proposed. GE would still require recirculation pump trip and the fast auto-boron injection system. It has not known if the high pressure makeup system could met the 10^{-5} safety goal in its present form. NRC noted that in general, mitigating system reliability would be on the order of 10^{-2}. If the safety goal was 10^{-7} per reactor year, Westinghouse would require 1-2 relief valves, CE and B&W would require 4-6 relief valves, and GE would require an additional high pressure makeup

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system, in addition to the modifications noted above. In general, mitigating system reliability would have to be on the order of 10^{-4}. Dr. Lipinski commented that he believes NRC must demonstrate a 10^{-3} reliability for the mitigating system, even if it is specified Safety Grade.

2. NRC was asked to discuss the accuracy with which measurements of moderator temperature coefficient at full power are known, and whether or not they have investigated the possibility of making the moderator temperature coefficients more negative. NRC stated that the uncertainty involved in the measurement of the moderator temperature coefficient (MTC) is on the order of 10%. Dr. Kerr was not aware that the MTC could be determined to such accuracy. It was also noted that the difference between the 95% and 99% MTC value (moderator temperature coefficient will be no less negative for either 5% or 1% of operation), results in a difference in peak pressure of about 100 pounds for W and CE reactors, about 400 psi for B&W reactors. NRC also noted that there is no Staff requirement to make the MTC more negative, and cited a vendor topical that suggests a fuel burnup penalty may result from attempting to do so. There was extensive discussion regarding whether or not the uncertainty in the measurement of MTC would be greater than the MTC quantity being measured. The Staff feels that while the uncertainty in this parameter is difficult to assess, it does effect the peak pressure calculations and must be addressed.

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3. In response to the question "to what extent does ATWS contribute to the probability of a LOCA, assuming that Staff fixes are implemented," NRC said they have not really looked at this question and can not quantify the answer at this time. NRC is more concerned with the functioning of equipment necessary for long term cooldown after the peak pressure has occurred.

4. The Staff discussed the alternatives to waiting until after rulemaking is completed before ATWS fixes are applied. NRC believes that the rulemaking procedure is preferable for the principal reason that rulemaking would be binding on both the NRC and the vendors. This would avoid the possibility of extensive litigation if a non-rulemaking path was followed. NRC believes the time required for implementation of the ATWS fix would ultimately be shorter, if litigation is avoided.

5. There was discussion of the Staff's statement "The Staff believes that common mode failures are likely to dominate reactor protection system unreliability, and the Staff's estimates do not weigh heavily the results of synthesis calculations". Dr. Kerr felt the synthesis calculations should be used for an estimate of RPS unreliability, since there is an insufficient amount of data available to calculate an unreliability figure. NRC countered that engineering judgement is used in lieu of data, and they do not believe synthesis models

are adequate for common mode failure situations. Dr. Saunders cited one mathematical model that he believed could be used for the common mode failure situation. Dr. Hanauer replied that the Staff did look at this model, but does not believe it could be relied upon to provide adequate results.

6. In response to a question regarding the difference between the US and the Federal Republic of Germany BWR ATWS mitigation systems, NRC noted that the FRG relies upon a dual shutdown system in conjunction with reduction in the speed of the recirculation pumps. The NRC also stated that the recently proposed GE 10-second-auto-boron injection system has the potential of meeting the NRC ATWS acceptance criteria.

7. A Working Group question requested a discussion of the appropriateness of using Part 100 dose guidelines for ATWS calcuations, in light of the extremely conservative source term used in LOCA calculations. NRC responded that it is their belief that an ATWS event has a very low probability, therefore the use of Part 100 dose guidelines is appropriate for this event. Dr. Kerr was of the opinion that the Part 100 dose term is nonmechanistic and has little relationship to physical reality. He found it difficult to apply this nonmechanistic source term to a mechanistic phenomena such as ATWS. The Staff replied that while they believed that the

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ATWS source term will be less than the nonmechanistic Part 100 source term, such an accident has a very low probability, and Part 100 dose guidelines are appropriate.

8. There was a series of Working Group questions regarding the uncertainties involved in the calculation of peak pressures seen during an ATWS. The following points were noted: (1) NRC performed a set of calculations for the peak pressure using the input from the three PWR vendors. Comparing the NRC results with the vendor calculated results gave a difference of between 50 and 200 psi. The Staff believes that they can't quantify the uncertainties in the peak pressure calculations, but they believe that the calculations provide a somewhat conservative prediction of the peak pressures expected during ATWS; (2) NRC will require the vendors to perform confirmatory tests on their ATWS evaluation models; the Staff is also attempting to collect information from tests being conducted overseas on such items as relief valves; (3) NRC analyzed the effect of equipment failure on the peak pressure seen during ATWS. The results indicate that equipment failure only accounts for a small amount of uncertainty in the determination of peak pressure.

9. The ATWS Working Group has scheduled two additional meetings on July 13 and August 1, 1978, to meet with Industry representatives and obtain their input on the ATWS issue. The Working

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Group also recommended that the NRC come before the full Committee in August to give a presentation on their ATWS position, with an Industry presentation to follow at the September 1978 meeting.

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ATWS Questions for Staff

1. Discuss changes that might occur in the fix or the method of approach if the ATWS goal were varied over the range 10^{-5} to 10^{-7} per RY.

2. Discuss appropriateness of using Part 100 dose guidelines for ATWS calculations in light of the extremely conservative source term used for LOCA calculations.

3. Discuss the accuracy with which measurements of moderator temperature coefficients at full power are known. Has there been any investigation of the possibility of making moderator temperature coefficients more negative?

4. How much does ATWS contribute to the probability of a LOCA, assuming staff fixes are implemented.

5. What is the accuracy with which calculations of transient peak pressures can be calculated? Are there significant unresolved discrepancies between vendor and staff calculations?

6. Can the conservatism in the staff's proposed fixes be estimated quantitatively?

7. Discuss alternatives to waiting until after rulemaking is completed before ATWS fixes can be applied. Has the Staff given thought to shorter term corrective measures that could be applied before rulemaking is concluded?

8. Although the Staff has accepted the probability of core melt as calculated in the Reactor Safety Study as an appropriate goal, the calculated consequences associated with this core melt probability were based on a specific containment. The consequences for different containments might be markedly different. How does the Staff propose to account for this possible difference?

9. What fraction of the anticipated transients are expected to be accompanied by lost of offsite power? Has this been considered in arriving at the probability of an ATWS event? Discuss further Comment 1.4 in Appendix XIII of NUREG-0460 concerning the increased probability of a LOCA with ECCS failure due to the increased number of relief valves suggested as an ATWS fix on two of the PWR types.

10. Please provide additional discussion of the statement on page 23, "The staff believes that common mode failures are likely to dominate reactor protection system unreliability, and the staff's estimates do not weigh heavily the results of synthesis calculations."

11. Please provide additional discussion of the statement on page 24, "Common mode failures are believed to be the most likely cause of multiple failures of rods."

12. On page 27 the following statement appears: "While the data do not exclude unreliabilities of the mechanical portion of the scram system in the order of 10^{-7}, the data are also consistent with much higher failure probabilities in the 10^{-4} to 10^{-5} range." What prevents this statement from being made about the fixed-up system?

13. On page 28 the following statement appears: "In assessing the additional requirements that might be necessary in order to meet the staff safety objective for ATWS events we have used a value of 3×10^{-5} per demand for this probability, which includes some allowance for the improvement in future reactor protection systems compared with the systems used to derive the estimate." Please provide some additional information on how this value was achieved.

