APPENDIXES
TO
MINUTES OF THE 238TH ACRS MEETING
FEBRUARY 7-9, 1980

ACRS-1717

CERTIFIED

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238TH ACRS MEETING

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

January 28, 1980

DETAILED SCHEDULE AND OUTLINE
FOR DISCUSSION
238TH ACRS MEETING
FEBRUARY 7-9, 1980
WASHINGTON, DC

Thursday, February 7, 1980, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 12:00 Noon — Executive Session (Open)

- 1.1) 8:30 A.M.-8:50 A.M.: Chairman's Report (MP/RFF)
  - 1.1-1) Status of low power test program for Sequoyah Nuclear Power Plant
  - 1.1-2) Report of NRC Special Inquiry Group on Three Mile Island
  - 1.1-3) Response from Comm. Ahearne regarding ACRS participation in rulemaking proceeding regarding storage and disposal of radioactive wastes
- 1.2) 8:50 A.M. - 12:00 Noon: Discuss ACRS Annual Report to Congress on the NRC Safety Research Program (CPS et al./TGM/DZ et al.)

12:00 Noon - 1:00 P.M. — LUNCH

2) 1:00 P.M. - 3:00 P.M. — Executive Session (Open)

- 2.1) 1:00 P.M.-2:00 P.M.: Discuss proposed ACRS report to NRC on proposed changes to NRC Criteria for Siting Nuclear Power Plants (NUREG-0625) (DWM/RM)
- 2.2) 2:00 P.M.-3:00 P.M.: Discuss proposed reply to U.S. House of Representatives Committee on Interior and Insular Affairs (Rep. Morris K. Udall, Chairman) regarding component failure rates and probabilistic assessment of specific incidents at nuclear facilities. (DO/GRQ/DJ)

3) 3:00 A.M. - 4:00 P.M. Meeting with Director, NRR (Open)

3.1) Report on proposed NRR action to implement lessons learned from the TMI-2 accident

4) 4:00 P.M. - 6:30 P.M. Executive Session (Open)

4.1) Reports of ACRS Subcommittees on:

4.1-1) Surry Power Station Unit 2 - steam generator replacement (HE/GRQ)

4.1-2) Wolf Creek Nuclear Plant - seismic design (DO/RS)

4.1-3) ACRS Procedures (MP/RFF) and Working Group on Report of NRC Special Inquiry Group on TMI (MWC/RFF)

Friday, February 8, 1980, Room 1046, 1717 H Street, NW, Washington, DC

5) 8:30 A.M. - 12:30 P.M. Three Mile Island Nuclear Station Unit 1 (Open)

5.1) 8:30 A.M.-9:00 P.M.: Report of ACRS Subcommittee on TMI-1 (HE/RM)

5.2) 9:00 P.M.-12:30 P.M.: Meeting with NRC Staff and applicant

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

12:30 P.M. - 1:30 P.M. LUNCH

6) 1:30 P.M. - 4:30 P.M. Proposed Acceptance Criteria for MK I Containment (Open)

6.1) 1:30 P.M.-2:00 P.M.: Report of ACRS Subcommittee on Fluid Dynamics (MP/ALB)

6.2) 2:00 P.M.-4:30 P.M.: Meeting with NRC Staff

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

Imple entation of NRC Bulletins and Orders resulting from the TMI-2 Accident (Open)

7.1) 4:30 P.M.-5:00 P.M.: Report of ACRS Subcommittee (WMM/PB)

7.2) 5:00 P.M.-6:30 P.M.: Meeting with NRC Staff

Saturday, February 9, 1980, Room 1046, 1717 H Street, NW, Washington, DC

8) 8:30 A.M. - 12:30 P.M.

Executive Session (Open)

8.1) Discuss Proposed ACRS reports/letters on:
- 8.1-1) NRC Safety Research Program
- 8.1-2) NRC Bulletins and Orders
- 8.1-3) Mk I Acceptance Criteria

12:30 P.M. - 1:30 P.M. LUNCH

9) 1:30 P.M. - 4:15 P.M.

Executive Session (Open)

9.1) 1:30 P.M.-3:00 P.M.: Discuss proposed ACRS reports/letters on:
- 9.1-1) Proposed operation of TMI-1
- 9.1-2) Proposed revision of NRC Siting Criteria
- 9.1-3) Component failure rates and probabilistic assessment of incidents at nuclear facilities (reply to Congressman M. K. Udall)

9.2) 3:00 P.M.-3:15 P.M.: Discuss Future Agenda Items
- 9.2-1) Anticipated Subcommittee activities
- 9.2-2) Anticipated Committee activities

9.3) 3:15 P.M.-3:45 P.M.: Reports of ACRS Subcommittees on:
- . Anticipated Transients Without Scram (WK/TGM)
- . Proposed rule on Fire Protection (MB/PST)
- . LaCrosse Nuclear Plant - spent fuel storage racks (WK/JCM)

9.4) <u>3:45 P.M.-4:15 P.M.</u>: Miscellaneous

9.4-1) Proposal regarding seismic qualification of 1E Control Panels (DO)

9.4-2) Proposed ACRS letter regarding qualification of personnel who operate radwaste systems (DWM)

9.4.3) Participation in AIF Workshop on Licensing and Technical Issues (MP) address and telephone number of the person.

James B. Roberts,
Executive Officer.

[FR Doc. 80-2571 Filed 1-25-80; 8:45 am]
BILLING CODE 6820-49-M

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, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on February 7-9, 1980, in Room 1046, 1717 H Street, N.W., Washington, D.C. Notice of this meeting was published on January 22, 1980.

The agenda for the subject meeting will be as follows:

Thursday, February 7, 1980

8:30 A.M.-12:00 Noon: Executive Session (Open)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities.

The Committee will discuss its annual report to Congress on the NRC Safety Research Program.

Portions of this session will be closed as necessary to protect information the premature disclosure of which would frustrate the Committee in the performance of its statutory function.

1:00 P.M.-2:00 P.M.: Meeting with NRC Staff (Open)—The Committee will hear and discuss proposed NRC Staff plans for study of additional engineered safety features for the Zion Nuclear Station Units 1 and 2 and the Indian Point Nuclear Station Units 2 and 3.

2:00 P.M.-6:00 P.M.: Meeting with NRC Staff (Open)—The Committee will hear and discuss presentations from members of the NRC Staff and consultants who may be present regarding proposed plans for implementation of NRC Bulletins and Orders resulting from the accident at the Three Mile Island Nuclear Plant, Unit 2.

Friday, February 8, 1980

8:30 A.M.-12:30 P.M.: Three Mile Island Nuclear Station, Unit 1 (Open)—The Committee will hear reports from and will discuss proposed plans for restart and operation of the Three Mile Island Nuclear Station, Unit 1 with representatives of the licensee and the NRC Staff.

Portions of this session will be closed as required to discuss Proprietary Information applicable to this matter.

1:30 P.M.-4:30 P.M.: Meeting with NRC Staff (Open)—The Committee will hear presentations and discuss proposed criteria for modification of containment systems making use of the Mark I pressure suppression containment concept.

Portions of this session will be closed as required to discuss Proprietary Information applicable to this matter.

4:30 P.M.-6:30 P.M.: Executive Session (Open)—The Committee will discuss its proposed report to the NRC regarding proposed changes in criteria for siting of nuclear power plants (NUREG-0625). The Committee will also hear and discuss reports of its Subcommittees on the Surry Nuclear Station steam generator replacement and the Wolf Creek Nuclear Plant seismic design.

Saturday February 9, 1980

8:30 A.M.-4:15 P.M.: Executive Session (Open)—The Committee will continue its discussion of proposed ACRS reports regarding matters discussed during this meeting including the NRC safety research program: NRC Bulletins and Orders; criteria for Mark I containment; startup and operation of TMI, Unit 1; proposed revision of NRC siting criteria.

The Committee will hear the report of its Subcommittee on Reliability and Probabilistic Assessment and will discuss a proposed report to the House Committee on Interior and Insular Affairs regarding equipment failure rates in nuclear facilities and probabilistic assessment of selected incidents at power plants.

The Committee will hear reports from its Subcommittees on Anticipated Transients Without Scram, proposed criteria for fire protection of nuclear facilities, and changes in fuel storage racks at the LaCrosse Boiling Water Reactor. The future schedule for Committee activities will also be discussed, and the Committee will complete discussion of items considered during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information related to matters being considered and to protect information the premature disclosure of which would frustrate the Committee in the performance of its statutory function.

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on October 1, 1979 (44 FR 56408). In accordance with these procedures, oral or written statements may be presented by members of the public, recordings will 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. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a telephone call to the ACRS Executive Director (R. F. Fraley) prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with Subsection 10(d) Pub. L. 92-463 that it is necessary to close portions of this meeting as noted above to protect Proprietary Information (5 U.S.C. 552b(c)(4)) and to protect information the premature release of which would frustrate the Committee in the performance of its statutory function (5 U.S.C. 552b(c)(9)(B)).

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

Date: January 22, 1980.

John C. Hoyle,
Advisory Committee Management Officer.

[FR Doc. 80-2603 Filed 1-25-80; 8:45 am]
BILLING CODE 7590-01-M

[Dockets Nos. 50-277, 50-278, 50-320, 50-354, and 50-355; STN 50-485]

Philadelphia Electric Co., et al.; Order for Further Evidentiary Hearing

January 21, 1980.

In the matters of Philadelphia Electric Company, *et al.* (Peach Bottom Atomic Power Station, Units 2 and 3); Metropolitan Edison Company, *et al.* (Three Mile Island Nuclear Station, Unit No. 2); Public Service Electric and Gas Co. (Hope Creek Generating Station, Units 1 and 2); Rochester Gas and Electric Corporation, *et al.* (Sterling Power Project, Nuclear Unit 1).

The further evidentiary hearing on the aircraft crash probability issue in the

Issue Date:

June 16, 1980

MINUTES OF THE
238TH ACRS MEETING
FEBRUARY 7-9, 1980
WASHINGTON, DC

CERTIFIED

The 238th 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, February 7, 1980.

[Note: For a list of attendees, see Appendix I. Mr. Bender was not present on Saturday, February 9, 1980.]

The Chairman noted the existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been made from members of the public to present either oral or written 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 in approximately 24 hours.

[Note: Copies of the transcript taken at this meeting are also available for purchase from International Verbatim Reporters, Inc., 499 South Capital St. S.W., Suite 107, Washington, DC 20002.]

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 M. W. Carbon and J. J. Ray as reviewers, and J. C. Ebersole as alternate reviewer for the 238th ACRS Meeting.

B. Sequoyah Nuclear Power Plant Unit 1 - Proposed Low-Power Testing

The Chairman informed the Committee that the NRC Staff appears to be in general agreement with the Committee's views regarding the proposed Low-Power Test Program for Sequoyah Nuclear Power Plant Unit 1.

J. C. Ebersole suggested that feed and bleed tests under saturated steam conditions would be useful. This method of emergency cooldown relies on non-safety grade equipment, such as the PORVs and block valves, rather than safety valves (safety-grade). Clarification is needed regarding the nuclear industry's intent with respect to feed and bleed. There is a need also to develop test information under saturated-steam conditions. R. F. Fraley suggested that an ad hoc subcommittee could be formed to consider the feed and bleed concept. There may be need for some research work also. He noted that the PORVs and block valves discharge into the quench tank, which also is not safety grade.

W. Kerr suggested that the NRC Staff be informed that the proposed tests should continue even if additional information is desired; the need for additional information should not interfere with the test program.

R. P. Savio noted that an NRC Staff task force under R. Baer is reviewing the proposed tests, and also a number of other useful tests that have not been proposed by the applicant.

M. Bender noted that the block valves in question meet ASME requirements (physically) for primary system boundary components, but are not considered to be safety grade because there is no redundant control system for the valves. He suggested that there is a need to review natural circulation system requirements.

D. Okrent noted that the proposed feed and bleed cooling is tied to the reliability of the auxiliary feedwater system, and should be reviewed.

C. Report of Special Inquiry Group

Following a discussion of the Report of the Special Inquiry Group on the Three Mile Island Accident, especially with respect to those portions of the report that deal with the operation of the ACRS, it was the consensus of the Committee that a detailed response from the Committee would be inappropriate until after the full report has been received and studied. The Committee agreed to defer a committee report on these matters until after Volume 2 has been received.

D. ACRS Participation in Waste Disposal and Processing Rulemaking

R. F. Fraley noted that the Commissioners have agreed to the additional time that the Committee requested regarding its participation in the proposed rulemaking on radioactive waste processing and disposal.

II. Annual Report to Congress on NRC Research Programs (Open to Public)

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

The Committee reviewed the executive summary and approved the entire annual report to Congress on the NRC's reactor safety research program (see Appendix XXIII).

III. Meeting on "Report of the Siting Policy Task Force", NUREG-0625 (Open to Public)

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

The Committee discussed Draft 2 of its comments on the Report of the Siting Policy Task Force (NUREG-0625) (see Appendix XXVI).

IV. Meeting with Members of the NRC Staff on Implementation of TMI-2 Lessons Learned (Open to Public)

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

H. Denton, NRC Staff, informed the Committee that the NRC Staff would require that Indian Point Station Units 2 and 3 and Zion Nuclear Station Units 1 and 2 will be required to conduct certain studies of filtered containment venting, hydrogen control, and containment cooling, with a view toward installing these systems within a couple of years. In the interim, they will be required to take several actions spelled out in the NRC Action Plan. Orders to effect these requirements will be issued within several days.

H. Denton also requested that the Committee form an ad hoc subcommittee to aid the NRC Staff in its review of the long-term TMI-2 lessons learned actions.

H. Denton noted that an RES study indicates that because of the population distribution around the Zion and Indian Point Sites, the two sites together represent about 30% of the total acute risk to the public from operating plants. He added that there is approximately a factor of 10 difference between the risks from these two sites and other nuclear plant sites.

H. Denton said that licensee submittals on the new NRC requirements are expected to be received around March 1. He asked that an ACRS subcommittee review these submittals. He offered a further personal opinion that plants located in areas of higher population must be demonstrably safer than plants located at sites with lower population densities.

H. Denton noted that 52 plants have either complied with all of the requirements of the short-term lessons learned, or have shut down to make the changes. Eighteen plants have certified that equipment was not available, and they have been given additional thirty days to receive this equipment. Further, Oconee has been granted an extension so that not more than one plant is down for modification at a time. However, Duke Power has proposed some compensatory measures to make up for not meeting the requirements by the deadline.

With respect to long-term lessons learned items, H. Denton said that the NRC Staff is trying to resolve some of the issues currently in dispute between them and the utilities, e.g. ATWS and MARK I containments. GE has proposed alternative 3 as their solution to ATWS, but the NRC Staff favors alternative 4. GE has been informed that the Staff would consider it reasonable to install alternative 3 as early as possible, but to make pipes from the liquid control system adequate to meet the alternative 4 requirements.

H. Denton said that the NRC Staff is almost ready to recommend to the Commission that Sequoyah be licensed for special low-power testing.

J. C. Ebersole suggested that additional tests should be performed at Sequoyah to obtain information on saturation experiments, natural circulation tests, and test the feed and bleed system for emergency decay heat removal.

R. Mattson, NRC Staff, informed the Committee that the review of the Report of the NRC Special Inquiry Group on the Accident on Three Mile Island has been approved by the acting EDO, and has been transmitted to the Commissioners (see Appendix IV). He said that this report actually contains more than the NRC analysis of the Rogovin report; it also contains a conclusion that the review performed to date is sufficient to identify any urgent matters that need to be applied to operating reactors or to near term operating-license (NTOL) reactors. He said that just prior to coming to the ACRS meeting, principal members of the NRC Staff had met with the Commissioners, and obtained their approval of the NTOL list of items to be applied. The Commissioners did defer arriving at a decision of whether this list provides them with sufficient margin to end the licensing pause. The Commissioners plan to describe steps they want the NRC Staff to take regarding the Action Plan before they make that decision.

The NRC Staff has categorized the Special Inquiry Group recommendations into four groups:

- A - It is covered by the Action Plan already,
- B - The Action Plan should be changed slightly in language to incorporate the specific recommendations.
- C - The Plan should be added to where an action item is missing, and
- D - The recommendation is a bad idea or has been considered before and rejected.

R. Mattson noted that in draft 2 of the Action Plan, the number of items to be addressed has been reduced from the 245 items listed in draft 1 to 190 items; in addition it is expected that 10 to 20 items will be added as a result of the Report of the Special Inquiry Group.

Forty-five items were endorsed by the Commissioners today. Sixty to seventy items have been studied further, and will be brought to the Commissioners soon. Forty to fifty items need longer term study. He said that draft 3 will be available late in February or early in March, and that it will be difficult for the Committee to review this draft before the 239th ACRS Meeting. He said that he believed that the Commissioners were anxious to reach decisions on the Action Plan.

R. Mattson said that the NRC Staff is preparing a cross-index of the ACRS reports on Three Mile Island through December, 1979, indicating how each specific recommendation is treated in the Action Plan. He said that the Commissioners have not asked that the Committee comment on draft 3 before the NRC Staff and the Commissioners concur on it.

In answer to a question, H. Denton said that the NRC Staff is requesting Indian Point to study the mechanisms of steam explosions to determine if they would be a factor in containment penetration. Beyond that, the NRC Staff is considering the use of a core ladle to delay the penetration of the bottom of a containment by the molten fuel from a reactor following an accident. He said that the advantages of a core ladle are that they are cheap, they seem to be well understood by the NRC Staff, and they may be effective in reducing the risk to the public.

The Committee agreed that the TMI-2 Action Plan Subcommittee should consider draft 3 of the NRCs Action Plan if available, prior to the 239th ACRS Meeting (March). The TMI-2 Implications Subcommittee was also assigned to consider proposed changes at the Zion and Indian Point Nuclear Stations resulting from the lessons learned as a result of the TMI-2 accident.

V. Meeting on Three Mile Island Nuclear Station Unit 1 (Open to Public)

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

A. Subcommittee Report

H. Etherington, Subcommittee Chairman, noted that at the time of the TMI-2 accident, Unit 1 was shut down for refueling and has been kept shut down since the accident. On July 2, 1979, the Commission issued a formal order for the unit to remain shut down. On August 9, the Commission issued a follow-up order giving the reasons for its July 2 order, and specifying requirements to be met as a condition for consideration of approval for continued operation. This second order requires an ASLB hearing. The technical and administrative requirements imposed by the order of

August 9 include post-TMI-2 short-term and long-term items applicable to all nuclear power plants, those applicable to only B&W plants, and those unique to TMI-1 because of the Unit 2 cleanup. The Subcommittee has heard the Metropolitan Edison report on the status of the restart requirements.

H. Etherington noted that one item that was not in the NRC Staff order concerns intergranular stress corrosion cracking (ISCC). He said that on April 4, 1979 a through-wall leak was observed in the 8 inch type-304 stainless steel piping of the spent fuel system. Subsequently five more leaks were found in the system. All failures were in the heat affected zone of the top welds. Failures of this type are not unusual in cold water; it was speculated that boric acid may play a part in this attack. Nearly 2000 welds have been examined by UT in seven systems containing borated water; 31 indications of cracking were found, of which 42 were identified by specially developed UT techniques for ISCC. Several repair procedures have been proposed.

H. Etherington noted that the review of TMI-1 will take more time than is available at this meeting; the NRC Staff has indicated that an interim report is not needed at this time. (For background material, see Appendix V).

W. Lipinski, ACRS Consultant, noted that the containment isolation valves are of 40 in. diameter, and that the signal required to activate the isolation is a differential of 4 psi, a pressure that would not be reached unless there was a major leak in a reactor system. He questioned the safety of such an arrangement. He noted that the control room for Unit 1 is similar to that in Unit 2, and that the plant computer is identical to that in Unit 2. The Licensee plans to improve the system over the problems noted during the TMI-2 accident by replacing the typewriters with higher speed units. This fix may be inadequate. He said also a question has been raised regarding the freedom of access for operators to move through the plant in the event of an emergency. He noted that the Licensee has not agreed to the installation of a purge valve in the reactor vessel head. He said that the compressed air system that operates the emergency feedwater and chemical systems is not seismically qualified. He also questioned the adequacy of the additional training of operators planned by the Licensee.

I. Catton, ACRS Consultant, registered his surprise that the TMI operating staff has concluded that information retrieval problems during TMI-2 accident had little impact on the accident severity. He also questioned the adequacy of the more intense operator training program, noting that the content of the 32 hour training program in thermodynamics, heat transfer, and fluid mechanics, is greater than the 62 hours training in an undergraduate course in hydraulics and thermodynamics. He questioned also the adequacy of the proposed changes for the hydrogen control system and containment. He said that it was not clear how Regulatory Guide 1.97, Instrumentation to Follow the Course of an Accident, would be satisfied.

B. Status of NRC Staff Review

R. Volmer, NRC Staff, stated that the NRC Staff Safety Evaluation Report was basically a response to the tchnical and administrative issues that were raised in the NRC's August 9 order, and that the Staff was desirous of receiving the Committee's views on these matters. He said that the NRC Staff is trying to resolve all of the generic backlog issues as they apply to TMI-1, e.g., the RPV purge valve issue and the manner in which TMI-1 will comply with Regulatory Guide 1.97. He recognized that the SER cantains a large number of open items at this time, but noted that the review has not been completed. He said that the requirements the NRC Staff has recommended prior to the issuance of the Action Plan will be required for restart, but that no decision has been made yet regarding the recommendations of the Special Inquiry Group and the items specifically identified in the Action Plan. He noted that the Commissioners have requested an expedited treatment of the restart hearing, and the NRC Staff is trying to meet this request. Because the NRC Staff is trying to pursue these parallel paths simultaneously, it has brought the matters to the Committee at an earlier time than normal.

H. Silver, NRC Staff, provided an updated status report regarding the issues that apply to the TMI-1 restart (see Appendix VI).

D. DiIanni, NRC Staff, presented a summary report on the open generic items that existed prior to the TMI-2 accident (see Appendix VII).

Mr. Bender questioned the benefit to safety of attacking so many different items simultaneously without identifying those that are the most important.

C. Licensee's Presentations

1. Introduction and Utility Organization

R. Arnold, General Public Utilities Co. (GPU), representing the Licensee, Metropolitan Edison Co., noted that although there are a number of major items still unresolved with regard to NRC Staff requirements relating to TMI-1, his company had requested that the Committee review those items that have been resolved. This will provide the licensee with an opportunity to work further on those items for which the Committee still has concerns. He offered the opinion that a review can be more thorough and orderly when it is spread out over time. He said that June 1 has been set as a target date for completion of all the items that were identified in the NRC's restart order.

R. Arnold noted that GPU has formed a new unit, GPU Nuclear Corporation, to provide a full time single-minded dedication with uniform policies and a maximum availability of technical resources to safely operate all nuclear units owned by GPU operating companies (see Appendix VIII). This reorganization has resulted in a tripling of the professional technical staff supporting the current TMI activities. This new organization will have responsibility for Three Mile Island activities and for Oyster Creek.

In answer to a question, R. Arnold said that GPU Nuclear Corp. plans to review TMI-1 for reliability, probably in 1981. D. Okrent requested that the GPU Nuclear Co. provide the Committee with its proposed schedule for this study in the near future.

Members raised the question of the type of level instrumentation that the licensee plans to install on the reactor pressure vessel and in other parts of the primary cooling system, and a number of suggestions were made. However, the licensee indicated that no decisions have been made yet regarding this instrumentation.

2. Questions on the Remainder of the Agenda

Members raised the question of the effects of a total loss of power at TMI-1. C. Hartman, GPU, described the problem that had been raised as one of a total loss of DC power, initiated by a loss of offsite power, and followed by subsequent inadequate cooling of the core. He noted that this is a multiple failure, and has been reviewed by the Licensee with respect to TMI-1. He said that there are some alleviating devices present, including redundant transformers, and the fact that the turbine driven emergency feedwater pump can function without electricity, allowing time to respond to the event. He said the Licensee believes that the TMI stiuation is not as severe as that described by the NRC Staff in the SER.

J. C. Ebersole suggested that this matter should be examined by the Licensee in greater detail.

D. W. Moeller suggested that at a later meeting, the Licensee should discuss its emergency plan, including the availability of specific highways for evacuation, regional demography, consideration of local conditions, review of the liquid pathways including consideration of the nearest downstream water users and available methods for interdiction of liquid releases, and the capability of field monitoring.

Because of time factors, the Committee agreed to defer consideration of other matters regarding the TMI-1 restart until later meetings. (For handouts provided by the Licensee to the Committee, but not discussed or considered at this meeting, see Appendix IX.)

3. Plant Security (Closed to Public)

R. Skelton, Metropolitan Edison, discussed the implications of an article appearing in the Guide, a weekly newspaper in Cumberland, Pennsylvania, which received national publicity. Late in December, a Guide reporter assumed the identity of a friend and was hired to work at TMI as a watchman. In his article, the reporter claimed that he learned a great deal about Three Mile Island, and that he had adequate access to perform, if he had desired, acts of sabotage. The NRC Staff believes he disclosed nothing that was not already known.

R. Skelton said that the bottom line is that there is, in fact, little protection against an insider committing sabotage. The NRC has deferred a decision on how to handle this subject until December, 1980.

R. Skelton also cited the incident at Surry in which sabotage was performed on stored fuel elements by two insiders.

The article claimed that there was faulty screening of personnel, and that the Licensee was left on his own devices. The Licensee counters that a security plan was being followed. The reporter alleged that there was inadequate attention to [redacted] The reporter alleges that there [redacted] The reporter also alleged that the Licensee [redacted]

[redacted] does not apply to Unit 2, and that this is one of the things the NRC currently is studying. The question was raised whether there were adequate maintenance programs in testing [redacted] and whether new shifts are familiarized with the current situations.

L. Bush, NRC Staff, said that the NRC intends to investigate this matter further. So far it has been unable to interview the reporter. He also said that the article referred to studies made by Los Alamos for the Kemeny Commission, in which it was concluded that [redacted]

L. Bush noted that the reporter involved was working for a contractor, and not for the Licensee directly. Some of the watchmen are contractor employees, while the security guards are Licensee employees.

R. Rice, Metropolitan Edison, said that the reporter was assigned to five separate posts, and was rotated every two hours. These posts are not manned by armed guards. 55 site protection officers. In hiring, they are interviewed by a panel, are given physical and psychiatric tests, references are checked, and they operate under the Pennsylvania Lethal Weapons Act. Finger print checks, performed under the auspices of the Pennsylvania State Police, are made by the FBI to determine whether there has been a felony conviction. The guards are trained under the Pennsylvania State Police. Metropolitan Edison employee guards have weapons permits and are armed. It takes approximately three to four months to get [redacted]. Watchmen are contracted in accordance with Regulatory Guide 5.20. [redacted]

P. Clark, GPU, said that the Licensee is sensitive to this matter and is investigating. The Licensee will take those steps that can legally be taken to correct the situation.

R. Rice said that steps are being taken to assure that key personnel are familiarized with the classified Sandia security reports.

R. Rice said that with respect to [redacted]

VI. Meeting on Proposed Acceptance Criteria for Mark I Containments (Open to Public)

[Note: Andrew L. Bates was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

M. S. Plesset, Fluid Dynamics Subcommittee Chairman, noted that Mark I containments for boiling water reactors have been operating on an interim basis for almost two decades. The NRC Staff and the Mark I owners group have been working together, and presumably are close to a resolution of this generic problem. There have been series of tests in different facilities: GE's 1/4 scale facility, EPRI's Small 3-D facility, NRC research has funded a series of tests in a 1/5 scale facility at Lawrence Livermore Laboratory, and the Mark I Owners Group has financed a series of full scale tests in their Mark I-type facility at Norco, CA. He briefly reviewed the history of the programs, and the research that has been generated by them (see Appendix X).

M. S. Plesset informed the Committee that the Owners Group had accepted an assumption that the NRC Staff proposed regarding loads under conditions of high mass flow, i.e., SRV activation, which they now believe may be unreasonably conservative. This is one of the areas on which the Committee should focus.

B. Background

C. Grimes, NRC Staff, noted that the acceptance criteria for Mark I containments is the NRC Staff's Generic Item A1. He identified the plants and utilities that are operating or constructing plants with Mark I containment systems, identified the important design features of these systems, and discussed the history and chronology of the development and construction of the plants (see Appendix XI).

C. Long-Term Program Summary

R. Logue, Philadelphia Electric Co., representing the Mark I Owner's Group, described the owners organization, identified the utilities and plants involved, and discussed the program milestones (see Appendix XII).

T. Mulford, General Electric, discussed highlights from the Mark I containment program (see Appendix XIV).

D. Mark I Long-Term Program Acceptance Criteria

C. Grimes discussed the NRC Staff's proposed acceptance criteria for Mark I containment systems (see Appendix XIV).

E. Implementation Programs

R. Smart, Northeast Utilities Co., representing the Mark I Owners Group, discussed the current programs to develop and implement the NRC Staff's criteria (see Appendix XV). He noted that the big problem is the characterization of safety relief loads. He claimed that the proposed NRC criteria do not give realistic response for this problem; the Mark I Owners Group is still working on it. The big modification made so far in Mark I systems is the installation of T or Y quenchers. The Owners Group has tried to develop a test program to obtain structural response data and load definition data, but so far has not been successful.

C. Grimes stated the current NRC Staff position requiring the use of the conservative values proposed by the Staff is being pushed at this time because the Staff believes that the development of the criteria has gone on long enough, and that the Staff has inadequate resources to continue the dialogue with the Owners indefinitely.

In answer to a question regarding the safety margins of plants that have not yet installed quenchers, R. Smart noted that there is adequate experience to show that these plants have blown down successfully without causing any problems in the wet well or dry well.

Several Members indicated concern regarding the effects of a failure of the SRV piping between the safety relief valve and the torus.

R. Logue requested that the NRC Staff assure itself that any orders they issue regarding the Mark I acceptance criteria does not needlessly cause plants to shut down to meet unnecessary requirements.

C. Grimes said that the NRC Staff believes that modifications required by the criteria should be completed within two years; the current goal for completion is December 1981.

R. Logue said that the current Owners schedule for modifications of plants is that all will be modified prior to mid-1981 except for those utilities that will have to modify more than one plant.

VII. Meeting on the Implementation of NRC Bulletins and Orders Resulting From the TMI-2 Accident (Open to Public)

[Note: Paul A. Boehnert was the Designated Federal Employee for this portion of the meeting.]

(For background material, see Appendix XVI.)

The Chairman noted that W. M. Mathis, Chairman of the TMI-2 Bulletins and Orders Subcommittee, was unable to attend the meeting. He suggested that Members make use of the background materials provided, and regretted that there would be no direct subcommittee report to the Committee.

A. Overview

W. Kane, NRC Staff, discussed the Bulletins and Orders Task Force activities completed since it reported to the committee at the 237th ACRS Meeting (see Appendix XVII).

The Chairman noted that the Committee expected that it would write a report on the implementation of the Bulletins and Orders but that this report would not be written at this meeting.

B. Strengthening Reliability of Auxiliary Feedwater Systems in Combustion Engineering and Westinghouse Plants

M. Taylor, NRC Staff, discussed a probabilistic analysis of the reliability of auxiliary feedwater systems in both Combustion Engineering and Westinghouse plants (see Appendix XVIII).

J. C. Ebersole noted that a prototype of a block valve, scheduled to be installed in the McGuire and Catawba plants failed to close during full-flow testing. He said that the bleed and feed method for emergency cooling is predicated on the belief that the PORVs and block valves work. He suggested that it is necessary to identify the characteristics of the relief mechanisms, and to identify that the capacity of PORVs is adequate to relieve the pressure.

D. Okrent raised the question that in some plants, PORVs are "gagged" and asked how these valves are opened when needed. There was no clear answer to the question.

VIII. Executive Sessions (Open to Public)

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

A. Subcommittee Reports

1. Surry Power Station: Steam Generator Replacement

A movie showing the replacement of the Surry Unit 2 steam generators was shown. H. Etherington, Subcommittee Chairman, noted that the Subcommittee reviewed the replacement program on October 18, and that the Committee had asked that the Subcommittee review the current status and report to the Committee. Accordingly, the Subcommittee met with the licensee on January 23, 1980.

H. Etherington noted that VEPCO had constructed a refurbishing building for use in the decontamination of piping, weld preparation, valve packing, etc. Further, a full scale mock-up of the piping was made to train welders, pipe fitters, riggers and laborers. The planning for the replacement included calculation of dose reduction obtainable by shielding, decontamination, special tools, tents, and glove boxes for specific grinding and cutting operation. Health physics and work training programs were developed, and the operation was divided into engineering task assignments. Progress has been reported by VEPCO in five bimonthly progress reports, the last covering the period from October 1 to November 30, 1979. A final report is due from VEPCO sixty days after the completion of the work.

H. Etherington noted that while the dose estimates were reasonably accurate, there was a major overrun in man-hours expended in the replacement.

J. Benton, VEPCO, explained this discrepancy. He said that the original man-hour estimates, made three years ago, were made only for those tasks where there would be some exposure to radiation. No estimates were included in those initial estimates for normal routine tasks in which workers would not be exposed. The final figures included all of the manpower expended.

H. Etherington said that the Subcommittee believes that Surry 2 can be permitted to restart provided other issues are settled to the satisfaction of the NRC Staff, such as the seismic show cause order, the I&E bulletin on anchor bolts, and verification of as-built piping supports. (For project status report, see Appendix XX.)

2. Seismic: Wolf Creek Seismic Design

D. Okrent, Subcommittee Chairman, discussed the issues raised by a petitioner regarding the design basis earthquake for Wolf Creek, and the issues associated with establishing a minimum 0.2g design basis earthquake for sites east of the Rockies (see Appendix XXI). He said that because of the lack of accurate information regarding the controlling earthquake at this site, and more generally in the Eastern U.S., the decisions regarding seismic design are a matter of judgement. He suggested that future experience may show the wisdom of having all plants with a higher seismic floor. He believes it would be wise for any plant that has the flexibility to do so, to qualify the safe-shutdown and decay heat removal equipment to withstand a 0.2g design basis earthquake.

R. Jackson, NRC Staff, offered his opinion that undue emphasis is being placed on the ground acceleration value. He said that this value is merely an anchor point for design, and that it might be wise for the NRC to get away from this type of terminology.

The Committee agreed to table a decision on the seismic issues regarding Wolf Creek Nuclear Plant. The Seismic Subcommittee was requested to review the seismicity of Eastern U.S., and to consider the proposed recommendation for seismic response floor of 0.2g for plants east of the Rocky Mountains.