14. On page 28 the following statement appears: "The staff believes that its current estimate of unreliability is appropriate for the electrical portion of the scram system, but recognizes that the lack of observed control rod or drive failures may make the estimate less applicable to the mechanical portion of the scram system." Is the difference that the staff observes between the electrical portion and the mechanical portion due to the fact that the staff interprets one failure as having occurred in the electrical system but no failures in the mechanical system? If this is the case, how does this difference of one failure make so large a difference in the staff's evaluation of performance of the two systems.

15. As a corollary to 5 above; if the shutdown system is thought to consist of the following: (1) trip signal, (2) electronic circuitry, and (3) mechanical components, what is the contribution of each to the overall system reliability? Is there a need for reliability improvement in any of the three areas noted?

16. Since incomplete insertion of control rods does not cause significant shutdown reactivity losses, could they be omitted from the scram system reliability assessment?

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17. What has the BNL ATWS study taught us regarding the uncertainty in the peak pressure calculation for PWRs? How do these uncertainties compare with the uncertainties inherent in the vendor calculations of peak pressures?

18. Are there actions that can be taken to remove uncertainties in the pressure calculations, especially during preoperational testing of a plant?

19. What time-response considerations are involved in the investigation of the ATWS transient if the recirculation pump trip is included in the BWR fix? Are the U.S. systems different from the FRG systems in the time-response needed for secondary shutdown in addition to recirculation pump trip following ATWS?

20. In the event of a turbine trip, would the provision for bypassing of steam flow to the condenser mitigate the demand on ATWS Plant Protection for BWRs?

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Appendix XXIV

RECOMMENDED IMPROVEMENTS

INCREASED PARTICIPATION IN LICENSING ACTIVITIES

CURRENT ACTIVITIES

- RETRAN Development Effort
- Fuel Behavior Modeling
- SIMULATE Improvements

POSSIBLE FUTURE ACTIVITIES

- Reactor Kinetics Model Development
- RELAP4 Model Development

BENEFITS

- Faster Response to NRC concerns
- Results more appropriate to VY
- Lower costs

BCS/1
5/19/78

- NRC REQUIRED ACTIONS

  - PROVIDE EARLY ACCESS TO INFORMALLY DISCUSS REQUIREMENTS, POSSIBLE SCHEDULES
  - PROVIDE NECESSARY PRIORITY TO REVIEW VY SAFETY ANALYSES
  - PROVIDE FOR VY PARTICIPATION IN NRC-VENDOR GENERIC MEETINGS
  - PROVIDE ACCESS WHEN APPROPRIATE TO VENDOR PROPRIETARY INFORMATION
  - PROVIDE VY ACCESS TO GENERIC BWR ISSUES AS THEY EMERGE FROM WORK AT NRC-FUNDED LABORATORIES

BCS/2
5/19/78

Appendix XXV

INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION

Stockholm, Sweden - May 22-27, 1978

D. W. Moeller

I. Introduction

This was the 50th Anniversary Meeting of the International Commission on Radiological Protection (ICRP) which was organized in 1928 by the Second International Congress of Radiology. Initially called the International X-ray and Radium Protection Committee, the ICRP assumed its present name in 1950, in order to cover more effectively the rapidly expanding field of radiation protection.

The main Commission consists of a Chairman plus twelve members, each elected for a 4-year term. Supporting the Commission are four Committees with responsibilities as follows.

1. Committee 1 on Radiation Effects

   This Committee has been assigned responsibility for assessing the risk of stochastic (non-threshold) effects and the induction rates of non-stochastic (threshold) effects of ionizing radiation. Included in its deliberations are the modifying influence of exposure parameters such as dose rate, dose fractionation, RBE, the spatial distribution of dose and any synergistic effects of chemical and physical factors.

2. Committee 2 on Secondary Limits

   The basic function of Committee 2 is to develop values of secondary limits based as the dose-equivalent limits recommended by the Commission. The Committee currently is devoting its entire attention to secondary limits for internally deposited radionuclides.

3. Committee 3 on Protection in Medicine

   This Committee has been established to enable the Commission to meet its responsibilities to the International Congress of

Radiology and to the medical profession. Matters currently being addressed include protection of the patient in radiodiagnosis and radiotherapy and protection in nuclear medicine. Committee 3 is also developing secondary standards for external radiation.

4. Committee 4 on the Application of the Commission's Recommendations

   Committee 4 provides advice on the Commission's system of dose limitation and on protection of the worker and the public. The Committee also serves as a major point of contact for the ICRP with international organizations concerned with radiation protection.

Joint meetings of the Commission and its Committees are held every two years and the Commission meets independently on an annual basis. In addition, the four Committees meet independently on alternate years when they are not meeting with the Commission.

The number of members of the Commission and its Committees totals 63 and they represent 18 different countries. The U.S. has 19 members; the UK has 10; France, 7; Federal Republic of Germany, 6; Sweden, 5; Japan, 4; U.S.S.R., 2; and one representative each from Argentina, Austria, Canada, Czechoslovakia, Israel, Italy, Netherlands, Norway, Poland, and South Africa.

II. Basic Philosophy

1. Recommendations of the ICRP are based on the assumption that, in the range of current occupational dose limits, there is a linear relationship between the stochastic (non-threshold) effects of radiation and the total dose. There are two significant implications associated with this assumption:

   a. If the risk is proportional to total dose, then the dose rate and any fractionation thereof need not be taken into account. There is no rationale for limiting the rate of dose accumulation within a given time span.

b. If the average dose is a measure of risk, then inequalities regarding the distribution of dose within a given tissue need not be taken into consideration.

2. The ICRP believes that the use of a linear extrapolation (based on the frequency of effects observed at higher doses) may suffice to provide an upper limit of risk at lower doses. It is acknowledged, however, that this approach may be conservative by a factor of 2 to 5 (most common estimate) to perhaps as much as 100 (upper quoted estimate). As a result, it is important to recognize that the assumption of linearity may lead to an overestimation of radiation risks which, in turn, could lead to the choice of alternatives that are more hazardous (if the alternatives have been evaluated on the basis of less conservative methods). For this reason, the ICRP recommends that realistic, not conservative, approaches be used for optimization in the selection of a choice among several alternatives. Although the Commission recognizes that conservatisms should be applied in setting dose limits, it does not believe that conservatisms should be used in evaluating the risk assessments that underlie the limits.

3. The main features of the ICRP recommendations for dose limitation are as follows:

   a. No practice shall be adopted unless its introduction produces a positive net benefit.

   b. All exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

   c. The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances. However, the degree of justification needed for any practice, and the point at which exposures can be said to be ALARA, depend on the number of exposed individuals and the dose distribution within the exposed group.

d. Dose commitments associated with current operations should be carefully considered so that allowance can be made for future expansions in nuclear activities without undue exposures to any members of the public.

III. Basic Recommendations

1. The latest recommendations of the ICRP on radiological protection have two basic goals. These are to:

   a. Prevent non-stochastic effects (where severity is a function of dose). Examples of non-stochastic effects are production of cataracts, erythema of the skin, and acute death.

   b. Limit stochastic effects to acceptable levels (where the probability of harm is a function of dose). Examples of stochastic effects are chromosomal aberrations, mutations in spermatagonia, ovarian tumors, and cancer production.

2. Dose Equivalent Limits have been set on the principle that the risk should be equal whether the whole body is irradiated uniformly or non-uniformly.

   a. Thus the standard limits are based on the total risk to all tissues (organs).

   b. They are related to the committed dose equivalent resulting from one year of practice.

   c. They include for individuals the hereditary detriment in the immediate offspring (1st two generations).

   d. They are to be regarded as upper limits, not the acceptable dose.

   e. They are not to be regarded as the dividing line between safe and unsafe conditions--that is, the ALARA principle should be applied at all times.