3. Procedures

The Committee concurred with the following recommendations of the Subcommittee (see minutes of the Procedures Subcommittee and attachments, February 6, 1980 meeting, Appendix XXII):

a. Proposed changes to NRC regulations which delineate procedures for ACRS participation in the rule making process.

b. Proposed procedures for ACRS handling of dissenting professional opinions, both for NRC Staff and ACRS Consultants.

c. Proposed procedures for the management of the ACRS Fellowship Program.

d. Recommendations resulting from comments of ACRS members J. C. Ebersole and W. M. Mathis aimed at improving full committee and subcommittee meetings by

(1) stating the specific purpose and objectives of each meeting in the meeting notice well in advance of the meetings;

(2) clearly informing ACRS consultants regarding what is expected of them at meetings;

(3) setting minimum time limits regarding receipt of documents prior to meetings;

(4) make better use during ACRS meetings of the time set aside to discuss anticipated Committee meetings so Members can identify items of concern/interest in advance of the meetings. The ACRS subcommittee chairman/cognizant staff engineer should provide a list of topics to be discussed and specific meeting objectives to facilitate this discussion;

(5) more recognition at full committee meetings of work accomplished at subcommittee meetings (e.g. Members should try to identify items of interest/ concern in advance of subcommittee meetings; familiarize themselves with information discussed/ developed at subcommittee meetings, etc. to better focus discussion during the full committee meetings;

(6) participate more actively in subcommittee activities or limit questions at full committee meetings that explore areas of personal interest; and

(7) subcommittee chairmen and cognizant staff engineers examine supplementary SERs and inform the Committee when ACRS recommendations are not implemented. The Committee should then take appropriate action. (Note: This is consistent with the existing system for handling category B reports provided to the ACRS.)

e. Proposed reorganization of the ACRS Technical Staff based on the assumption that ten additional, permanent, full-time staff members, as requested by the Committee, are approved.

f. To improve contact and communication with the Commissioners and the EDO by inviting them to monthly ACRS meetings.

4. ATWS

W. Kerr, Subcommittee Chairman, described the status of the ATWS problem as follows:

a. S. H. Hanauer has notified the Subcommittee that the NRC Staff's next ATWS report will not be available before the end of February.

b. The subcommittee does not plan to hold another meeting before the document has been received and the subcommittee members have a chance to review it.

c. The subcommittee has not received a proposal regarding the resolution for Combustion Engineering plants yet.

d. S. H. Hanauer has indicated that he would like a report from the Committee during the 240th (April) ACRS Meeting.

B. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 239th ACRS Meeting (March) (see Appendix II).

2. Schedule for ACRS Meetings and Tours

A schedule of Future ACRS Meetings and Tours was distributed to ACRS Members (see Appendix III).

C. Quality of NRC Staff's Safety Evaluation Reports

C. P. Siess was requested to represent the Committee on a group (D. Vassallo, Chairman) organized by the NRC Staff to review and improve the content and quality of the NRC Staff's safety evaluation reports (SERs). M. W. Libarkin will assist in this effort.

D. Emergency Decay Heat Removal

The Committee agreed to set up an ad hoc subcommittee to review proposed methods for emergency decay heat removal such as natural circulation and feed and bleed, and to follow those portions of the Sequoyah Unit 1 low-power testing programs that apply to these methods. Named to the Subcommittee were M. W. Carbon, Chairman and M. Bender, J. C. Ebersole, H. Etherington, M. S. Plesset, J. J. Ray; with A. L. Bates as cognizant engineer, and E. Abbott and G. Young, ACRS Fellows, to provide support.

E. Employment of Foreign Nationals as ACRS Fellows

The Committee considered a proposal to employ a foreign national as an ACRS Fellow as part of an international exchange program. The Members agreed that his technical qualifications should be considered to determine if he can contribute to support of ACRS activities.

F. ACRS Reports and Letters

1. ACRS Annual Report to Congress on NRC Safety Research Program

The Committee completed its annual report to Congress on the NRC Safety Research Program, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981, NUREG-0657 (see Appendix XXIII).

2. Consistency of Component Failure Experience with that Projected in WASH-1400

The Committee prepared a letter to Representative Morris K. Udall regarding the consistency of component failure experience with that projected in WASH-1400 and probabilistic analysis of selected incidents at the Davis-Besse and Rancho Seco nuclear stations (see Appendix XXIV).

3. Commission Adoption of Parts of NUREG-0660

The Committee prepared a letter to the Commissioners regarding the adoption of parts of NUREG-0660, Draft 2, Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident (see Appendix XXV).

4. Report on NUREG-0625, Report of Siting Policy Task Force

The Committee completed a report to the Commissioners providing the Committee's comments on NUREG-0625, Report of the Siting Policy Task Force (see Appendix XXVI).

5. Acceptance Criteria for Mark I Containment Long-Term Program

The Committee prepared a report to the Commissioners providing the Committee's comments on the NRC Acceptance Criteria for the Mark I containment long term program (see Appendix XXVII).

6. Qualifications of Radioactive Waste System Operating Personnel

   The Committee prepared a letter to the Commissioners regarding the qualifications of radioactive waste system operating personnel (see Appendix XXVIII).

7. Low-Pressure Turbine Disk Cracking

   The Committee approved a memorandum to the Acting Executive Director for Operations noting reports of cracks in Westinghouse low-pressure turbine disk assemblies, and requesting that the NRC Staff reevaluate the probability of failure and consequences from turbine missiles (see Appendix XXIX).

The 238th ACRS Meeting was adjourned at 3:20 p.m., Saturday, February 9, 1980.

Appendix I

ATTENDEES
238th ACRS Meeting
February 7-9, 1980

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
James M. Jacobs, Technical Secretary
Herman Alderman
Andrew L. Bates
David E. Bessette
John Bickel
Paul A. Boehnert
Sam Duraiswamy
Elpidio G. Igne
David H. Johnson
William Kastenberg
Morton W. Libarkin
Richard K. Major
Thomas G. McCreless
John C. McKinley
Robert E. McKinney
Ragnwald Muller
Gary R. Quittschreiber
Richard P. Savio
John Stampelos
Peter Tam
Hugh E. Voress
Gary Young
Dorothy Zukor

CONSULTANTS

I. Catton
W. Lipinski

NRC ATTENDEES

238TH ACRS MEETING

Thursday, February 7, 1980

Nuclear Reactor Regulation

Frank Miraglia
Gary Zech
G. Gahoche
A. Marchese
R. E. Jackson
E. A. Licitra
P. Sobel
P. Justus

Standards Development

R. Gall

NRC STAFF ATTENDEES

238th ACRS MEETING

February 8, 1980

Div. of Operating Reactors

A. Taboada
R. Vollmer

Div. of Systems Safety

J. Voglewede
G. Mazetis
S. Newberry

Nuclear Reactor Regulation

R. Fitzpatrick
B. A.Boger
J. S. Wermiel
W. L. Jensen
D. Pickett
J. Roe
C. Grimes
W. Kane
B. Wilson
Z. Rosztoczy
W. Hodges
C. Thomas
P. Matthews
D. DiIanni

Inspection and Enforcement

D. Haverkamp
L. Bush
C. Schwan

Div. of Project Management

J. Villalva
H. Silver
L. P. Crocker

Nuclear Reactor Regulation

P. D. O'Reilly
W. Hodges

Probabilistic Analysis Staff

M. Taylor
R. Beimer

LPDR
J. Souder

INVITED ATTENDEES

238TH ACRS MTG.

February 8, 1980

Metropolitan Edison

J. D. Bigber
M. J. Ross
C. E. Hartman
J. G. Herbein
L. Lawyer
J. Thorpe
R. Dubril

General Public Utilities

P. Clark
R. C. Arnold
R. W. Keatea
D. K. Croneberger
D. Slear
R. Rice
E. Wallace
C. Smyth

Public Service Electric & Gas

F. Marian

Shaw, Pittman
D. Ridgway

Babcock & Wilcox

E. Kane

PUBLIC ATTENDEES

238TH ACRS MEETING

Thursday, February 7, 1980

Acate, Nuclear Research Inst.
S. Wither, Nuclear Research Inst.
R. Borsum, Babcock and Wilcox
B. Horin, D&L
L. Connor, Doc-Search Associates
L. S. Gifford, General Electric
Mr. Leyse, Electric Power Research Inst.
Ichiro Cabe, NCRA
Kunihiro Ota, KEPCO
Henry Myers, Congress
Hiroyoshi Hamada, TEPCO
Frank S. Beal, Westinghouse
Takayuki Shirao, Embassy of Japan
Toshiaki Kikuchi, Embassy of Japan
Clark Downs, Isham, Lincoln andBeale
K. C. Fortino, Lowenstein, Newman
D. Walker, Offshore Power Systems
Whayne Dillehay, Critical Mass Energy Project
Frank Marian, PSEG
R. L. Stright, SNUPPSS
J. S. Trapp, Dames and Moore
D. F. Fenster, Dames & Moore
John T. Benton, Virginia Electric Power Company
H. Stephen McKay, Virginia Electric Power Company
Thomas E. Stenzel, Virginia Electric PowerCompany
Frank S. Beal, Westinghouse
Gene Rathban, Kansas Gas & Electric
Frank Avlicino, Self

PUBLIC ATTENDEES

238TH ACRS MEETING

Friday, February 11, 1980

Koroscik, Polish Embassy
R. S. Bowing, BETA
C. Eranomond, Brookhaven National Lab.
L. J. Sobon, NUTECH
R. Jabol, ABI
R. Smart, NUSCO
R. Mulford, General Ele-tric
L. Stewinert, General Electric
R. Gilasby, Bechtel Corporation
I. Cabe, NSRA
V. Montague, Med. News Service
R. H. Logue, PECo
R. J. Ross, Dames and Moore
R. Leyse, Electric Power Research Inst.
G. Maise, Brookhaven Nat'l Lab.
L. Sonin, Mass. Inst. of Technology
R. Berks, Teledyne
G. Balland, NUTECH
G. Kosi, Bechtel
D. Lehnert, Detroit Edison
R. Broman, Bechtel
H. Hamada, TEPCO
O. Mallou, Power Authority of New York
R. Swenson, PASNY
M. Mosier, NMPC
L. Connor, Doc-Search Associates
J. Kikuchi, Embassy of Japan
C. Giahmal, Stone and Webster
M, Liss, Mayer Brown & Platt
R. Borsum, Babcock and Wilcox
M. Schock, Self

Appendix II

APPENDIX II

FUTURE AGENDA - 2/11/80

IP/Zion changes from TMI-2	3 hours
Rancho Seco Transient (Light Bulb Incident)	1 hour
Implication re IP-3 and Zion 1 and 2, Seismic, etc.	
Draft 3 of Action Plan	4 hours
Future Case/Work	
Response to Commissioner Gilinsky re Reliability	3 hours
FNP Core Ladle	3 hours
10 CFR 50 Clad Ballooning	2 hours
B&W SG Sensitivity	2 hours
NRC Staff Re-evaluation of Turbine Missle	2 hours
B&O	4 hours
Method of Generic Item Resolution	1 hour
Seismic Qualification	
G. Young Memo on Systems Interactions	
Follow-up of ACRS Recommendations	
Meeting with NRC Chairman	1 hour
Meeting with EDO	
Rogovin Report, Response re ACRS items	
Discussion of proposed revisions to nuclear energy legislation introduced by Congressman Morris K. Udall (HR-6390)	

Appendix III

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

APPENDIX III

February 9, 1980

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 forefeiture 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
J. Jacobs

<u>NOTE</u>: During their February 6 meeting, members of the Procedures Subcommittee asked that we provide as much information as possible on the background, purpose, etc. of subcommittee meetings as early as possible. I have tried to do this for the current list. Any comments you may have as to format, content, etc. would be appreciated.

FEBRUARY

14	Emergency Core Cooling Systems/Reactor Fuels (Boehnert) - MP, HE, PS
20-21	Plant Arrangements (Tam) - MB, JE, SL, CM, JR, DWM, HE

MARCH

3	Bulletins & Orders (Boehnert) - WM, HE, PGS, MP, JE(tent.)
4	Babcock and Wilcox Water Reactors (Tam) - HE, JE, WM, JR
5	Regulatory Activities (Duraiswamy) - WK, HE, JR
5	Three Mile Island-2 Accident Implications (Major) - DO, MC, WM, JR
6-8	239th ACRS Meeting
25	Natural Circulation Heat Removal (Bates) - MC, MP, HE, JR, JE
25-26	Concrete and Concrete Struct. (Igne) - CPS, JE, PGS

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
2/14/80	Combined Reactor Fuel/ECCS	(PB) M.Plesset, P.Shewmon, H.Etherington - Cons: A. Acosta, I.Catton, J.Lienhard, F.Nichols

BACKGROUND, ETC.

The purpose of the meeting is: (1) discussion of the draft NRC NUREG report "Cladding Swelling and Rupture Models for LOCA Analysis" - NUREG-0630; and (2) discussion of the results of recent small-break LOCA tests in LOFT as well as the results of PBF tests which examined the effect of fin thermocouples on LOFT fuel temperature performance. NRC is requesting ACRS comment on item 1 at the March 1980 meeting. Following ACRS comment, the NRC will issue a final version of the report.

Item (2) is being presented for the Subcommittee's information and no Committee action is expected on this material at this time.

The NRC NUREG report noted in (1) above was published in early November 1979 when it was thought by NRC that the vendors clad swell and rupture models may be non-conservative and in violation of Appendix K. This turned out not to be the case, nevertheless NRC believes that some modifications in the clad swell and rupture curves are warranted, given the receipt of new test data from the US and overseas in recent years.

The following information has been provided:

1. Draft NUREG-0630 "Cladding Swelling and Rupture Models for LOCA Analysis.

2. NRC Memo dated February 2, 1980 from R. Meyer to P. Boehnert providing written comments received on draft NUREG-0630.

3. NRC Memo dated November 1, 1979 from G. McPherson to T. Murley discussing the PBF T/C Test Series Results.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
2/20-21/80	Plant Arrangements	(TAM) M.Bender, J.Ebersole, S.Lawroski, C.Mark, J.Ray, D.W.Moeller, H.Etherington

BACKGROUND, ETC.

The Subcommittee will meet on February 20 to discuss with the NRC Staff and Sandia Laboratories the "Final Report (DRAFT), Phase I, Systems Interaction Methodology Applications Program" (distributed to all members and subcommittee consultants). This item is a part of the effort on unresolved safety issued A-017, Systems Interaction. The Subcommittee will be updated on the status of this issue and have an opportunity to comment on the progress and future goals.

On the 21st the Subcommittee will discussion with the NRC Staff the status of thirteen ACRS generic items assigned to the Plant Arrangements and Combination of Dynamic Loads Subcommittees. A background material prackage was distributed on February 6, 1980. The meeting is aimed at resolving, redefining or updating the status of items: 6. Fuel Storage Pool Design Bases, 8. Protection Against Industrial Sabotage, 70. Design Features to Control Sabotage, 30. ECCS Capability of Current and Older Plants, 60. Primary Coolant Pump Overspeed During LOCA, 62. ECCS Capability for Future Plants, 52. Safety Related Interfaces, 58. Non-Random Multiple Failures, 23. Quality Group Classification for Pressure Retaining Components, 22. Seismic Design of Steam Line, 28. Protection Against Pipe Whip, 41. Seismic Category I Requirements for Auxiliary Systems, and 73. Vessel Support Structures.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/3/80	Bulletins and Orders	(PB) W.Mathis, H.Etherington, P.G.Shewmon, M.Plesset, J. Ebersole (tent.)

BACKGROUND, ETC.

Not fully developed as of this time. Will include the folowing:

(1) Impact of folding B&O Recommendations into Action Plan

(2) problem of qualification of block valves upstream of PORVs to close against design pressure

(3) development of criteria for use of "feed and bleed" mode for plant cooldown

(4) other questions from ACRS members and consultants growing out of review of B&O NUREG reports

(5) effectiveness of B&O Recommendations in reducing small break LOCAs due to stuck open PORVs

References

NUREG-0645 - B&O Summary Report
NUREG-0611 - W Generic Report
NUREG-0565 - B&W Generic Report
NUREG-0635 - CE Generic Report
NUREG-0626 - GE Generic Report

All of these documents have been distributed recently to ACRS.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/4/80	Babcock & Wilcox Water Reactors	(TAM) H.Etherington, J.Ebersole, W.Mathis, J.Ray

BACKGROUND, ETC.

The purpose of the meeting will be to complete the Subcommittee review of the Staff study to determine whether construction should be halted on certain B&W plants because of sensitivity of the Once-Through-Steam Generator (OTSG) to feedwater transients. Tom Novak has indicated that the Staff would like an ACRS report on the matter in March. At present no documents other than those available at the January 8 subcommittee meeting are available. (Replies to the October 25 Denton 50.54 letter from WPPS, [1,4] TVA-[Bellefonte], and Consumers-[Midland] were available at that time). By February 15 we expect a copy of Novak's testimony prepared for the SMUD hearing related to OTSG sensitivity. Joe Murphy, Probabilistic Analysis Staff expects to have more data from the IREP Study on Crystal River-3, just prior to March 4, however a written report on the subject is behind schedule and will not be available by March 4. B&W, which did not make a presentation January 8, has asked for time March 4.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/5/80	Regualtory Activities	(SD) W.Kerr, H.Etherington, J.Ray

BACKGROUND, ETC.

Items to be Discussed:

1. Regulatory Guide 1.58, Revision 1, "Qualifications of Inspections, Examination and Testing Personnel for Nuclear Power Plants" (Post comment)

2. Proposed Regulatory Guide 1.XXX, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Copies of these Guides will be forwarded to you as soon as they are available.

Status:

A draft copy of Regulatory Guide 1.58, Revision 1 (Item 1) was reviewed by the Subcommittee at the April 4, 1979 meeting. It was issued for public comment in July 1979. The present version of this Guide reflects consideration of the public comments. The NRC Staff requests ACRS concurrence in the Regulatory Position of this Guide. Subject to the concurrence of the Regulatory Activities Subcommittee, this Guide will be submitted to the full Committee for concurrence with the Regulatory Position of this Guide during the 239th ACRS meeting.

Item 2 is a draft Guide. Subsequent to the review of the Regulatory Activities Subcommittee, the NRC Staff may issue this for public comment.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/5/80	TMI-2 Accident Implications (TENTATIVE)	(RKM) D.Okrent, M.Carbon, W.Mathis, J.Ray

BACKGROUND, ETC.

The purpose of this meeting would be to discussion the studies on proposed additional ESF at the Indian Point Units 2/3 and Zion Units 1 & 2. Memo regarding Task Force Review of IP and Zion for Add'l ESF distributed 1/24/80. It is unclear whether ACRS participation will be requested.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/25/80	AD Hoc Subcommittee on Natural Circulation Heat Removal	(AB) M.Carbon, M.Plesset, H.Etherington, J.Ray, J.Ebersole

BACKGROUND, ETC.

The Subcommittee will meet to review the information presently available on natural circulation and bleed and feed systems. Background documents are being assemblied and will be provided to Subcommittee members. The areas where there is inadequate information available will be examined. Planned programs in NRC Research and test at Sequoyah and North Anna will be reviewed to determine their ability to meet the information needs.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
3/25-26/80	Concrete and Concrete Structures	(EI) C.P.Siess, J.Ebersole, P.G.Shewmon

BACKGROUND, ETC.

Objectives of Meeting

1. To review "user needs" in structural engineering and the way in which these have been and are being met by

    a) research projects in Structural Engineering Branch (SEB)
    b) Technical Assistance Program (TAP) by "user" offices

2. To review past, present, and proposed research programs in SEB, including those related to the SSMRP.

3. To review long-range plans of SEB and their relation to user needs and/or to needs perceived within RES.

4. To explore the procedures, approaches, and philosophy of the SEB as they related to user need requests, priorities, RES initiatives, short vs. long term needs and projects, research by industry, by DOE, or by other federal agencies vs. research by NRC, unsolicited proposals, sole-source contracts (or grants), etc.

5. To consider research needs as visualized by members of the subcommittee and consultants.

Participants

Subcommittee members
Subcommittee consultants: Zudans, Pickel, White
RES: Bagchi, Chief SEB; Shao, AD for General Reactor Safety Research
NRR: Schauer, Chief, SEB
I&E: (possibly Shewmaker)
et al

Information to be Provided in Advance of Meeting

1. Bagchi will be asked to provide:

    a) a summary of current and proposed research projects n SEB-RES, with current and proposed budgets.

(continued)

b) a roster of SEB-RES staff with brief biodata including education and experience in design, construction, or research, together with their present assignments and duties.

c) copies of user need requests covering current and future activities.

d) copies of program plans, RFPs, proposals, etc., related to user needs.

2. Representatives of user offices will be asked to provide information on past and present user needs that have been or are being met by TAPS.

3. All above should be provided at least two weeks in advance of the meeting. It should be presented in as condensed a form as possible with the idea that it can be elaborated on during the meeting.

Appendix IV

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

FEB 6 1980

MEMORANDUM FOR: Chairman Ahearne
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Hendrie
Commissioner Bradford

FROM: William J. Dircks
Acting Executive Director for Operations

SUBJECT: STAFF REVIEW OF THE REPORT BY THE NRC SPECIAL INQUIRY GROUP ON THE ACCIDENT AT THREE MILE ISLAND

The staff has completed a preliminary review of the report of the NRC's Special Inquiry Group (SIG) on the Accident at Three Mile Island. Their analysis is presented in the attached staff report.

The review was conducted from two different perspectives within the staff. The individual offices conducted a review of the technical and policy content of the report from the point of view of their particular line management responsibilities. Simultaneously, the TMI Action Plan Steering Group conducted a review of the report from the point of view of how it compared with the Action Plan.

The Office Directors and the Steering Group met on February 5 to discuss the attached staff report. They have concurred in the revised list of requirements for near-term operating license (NTOL) applicants discussed in Chapter 3 of the staff report. The Office Directors have also agreed that March 1, 1980 is a reasonable deadline for draft 3 of the Action Plan that will include the priorities of the TMI action items relative to the existing operating plan and factor in the recommendations of the report of the Special Inquiry Group.

At our meeting with the Commission on February 7 to discuss this report, we will be seeking two things: (a) approval of the NTOL list and (b) comments on our plan to continue to review the SIG report in the context of the development of the Action Plan.

E Kevin Cornell
for William J. Dircks
Acting Executive Director
for Operations

Enclosure:
Preliminary NRC Staff Analysis of the Report of NRC's Special TMI Inquiry Group and Its Effect on the TMI Action Plan

cc: Office Directors
Steering Group Members
Action Plan Task Managers
SECY
OPE
OGC

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

FEB . 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Victor Stello, Jr., Director
Office of Inspection and Enforcement

SUBJECT: COMMENTS ON ROGOVIN REPORT

We have reviewed both volumes of the Report of the Special Inquiry Group (SIG) -- the "Rogovin Report." Our review was aimed at developing comments on all of the Findings and Recommendations of the SIG. This proved to be a difficult task because of the dispersion of explicit and implied SIG comments and suggestions throughout the report,

Principal IE staff reviewed the report to determine; (1) if any action is needed in the short term for operating reactors; and (2) if additional issues need to be resolved before the "pause" should be lifted. Our review did not identify any such action or issue beyond those already identified, other than increasing the priority on reviewing control room design. This action appears appropriate. Generally, the issues identified in the SIG Report have been noted in previous studies; however, the report's findings, conclusions and recommendations did include variations, some significant, from those in previous studies. Preliminary comments on these issues are presented in Enclosure 1. Analyses of all of the findings, conclusions and recommendations will be coordinated through the IE representative on the Task Action Plan steering group.

There is an underlying theme that seems to be fundamental to all of the studies; namely, the communication, and appropriate followup of important safety information within the nuclear industry. While steps have been taken to improve communications, additional steps for improvement appear to be possible. The consolidation of operating reactor functions into one organizational unit, as suggested by SIG, would be a mechanism to further improve communication of important safety matters. This issue is discussed further in Enclosure 1. I urge consideration and prompt resolution of this issue. It may be desirable to request OMB to include sufficient flexibility in the proposed Presidential reorganization of NRC to accommodate such an approach.

Although the SIG report addressed the subject of improving training and qualifications of personnel, we believe that this subject deserves more than the SIG proposed.

Many proposals for safety improvements have been made in the ten-month period following the Three Mile Island accident. The single proposal that appears to hold the most promise for the future is the industry's effort to upgrade the

PRELIMINARY NRC STAFF ANALYSIS OF THE
REPORT OF NRC'S SPECIAL TMI INQUIRY GROUP
AND ITS EFFECT ON THE TMI ACTION PLAN

1. COMMENTS OF LINE MANAGEMENT OFFICES

The NRC staff offices have reviewed and commented on the report of the NRC Special Inquiry Group. The summary memorandums of the Office Directors are reproduced below. The attachments to their memorandums are provided in Enclosure 1.

training of their personnel (INPO). Now, following assessment of the SIG Report, it appears proper that the Commission undertake a similar task of upgrading and maintaining the technical and managerial competence and safety attitudes of individuals and groups that either caused or prolonged the accident. The Commission should consider establishing an NRC School for Public Safety. Such a school would require a considerable resource commitment; however, it would have significant potential to educate all segments of the NRC staff; other Federal Government officials, State and local officials, senior management and key operating officials in the industry, members of the media, and other individuals and groups supporting or opposing activities licensed by the NRC. This proposal would also help solve a growing safety issue - providing and properly educating the large number of personnel required by the industry and NRC. This major technical and administrative task, well within expected capabilities, holds the potential for being our most important response to the lessons of TMI.

The resource requirements on NRC and the industry arising from all the recommendations from the various study groups are staggering. The normal operation of a nuclear power plant requires day-to-day contributions from a large number of well trained and experienced engineers and technicians. In the aftermath of Three Mile Island, these engineers and technicians have been heavily involved in many new activities that have diluted their capabilities for accomplishing normal day-to-day tasks as well as for completing the new tasks. We are concerned about this growing problem and recommend that each new NRC staff requirement be weighed against this negative safety impact. The solution appears to be in presenting to the industry a balanced "Action Plan" that utilities can properly understand and plan for. This certainly includes allowing time for hiring, training and for obtaining necessary in-plant experience for plant staffs. This action is planned by the Commission but the findings from the recent Commission-directed review of eight nuclear facilities, and from our senior IE managers, indicate we should consider moving even more slowly in imposing requirements that demand substantial in-plant resources and which may have limited short-term safety benefits. Guidance from the Commission on this matter is desirable because it affects several offices and functions within the NRC, many of which have hundreds of recommendations to evaluate and, perhaps, to impose. Almost all of these recommendations impact to varying degrees on both NRC staff and in-plant engineers and technicians. We recommend that the Commission encourage the industry to establish a system to facilitate interfacing with NRC in assessing the impact of these new requirements.

The tone of the SIG analyses presented in Volume I suggests that action or inaction of the Commission is a major underlying cause of the accident. In particular, the critical tone of the Epilogue is understandably upsetting. We believe that the Epilogue does not accurately portray the status of the NRC or the industry ten months after the accident. In those instances in which there is acknowledgement of positive post-accident activities, they are presented in a context that belittles their significance.

Volume II partially offsets this criticism. Clearly the Commission and its staff must share some of the blame for the accident. Vendors, owners and operators of the facilities must also share the blame. Undue emphasis on action needed by the Commission must be avoided so that others in the nuclear industry also recognize the need for reform. More attention to design, analyses, fabrication and operations are all needed. We believe that a brief overall Commission statement consolidating the results of all of the studies is needed. This statement should recognize that the accident was the result of inadequacies in the regulatory, designer, owner and operator institutions. Findings, conclusions, and recommendations from these studies should be accepted and programs (perhaps the Task Action Plan) for their implementation should be identified.

Considerable resources have been devoted to studies of the Three Mile Island Accident. These studies have examined the cause of the accident, and identified flaws in the industry and regulatory processes that contributed to the conditions that permitted such an accident to occur. All of the studies have concluded that the accident was serious but that the basic defense in depth concept prevented this accident from having serious health effects. These studies include many recommendations that, if implemented, will reduce the probability of another accident. Many of these recommendations have already been implemented. Consolidation of other recommendations is in process in the Task Action Plan.

Decisive action by the Commission will be needed on the Task Action Plan to set the policy and programmatic guidance for the staff and the industry. Timely decisions are needed to bring stability and predictability to the nuclear industry and to the NRC staff. We can never put the Three Mile Island Accident behind us until these decisions are made and implementation is well underway. We are confident, however, that when these decisions are made, the safety of nuclear facilities will be substantially improved; public confidence in NRC can be improved; the concerns of the public, the industry and the NRC can be effectively addressed; and the preoccupation with the aftermath of Three Mile Island can be relegated to a more realistic position in the list of priorities for the attention of all these groups.

Victor Stello, Jr.
Director
Office of Inspection
and Enforcement

Enclosure:
As stated

cc: H. R. Denton, NRR
J. G. Davis, NMSS
R. B. Minogue, SD
R. J. Budwitz, RES
N. M. Haller, MPA
R. J. Mattson, NRR

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

February 4, 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: COMMENTS ON NRC SPECIAL INQUIRY GROUP REPORT

I have reviewed the report of the NRC Special Inquiry Group (SIG) and have had the benefit of comments provided to me by the NRR Division and Task Force Directors. In general, it is our conclusion that most of the problems and issues discussed in the SIG report have previously been identified and highlighted by the other investigations and reviews conducted since TMI. Thus, the most important contribution of the SIG report is that it corroborates and reinforces to a considerable degree both the general thrust of other reviews and the related corrective actions currently underway in the NRC and industry.

The findings and recommendations of the SIG will in the future be reviewed by my staff and me in much more detail than has so far been possible. However, almost all of the SIG recommendations dealing with NRR reactor licensing programs appear to be generally consistent with the TMI Action Plan. The Action Plan Steering Group and its task managers (many of whom are from the NRR staff) are presently reviewing the SIG recommendations in detail to identify those that warrant changes in the Action Plan. I will, of course, review and comment on the Steering Group's recommendations when they are available. Many of the SIG recommendations are more detailed and prescriptive than the Action Plan items; the SIG details should be considered by the people who will carry out the plan.

I have noted three of the SIG recommendations that I believe merit particular consideration for immediate implementation on near-term operating licenses:

- In commenting on the suspension of licensing reviews, the SIG recommended that every applicant for an OL examine the control room to identify outstanding human factors deficiencies and any instrumentation problems, and that NRC should conduct field inspections to determine whether the applicant's self-examination was adequate. NRR is already planning to implement this recommendation for the Sequoyah and other near-term OL reviews pending issuance of criteria for use by other OL applicants with plants under construction.

- In Volume 2, Section I.E., the SIG recommends increased NRC scrutiny of the power ascension test program and the listing of such tests in the FSAR to prevent any compromising of safety. We believe such increased scrutiny is particularly important in view of the proposed expansion of startup test programs and the economic incentives to achieve the already delayed commercial operation of new plants. We will work with IE to assure that this potential problem is watched closely.

- In Volume 1, Page 127, the SIG recommended that immediate improvements could be made in control rooms by installation of the equivalent of a "reactimeter" in every plant. I believe that the provisions for plant safety status instrumentation in the Action Plan are a better, but longer range, requirement. However, if "reactimeter" hardware and associated software are, in fact, readily available in sufficient quantity, they could provide worthwhile interim improvements.

In some important respects the SIG recommendations are different from those already reflected in the NRC draft Action Plan. My comments on these are given below:

- The SIG report recommends consolidation of "all the agency's resources devoted to monitoring the safe operation of existing reactors in a single office -- probably the current Office of Inspection and Enforcement." I agree that improvements are needed in the interactions and communications between NRR and IE, but it is equally important, in my view, to maintain the checks and balances that are provided by the present separation of licensing and inspection functions, to maintain and improve the feedback from operating experience into design reviews and vice versa, and to minimize the duplication of technical centers of excellence. Exchanging one set of troublesome interfaces for another does not necessarily improve the strength of the overall organization. Although many organizational structures can be made to work, given proper management, for given specific missions and priorities and available personnel, some structures may work more easily and efficiently than others. As we move forward to implement the TMI Action Plan and approach the end of the licensing pause, it is important that we maximize the efficiency of our organization to deal with this intensive workload. With this in mind, some weeks ago I proposed a reorganization of NRR that I hoped to put in place by the time Commission decisions were made regarding the Action Plan and the resumption of licensing. The SIG recommendation, if adopted by the Commission, would obviously affect the much needed NRR reorganization. An indefinite period of uncertainty regarding the SIG recommendation would, to the extent that it delays the restructuring of NRR, further postpone an effective NRR organization compatible with the pertinent Lessons Learned and Presidential Commission recommendations. These considerations argue for a prompt Commission decision on whether or not to implement the SIG recommendation. On the other hand, the potential advantages and disadvantages of the recommendation warrant careful and deliberate consideration before making such an important decision. Therefore, given the need for careful consideration of this far-reaching recommendation, and the potentially disruptive and morale-degrading effects of an indefinite period of uncertainty, I recommend that, within the next few weeks, the Commission make an explicit determination that, pending further consideration, no decision on the SIG reorganization recommendation will be made for a definite period of time (say 18 months), and proceed to approve the needed reorganization of NRR as soon as possible.

- Two of the SIG recommendations concern the establishment of an Independent Nuclear Safety Board and the procedures and membership of the Regulatory Requirements Review Committee. Many of the underlying objectives of these recommendations could, in my judgement, be accomplished by the Division of Licensing Requirements proposed to be established in the NRR reorganization.

- The recommendations of the SIG report concerning the potential advantages of a consortium for operation of some reactors and the institution of a mandatory one-step licensing process merit careful consideration by the Commission and the industry but do not seem to warrant immediate decisions.

In the enclosure I have provided some additional comments by my senior staff. Although I have not attempted to provide a detailed review and comments on the many insights and recommendations contained in Volume 2 of the SIG report, they deserve and will receive careful consideration over the next several months.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Additional Staff Comments

cc: V. Stello
R. Minogue
J. Davis
R. Budnitz
R. Mattson

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

FEB 4 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Ronald M. Scroggins, Director
Administration & Resource Control Staff
Office of Nuclear Regulatory Research

SUBJECT: COMMENTS BY THE OFFICE OF RESEARCH ON THE NRC's TMI SPECIAL INQUIRY GROUP REPORT (ROGOVIN REPORT)

The results of the current RES review of the NRC's TMI Special Inquiry Group (SIG) report is being forwarded from me, instead of R. Budnitz, Director, RES, because of the potential conflict of interest (Ref. Memo R. Budnitz to L. Gossick dated January 21, 1980). The comments contained in this memorandum represent a consensus summary of the SIG report by the RES senior management. In general, RES believes that the SIG did a credible review and examination, and we are in general agreement with their diagnosis of the problems in the industry and NRC. Also, there is general agreement on the finding and recommendations of the SIG. In this regard, RES management supports the SIG recommendations for a strong executive function to direct and control the day to day operations of the agency.