3. The units for expressing the dose from ionizing radiation are:

   a. The Gray

      1 Gy = 1 J/kg = 100 rad

   b. The Sievert

      1 Sv = 1 J/kg = 100 rem

      1 mSv = 0.1 rem = 100 mrem

4. The goal of the ICRP for protecting workers in radiation environments is to keep their risks comparable to those in "safe" industries. The Commission expresses this quantitatively as seeking a goal of risk of mortality of no more than 10^{-4}/year.

   a. On this basis, the following whole body dose equivalent limits have been recommended:

      Non-stochastic -- 0.5 Sv/y (50 rem/y);
      Stochastic -- 50 mSv/y (5 rem/y).

   b. The listed dose equivalent rate limit of 50 mSv (5 rem) per year for whole body occupational exposure is estimated to have an associated risk of about 10^{-4}/y. The ICRP assumes, however, that with this limit the actual dose equivalent rates received by workers will average about 5 mSv/y (0.5 rem/y). This reduced dose equivalent rate is assumed to carry an associated risk of mortality of about 10^{-4}/y.

5. Based on biological studies with animals and humans, estimates can be made of the risk of cancer and/or genetic effects for given levels of dose to specific body organs. For exposures to single body organs (such as will occur due to internally deposited radionuclides), the ICRP has recommended dose limitations on the principle that the risk should be equal whether the whole body, or only a portion thereof, is irradiated. Listed in Table 1 are the assumed risks for irradiation of single body organs or tissues on an individual basis and the resulting dose equivalent limit to that body portion, assuming the risk should not be greater than that associated with a dose equivalent rate of 5 rem per year to the whole body.

Table 1

Risk of Exposure of Single Body Organs and
Associated Dose Equivalent Rate Limits

Body Organ or Tissue	Effect Considered	Risk (per 100 rem)	Weighting Factor*	Dose Limit (rem/y)
Gonads	Genetic**	4×10^{-3}	0.25	20
Breast	Cancer	2.5×10^{-3}	0.15	30
Red bone marrow	Leukemia	2×10^{-3}	0.12	40
Lung	Cancer	2×10^{-3}	0.12	40
Thyroid	Cancer	5×10^{-4}	0.03	170***
Bone Surfaces	Cancer	5×10^{-4}	0.03	170***
Remaining Organs	Cancer	5×10^{-3}	0.30	17
TOTAL		16.5×10^{-3}	1.00	

*The weighting factor represents the proportion of the stochastic risk resulting from irradiation of the given tissue or organ compared to the total risk when the whole body is irradiated uniformly.

**Serious hereditary ill health within the first two generations. The total effect for all succeeding generations is estimated to be about twice this amount.

***For these two cases, the non-stochastic limit of 50 rem/y will govern. This carries with it the implication that for radionuclides, such as plutonium and strontium (which cause irradiation of the bone surfaces), it is the non-stochastic (non-cancer) effects that govern.

6. Dose Equivalent Rate Limits for specific groups within the population include the following:

   a. Individual members of the public--5 mSv/y for individuals within the critical group. Following this approach, the ICRP estimates that:

      (1) the average lifetime dose equivalent rate to individual members of the public will not exceed 1 mSv/y;

      (2) the average dose equivalent rate to the population will not exceed 0.5 mSv/y.

   b. Population groups--there is no specific limit. Their limit is the summation of the minimum that is necessary. If the doses are necessary, then the sum of all contributors becomes the population dose limit. The 5 rem/30 years (old National Academy of Sciences recommendation) has been discarded.

   c. Women of reproductive capacity--50 mSv/y at a uniform rate. On this basis, it is unlikely that any embryo could receive more than 5 mSv during the first two months of pregnancy.

      Pregnant women--15 mSv/y (following the first two months).

7. For emergency situations, the recommended limits (where you can plan ahead or have control of the situation) are as follows: (Note, however, that these dose limits are not for life saving exercises, nor are they to be applied to women):

   Single Event--100 mSv (10 rem);
   Lifetime Limit--250 mSv (25 rem).

8. Medical Exposures--are subject to justification and optimization--but there are no specific dose limits.

9. Natural background--no limit on normal radiation from this source; however, technologically enhanced natural background may be subject to limits.

IV. Assessment of Internal Exposures

In what represents a major change, the ICRP has established a new approach for limiting occupational doses from internally deposited radionuclides.

1. The ALI

The basis for this approach is the designation of an Annual Limit of Intake (ALI) which is the quantity of a given radionuclide which, if ingested or inhaled in a single event, will result in an uptake that will yield a committed dose equivalent over the subsequent 50-year period equal to the annual dose limit.

For a given radionuclide, there will be one range of values of the ALI for ingestion and one for inhalation, depending on whether the material is in soluble or insoluble form.

If a radionuclide causes exposure of the total body, the applicable dose limit for the ALI is 0.05 Sv (5 rem). If it causes exposure predominantly to a single organ, the dose limit is as shown in Table 1.

2. The DAC

Using the ALI, it is possible to calculate Derived Air Concentrations (DACs) for purposes of limiting airborne intakes of ratioactive materials via this avenue of exposure. The DAC for occupational exposure to any radionuclide is that concentration in air which, if breathed by Reference Man for 2000 hours of work per year, will result in the ALI for inhalation.

$$DAC = \frac{ALI}{(2000 \text{ hr/y})\ (60 \text{ min/hr})\ (20{,}000 \text{ cm}^3\text{/min})}$$

where 20,000 cm^3/min equals the breathing rate of Reference Man and the ALI for inhalation is expressed in Becquerels/y where 1 Bq = 1 disintegration/sec.

The ICRP plans to publish in about two years complete listings of ALIs and DACs for about 200 isotopes of some 50 elements. This report is expected to have a major impact on evaluation procedures for radionuclides subject to ingestion and inhalation.

3. Comparison of New and Old Approaches

Formerly, limitations for the intake of radionuclides were expressed in terms of Maximum Permissible Concentrations (MPCs) for air or water. For radionuclides with short and intermediate effective half lives, continuous intake at the MPC for a period of one year (assuming equilibrium conditions) resulted in an annual dose equivalent rate at the maximum permissible level (5 rem/y). This contrasts to the new ALI which is calculated on the basis that the ALI will yield a committed 50 year dose equivalent equal to the one year dose limit. Although calculated on a different basis, it is not anticipated that DACs will be significantly different from the current MPCs for many of the radionuclides. Where differences do exist, they are frequently due more to improvements in the basic supporting data on the biological behavior of specific radionuclides within the human body, than to changes in the dose limits or calculational approach. In the main, the dose limits given in Table 1 are not that different from those currently being applied.

4. Monitoring of Internal Exposures

The quantity of a given radionuclide inside the body can be estimated through whole body counting (for gamma emitters) or through analyses of excreta and other biological specimens. Specific examples of such specimens include urine, feces, exhaled breath, nasal discharges, sputum, saliva, sweat, blood and hair. Of these, first in importance is urine; second is feces. The other materials are generally analyzed only in special cases.

In a report, to be issued in about two years, the ICRP plans to make recommendations for appropriate sampling and bioassay techniques for a variety of radionuclides. These recommendations will include data on the range of biological variability to anticipate in such assay procedures as well as guidance on the times (subsequent to intake) for optimum collection of samples.