We would like to note that a number of the significant recommendations of the Rogovin Report have been acted upon by the agency in both the short- and long-term NRR Lessons Learned Reports, the NRC Response to the President's Commission, the I&E Special Investigations, the Bulletins and Orders Task Force Report, and in the development of the RES FY 1981 program and budget request. Also, most of these recommendations are being covered in the TMI Action Plan, and a in the case of RES, there has been a general reorientation of priorities. While we believe that there exists generally throughout the agency a strong sense of awareness of the problems of the past and an urgent desire to correct them and not to have our efforts degenerate into the "business as usual" attitude of the past, it is clear that continuing effort on the part of the program offices is required to ensure improved interoffice cooperation and coordination.

Our following comments relate to some of the general recommendations of the report, especially as they relate to RES, and will not touch on some of the specific technical aspects raised in the report.

i. The Rogovin Report recommends that "present NRC staff functions devoted to performing quantitative risk assessment of reactors should probably be relocated in AEOD," (Office of Analysis and Evaluation of Operational Data). The implication of this would be to combine RES's Probabilistic Analysis Staff (PAS) and AEOD. We disagree with this recommendation. Although the collection and analysis of data does correlate well with some aspects of the risk assessment function, we see a number of possible deficiencies arising which could outweigh any gains of such a merger. Primarily, AEOD's effort will be on operating reactor experience, which could dilute PAS efforts on reliability engineering, probabilistic analysis and the application of risk assessment techniques to other areas, such as siting/consequence modeling, fuel cycle risk, and transportation risk. Also, methodology development could suffer, along with PAS's role to educate other Offices on the use and applications of the technology. There exists the possibility of some overlap in functions between the AEOD and PAS in the early stages of operation of the AEOD; however, we feel this early overlap will be worked out as we assure together that all the important areas of data evaluation and interpretation are covered.

Another recommendation in this area was the "AEOD Office should be staffed in part on a rotational basis from all the other offices and branches of the NRC staff, at a level of no less than 35 to 40 professionals." We feel that the AEOD Office should have a permanent, dedicated staff in order to operate most effectively. The idea of a rotating staff sets up the possibility of "mixed loyalty" for individuals who know they're only on loan for a short time period.

ii. It was recommended that quantitative risk assessment techniques be used more and that more emphasis on human factors be included in the design review process and in other areas of the licensing process. Also, the SIG recommends that the spectrum of the design basis accidents used for safety assessment be expanded by using operational experience, research results, lessons learned from accidents, and advice from the ACRS, all studied through the use of risk assessment. Additionally, risk assessment could help the agency to establish a safety objective for nuclear power plants. We agree with these recommendations and strongly support the use of quantitative risk assessment methodology throughout the decision making process, such as in establishing priorities for the research programs.

iii. It was recommended that the NRC retain the Executive Management Team (EMT), but that it should have a single director who would exercise the authority of the entire agency during an emergency. The SIG also proposed that FEMA and other Federal agencies involved should have senior representatives present at the NRC Incident Response Center during an emergency. We believe that this is an area of immediate concern for the Commission and that the NRC should develop a position that defines its authority and responsibility during an emergency. What will be the structure of the EMT, what authority can and cannot be delegated to whom, etc? This position statement should include a definition of NRC's authority at the site under projected emergency situations.

iv. The Rogovin Report suggests that a Nuclear Safety Board be established "to be responsible for observing, evaluating, and making recommendations to improve the quality of the overall performance of the regulatory staff." It further suggests that this Nuclear Safety Board consist of five full-time members of the Advisory Committee on Reactor Safeguards (ACRS). We agree with the intent and objectives of such a board; however, we disagree with the proposed board membership. We feel this suggestion would inevitably fragment the ACRS. The ACRS would lose its collegiality as power would flow to the five members of the Board, who would have greater control over the supporting staff and more extensive contacts with the licensing staff. We believe, the ACRS could, as presently constituted, satisfy the needs outlined for the proposed Nuclear Safety Board if closer attention were paid to their recommendations by the Commission.

v. We concur in the objective of the recommendation to rotate the senior staff throughout the various Offices of the NRC to gain greater breadth of experience and foster an agency-wide attitude. We believe it would have been beneficial to both the agency and the staff if various senior managers had been rotated to other Offices of the agency. We are not recommending any prescribed formula for this rotation of senior staff, but the plan should be consistent with the Charter of the Senior Executive Service. The number of staff involved and the period of time could vary as necessary. This is a recommendation which should be pursued further by the agency, as it would afford the senior managers a better perspective of the overall agency operation and could help to foster a closer cooperation among Offices.

vi. We strongly support the recommendation for a single location for the agency. The present physical separation of RES from NRR, IE, ACRS and the Commission greatly inhibits communication between organizations and people that should be in daily touch with each other. Certainly we feel it would enhance our ability to transfer research results and to be closer in line with the licensing and regulatory needs of the agency. Better communication through closer contact should help in the overall efficiency of the agency in meeting its goals. The Commission and staff should do as much as possible to have a single location found for the agency as soon as possible.

vii. Last, but not least in importance, is the recommendation for a good staff training program in reactor power plant design, construction and operation and in problems of radiation protection. We agree with this recommendation and feel it could be very beneficial to RES and other Offices of the agency. The training would help to broaden the technical expertise of the staff, many of whom have backgrounds in highly specialized technical areas, and give them a better perspective of some of the licensing and regulatory issues. The training should provide a systems approach to reactors, such that individuals concerned with certain components could develop a feel for how the failure of certain components effect various systems and the overall operation of the plant. This idea of a systems approach to safety is important and should be emphasized.

Staff comments were solicited and carefully considered as part of RES review of the SIG report. As noted earlier, this memo represents a consensus of senior office management. Copies of the individual staff comments are available. Please contact me, if there are any questions on the above comments.

R M Scroggins

Ronald M. Scroggins, Director
Administation & Resource Control Staff
Office of Nuclear Regulatory Research

cc: N. Haller, MPA
H. Denton, NRR
V. Stello, IE
R. Minogue, SD
H. Shapar, ELD
R. Mattson, Director
TMI Action Plan Steering Group

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

February 4, 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Robert B. Minogue, Director
Office of Standards Development

SUBJECT: STAFF REVIEW OF REPORT OF THE NRC SPECIAL INQUIRY ON TMI-2

In accordance with the EDO memorandum dated January 21, 1980, the SD staff and I have reviewed the subject report. Because of the length of the report, and the fact that much of the substance is in the rather unstructured second volume, I found it necessary to focus our initial review in the following way:

(1) Volume I was reviewed for general policy questions, and

(2) Both Volumes I and II were reviewed for programmatic questions related to:

(a) items that might affect SD's Congressional testimony, i.e., broad program planning for SD,

(b) items that might affect the post TMI-action plans now being developed, and

(c) items that might affect rulemaking activities currently in progress.

To assist in this effort, I had a senior SD person who was on the Special Inquiry Group review the entire report and make up a "Road Map" to identify areas that may potentially impact SD's program. As background information, the list he developed is appended as Enclosure 2.

In the general policy area, some of my reactions are as follows:

1. The report takes little note of the fact that the NRC regulates a number of nuclear activities which are quite different from reactor operations and which involve substantial public health risks. This becomes a serious flaw when the report speculates about improved organizational structures.

CONTACT: Robert B. Minogue
443-5936

2. The report recommends replacing the Commission with a single Administrator. Underlying this recommendation is the perception of a pressing need for more effective day-in day-out executive management of agency operations. I agree that there is such a need. However, I do not agree with the premise that effective management of operations is inherently incompatible with a Commission structure and that strong executive direction cannot be obtained by a structuring of agency operations for tighter management which retains the important contribution that a diversely constituted Commission makes to the formulation of balanced and thoughtful public policy in such a complex and controversial area. I think the agency would lose a great deal if it lost the forum for public discussion from diverse viewpoints of sensitive important issues which the present Commission provides with its dedication to open operations.

3. A number of specific technical recommendations are made in Volumes I and II; many are of great merit and should be taken into account in the formulation of action plans. One which I would particularly support is that part of the first recommendation in Volume I which speaks to systematic evaluation of operating experience. It is imperative that the agency and the industry promptly upgrade and expand their programs to gather and to assess operating experience so that a better understanding of the operating and other characteristics of current nuclear power plants can be developed and fed back into the design, construction, and operation of these plants to enhance their safety.

4. The report recommends funding of intervenors who "contribute materially". I support the concept of intervenor support but believe that the criteria should be more liberal, for example only that the intervenor has a substantially affected interest and that he could not otherwise afford to participate. I also support the creation of an Office of Public Counsel. To be effective, such an office would need a technical as well as legal staff independent of the NRC regulatory groups. Because of the uncertainty of the type and quantity of such staff needs, this independent capability, initially, might be best achieved by supplementing a small permanent staff with outside consultants as required. Access to the services of such a group would be, as I see it, the major mechanism for providing support to intervenors.

5. An important recommendation of the report is to structure a much more effective program of inspection and enforcement. (And, I might add, one which takes into account the realities of the industry which designs, constructs, and operates the facilities which we are

tasked to regulate.) Inspection and enforcement programs, to be effective, must face the issue of "inspect against what?" The transfer of DOR to I&E which the report recommends appears to be intended to come at this problem; it is but one way, and perhaps not the best. There are others as well, -- improved industry standards (i.e., more enforceable standards like ASME Section III); programs of certification of procedures, laboratories, and personnel; more attention to stating essential licensing decisions in enforceable form; more attention to enforceability in regulations and guides. These are all necessary to give the inspection and enforcement staff the tools it needs.

Most specific technical issues raised in the report had already been identified in the Task Action Plan and, to the extent that these issues broadly affected SD's program, they had already been considered. Consequently, no further changes in SD's Congressional testimony are required as a result of the subject report.

Specific standards development actions related to the Action Plans are being developed in parallel with and through participation in the work of the Steering Group. We have concluded from our review of subject report that the crucial items affecting SD will be considered in TMI-2 Steering Group and NRR reviews, with which we are quite familiar, and the revised Plan will include consideration of the specifics of the report.

Regarding the recommendations in the report that could affect rulemaking that is presently being pursued, I have the following comments:

1. Appendix E Revision (Emergency Plans) - Part 100 Revision (Reactor Siting). These topics are inherently closely related. In particular demographic factors can be treated in site assessment, in emergency planning, or in both. Both of these rulemaking activities involve demographic considerations such as accident scenarios affecting people and the effectiveness of emergency actions and must be closely coordinated.

   (a) The report recommends that future reactors should be located only at sites that are at least 10 miles, and perhaps more, from any significant center of population.

   We are in complete agreement that a minimum distance from centers of population should be stipulated in NRC's siting regulations. This recommendation is virtually identical with one of the changes in Part 100 recommended last August by the Siting Policy Task Force.

We are preparing an Advance Notice of Proposed Rulemaking addressing amendment of the NRC reactor site criteria (Part 100) in accordance with the Siting Policy Task Force recommendations. However, in order to implement the recommendation effectively, it is necessary to specify what is meant by a "significant" population center in terms that are measurable and capable of being applied in the regulatory process. The principal factor in arriving at the specification of a significant population center is the diminished capability for taking effective emergency action as the population density exceeds some limiting value.

The distance of "at least 10 miles" to a significant center of population is similar to the recommendation of the Siting Policy Task Force and entirely consistent with the proposed emergency planning rules recently published for public comment and discussed with State and local government officials, utility representatives, and the public at four regional workshops. Any limitation in distance implies acceptability of some non-zero residual risk for population centers beyond that distance.

(b) The report recommends that specific criteria for reactor siting should be developed promptly by the NRC in conjunction with other Federal and State agencies with experience in emergency evacuation, and that consideration should be given to the specific characteristics of the area that influence the effectiveness of evacuation, such as population density, population centers beyond 10 miles, and evacuation routes.

As noted above, the staff is developing the technical bases for revising Part 100 to include demographic criteria, and the bases will include consideration of the factors affecting the capability to take effective emergency action. Since State and local governmental officials are the persons responsible for planning and executing actions, their participation in the development of the technical bases is essential.

It is our intent to develop demographic criteria that are generic and not "tailor made" to each site as the report recommendation implies. However, each site would need to be reviewed to assure that the characteristics important to effective action are within the bounds used in developing the generic criteria. Further, following the evaluation of the site against generic criteria, both the utility and state/local emergency plans as well as the utility emergency implementing procedures will consider site specific characteristics. "Fine tuning" for special site and regional factors which might inhibit protective actions would be considered in the review of emergency plans.

(c) With respect to the general thrust of Chapter 6 of Volume I which deals with Emergency Planning, I agree that the emergency planning area must be upgraded, and NRC has taken several actions to do this which have not been adequately reflected in this report. The Commission presently has in progress a major rulemaking. As part of this rulemaking process, the Commission staff has held several workshops to get feedback from public, State, and local authorities on this proposed rule. Items identified in the report, such as funding support for preparation of plans, FEMA approval of State/local plans, and the appropriate distance for emergency planning zones, are all similar to a number of comments received and discussed during these workshops. The staff will consider these comments and others made in the report as the final rule is developed for consideration by the Commission.

Further, the NRC and FEMA staff are conducting a series of site specific reviews of utility and State/local emergency preparedness. Also, the NRR staff is conducting specific analyses of protective features such as filtered containment venting and core ladles for a few sites which are located near densely populated areas. Both the Commission and its staff are committed to significantly upgrade the requirements for nuclear power plants related to emergency planning.

Recently, NRC and FEMA have completed a Memorandum of Understanding in this area and a joint staff group has prepared the specific criteria which will be used in assessing the adequacy of utility and State/local emergency plans.

Several detailed errors of fact appeared in the report. For example, the proposed rule would require <u>concurrence</u> by NRC of State/local emergency plans as a condition of nuclear power plant operation rather than approval. Also, the requirement for concurrence would apply on or about January 1, 1981, not July 1, 1981, and the Commission could grant <u>exemption</u> under certain conditions - not exceptions.

2 Part 21 (on which the books are just being closed on a reassessment). Volume II, Part 1, page 37 of the report, in addressing 10 CFR Part 21, "Reporting of Defects and Noncompliance," states this regulation is ambiguous regarding its applicability to architect-engineering firms and to information based on experience with a reactor located outside the United States. Page 79 of the report states that "because the NRC regulations do not apply directly to licensees' vendors and contractors, they are not subject to enforcement actions ..."

Part 21 is clear that it imposes enforceable reporting requirements on the "responsible officers" (the wording of the statute) of an architect-engineering firm. Part 21 also clearly includes some implementing provisions applicable to architect-engineering firms but, because of the quote cited above from page 79 of the report, NRC cannot enforce compliance with these implementing provisions.

Part 21 was deliberately silent in regard to the source of the information (foreign or domestic) but purposefully related the required reporting to the effect of the information on facilities and activities "within the United States."

We believe there may be points of concern regarding Part 21 which are not stated in the report, and when the author of that section of the report returns from extended leave, we intend to inquire further. In the meantime we are advising interested parties that there may be a further reexamination of Part 21.

3. Improved rulemaking procedures. The report recommends improvements in rulemaking procedures, including designation of an organization to have primary responsibility in the rulemaking area (Volume II, Part 1, page 41). Executive Order 12044 and legislation currently being considered by Congress speak to a number of improvements in rulemaking procedures. In recent meetings, the Commission has discussed an OGC analysis of regulatory reform legislation and an OPE/OGC staff paper "Review of Delegation of Authority" dated October 4, 1979. The Commission has decided to become more involved and at an earlier point in major policy rulemaking and to delegate the more routine, technical rulemaking to the Director of Standards Development. It seems to me these actions by the Commission meet and go well beyond the recommendations of the report.

I have included some additional comments by SD staff in Enclosure 1.

Robert B. Minogue, Director
Office of Standards Development

Enclosures:
1. Additional comments
2. List of areas that may potentially impact program

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

FEB

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: William J. Dircks, Director
Office of Nuclear Material Safety
and Safeguards

SUBJECT: NMSS COMMENTS ON REPORT OF THE NRC SPECIAL INQUIRY ON TMI-2

We have reviewed the report of the NRC Special Inquiry Group. Our views on the recommendations of the report are enclosed.

William J. Dircks, Director
Office of Nuclear Material Safety
and Safeguards

Enclosure: As stated

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

FEB 4 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Howard K. Shapar
Executive Legal Director

SUBJECT: REVIEW OF REPORT OF THE NRC SPECIAL INQUIRY ON TMI-2

As requested in the January 21, 1980 memorandum of the Executive Director for Operations, we have reviewed the report of the NRC Special Inquiry Group--to the extent feasible within the relatively short time available--and formed some preliminary impressions set forth below. Because of time constraints, we have not attempted to duplicate the work of the major programmatic offices which, we expect, will focus in detail upon the more technical portions of the report. (We will, of course, perform a legal review of the work product of those offices.) Rather, our review has concentrated upon the major legal and policy issues which stand out in the report.

As a preliminary matter, some overall perceptions should be pointed out. The report of the Special Inquiry Group closely parallels the report of the President's Commission, both in major conclusions regarding reorganization of the NRC, and in many of the recommendations concerning the need for specific safety improvements. There do not appear to be any startling revelations in the new report, unheralded by the work of the President's Commission. At the same time, we do not wish to minimize the value of the detailed backup analyses in Volume II of the report. These appear to be thorough and, for the most part, competently done, and should contribute to improved understanding of the accident at Three Mile Island, its causes, and the corrective actions which remain to be accomplished.

An additional overall impression, unfortunately, is that the report suffers from inadequate analysis in many places. Conclusions and recommendations are not always shown to be related, directly or indirectly, to the supporting material in the report. Articulation of the reasoning process which leads from a particular finding or conclusion to a recommendation is not always shown. Specific examples of these failings will be discussed in connection with some of our more specific observations which follow.

Our final observation of a general nature is that Volume II is not organized in a manner that permits ready access to particular portions of its contents. Recommendations, for example, are scattered throughout the three parts of Volume II, sometimes being listed in the table of contents and sometimes not. While this is consistent with the authors' hopes that the report be read in its entirety (Volume I, Foreword), it makes use of the document extremely difficult by the researcher interested in specific issues. (This difficulty has been somewhat alleviated by the availability of the compilation of Volume II recommendations prepared by the TMI Action Plan Steering Group.)

SPECIFIC COMMENTS - VOL. I.

1. Chartering of a national operating company or consortium. (Vol. I, pp. 109-111.)

This recommendation of the Special Inquiry Group is illustrative of the failure to relate the recommendation to the supporting discussion in Volume II. While the need to eliminate the "wide spectrum in the capability of the various nuclear utilities to operate existing plants in a safe fashion" and attain a uniformly high standard of excellence can readily be discerned from the report, we have not found any discussion in Volume II of the pros and cons of the means proposed to attain this objective-the operating consortium. Considering the radical (though, as noted on p. 111, not original) nature of the proposal more extended discussion of its merits and feasibility is warranted.

This recommendation is also based in part on the erroneous premise that "the only weapons in the NRC's regulatory arsenal [to deal with inadequate technical and managerial capability] are rather trivial fines on the one hand, or the authority to close down a plant on the other." This is simply wrong; the NRC has a host of intermediate remedies through the vehicle of license amendments. Finally, given the premise (erroneous as it is), one has to wonder why the special inquiry group did not discuss less drastic solutions, such as obtaining legislative authority to levy greater civil penalties. In fact, it is likely that NRC will soon have such authority.

2. Improved NRC Management and Reorganization to a single administrator agency; Establishment of an Independent Reactor Safety Board (Vol. I, pp. 112-121).

The Commission, the Administration, and some members of Congress took strong positions on the merits of the proposal to replace the five-member Commission with a single administrator after it was made by the President's Commission. Though the Special Inquiry Group has made the same recommendation, it has not presented new arguments which call for a different reaction. Nonetheless, several observations on this chapter of the report are in order.

In an apparent effort to support the recommended organizational changes, the report contains some overstatements which should be noted: (1) at p. 115 it is observed that "the present Commission does not involve itself in [the licensing] process anyway," and (2) at p. 118 is the finding that "there is really no existing organization within the agency that has either the responsibility for or the capability of monitoring the effectiveness of the regulatory staff and of making recommendations for actions needed to establish and maintain a safety review process of the requisite level of quality." The first of these statements is incorrect, though there is obviously room, as the Commission has recognized, for greater Commission involvement in the actual licensing process.

The second statement, made in support of the recommendation for an independent Nuclear Safety Board, is also incorrect-clearly the Commission itself has the "responsibility for . . . monitoring the effectiveness of the regulatory staff" and also the "responsibility for [initiating, not recommending] actions needed to establish and maintain a safety review process of the requisite level of quality." Perusing the statement, and the remaining arguments for a Nuclear Safety Board, a little more closely, however, it becomes difficult to distinguish the proposed Board from the present Commission (except for the feature that Board members would also be ACRS members); the distinction appears possible only if we assume (as have the authors) that the Commission has been replaced with an Administrator.

Conversely, the proposal for a single administrator "with the clear authority to supervise and direct the entire NRC staff" has not been developed sufficiently to demonstrate that the perceived need could not be accomplished (as currently proposed) by strengthening the authority of the Executive Director for Operations. In sum, the report has presented an alternative for reorganization involving replacement of the present Commission with a single administrator and creation of a new Nuclear Safety Board. It has not, in our view, presented a convincing case that the desired objective of enhanced management capability and efficiency cannot be largely or entirely accomplished with the Administration's proposals of December 7, 1979 with respect to improvement of NRC organization and management.

3. Overhaul of the licensing process: One stage licensing, increased standardization, increased use of rulemaking, establishment of an office of public counsel, and intervenor funding (Vol. I, pp. 138-146).

The procedural recommendations of this section are quite similar to those of the President's Commission though the basic premise is much more starkly stated by the Special Inquiry Group:

> Insofar as the licensing process is supposed to provide a publicly accessible forum for the resolution of all safety issues relevant to the construction and operation of a nuclear power plant, it is a sham. (Emphasis added.)

While some of the proposals to remedy this situation appear to be generally meritorious, the analysis in the report contains some inconsistencies. Greater public participation, for its own sake, seems to have overshadowed observed deficiencies, specifically the consensus that "the formal licensing process does little to enhance the quality of reactor safety" and the view of some (including the ACRS) that "these formal proceedings discourage applicants and the NRC staff from dealing candidly with all sides of controversial safety issues in their analyses and evaluations." Having raised the problem, referred to by many as "overjudicialization" of the licensing process, the report does not fully come to grips with it.

Another inconsistency in this section of the report is found in the recommendations--contained in the same paragraph--that the ACRS play a more formal role as a party in licensing and rulemaking proceedings while, at the same time, reducing the time commitments demanded of its members.

Finding that three levels of appellate review is "completely unnecessary" the Special Inquiry Group recommends abolition of the Appeal Board if the Commission is retained, requiring the Commission itself to consider and finally approve every new reactor license. (Conversely, if the Commission were replaced with a single administrator, the report recommends retaining the Appeal Board which would be the final licensing authority.) The further suggestion is made that if the Appeal Board is abolished, its "members could be transferred to a support office to assist the Commission in this work, which would permit the outstanding quality reflected in past Appeal Board decisions to be perpetuated in the decisions of the Commission." This may be extremely naive. In our view it is doubtful that many of the present Appeal Board members, who enjoy outstanding professional reputations and occupy positions of recogized prestige, would accept transfer to a role comparable to that of a law clerk. Beyond that, the proposal implicitly recognizes that the Commissioners themselves will <u>never</u> actually sit down, review records and write licensing decisions; they simply have too many other responsibilities.

The recommendations of the report dealing with an office of public counsel and intervenor funding, while of particular interest to this office, do not require extended discussion. The Commission has already sought funding for a pilot program of funding for intervenors. The TMI Action plan calls for a study of the concept of an office of public counsel. In this regard, we notice that the public counsel proposal of the Special Inquiry Group is somewhat more clearly articulated than is the counterpart proposal of the President's Commission.

The proposal for one stage licensing "in which design plans that are as detailed as possible should be considered and approved" is not new--the basic idea, on an optional basis, is reflected in proposals for "licensing reform", which have appeared during the past few years.

SPECIFIC COMMENTS - VOL. II.

4. <u>Vol. II, Part I, p. 0024</u>

The report misleadingly states that "A substantial array of other licensing actions taken by the staff typically neither go to hearing nor receive review by anyone outside the Office of NRR," listing several examples including license amendments. Many license amendments--particularly those involving major safety questions--do indeed go to hearing. Beyond that, many actions taken by the Director of NRR within his delegated authority are nonetheless reviewed by other offices including OELD and, as appropriate, the ACRS.

5. Vol. II, Part I, pp. 0041-41.

The report recommends that "an organization should be designated to have primary responsibility in the rulemaking area to assure that the quality of the regulations are adequate." The premise for the recommendation appears to be the finding (at p. 0036) that

> The regulations are almost completely lacking in any criteria relating to the operational aspects of nulcear reactor safety. Moreover, the regulations do not contain well-defined safety criteria and requirements. Many are ineptly drafted-some to the point of being virtually incomprehensible. Others appear to be of questionable merit in view of the changes that have occurred since their publication. Still other regulations have quite obvious gaps. No organizational unity [sic] is charged specifically with the responsibility of assuring that the regulations are adequate, or alerting the NRC to problems in the regulations themselves.

We can readily agree that the quality of the regulations needs improvement, but we do not believe that creation of a new organizational element is the appropriate way to achieve this objective. Since quality of regulations reflects technical, legal, policy and editorial considerations, the proposed new quality control group would be largely duplicative of the very considerable resources already involved in rulemaking (including SD, OELD, one or more affected "program" offices, Commission staff offices, and the Commission). Moreover, public comment on proposed rule changes will have a beneficial effect on quality.

We believe that the intent of these recommendations can be more satisfactorily accomplished through the process contemplated in Task IV.E of the TMI Action Plan which includes (1) development of a public agenda for rulemaking, and (2) a periodic and systematic reevaluation of existing rules. Dedicated effort at accomplishing this task (with adequate resources as set forth in the Action Plan) will do a great deal to improve the quality of our regulations.

6. Vol. II, Part I, p. 0042. Abolish limited work authorizations.

The first recommendation on page 0042 calls for abolishing limited work authorizations, along with the two-stage licensing process and the immediate effectiveness rule and "replaced with a system that provides incentives for more design and siterelated safety and environmental issues to be resolved before construction begins" (emphasis added). The recommendation is based on the erroneous premise that construction can begin under the present rules before all environmental issues have been resolved. This is not correct, even in cases where limited work authorizations are issued. 10 C.F.R. 50.10(e)(2).

7. <u>Vol. II, Part I, p. 0190. Combining I&E and Division of Operating Reactors.</u>

The Special Inquiry Group recommends that the organizational separation that exists between I&E and NRR "be reduced by integrating I&E and the Division of Operating Reactors into a single group." While we think that there may be a number of organizational options which would help to correct the problems caused by the separation, the particular option recommended probably cannot be accomplished without legislation (unless all of I&E were moved into NRR). This is so because NRR is an office specifically established by the Energy Reorganization Act of 1974 to perform the "principal licensing and regulation" of nuclear reactors. It thus appears likely that the Division of Operating Reactors cannot be removed from NRR without a statutory change. In any event, we believe that serious consideration should be given to various other organizational changes to accomplish the desired objective.

8. <u>Vol. II, Part I, p. 0355. Establishment of a "Financial Analysis" office.</u>

The recommendation that NRC should establish an expanded Financial Analysis office to monitor situations in which business considerations may impact on nuclear safety is based on a very weak and incomplete discussion. The merits of the proposal cannot be seriously debated until more adequate investigation of the problem and exploration of alternatives is undertaken.

9. <u>Vol. II, Part I, p. 0358. "Legal ambiguity regarding the status of the FSAR."</u>

The report identifies a question as to whether a licensee can delete tests listed in the FSAR without NRC approval and if so (i.e., if permissible under 10 C.F.R. 50.59) whether NRC is informed after the fact. The recommendation is that "the NRC should--at a minimum--be informed of that decision." If other offices agree that the recommendation is sound, as it appears to us at first blush, it can be quickly and easily implemented with a minor rule change.

Howard K. Shapar

Howard K. Shapar
Executive Legal Director

A-44

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

FEB 4 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: C. J. Heltemes, Jr., Interim Director
Office for Analysis and Evaluation of Operational Data

SUBJECT: AEOD COMMENTS ON THE NRC SPECIAL INQUIRY GROUP (SIG) REPORT

The SIG Report emphasizes the need for and importance of an effective program to assess operating experience and to feedback the lessons of experience to the NRC licensing, standards, and inspection activities and to NRC licensees. In this regard, the SIG made a number of recommendations regarding the authority, size, and scope of AEOD and had several suggestions for improvement in the reporting of operating experience. The principal recommendations concerning AEOD were:

(1) AEOD recommendations for actions should be required to be rejected, modified, or imposed as recommended by the appropriate program office of NRC within a fixed period of time.

(2) NRC staff functions devoted to performing quantitative risk assessment of reactors should be relocated in AEOD.

(3) AEOD should be staffed, in part, on a rotational basis from all the other offices and branches of the NRC staff.

(4) AEOD staff level should be no less than 35 to 40 professionals.

Each of these recommendations is discussed in detail in the enclosure.

We support the SIG's strong emphasis on the importance of an integrated, systematic, and thorough assessment program of operating experience. The observations and comments made in the report are generally well-founded. The SIG correctly calls for a broad, yet well directed and coordinated, program by the involved organizations and identifies a number of problems and deficiencies that must be overcome for the proper feedback of operating experience. One important subject requiring future attention is the adequacy of resources and the wide difference between the current AEOD allocation and the staff level recommended by the SIG. This aspect is further discussed in the enclosure.

While programs to assess operating experience existed prior to the TMI-2 accident, the NRC and the industry have recognized that substantial improvements and expansion in these programs are required. Over the past several months, a number of specific actions have been initiated within the NRC and the industry in order to get the resources, attention, and results that are needed. Yet, the SIG report did not recognize many of these actions. For example, no mention is made of the dedicated groups established and working in NRR and I&E on operating experience assessment, nor of the actions taken to assure that reactor licensees have an onsite assessment program, nor was the industry program in progress at the Nuclear Safety Analysis Center (NSAC) recognized. In addition, the SIG suggestions for improvements in data collection and analysis do not involve a substantial change from current planning and are already part of the ongoing activities or have been incorporated in the NRC Action Plan. This may be a reflection of the fact that the SIG, in the rush of completing its report, did not meet and discuss the ongoing and planned operational data assessment activities with members of the AEOD interim office.

Please let me know should you desire clarification or additional information.

C. J. Heltemes, Jr., Interim Director
Office for Analysis and Evaluation of
Operational Data

Enclosure:
Comments on the SIG Recommendations
Involving AEOD

cc w/enclosure:
R. Mattson
S. Boyd

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

FEB 5 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: James R. Shea, Director
Office of International Programs

SUBJECT: IP COMMENTS ON THE ROGOVIN REPORT

In response to Lee Gossick's memorandum of January 21, IP has the attached comments on the Rogovin Report on the TMI accident. Since the conclusions and recommendations of this report with regard to export matters are similar to those of the Kemeny Commission, our comments are along the same general lines as our comments on the Kemeny Report, which were submitted as part of the package of NRC comments sent to the Commission on November 2, 1979.

J R Shea
James R. Shea, Director
Office of International Programs

Enclosure:
IP Comments on the Rogovin Report

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

February 4, 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Norman M. Haller, Director
Office of Management and Program Analysis

SUBJECT: MPA VIEWS ON THE ROGOVIN REPORT

The EDO requested Office Directors' views on the Rogovin Report.

The importance of the report lies principally in its frank discussion of NRC's management. An obvious strength of the Rogovin recommendations is that a single administrator could more easily establish a strong link between policy development, program planning, resource allocation, and program implementation. However, we believe that the Commission can do much to establish and participate in such a management structure which would have the following key ingredients: a direct line of authority from the Chairman through the EDO to Office Directors, and the holding of line managers throughout the organization accountable for results to those above them.

The Rogovin Report highlights the importance of management setting goals describing the level of protection that is to be attained through NRC's health and safety, safeguards, and environmental regulatory activities. We agree with Rogovin that NRC should formulate such standards. We are encouraged by processes like the Policy, Planning, and Program Guidance (PPPG), but we do not see progress that will lead to establishing these goals, at least in the next few years. While a single administrator would undoubtedly move faster in this regard, we see no inherent reason why the present Commission cannot also move faster, and we encourage this step.

Rogovin also addresses organization approaches for achieving the goals once they are formulated. We recognize that NRC's present organizational structure may hamper effectiveness and efficiency. Nevertheless, we believe that -- by building on the attributes of Rogovin's single administrator proposal -- NRC can attain the needed management improvements through an executive authority for the Chairman and a strong EDO. But we do not believe that a five-member collegial body overseeing the efforts of five independent program offices can ever provide satisfactory management. We therefore urge that the Commission (a) do what it can, within the limits of the law as currently written, to strengthen the role of the Chairman and the EDO, and (b) support similar efforts that may result in changes to the law.

We feel there is an important warning in Rogovin's Epilogue -- particularly "The Prognosis." Some of NRC's programs are at a virtual standstill nearly a year after the TMI-2 accident, and with almost half of our FY 1980 resources consumed. Yet we still face the difficult tasks of deciding comprehensively what to do as a result of the accident, and carrying out the necessary program changes and resource reallocations to do it. While we recognize the importance of careful deliberations over what to do, we also believe the Commission and EDO should begin to set in motion the process for carrying out the necessary reprogramming that inevitably will come.

Finally, with respect to Rogovin's proposal to consolidate NRC's resources for operating reactors into a single office, probably IE, we offer three thoughts. First, such a move must take into account other initiatives (either underway or proposed) to improve the inspection program or to add functions to IE. Too much too fast may overload the IE headquarters and result in a decrease in capability rather than an increase. Second, NRC's efforts for analysis of operating data must be focused and highly coordinated; spreading the resources for such analysis among several offices appears inconsistent with the recommendation to consolidate all operating reactor activities into a single office. Third, cooperation between NRC's licensing and inspection functions could be improved through more direct contact between the Regions and the licensing offices.