V. Assessment of Environmental Releases

The ICRP has approved the publication of a report on principles and methods for use in assessing environmental releases. The report is designed for application prior to operation of a facility and outlines mathematical models that can be used to assess the predicted radiation doses to the neighboring population. Noteworthy items contained in the report, or expressed philosophically, include:

1. Because of large seasonal variations in radionuclide transfer factors within the environment, values applicable to chronic long-term releases may not be applicable on an acute short-term basis. Even for routine releases, if the rate of discharge varies substantially, it will sometimes be important to consider the combined effects of transient high discharge rates, and unfavorable environmental situations.

2. Basically, recommendations provided by the ICRP are designed to assure that no individual within a population group receives more than the applicable dose limit. To facilitate this approach in evaluating releases from a given facility, selection is made of a so-called "critical group" which, because of living habits or unusual circumstances, receives a dose greater than that of any other group. If the dose to this group is within limits, the assumption is made that the doses to all other groups will be acceptable.

   Application of the ALARA criterion, however, requires that consideration be given to the total number of people exposed as well as the dose distribution among them. The ICRP urges that those responsible for the evaluation of environmental releases be aware of the fact that, in some cases, the total population impact may be greater outside the critical group than within it. Although the dose to individual members of the critical group may be larger, the sum of the doses to the much larger number of individuals within the non-critical groups may make them more significant.

3. The establishment of a food-chain or inhalation pathway model requires that:

   a. The objective of the modeling effort be clearly defined.

   b. The system to be modeled be outlined in detail.

c. The transfer factors within the system be identified and quantified.

d. Calculations be made to determine the response of the system for specific inputs.

e. This response be analyzed to determine the c itical nuclides and pathways and the effects of parameter uncertainties.

Two situations must be considered in modeling the pathways of radionuclides within the environment. One is that in which an equilibrium exists between the rate of discharge and the steady-state concentrations of radioactive materials within the environment. The second is where no such equilibrium has been established. Although the former situation is relatively easy to model, the latter non-equilibrium situation is far more complex. Care must be taken not to apply the simpler equilibrium models to non-equilibrium situations.

4. Projections of population doses from routine environmental releases may be important in site selection, particularly in those cases where the individual or collective doses per unit release are much smaller for one location than another. The technology of radioactive waste management, however, is now such that only in a few circumstances will planned releases have a decisive influence on the choice of a site. Where a decisive influence is exercised, it is more likely to be because of public relations implications than because of the radiological implications of predicted doses to members of the public.

VI. Existing and Future Reports

Presented in Table 2 is a list of the publications of the ICRP. Publications in preparation include:

1. Principles Concerning Emergency and Accidental Exposures (Medical Handling of Patients).
2. Limits for Intake of Radionuclides by Workers.
3. Assessment of Doses from Radionuclide Releases into the Environment.
4. Monitoring for Internal Contamination.
5. Biological Effects of Inhaled Radionuclides.

In addition, Committee 4 of the ICRP has initiated studies to prepare reports on:

1. The Principles and Methods for Application of the Optimization Requirement to Dose Limitation (Application of the ALARA Criterion).

2. Evaluation of Practices Which May Influence Exposure to Natural Background.

3. Protection of the Public in the Event of Radiation Accidents.

TABLE 2

PUBLICATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION

PRINCIPLES OF ENVIRONMENTAL MONITORING RELATED TO THE HANDLING OF RADIOACTIVE MATERIALS. A report prepared by a Task Group of ICRP Committee 4. ICRP Publication 7, Pergamon Press, Oxford (1966).

REPORT OF COMMITTEE 4 ON EVALUATION OF RADIATION DOSES TO BODY TISSUES FROM INTERNAL CONTAMINATION DUE TO OCCUPATIONAL EXPOSURE. ICRP Publication 10, Pergamon Press, Oxford (1968).

THE ASSESSMENT OF INTERNAL CONTAMINATION RESULTING FROM RECURRENT OR PROLONGED UPTAKES. A report of ICRP Committee 4. ICRP Publication 10A, Pergamon Press, Oxford (1971).

A REVIEW OF THE RADIOSENSITIVITY OF THE TISSUES IN BONE. A report prepared by a Task Group for ICRP Committees 1 and 2. ICRP Publication 11, Pergamon Press, Oxford (1968).

GENERAL PRINCIPLES OF MONITORING FOR RADIATION PROTECTION OF WORKERS. A report prepared by a Task Group of ICRP Committee 4. ICRP Publication 12, Pergamon Press, Oxford (1969).

RADIATION PROTECTION IN SCHOOLS FOR PUPILS UP TO THE AGE OF 18 YEARS. A report by Committee 3 of ICRP. ICRP Publication 13, Pergamon Press, Oxford (1970).

RADIOSENSITIVITY AND SPATIAL DISTRIBUTION OF DOSE. Reports prepared by two Task Groups of ICRP Committee 1. ICRP Publication 14, Pergamon Press, Oxford (1969).

PROTECTION AGAINST IONIZING RADIATION FROM EXTERNAL SOURCES: A report of ICRP Committee 3. ICRP Publications 15 and 21, Pergamon Press, Oxford (1976).

PROTECTION OF THE PATIENT IN X-RAY DIAGNOSIS. A report prepared by a Task Group of ICRP Committee 3. ICRP Publication 16, Pergamon Press, Oxford (1970).

PROTECTION OF THE PATIENT IN RADIONUCLIDE INVESTIGATIONS. A report prepared for ICRP and adopted by the Commission in September 1969. ICRP Publication 17, Pergamon Press, Oxford (1971).

THE RBE FOR HIGH-LET RADIATIONS WITH RESPECT TO MUTAGENESIS. A report prepared by a Task Group of ICRP Committee 1. ICRP Publication 18, Pergamon Press, Oxford (1972).

THE METABOLISM OF COMPOUNDS OF PLUTONIUM AND OTHER ACTINIDES. A report prepared by a Task Group of ICRP Committee 2. ICRP Publication 19, Pergamon Press, Oxford (1972).

ALKALINE EARTH METABOLISM IN ADULT MAN. A report prepared by a Task Group of ICRP Committee 2. ICRP Publication 20, Pergamon Press, Oxford (1973).

IMPLICATIONS OF COMMISSION RECOMMENDATIONS THAT DOSES BE KEPT AS LOW AS READILY ACHIEVABLE. A report of ICRP Committee 4. ICRP Publication 22, Pergamon Press, Oxford (1973).

REPORT OF THE TASK GROUP ON REFERENCE MAN. A report prepared by a Task Group of Committee 2 of ICRP. ICRP Publication 23, Pergamon Press, Oxford (1975).

RADIATION PROTECTION IN URANIUM AND OTHER MINES. A report of ICRP Committee 4. ICRP Publication 24, Pergamon Press, Oxford (1977). (Annals of the ICRP vol 1, no. 1).

THE HANDLING, STORAGE, USE AND DISPOSAL OF UNSEALED RADIONUCLIDES IN HOSPITALS AND MEDICAL RESEARCH ESTABLISHMENTS: A report of a Task Group of ICRP Committees 3 and 4. ICRP Publication 25, Pergamon Press, Oxford (1977). (Annals of the ICRP Vol. 1. No. 2).

RECOMMENDATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION. (Adopted January 17, 1977). ICRP Publication 26, Pergamon Press, Oxford (1977). (Annals of the ICRP vol. 1, no. 3).

PROBLEMS INVOLVED IN DEVELOPING AN INDEX OF HARM: A report prepared by a Task Group of the International Commission on Radiological Protection. ICRP Publication 27, Pergamon Press, Oxford (1977). (Annals of the ICRP Vol. 1 No. 4).