Norman M. Haller

Norman M. Haller, Director
Office of Management and Program Analysis

cc: Roger Mattson, DSS

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

February 4, 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director for Operations

FROM: Joseph J. Fouchard, Director
Office of Public Affairs

SUBJECT: COMMENTS ON ROGOVIN REPORT

Enclosed are the detailed comments of the Office of Public Affairs on public affairs aspects of the Special Inquiry Report. There is one major difference between the public affairs recommendations of the Presidential Commission and those of the NRC Special Inquiry. The Presidential Commission recommends that the utility take the lead in providing information to the media and the public, while the Special Inquiry Group recommends that this lead task be the responsibility of a senior NRC official. In my view, this must be a cooperative effort involving all agencies and the utility. Whether or not NRC is in charge will depend in large measure on the role the Commission determines it will have in an emergency--investigative, operational or both.

J Fouchard

Josepn J. Fouchard, Director
Office of Public Affairs

Enclosure

cc: Roger Mattson, DSS

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

FEB 1 1980

MEMORANDUM FOR: William J. Dircks
Acting Executive Director
for Operations

FROM: G. Wayne Kerr, Acting Director
Office of State Programs

SUBJECT: COMMENTS ON ROGOVIN REPORT

We have reviewed Volume I of the subject report and certain supporting documents. Our review related to those recommendations that might impact residual OSP responsibilities or interests.

1. Recommendation No. 6 - More Remote Siting and Improved Emergency Planning, Including Workable Evacuation Planning As A Condition of Reactor Operation.

   Comment: The full effect upon NRC of the transfer of lead EP work to FEMA is not known. The NRC State Liaison Officers in Regions I and V have been spending 75-100% of their time since July 1979 in EP related work. We are attempting to better define the extent of their continued level of effort to be expended in this area. The recruitment of SLO's to the other 3 regions is currently under way. The SLO's should not continue in full time EP work in the long term and alternate methods of handling this work will have to be addressed.

   Comment: There is some discussion of financing of development of State and local plans in the supplementary document NUREG/CR-1225 (see Sec. 4.07 on Page 8, Issue Four - pp. 25-31, Sec. 9.10 on Page 44, and Sec. 1.23 - 1.26 on pp. 50 and 51). We believe legislation providing for a combination of development grants to States and State imposed fees on the licensee would provide the best resolution of this matter. Our principal concern is that in the absence of an adequate funding mechanism for the States for emergency planning work, their other regulatory programs (e.g., regulation of agreement materials) can be severely impacted since the radiation control staffs are heavily involved in the development of the plans, keeping them current, and participating in the drills.

G. Wayne Kerr

G. Wayne Kerr, Acting Director
Office of State Programs

2. ANALYSIS BY THE ACTION PLAN STEERING GROUP

The TMI Action Plan Steering Group, reporting to the Executive Director for Operations, has also performed an initial analysis of the report by the NRC's Special Inquiry Group (SIG) on the Accident at Three Mile Island (see Enclosure 2). The analysis was conducted from the perspective of draft 2 of the Action Plan, NUREG-0660. The analysis involved input from the 20 managers and senior staff from six Offices of the NRC who have been designated as Task Managers for the various sections of the Action Plan, reporting to the Steering Group.

The comments developed by the Steering Group are intended to indicate how the plan should be revised to reflect the recommendations of the Special Inquiry Group. The comments are preliminary because of the short time available for their preparation. More thought and careful coordination needs to be given to the details of the SIG report, especially to some of the findings and conclusions of Volume II of the report which have no accompanying recommendations. The Steering Group will give this added attention to the SIG recommendations during its coordination of the development of draft 3 of the Action Plan, now scheduled for about March 1, 1980.

It is the Steering Group's opinion that although the time for review has been short, we and the principal line offices have been able to afford a number of people a good opportunity to view the several hundred specific recommendations of the SIG for the purpose of identifying items that should be considered for prompt action on operating reactors or for inclusion on the list of near-term operating license (NTOL) requirements. Those identified by the Steering Group and its Task Managers or by the Office Directors and approved for inclusion in the NTOL lis are discussed in Section 3 of this report.

The Steering Group, on the basis of input from its Task Managers, prepared four documents, as follows:

a. A detailed list of recommendations in Volume I of the SIG report, annotated with the corresponding section of the Action Plan, if any, and a preliminary indication of how the plan should be revised in draft 3 to incorporate each SIG recommendation (see Enclosure 3).

b. A detailed list identical to the one above, except comprised of the recommendations of Volume II of the SIG report (see Enclosure 4).

c. Proposed additions to the list of NTOL requirements, in light of the SIG report, for consideration by the Office Directors.

d. A narrative summary of our comparison of the Action Plan and the SIG report (see Enclosure 2).

3. REVISED LICENSING REQUIREMENTS LIST FOR NEAR TERM OPERATING LICENSE APPLICANTS

Pursuant to the Commission's direction in the Secretary's memorandum of January 18, 1980, the list of TMI-related requirements for pending operating license applications proposed by the EDO and staff in the January 5,.1980 memorandum to the Commission has been reviewed and revised. The revised list of NTOL requirements and a cross index to the January 5 list are provided in Enclosure 5.

In parallel with performing the studies and analysis requested by the Commission for revising the NTOL requirements, the staff has been reviewing the report of the Special Inquiry Group, as described above. A high priority was given in that review to the identification of SIG recommendations requiring prompt action on operating reactors or addition to the NTOL requirements list. The Office Directors and the Steering Group have agreed that two items should be added to the NTOL list but need not be applied to operating reactors. In the case of one item, the operating reactors will be required to do something later, and the other item applies only to power ascension testing.

In summary, the list of NTOL requirements has been revised since January 5, 1980 to reflect the following considerations:

a. evaluation of the results of the Operator Feedback Review requested by the Commission (see Enclosure 6),

b. evaluation of which items would require changes to the Commission's rules in order to implement (see Enclosure 7),

c. an approximate evaluation of the impact of the NTOL list on available resources,

d. a consideration of the priority of the items on the NTOL list relative to one another and other items in the Action Plan using the numerical ranking scheme approved by the Commission on December 21, 1979 (see Enclosure 5),

e. incorporation of changes caused by more general consideration of the Action Plan in its revision from draft 1 to draft 2, especially the addition of items deriving from the work of the Bulletins and Orders Task Force in NRR that had been previously omitted,

f. preliminary analysis of the recommendations of the Special Inquiry Group, and

g. unsolicited input from the regulated industry (EPRI and AIF) requesting that the development of the Action Plan in general and the NTOL list in particular give specific attention to resource priorities so that industry resources will not be inadvertently diverted from other higher priority, safety-related activities.

Having considered the information listed above, the Executive Director, the Office Directors, and TMI Action Plan Steering Group have reviewed and approved the revised NTOL requirements list provided in Enclosure 5, and recommend its approval by the Commission. Part 5 of Enclosure 5 lists some NTOL requirements contained in draft 2 of NUREG-0660 that were reconsidered by the Steering Group and Office Directors and removed from the NTOL list but retained in the Action Plan for later application to all plants.

4. ONGOING AND FUTURE WORK ON THE ACTION PLAN

The Steering Group and Office Directors are proceeding along lines previously described to the Commission to develop priorities among the TMI action items relative to the previous NRC operating plan and to develop proposed reprogramming steps to accommodate any higher priority, presently unbudgeted TMI actions. The Offices have identified reprogramming candidates in inverted priority order, as described in the EDO's January 18 memorandum to the Commission. The Office Directors and the Steering Group are continuing to review and refine the NRC resources identified in draft 2 of the Action Plan. The Steering Group has finished a numerical ordering of priorities for the 185 items in draft 2 and will use the same method for any additions or revisions of the plan in light of the report of the SIG. The Atomic Industrial Forum has provided preliminary industry resource estimates for the action items and has asked the Steering Group to meet with the TMI Steering Committee of AIF to discuss refinements of these estimates and other Action Plan implementation subjects on February 6. The ACRS will be provided an Action Plan status report at its February meeting. Proceeding on current assumptions, the staff estimates that a priority ordered revision of the Action Plan (draft 3) can be provided that accounts for the recommendations of the SIG by about March 1, 1980.

ENCLOSURE 1

OFFICE DIRECTOR REVIEWS OF SIG REPORT

OFFICE OF INSPECTION AND ENFORCEMENT

ENCLOSURE 1

DETAILED IE COMMENTS ON REPORT OF THE SPECIAL INQUIRY GROUP

Of the fifty Recommendations set forth in Volume I, only six have not appeared in earlier studies. Our comments on each of these six are set forth below:

1. Proposed Industry-wide Operating Consortium

   This is a provocative idea and certainly worthy of careful study. Our experience through the years indicates that small utilities frequently have problems with providing the technical resources required to operate a large nuclear power plant. On the other hand, we should be careful not to cause the creation of another layer of administrative organization. Further consideration of the proposal is warranted. We do not believe the priority of this effort is high.

2. Increased Emphasis on Project Management

   The desirability of strengthening NRC project management is clear. We believe that the changes the SIG proposed for operating reactors. which we discuss later, will provide substantial improvement in this area.

3. Periodic Manager Reassignments

   We endorse the concept of more preplanning of management interchange programs among NRC offices and program divisions within the offices. Post-TMI experiences have been relatively positive and enlightening to many. However, a more disciplined approach is desirable. It is easy, moreover, to lose overall management effectiveness in such efforts if the desirability of cross-fertilization is allowed to outweigh the need for knowledgeable management of vital programs on a continuing basis. We are concerned over the inertia of such systems in themselves.

4. Liberalization of the Ex Parte Rule

This is clearly a desirable change. In our zeal to be fair and impartial to all the parties in matters subject to possible Commission review, we have overreacted to the point of isolating the Commission. The extent to which the present ex parte restrictions could be responsibly relaxed is, in our view, a question that should be promptly addressed by OGC and OPE.

5. Formalizing RRRC Functions

We recognize the appeal of formalizing the activities of the Regulatory Requirements Review Committee. The SIG proposal concerning staffing for the RRRC deserves particular attention. In conjunction with our review of such RRRC staff augmentation, we believe we should assess the desirability of returning many RRC functions to the line organizations.

6. Improvement in NRC Evaluation of Utility Finances

This recommendation, in various forms, appears in several places in both volumes of the report. The relationship between regulatory requirements based on the safety of the plant and their financial impact is extremely complex. That there is such a relationship is clear. We agree that the general subject is worth further study. We do not recommend a high priority.

We feel obligated to comment here on a several SIG Recommendation that reflect concerns previously identified in other studies:

- Consolidation of DOR-NRR and IE Activities in a New Line Office

This general subject also was adddessed in NUREG-0585 in a slightly different and more limited context. That is, the Lessons Learned Task Force simply recommended consolidation of all NRR activities involving reactor operations; reactor operations evaluation; operational QA; human factors evaluation; personnel qualifications standards; and personnel licensing and certification. We agree with both groups that operations must have a stronger voice in NRC's evaluations and decisions.

We endorse the concept as a significant step to improve both the efficiency and effectiveness of the staff, with possible resource economies. The point is less whether the combination is located in one of the existing line offices than it is of the consolidation of operational activities for all licensees, not just nuclear power plants. Such an organization would be separate from the pressures of CP and OL licensing priorities. In addition, integration of post-OL licensing activities would be enhanced. We agree with the SIG position on combining DOR and

IE; we agree with the Lessons Learned position of incorporating in the combined office the other functions they proposed.

We believe that the logic that led the SIG to this recommendation extends beyond the move they proposed. All operating responsibility for major licensees, including fuel facilities and major materials licensees, should reside in the new organization, including project management. The corollary of consolidating pre-OL activities in NRR and NMSS also follows; i.e. consideration should be given to transferring responsibility for construction inspection management from IE to the licensing offices. The resulting organizations would provide considerable emphasis to project management, as advocated by the SIG.

As you are aware, I have previously expressed my concern about the limited staff capability in IE headquarters. Stressing a separate operating organization would not only provide the benefits proposed but also ameliorate staffing difficulties I have described separately. In addition, we believe NRC's ability to respond effectively to emergencies would be improved.

Such an organizational change would entail substantial rearrangement of the infrastructure of the involved offices. I have not yet initiated staff effort in this area, but I am prepared to do so expeditiously. I urge a prompt Commission decision on this matter.

Status of Radiation Protection Programs

We agree with these recommendations -- the thrust of which is that radiation protection programs be given greater emphasis by both the industry and by the NRC. This means resources as well as verbal exhortations. The NRC resources available for inspection of the HP areas, not only in reactors but in the fuel facility and material licensing areas as well, are not commensurate with the job to be done.

One-Step Licensing and Increased Use of Standardization

The SIG supports one-step licensing and increased use of standardization. Adoption of this recommendation would improve the enforcement process during construction by providing more enforcable design requirements. A number of actions could be taken in this area to provide more formalized designs at an earlier stage, such as standardization; defining principal architectural and engineering criteria; and providing greater detail in Appendix A, 10 CFR 50. We support such actions.

Steps Needed to Permit Reinitiation of Licensing

The SIG report explicitly refrains from advocating a "moratorium," but notes that new OL's should not be issued, nor new construction permit applications be accepted until certain actions are completed. We believe that the Task Action Plan adequately addresses the necessary steps that no additional measures are required separately as a result of the SIG report.

. Field Emergency Response

We endorse the SIG recommendations concerning the need for clearer authority for the senior NRC official responding to the scene of an accident. Our present procedures call for immediate dispatch of the Regional Director upon activation of the Operations Center. However, clarification of his authority is clearly needed.

. Evaluation of Utility Management

We agree with the general thrust of the SIG recommendations for increased attention to NRC evaluation of utility management. We are scheduled to brief the Commission on this matter in the near future and will set forth our plans for the program at that time. Our plans are in accordance with the NRC's Action Plan.

Finally, the issues of: (1) the adequacy of the licensee's reporting of critical plant parameter information during the early hours of the accident; and (2) possibly deficient Part 21 reporting by Westinghouse concerning PORV problems are undergoing staff analysis, in view of the SIG comments on these matters. As you know, Congressman Udall has also requested further information on the subject of Met Ed's reporting. We understand that the SIG will respond to the question raised in Congressman Udall's letter. If requested, we shall report separately the results of our analyses.

OFFICE OF NUCLEAR REACTOR REGULATION

ADDITIONAL STAFF COMMENTS

1. The central theme, raised in virtually every section of the Report, and represented as the most serious problem, is the NRC's and industry's management problems. It is difficult to completely disagree with this issue, since arguing against the belief that we should all improve our management capability and discipline is arguing against "motherhood". This is true both organizationally and individually. It does not seem justified, however, to present "management problems" as the principal deficiency in safety today.

2. We should move promptly from a position of developing plans to implement the various Lessons Learned, Kemeny, and Rogovin recommendations to one of the implementing at least the most important of these plans. This point is important for two reasons. First, our past experience in identifying, planning and prioritizing the Unresolved Safety Issues has been outstanding but when the effort turned toward solving problems, the results were less than completely successful. We again appear to be more preoccupied with planning rather than with doing. Second, relatively few NRC people, but almost all management-level people, are actively engaged in the planning process for TMI response and thus feel some sense of urgency. The remainder of the staff, mostly at the worker level, are continuing work on some assignments that are perceived as having lower priority. It would be a big boost to staff morale if they were to be instructed to move forward on at least the high priority tasks: While there is some potential for inefficiency in taking what might appear to be presumptive actions, the overall perception of getting on with the job and having all of the staff employed in the most important activities would outweigh potential inefficiencies.

3. With respect to emergency planning, the SIG raises two policy issues. First, a recommendation is made that approved local emergency plans not be a condition of licensing. This can be considered during the pending rulemaking proceeding. Second, an effort to develop a new methodology for determining emergency planning distances and evacuation times is recommended by SIG, based on an acceptable risk criterion. While this area is worth exploring, it does not hold much hope of being useful in the near future since it would require (1) the definition of an acceptable risk, and (2) a WASH-1400 type of analysis for each plant and site to determine risks with and without protective actions such as evacuations and to determine acceptable evacuation times. This recommendation of Volume 1 is at variance with the Volume 2 recommendation (item 4.C., p. 319, Part 3) which endorses formal evacuation planning for the 10 mile distance recommended by the EPA/NRC task force.

4. Another policy aspect related to emergency planning raised by the SIG is whether extended bad weather should result in plant shutdown because emergency plans may not be fully implementable. The Report recommends that a decision on this aspect be made when the event occurs. Because this is a known and periodic problem at almost all sites, a generic position would seem more appropriate, taking into account the likely continued need for the electricity from the plant during such times.

5. The SIG recommends broader NRC staff training. The TMI-1 Site Office offers an excellent opportunity for such training, not only for practical plant experience, but for blending of the traditional NRR and IE roles.

6. Like several other reviews, the SIG Report recommends much broader use of probablistic risk assessment than is present practice in the NRC and the nuclear power industry. Expanded use of these methods is planned and the Action Plan delineates some proposals to do this. However, the Commission Policy Statement of January 18, 1979, is presently interpreted by some to severely limit such applications. A NRC-wide symposium was held recently on this topic, with general agreement toward increased utilization.

7. In Volume 2, a reexamination of surveillance procedures to prohibit simultaneous defeat of redundant systems important to safety was recommended. The staff intends to require that all surveillance testing be performed with the intent of minimizing the potential for complete outage of a safety system function while in a test mode. It is expected that most if not all redundant elements of a safety system can be tested in staggered sequence preserving at least one train of a system available to respond to an accident signal.