Appendix XXVI

APPENDIX XXVI
Reorganization of ACRS Generic
Subcommittees

Revisions:MWL:RFF:bjw:3/30/78

PROPOSED
ASSIGNMENT OF STANDING SUBCOMMITTEES

Standing Subcommittees will be responsible for design and performance of systems, components, and related materials in designated areas; the technical content of related criteria, Regulatory Guides, and Staff Action Plans for resolution of generic matters and criteria for backfit; reactor safety research in designated areas and preparation of appropriate portions of the periodic ACRS reports on Unresolved Generic Items and the RSR Program; follow-up with respect to the implementation of resolved generic items with respect to already-licensed facilities.

The Standing Subcommittee Chairman may organize his Subcommittee into smaller working groups to handle specific matters within the broader range of Subcommittee responsibility.

Advanced Reactors: Standardized advanced reactor designs (e.g., LMFBR, GCFBR, 1000 MW HTGR) proposed for non-water cooled reactors; advanced or non-water cooled or moderated reactor design bases, criteria, regulatory guides; preparation of Chapter 6 of Annual RSR Report to Congress.
~~MB~~, MC, JCM, PGS, CPS, WK, MB

Core Performance

Core physics; power distribution measurement and control; effect of positive moderation coefficient, fluence at pressure vessel wall, reactivity effects (e.g., calculation of rod drop and rod ejection accident.)
WK, HSI, JCM, DO, ~~DC~~, MC

Concrete and Concrete Structures

Concrete containment and reactor pressure vessel design bases and criteria; fuel storage pool design (e.g., structural integrity).

CPS, MB, PGS, DO, ~~DE~~, JE

ECCS

Design of current ECCS; improved ECCS designs; thermal-hydraulic performance of primary system during LOCA; ECCS research program; preparation of Chapter 2 of Annual RSR Report to Congress.

HSI, MC, JE, HE, DO, MP

Enrichment Plants

Enrichment plant design criteria and design bases.

JCM, MC, HSI, SL, PGS, ~~HE~~

Extreme External Phenomena

Criteria for extreme external phenomena, such as earthquakes, tornadoes, tsunamis, seiches, hurricanes, floods, explosions, airplane crashes, release of noxious chemicals; effects of LNG or other fires. Preparation of Chapter 5 of Annual RSR Report to Congress.

DO, ~~MB~~, JCM, CPS, ~~DE~~, DWM

Fluid Dynamics

BWR containment programs; RPV asymmetric loads; containment subcompartment pressures and dynamic loads during LOCA blowdown, relief valve operation, etc., water hammer.

MP, JE, ~~DE~~, HSI, CPS, ~~MA~~

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Follow-up Activities

Prepare response, as appropriate, re periodic NRC Staff report on status of ACRS recommendations and requests.

DWM, WK, MP, PGS, ~~JM~~, HE

Generic Items

Preparation of periodic (semi-annual) report on Generic Items; coordination and review[1/] of NRC task action plans, including referral to cognizant topical or project Subcommittees, where appropriate; review of implementation of resolved generic items, including referral to cognizant topical or project Subcommittees where appropriate.

MB, WK, DO, PGS, CPS, DWM

Metal Components

Design and performance of metal components including the reactor pressure vessel, and other components such as valves, pumps, snubbers, rod drives and piping; radiation damage and material properties, materials performance and load limits; primary and secondary system corrosion and water chemistry including steam generator tube degradation, effects of containment sprays on the primary plant, etc.; preparation of Chapter 4 of Annual RSR Report to Congress.

PGS, MB, HE, HSI, DO, ~~CPS~~

Plant Arrangements

Separation criteria; missile protection; post-accident environmental qualification; high-energy line restraints; systems interaction (mechanical).

MB, JE, SL, ~~CPS~~, MP, ~~JM~~, JCM

1/ Generic Items Subcommittee will review items not assigned to other Standing Subcommittees.

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Power and Electrical Systems: Design and performance of normal and emergency power supplies; plant computers; plant instrumentation and control system, safety actuation systems, leak detection

WK, ~~MS~~, JE, ~~ME~~, JCM, MP, DO

Procedures and Administration

Procedures and ACRS Bylaws; Fellowship program and assignments, new members.

SL, MB, MC, WK, DWM, DO, CPS

Radiological Effects and Site Evaluation

Population dose calculation - accident situations and normal operation; ALARA criteria on- and off-site; environmental monitoring; emergency procedures; ultimate heat sink design; protection for affected populations outside the LPZ; source term definition; preparation of Chapter 7 of Annual RSR Report to Congress.

DWM, JE, HSI, SL, ~~MS~~, ~~CPS~~, DO

Reactor Fuel

New and modified fuel design and proof testing; thermal-hydraulic and mechanical fuel performance during normal and abnormal conditions; pellet-cladding interaction; fuel failure propagation; end-of-life fuel performance preparation of Chapter 3 of Annual RSR Report to Congress; evaluation of replacement fuel designs and qualification testing.

PGS, HE, HSI, SL, DO, JCM

Reactor Operations

Systematic evaluation program; organization of operating group and of plant review and audit committees; stretch power increases, incident evaluation and responses (e.g., primary system blowdown transients, overpressuring action of primary systems, etc.) reload and FTOL reviews; spent fuel storage capacity; backfitting policies and practices; inservice inspection and testing; operational QA.

HE, JE, MB, DO, ~~CPS~~, DWM

Reactor Safety Research

Scope and balance of RSR program, coordination of Annual RSR Report to Congress.

DO, HSI, HE, PGS, CPS, MB, DWM, JCM, WK

Regulatory Activities: Conduct and coordinate review of Reg Guides as appropriate; ad hoc review of Reg. Staff proposals for new approaches, and referral as appropriate.

CPS, MB, HE, WK, MP, DWM

Reliability and Probabilistic Assessment

Reactor Safety Study; reliability assessment of systems and components (e.g., isolation of low-pressure from high-pressure systems); containment isolation provisions; containment isolation provisions (e.g., steam line isolation valve seal systems); functional systems interaction (probability of DC power supply of component cooling water failure compromising other systems); preparation of Chapter 9 of Annual RSR Report to Congress.

WK, HSI, JCM, DO, HE, MB

Safeguards and Security

Industrial security/access control; design features to preclude or mitigate effects of sabotage; material accountability - SNM; anti-diversion measures; preparation of Chapter 8 of Annual RSR Report to Congress.

JCM, MB, PGS, HE, SL, CPS

Waste Management

Plant decontamination decommissioning criteria and procedures; rad-waste management and long-term disposal; in-plant radwaste system design; effectiveness of containment sprays or removal of radio-nuclides.

DWM, MC, ~~HE~~, WK, MP, SL ,JCM

AD HOC SUBCOMMITTEES

Ad hoc Subcommittees will be set up as required to handle specific generic type problems which involve an interdisciplinary approach in areas assigned to a number of Standing Subcommittees. They will normally be disbanded when their specific assignment is completed. Ad hoc Subcommittees will be responsible for review of criteria and guides and backfit criteria for the matters they are established to review.

Fire Protection

Criteria and guide; separation criteria; backfitting criteria for fire protection.

MB, HE, JE, CPS

Long-Range RSR Program on Improved Concepts

CPS, MB, JE, HSI, MP

Transportation of Radioactive Materials

Transportation through urban areas.

CPS, MB, HE, JCM, DWM

Single-Failure Criterion

Reevaluation of the single-failure criterion, e.g., DC power supply, residual heat removal systems, design basis loading combinations, etc.

MB, CPS, JE, HE

ATWS

Generic reviews of ATWS "fixes", e.g., pending ATWS report; backfitting criteria for ATWS "fixes".

WK, JE, HSI, DO, HE, MB

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APPENDIX XXVII
Proposal for ACRS Generic Items
Evaluation effort

DRAFT/MB - 4/23/78

Proposal for ACRS Generic Items Evaluation Effort

Background

Presently the ACRS has a list of 28 generic safety items which are listed as unresolved. In addition, there is a grouping of 48 that are stated to be resolved in accord with the ACRS definition below:

> *"Resolved as used in the generic items report refers to the following: In some cases, an item has been resolved in an administrative sense recognizing that technical evaluation and satisfactory implementation are yet to be completed. 'Anticipated transients without scram' represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable as practical or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system and of improved methods of augmented scope to inservice inspection or reactor vessels."*

The above definition leaves open the question of appropriate implementation of the resolution actions.

There has been considerable public criticism of the manner in which the ACRS generic items list is treated by the NRC. Some critics suggest that the list is used as a way of tabling important safety questions when no prompt resolution action is planned. Others claim that the generic list is a means of permitting licensing of nuclear power plants when open safety issues exist. The ACRS has never considered these criticsms to be valid. Nevertheless, the NRC posture would be much improved if the ACRS could establish the means by which generic safety questions are eliminated from the licensing qualifications in ACRS reports. It is worth noting that several matters on the resolved generic items list are also shown as unresolved items on subsequent lists. These include sabotage protection, pump flywheel missiles, and ECCS capability. Other items such as "instrumentation to follow the course of an accident" and fire protection are listed as resolved, but the implementation action is still unclear. Most were stated to be resolved because the NRC staff has developed either a Regulatory Guide or regulatory Branch Technical Position that provided a basis for regulation, but the manner in which the regulatory documents are applied is sometimes obscure, as for example, preoperational testing and ATWS. The ACRS has thus continued to raise questions concerning the NRC staff position on many "resolved" generic issues.

Approach to Eliminating Generic Qualifications

In order to eliminate the ACRS qualifications concerning generic safety matters, the NRC staff must either show that the issue does not warrant public concern or that actions can be taken to change the physical plant design in a manner that eliminates the safety concern. Alternatively, in some cases a technical specification change can serve as the equivalent of a plant design change. Some of the items on the ACRS list might be clarified by more thorough discussions with the NRC staff and the applicants concerning actions that could and would be taken to resolve the issues.

In the attachment, the issues are categorized into seven groups and it is suggested that each category be assigned to an ACRS working group to develop a resolution and implementation approach that would ultimately serve to eliminate the items listed. Where the ACRS could not reach agreement with the staff, it would be appropriate to take the matters up with the Commissioners. In those instances where the ACRS does not expect a short term solution, it could establish milestones for a longer term action. In a few cases, such as common mode failures, the issue is so general that its appropriateness as a generic issue is subject to question. Unless ACRS can select explicit matters for examination in this area, the item should be eliminated from the list since no one could define a path of action to meaningful resolution.

The attachment shows the proposed categorization to be used in assigning the ACRS work groups. Many of these parallel the groupings suggested by C. P. Siess for the Safety Research Review and these working groups might accept these generic matters as a part of their responsibilities if these groups are established. Only matters that need active attention are included on the list.

Legend

S	- Structural	PD	- Plant Design
I	- Instrumentation	O	- Operations
T-H	- Thermal Hydraulics	P	- Probabalistic
M	- Materials	R	- Resolved by ACRS Definition
		RP	- Resolution Pending

ACRS GENERIC ITEMS

ACRS Designation No.	Title	Status	Category S	I	T-H	PD	M	O	P	Needs Implementation*	Comments
I-7	Protection Against Pump Flywheel Missiles	R	x							No	Missile threat undefined & depends on pump runaway characteristics
I-8	Protection Agsinst Industrial Sabotage	R				x				Unclear	The needed action should be clarified by ACRS
I-9	Vibration Monitoring	R	x	x						Yes	Vibration monitoring methods are not defined for all areas
IA-2	Detection and Location of Primary System Leaks	R		x				x		Yes	Leak detection methods are available but adequacy is unclear
IA-5	ECCS Capability of Current and Older Plants	R				x				Unclear	License plants meet Appendix K. Is ACRS satisfied?
IB-3	Environmental Performance of Critical Components (pumps, cables, etc.) Post-LOCA	R				x		x		Yes	Covered by Reg. Guide but qualification status of components is still open
ID-2	Instrumentation to Follow the Course of an Accident	R		x				x		Yes	Covered by Reg. Guide 1.97, Rev. 1 but no licensee has responded yet
ID-4	Fire Protection Concerning Adequacy of Branch Technical Position 9.5.1, and Regulatory Guide 1.120	R				x				Yes	Fire protection approach is still under review for several plants BTP-9.5.1 and Reg. Guide 1.120 may need modification
II-1	Turbine Missiles	RP							x	Unclear	Problem applies only to some plant arrangements. Is backfit necessary?
II-2	Effective Operation of Containment Sprays in a LOCA	RP				x		x		Unclear	Does ACRS want to insist on a specific chemical additive?
II-3	Post-LOCA Pressure Vessel Failure by Thermal Shock	RP	x		x					No	Is thermal shock question still open Is so, are experiments needed?
II-4	Instruments to Detect (severe) Fuel Failures	RP		x	x		x	x		No	PBF tests show need for instruments. What else is needed?

*"Yes" indicates plant modification, experimental work, instrumentation or surveillance and inspection activity is needed but not in place. "No" indicates activity is in place or none is needed. "Unclear" means the implementation need has not been established and should be examined.

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Legend

S - Structural
I - Instrumentation
T-H - Thermal Hydraulics
M - Materials
PD - Plant Design
O - Operations
P - Probabalistic
R - Resolved by ACRS Definition
RP - Resolution Pending

ACRS GENERIC ITEMS

ACRS Designation No.	Title	Status	Category S	I	T-H	PD	M	O	P	Needs Implementation*	Comments
II-5A	Monitoring for Loose Parts Inside Primary Coolant Circuit	RP		x				x		No	Effectiveness of installed equipment needs appraisal
II-5B	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	RP		x				x		Unclear	Some methods are available but their suitability is undefined
II-6	Common Mode Failures of Safety Related Components	RP							x	No	Inexplicit requirements. Item needs further definition
II-6A	Reactor Scram Systems	RP							x	Unclear	Single failure criteria need review for these items. Is other action needed?
II-6B	Alternating Current Sources Onsite and Offsite	RP				x			x		
II-6C	Direct Current Systems	RP				x			x		
II-7	Fuel Behavior Under Abnormal Conditions	RP					x	x		No	PBF tests should answer questions (relates to II-4)
II-8	BWR Recirculation Pump Overspeed During LOCA	RP	x			x				Unclear	Similar to I-7. Issue needs clarification
II-9	The Advisability of Seismic Scram	RP	x					x	x	No	Basis needs clarification. Is Japanese practice relevant?
II-10	Emergency Core Cooling System Capability for Future Plants	RP			x	x			x	Unclear	Position needs clarification. Which plants need improvement and when?
IIA-1	Ice Condenser Containments	RP			x					No	Applies to small ice condenser containments such as FNP and to non-UHI plants
IIA-2	PWR Pump Overspeed During LOCA	RP	x				x			Unclear	Awaits analysis of CE tests (Similar to Item II-8)

*"Yes" indicates plant modification, experimental work, instrumentation or surveillance and inspection activity is needed but not in place. "No" indicates activity is in place or none is needed. "Unclear" means the implementation need has not been established and should be examined.

Legend

S	Structural	PD	Plant Design
I	Instrumentation	O	Operations
T-H	Thermal Hydraulics	P	Probabalistic
M	Materials	R	Resolved by ACRS Definition
		RP	Resolution Pending

ACRS GENERIC ITEMS

ACRS Designation No.	Title	Status	Category S	I	T-H	PD	M	O	P	Needs Implementation*	Comments
IIA-3	Steam Generator Tube Leakage	RP						x	x	Unclear	NRC has policy position. Does ACRS agree?
IIA-4	ACRS/NRC Periodic 10-Year Review of All Nuclear Power Plants	RP						x		No	Reactor Operations Division is initiating action. What does ACRS expect?
IIB-1	Computer Reactor Protection System	RP		x		x		x		Unclear	Has ANO-2 resolved this issue?
IIB-2	Qualification of New Fuel Geometries	RP			x					No	Systematic review of fuel geometries is almost complete. Is more needed?
IIB-3	Behavior of BWR Mark III Containments	RP	x		x	x				Yes	Awaits GE test results and subsequent design response
IIB-4	Stress Corrosion Cracking in Operating BWRs	RP					x	x		Unclear	Requires continuing surveillance
IIC-1	Locking Out of ECCS Power Operated Valves	RP						x	x	No	NRC staff logic concerning valve lockout is questioned. Needs further discussion
IIC-2	Design Features to Control Sabotage	RP				x				Unclear	Design improvements need discussion Is SWESSAR or Cherokee design acceptable?
IIC-3A	Decontamination of Reactors	RP					x	x		Unclear	Is this a public safety issue? Will industry R&D work results apply
IIC-3B	Decommissioning of Reactors	RP				x		x		Unclear	What is expected as a basis for resolution?
IIC-4	Vessel Support Structures	RP	x							No	Should be resolved by analysis. Would probability study help?

*"Yes" indicates plant modification, experimental work, instrumentation or surveillance and inspection activity is needed but not in place. "No" indicates activity is in place or none is needed. "Unclear" means the implementation need has not been established and should be examined.

Legend

S	- Structural	PD	- Plant Design
I	- Instrumentation	O	- Operations
T-H	- Thermal Hydraulics	P	- Probabalistic
M	- Materials	R	- Resolved by ACRS Definition
		RP	- Resolution Pending

ACRS GENERIC ITEMS

ACRS Designation No.	Title	Status	Category S	I	T-H	PD	M	O	P	Needs Implementation*	Comments
IIC-5	Water Hammer	RP	x		x					Yes	Will approaches by NSSS vendors cure the problem?
IIC-6	Maintenance and Inspection of Plants	RP				x		x		Yes	What evidence exists that maintenance and inspection capability is inadequate? What improvement is desired?
IIC-7	Behavior of BWR Mark I Containments	RP	x		x					Yes	Are Mark I owners group efforts adequate
IID-1	Safety Related Interfaces Between Reactor - BOP	RP				x				No	Now included in SRP and should be listed and resolved
IID-2	Assurance of Continuous Long-Term Capability of Hermetic Seals During Post-Accident Conditions	RP					x			Yes	May be resolved by qualification & surveillance program. Staff position not fully defined.
IIE-1	Soil Structure Interactions	RP	x							No	Is current staff interpretation & methodology an acceptable resolution

*"Yes" indicates plant modification, experimental work, instrumentation or surveillance and inspection activity is needed but not in place. "No" indicates activity is in place or none is needed. "Unclear" means the implementation need has not been established and should be examined.

Appendix XXVIII

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

June 1, 1978

Docket No.: 50-70

APPENDIX XXVIII
Request for ACRS Review of GETR Seismic Issues

Dr. Stephen Lawroski
Chairman, Advisory Committee
on Reactor Safeguards
U. S. Nuclear Regulatory
Commission
Washington D.C. 20555

Dear Dr. Lawroski:

At the February 10, 1978 ACRS meeting, the NRR staff briefed the Committee on the seismic concerns associated with the General Electric Test Reactor (GETR). At that meeting, copies of the Show Cause Order, which required shutdown of GETR on October 27, 1977, were distributed.

On February 13, 1978, the Commission designated an Atomic Safety and Licensing Board to consider the following issues concerning the GETR:

(1) What the proper seismic and geologic design bases for the GETR facility should be;

(2) Whether the design of GETR structures, systems, and components important to safety requires modification considering the seismic design bases determined in issue (1) above, and, if so, whether any modification(s) can be made so that GETR structures, systems, and components important to safety can remain functional in light of the design bases determined in issue (1) above; and

(3) Whether activities under Operating License No. TR-1 should continue to be suspended pending resolution of the foregoing.

Due to the safety significance associated with the seismic issues involved in this show cause proceeding, we request that the ACRS review the GETR with respect to these issues, and provide its recommendations to the Commission.

The staff currently expects to issue its Safety Evaluation Report on these issues by July 1, 1978.

Sincerely,

Edson G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor
Regulation

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Appendix XXIX

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

June 2, 1978

APPENDIX XXIX
Schedule of ACRS Subcommittee Meetings and Tours

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS AND TOURS

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

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

M. W. Libarkin
Assistant Executive Director
for Project Review

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

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JUNE

8	NPRDS (JCM) - WK, MB
14-15	Diablo Canyon (JCM) - CPS, MB, SL, HE
16	Seismic/Indiana Point 3 (RS) - CPS, JCM, HE
21-22	Diablo Canyon (JCM) - CPS, HE, JCM
28	Naval Reactors, Schenectady, NY (GRQ/AB) - WK, MB, HE
28-29	New England Power, 1&2, Providence, RI - (RW)- DM, CM
29	Electrical Systems, Control & Instru. (GRQ) - WK, MB, HE
30	Davis Besse, 2&3 (RM/RKM) - CS, HE, JE, DM

JULY

5	Reg Activities (GRQ/SD) - CS, HE, WK
6-7	219th ACRS Meeting
11	Radiol. Eff. & Scte Eval. (RM) - DWM, HSI, PGS
13	ATWS (TGM/PB) - WK, JCM, HSI
14	External Phenomena (RS) - CPS, DWM, JCM
18	Erie, 1&2, Sandusky, Ohio (RM) - WK, JE, HE, CS
18	ECCS - Los Alamos, NM (AB) - HSI, MP
20	Electrical Syst., Control & Instru., Los Angeles, CA (GRQ) - WK, MB, HE, MP
21-22	GETR - San Jose, Calif (RS) - CPS, WK, JCM, MB (Tent.)
24	Siting (TGM/SD) - DWM, CPS
24	RESAR 414 (RS) - PGS, HSI, WK
24-25	Waste Mgmt. (1:00 pm) (RM) - DM, HI*, WK*, SL (* = 25th only)
27-28	HCDA, Los Alamos, NM (TGM/PB) - WK, MC, MP, PS (Tent.)
12	FFTF, Wash DC, WK, DWM, HE

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AUGUST

1 ATWS (TGM) - WK, JCM

2 Reg. Activities (GRQ/SD) - CS, MB, HE, WK

3-5 220th ACRS Meeting

10 FFTF (AB) - WK, MP, JCM, MB (tent.)

11 Adv. Reactors (RS) - MC, JCM, CPS, PGS (tent.), MB (tent.)

14 ECCS - Idaho Falls, ID (AB) - HSI, MP, HE

15 Fluid Hyd/Dyn. Eff. Idaho Falls, ID (AB) - HSI, MP, HE

16 INEL Tour, Idaho Falls, ID - (TGM) - MP, HE, PGS, JCM

Idaho Falls

17-18 Reactor Fuels, (TGM/PB) - PGS, HE, JCM

29 Fluid Hyd/Dyn, Eff. - Los Angeles, CA (AB) - MP, HSI

SEPTEMBER

7-9 221st ACRS meeting

14-15 Adv. Reactors, Albuquerque, NM (RS) - MC, JCM, CPS, PGS

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Appendix XXX

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

June 7, 1978

APPENDIX XXX
Report on Maine Yankee Power Station

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

REPORT ON MAINE YANKEE ATOMIC POWER STATION

Dear Dr. Hendrie:

During its 218th meeting, June 1-2, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application by the Maine Yankee Atomic Power Company for authorization to operate the Maine Yankee Atomic Power Station at power levels up to 2630 MW(t). A subcommittee meeting on this matter was held in Washington, D. C. on May 25, 1978. The Committee had previously reported favorably on operation of the Maine Yankee Atomic Power Station at power levels up to 2440 MW(t) in its report of January 13, 1972. During this review, the Committee had the benefit of discussions with representatives of the Maine Yankee Atomic Power Company, Yankee Atomic Electric Company, Combustion Engineering Incorporated, and the Nuclear Regulatory Commission Staff. The Committee also had the benefit of the documents listed.

In the NRC Staff review of the request to increase power, analyses of accidents and transients, physics tests, fuel performance and site meteorology were carried out. Modifications to the Technical Specifications were also considered. In addition, the NRC Staff reviewed the operating history of the plant. In evaluating the proposed power increase in each of these areas, the NRC Staff used current NRC criteria. The NRC Staff has concluded that operation at the proposed power level in accordance with the proposed Technical Specifications is acceptable. The ACRS concurs.

The Advisory Committee on Reactor Safeguards believes that there is reasonable assurance that the Maine Yankee Atomic Power Station can be operated at power levels up to 2630 MW(t), without undue risk to the health and safety of the public.

Sincerely,

Stephen Lawroski

Stephen Lawroski
Chairman

REFERENCES

1. Letter from W. P. Johnson, Maine Yankee Atomic Power Company to NRC, Office of Nuclear Reactor Regulation, concerning a proposed license amendment, on power level increase to 2630 MW(t), dated August 1, 1977.

2. Letter from W. P. Johnson, Maine Yankee Atomic Power Company to the Office of Nuclear Reactor Regulation, modifying the power level increase in two steps, dated December 9, 1977.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation Concerning Power Level Increase of Facility Operating License No. DPR-36, Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station, Docket No. 50-309, dated January 17, 1978.

4. Letter from D. W. Edwards, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning additional information regarding Maine Yankee power level increase, dated March 1, 1978.

5. Letter from R. H. Groce, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation concerning information for the preparation of the SER, dated April 5, 1978.

6. Letter from R. H. Groce, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning additional information on power level increase, dated April 10, 1978.

7. Supplement No. 1 to the Safety Evaluation by the Office of Nuclear Reactor Regulation, concerning Power Level Increase of Facility Operating License No. DPR-36 Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station, Docket No. 50-309, dated April 11, 1978.

8. Letter from W. P. Johnson, Maine Yankee Atomic Power Company, to the Office of Nuclear Reactor Regulation, concerning Technical Specification changes for power level increase, dated April 28, 1978.

9. Memorandum from Edson Case, Chairman, Regulatory Requirements Review Committee to L. V. Gossick, Executive Director for Operations, dated May 12, 1978, concerning an interim approval of Draft Regulatory Guide, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," dated February 3, 1978, and "Atmospheric Dispersion Model for Accident Evaluations," dated April 18, 1978.

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Appendix XXXI

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

June 8, 1978

APPENDIX XXXI
Letter to Representative M.K. Udall

The Honorable Morris K. Udall, Chairman
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Udall:

The Advisory Committee on Reactor Safeguards (ACRS) has considered the suggestion in your letter of January 27, 1978 for establishment of an independent, quasi-judicial board, patterned after the National Transportation Safety Board (NTSB), for accident analysis within the context of the current nuclear regulatory process. The Committee considered also the questions which you raised concerning the role of the ACRS vis-a-vis such a Board, should it be created.

Discussions with representatives of the NTSB's Bureau of Accident Investigation have indicated that, although the NTSB is responsible for investigating accidents in surface, air, and marine transportation, the criteria, procedures, and scope of the investigations vary depending on the specific mode of transportation involved. Air transport events, however, represent the bulk of NTSB work and range from minor incidents to serious accidents. It probably is the most well established area of NTSB's responsibility. In response to your inquiry the ACRS compared the nuclear power program requirements with air transportation investigation procedures.

While the NTSB reports on all aviation accidents, the bulk of the investigations, which are concerned with minor accidents or incidents, are delegated to the FAA, the involved regulatory agency. NTSB investigations are reserved for major accidents, generally involving fatalities. Analogous major accidents have not occurred in commercial nuclear power plant operation. Indeed, the Nuclear Regulatory Commission (NRC) and the ACRS devote a significant effort to reviewing operational experiences, proposed changes in operating procedures, and plant design features intended to forestall such accidents and continuing discussion of this process with the NRC Staff is planned. For this reason, the ACRS believes that existing institutional arrangements are adequate for the range of incidents thus far experienced in nuclear power plant operation. Should there be an accident comparable in magnitude and significance to those now investigated by NTSB, it is within the mandate of the ACRS to conduct a comprehensive and independent investigation of it. Therefore, our opinion is that no need exists to establish an independent board to carry out this function.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

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Appendix XXXII

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

June 8, 1978

APPENDIX XXXII
Regulatory Guides

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

SUBJECT: REGULATORY GUIDES - ACRS ACTION

Dear Mr. Gossick:

During its 218th meeting, June 1 and 2, 1978, the ACRS concurred in the regulatory position of Regulatory Guide 1.136, Revision 1, "Material for Concrete Containments."

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

cc: E. G. Case, NRR
R. Minogue, OSD
G. Arlotto, OSD
S. J. Chilk, SECY

bcc: ACRS Members
H. Voress
J. Jacobs

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Appendix XXXIII

Additional Documents Provided for ACRS' Use

1. Letter, M.K. Udall to S. Lawroski, relating to suggestion for establishment of an independent quasi-judicial board for review of nuclear reactor accidents, dtd Jan. 27, 1978.

2. Letter, H. W. Lewis to Rep. M.K. Udall, regarding a suggestion for establishment of an independent quasi-judicial board for the review of nuclear reactor accidents, dtd Nov. 23, 1977.

3. Memorandum, R.H. Vollmer to R.F. Fraley, Comparison of LOCA Radiological Evaluation Models, NRC vs. RSK, dtd May 24, 1978.

4. Letter, H.W. Lewis to L.V. Gossick, regarding "WASH-1400 Methodology," dtd. May 10, 1978.

5. Collection of Position Papers provided by members of the Interagency Nuclear Waste Management Task Force for its meeting, Apr. 20, 1978.

6. Paper, The Role of Risk Assessment in the Nuclear Regulatory Process, S. Levine, presented at the Atomic Industrial Forum Workshop on Reactor Licensing and Safety, Apr. 7, 1978.

7. Memorandum and Attachment, R.F. Fraley to ACRS Members, proposed ACRS Review of GETR, dtd June 1, 1978.

8. Minutes of Maine Yankee Subcommittee Meeting, May 30, 1978.

9. Minutes of Vermont Yankee Nuclear Generating Station, May 19, 1978.