8. In Volume 2, an interim requirement for licensees to assure availability of prompt expert technical advice to operation personnel in order to better assess and respond to emergency situations is recommended. It is expected that the joint NRC/FEMA criteria for emergency preparedness will require the operator to make provisions for obtaining offsite technical assistance and that implementation of this provision will be verified in the emergency preparedness evaluation team reviews. In addition, the AIF is now developing a catalog of available equipment and technical manpower resources.

OFFICE OF STANDARDS DEVELOPMENT

ENCLOSURE 1

ADDITIONAL SD STAFF COMMENTS

Vol. I, Pages 136 & 137 - Lead Monitoring Responsibility and On-Line Monitoring Systems

"EPA should be assigned long-term monitoring responsibilities--and HEW should be given the lead responsibility for population dose assessments and calculations of health impacts."

This recommendation ignores the fact that most of HEW's expertise in environmental radiation monitoring and dose assessment (including its Nuclear Facilities Branch) was transferred to EPA in 1970 by Reorganization Plan No. 3. HEW would be weak in this area as there are probably only a dozen or so people in HEW with this expertise.

Vol. I, Page 153 - Health Effects from Radioactive Releases During the Accident, and Occupational Health Physics at the Site

"The effects on the population in the vicinity of Three Mile Island from-- will certainly be nonmeasureable and nondetectable."

The write-up on health effects of TMI makes too much of the average dose within 50 miles (e.g., the average dose within 10 miles is 8-10 mrem 7 times higher) and also arrives at lower health effects numbers than the Presidential Commission or Ad Hoc Group.

Vol. I. Pages 147-152 - Improvement in the Basis for Safety Review of Reactor Design and Increased Use of Quantitative Risk Assessment Techniques

The suggestion that Congress is the body to set an acceptable level of risk for accidents is excellent. In preparing a rationale for setting a safety goal, the following suggestions are offered:

1. The potential consequences of a large reactor accident need to be presented more clearly. Greater emphasis needs to be placed upon the total impact of such accidents in addition to the average risk to an individual (which was shown in the Reactor Safety Study). Nuclear reactor accidents are not very different from other man-made hazards such as large fires or explosions in that they have a potential for killing large numbers of people. However, the expected frequency of occurrence of large reactor accidents is much lower than large fires or explosions. The public needs greater awareness of the risks and magnitude of the consequences of all industrial hazards.

2. Current risks from other activities probably cannot be used as a basis for setting allowable risks for nuclear power for three reasons:

   a. Presumption that existing risks are acceptable may be fallacious - more likely the public is often unaware of actual risks - the risks may be accepted (borne) but not acceptable (voluntarily borne with knowledge of risk).

   b. The public may not accept the same risk level from nuclear as is accepted for other hazards.

   c. Most man-made hazards (fires, explosions, aircraft crashes) do not have long-term consequences. Radioactive materials and other toxic chemcials can result in long-term restrictions on land use and may affect future generations through genetic effects. More research is needed on methods for evaluating these risks and for incorporating

SD

future risks into decisions on control methods. In particular. a uniform approch is needed for assessing the accident consequences and disposal requirements for all long-term toxic materials including long-lived radioactive materials.

3. In examining "value-impact" tradeoffs of safety systems. a 'cost-effectiveness" approach for comparing alternatives may be preferable to an absolute "cost-benefit" approach because an explicit dollar-to-health impact (e.g., dollar/manrem) number is not required. This factor is apt to be considerably higher for the real effects of a potential accident than for the potential effects of real (routine) effluents.

Vol. I, Page 145 - 5th Paragraph that begins "The pressurizer relief valve ... was not categorized as safety related, ..."

There appears to be some confusion between safety-related as used in 50.55a and R. G. 1.26 for pressure boundary components and safety grade for Class 1E electrical components. The PORV was quality group A (safety-related) per 50.55a, but apparently the controls were not Class 1E (safety-grade). This example highlights a significant weakness in our regulatory practice. "Safety-related" must be a graded concept relating the required quality in design, construction, operation and inspection to the safety significance of the system or component. The ASME Code has, to a large extent, developed a graded approach for pressure boundary components. This must now be improved and extended to cover active components (e.g., pumps and valves) and electrical equipment. Continuation of the present concept of safety-related vs. non-safety-related would result in regulatory overkill when maximum controls are applied to systems with marginal safety significance or underkill where such controls are ignored entirely as in the case of the PORV controls cited above.

Vol. I, Page 156 - Information Made Available to the News Media

The NRC staff (principally Office of Public Affairs) has initiated action toward a program for helping to better inform news media personnel on nuclear safety matters, particularly regarding nuclear power plants and associated safety issues. While the direction of this program will take is still being studied, a pilot phase is envisioned to gain insight into media personnel interest and problems, to be followed by periodic educational seminars throughout the country. The pilot phase, for which planning has already begun, is to be conducted by Inspection and Enforcement personnel from one or more NRC regional offices; at least two seminars will be offered. Based on information gained from this experience, it is most likely that a contract will be awarded to an appropriate organization to plan and conduct subsequent seminars on a continuing basis. One possible contractor, the Health Physics Society (HPS) has been contacted to determine their degree of interest. The current HPS president, Mr. Mel Carter of the Georgia Institute of Technology, has ageed to respond during the first week in February. The HPS has almost 40 chapters in the United States, located for the most part near major industrial nuclear centers. It would appear that these chapters would be in an excellent position to conduct these seminars.

Better programs of public information on the much broader topic of radiation hazards in general are anticipated as an early topic to be addressed by the Federal Radiation Policy Council, when it is established.

ROAD MAP OF THE NRC SPECIAL INQUIRY GROUP (SIG) REPORT OBJECTIVES THAT MIGHT IMPACT SD ACTIVITIES

The SIG report objectives that might impact SD activities are listed in abbreviated form and grouped below under subject headings. They are indexed as to their location in the report. Recommendations from Volume I are identified by the letters vI. The page locations in Volume I of the recommendations are not listed. Recommendations from Volume II are identified by the letter "p" followed by the section (part) of Volume II and the page number. For example, p1-41 and p2-100 indicate recommendations found respectively on page 41 of section (part) 1 of Volume II and on page 100 of section (part) 2 of Volume II.

Location

RISK OBJECTIVE

vI Strive to establish a substantive Risk Objective for nuclear power plants - provide clear guidelines on how safe is safe enough.

- propose a substantive quantitative standard for public discussion and Commission consideration.

p1-41 - develop a statement on regulatory objectives as well as risk objectives.

- develop methods for determining if risk objective is met.

vI Express new requirements (based on use of best available risk assessment techniques) for meeting risk objectives in Std. Review Plan.

INFORM PUBLIC OF RADIATION RISKS

Inform public fully of the manner in which nuclear power plants are designed, licensed and operated and of actual risks associated with radioactive materials.

DESIGN

p2-180 &vI Reconsider design basis of plants

- level of safety required
- type of accident for which plant designed
- method by which design basis established

Location	DESIGN (continued)
vI	- criteria for determining "safety grade" - use risk related scheme for classifying safety significance of equipment
	- magnitude of accident
p2-181	Use of Human Factor Principals
vI	- Control room design
vI	- Instrument display
p2-190	Determine principal sources of H_2
"	Shielding
"	Containment isolation
"	Diesel generators (lock out)
"	Etc.
p2-232	Require installation of malfunction detection analyzers
p2-232	Update RG 1.97 and ANS 4.5 to require administrative review of repair records.
vI	Expand SRP in areas --
	- Op. training
	- Plant emergency op. procedure
	- Control room design
	- Appl. tech. qual.
	- Plant techn. specs.
	- QA
vI	On a selective basis, determine whether some design features should be required to mitigate consequences of Class 9 accidents.

e.g. - vented and filtered containment

- redesign of waste gas and filtering systems that will get water and gas from primary system in an accident containing high concentration of radioactive material.

<u>Location</u> DESIGN (continued)

vI Include effect of multiple equipment and human failures where risk is significantly high -- in expanding and evaluating the spectrum of design basis accidents.

vI Human factors and operational procedures should be included in license review process.

Review of operating experience and equipment malfunction - on continuous basis by Industry.

p1-41 Increased use of standardization.

vI Establish explicit rationale - as quantitative and objective as possible for evaluating new safety requirements against the criteria "substantial additional protection required for public health and safety".

vI Assure comprehensive analysis and application of operating plant experience to development of new regulatory requirements.

RADWASTE SYSTEM

p2-71 Reexamine and determine appropriate radqaste design criteria for expected levels and volumes of radioactivity in normal and accident conditions.

- include all related systems - e.g., industrial waste system.

VENTILATION SYSTEM

p2-71 Prepare test procedures for inplace testing of ventilation system -- filters, etc.

Develop criteria for use of ventilation systems in normal and accident conditions.

Location

QA SHORTCOMINGS

p2-41 Lack sufficient definition for "safety related" as applied to equipment, system and structures to assure consistent implementation of Appendix B.

No QA standards for comparison commensurate with safety function as required by General Design Criterion #1 (Appendix A).

Appendix B lacks specific criteria for maintenance and other operations and certifying personnel performing maintenance or other op.

No quantitative reliability methodology in QA program requirements.

Section 17.1 and 17.2 of Standard Review Plan lack acceptable criteria and review procedures for list of items that conform to Appendix B standards.

QA program not a condition for OL.

OPERATOR TRAINING, QUALIFICATIONS, REQUIREMENTS

p2-66 Strengthen onsite technical capability and management of utility at reactor sites - (upgrade emergency response capability)

- improved operator training
  - certified training program
  - certified instructor

p2-66 - qualified engineer supervisor
  - degree in technical discipline

p2-66 - supervisors up to Unit supervisor should have SRO licenses

p2-66 - reassessment of duties, responsibilities and training of all support personnel

Location OPERATOR TRAINING, QUALIFICATION, REQUIREMENTS (continued)

p2-66 - increase shift manning levels

p3-159 - offsite safety review committee personnel qualification should be established

p3-159 - offsite safety review committee should make timely review of personnel changes

p3-159 - NRC should make timely review of personnel changes

p3-159 - qualifications and experience requirements for people managing utility emergency response

p3-183 - revision of RG 1.101 re training and drill requirments

HEALTH PHYSICS AND RADIATION PROTECTION

p2-104 Reevaluate NRC requests for radiological monitoring - normal and accident conditions.

- TLD locations, airborne activity monitors, etc.

p2-147 Establish standards for licensee radiation protection programs and competency of rad protection personnel.

p2-147 Guidance regarding use and training of "Rent-a-Techs" at licensed facilities.

p2-147 Appoint group of experts to examine feasibility and advisability of licensing or certifying rad prot. personnel at nuclear power plants (6-mo. study).

p2-157 Requirements for inplant fixed rad monitoring instruments.

p2-157 Requirements for operational portable radiation survey equipment at plants (type, quality and quantity).

p2-158 Requirements for respiratory equipment.

Location HEALTH PHYSICS AND RADIATION PROTECTION (continued)

p2-172 Utility management and organization for rad prot. function.

p2-172 Develop a regulatory base for assuring inplant radiological conditions resulting from an accident are considered in emergency planning procedures.

vI Specific requirements for Occupational Health program at plants.

vI High level management of Occupational Health Physics Program is required and must be independent of operation management.

SITING

vI Specific criteria should be developed promptly.

- consider population density, population centers, evacuation feasibility, evacuation routes.
- population centers within 10 miles must be evacuable.
- consider max dose levels, probability factors, associated time limits.
- must be 10 miles and maybe more from significant population centers.

EMERGENCY PLANNING

vI Develop specific criteria for determining minimum evacuation planning zone around each existing plant.

p3-278 Develop protective action guides to aid in evacuation decisions under various plant circumstances.

p2-95 Plans should be definitive and should include adverse conditions such as inclement weather (blizzard - can't evacuate) minimum allowable staff and rapidly developing accident.

Location	EMERGENCY PLANNING (continued)
p2-95	Clear chain of command for prolonged radiological response.
p2-95 & vI	Real time on-line radiation monitoring equipment installed arr nd power plants.
p2-95	Inplant and portable radiation monitoring equipment and trained personnel.
p2-95	Communications equipment - for communication between inplant and offsite people.
p2-123	Prompt technical support from industry.
p2-123	Adequate technical and managerial people preplanned into emergency organization.
	Offsite Data Center manned by industry.
vI	DOE should have lead responsibility for offsite radiation monitoring.
p2-66	Plans to include organization and use of offduty personnel.
p2-66	Prompt expert advice available to operation staff.
p2-66	Procedural guidance for situations that go beyond normal.
p2-66	Utility plants to mobilize and use industry resources for accident mitigation and recovery.
p3-183	Expedite review and upgrading of existing emergency planning and preparation requirements.

Location	MISCELLANEOUS
pl-41	Increased use of rulemaking - generic safety issues and important policy issues - decisions tha' lead to required safety levels.
pl-41	Designate an organization to have primary responsibility in the rulemaking area to assure that the quality of the regulations is adequate.
pl-41	Abolish 2 step licensing process and provide incentives that will result in more information available prior to construction -- less variety in design of important systems and fewer unresolved issues.
pl-41	Establish backfitting criteria -- use risk assessment to make plant operation judgment.
pl-354	Examine status of FSAR testing requirements and require the listing of much more test information and details.
	Ratchet Committee
pl-41	- require lower level voting members than Office or Div. Director.
pl-41	- require preliminary screening and review by a task group made up of 1 member for each organization providing a voting member.
pl-41	- provide additional steps to increase the opportunity for public and industry involvement -- and early ACRS involvement.
pl-41	- report R^3C deliberations in depth.
pl-41	- intervenor funding and Office of Public Counsel.

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

COMMENTS ON RECOMMENDATIONS CONTAINED IN
REPORT OF NRC/TMI SPECIAL INQUIRY GROUP

1. Systematic Evaluation of Operating Experience and Improvement in the Regulation of Operating Reactors

   We agree with most of this recommendation but caution against concentrating NRC's "resource devoted to monitoring the safe operation of existing reactors in a single office -- probably the current Office of Inspection and Enforcement (IE)," as proposed on page 99. We believe that good reason exists for a separation of inspection and evaluation functions. We strongly feel that inspection and licensing functions should be separated to assure a proper system of checks and balances.

   An alternative approach would be to consider establishment of licensing offices in the existing NRC Regional Offices to work with IE inspectors more closely on a day-to-day basis. In any event, we recommend that the Regional Offices report to the Executive Director for Operations rather than to the Director of IE.

2. Strengthening the Onsite Technical and Management Capability of the Utility: Improved Operator Training and New NRC Requirements for Qualified Engineer Supervisors on Every Shift

   We have no comments to offer on this recommendation other than to express the belief that an industry-run offsite data center, as proposed on page 107, is probably not needed for nuclear facilities licensed by NMSS. The single exception might be a chemical reprocessing plant, but no such plant is presently licensed to operate.

3. Chartering of a National Operating Company or Consortium

   We have no comments to offer on this recommendation.

4. Improved NRC Management and Reorganization to a Single Administrator Agency: Establishment of an Independent Reactor Safety Board

   We do not agree with this recommendation for the reasons stated in the NRC response to the recommendations of the Kemeny Report.

5. Greater Application of Human Factors Engineering, Including Better Instrumentation Display and Improved Control Room Design

   We have no comments to offer on this recommendation.

6. More Remote Siting and Improved Emergency Planning Including Workable Evacuation Planning as a Condition of Reactor Operation

We believe that siting criteria and improved emergency planning are needed also for nuclear operations licensed by NMSS. NMSS believes that improved emergency planning is needed for many byproduct, source and SNM materials and transportation activities regulated by NMSS. Siting criteria for these activities are needed, and should be integrated with off-site emergency response capabilities and facility design and operational features in overall safety assessments to ensure that those activities present no untoward risks to their neighbors and environs. Many of these activities, especially byproduct materials licensees, do not presently have formally approved emergency plans. Many have also not had rigorous site/facility safety assessments. NMSS plans to initiate rulemaking proceedings to correct those deficiencies. That proceeding will involve and depend upon development of criteria for determination of which licensees must have formally approved emergency plans, the scope of those plans, siting criteria, and overall safety afforded by siting, facility, operations and emergency preparedness. A regulatory capability including staff review procedures, acceptance criteria, regulatory guides, and inspection procedures and acceptance criteria will also be developed.

NMSS is examining licensee activities at this time to scope the values and impacts of proposed rulemaking. Contractor support to do evaluations of risks to the public from licensee activities is being sought; funds to do the evaluations have been provided. Related activities such as the various TMI studies and rulemakings and byproduct material ALARA and indemnification studies are being monitored for import to the subject rulemaking.

We do not agree with the recommendation on page 137 concerning "real-time, online monitoring devices around every nuclear plant... that can be read from the plant control room or some other remote site." We do not believe the report contains adequate justification for this recommendation and, in particular, we do not believe this system is needed for UF_6 conversion, fuel processing, fuel fabrication and similar plants.

7. Overhaul of the Licensing Process: One-Stage Licensing, Increased Standardization, Increased Use of Rulemaking, Establishment of an Office of Public Counsel, and Intervenor Funding

We question whether the recommendations to establish an Office of Public Counsel and intervenor funding are adequately supported in the report.

8. Improvement in the Basis for Safety Review of Reactor Design and Increased Use of Quantitative Risk Assessment Techniques

   We have no comments to offer on this recommendation.

9. Health Effects from Radioactive Releases During the Accident, and Occupational Health Physics at the Site

   We have no comments to offer on this recommendation.

10. Information Made Available to the News Media

    We have no comment to offer on this recommendation.

11. Sabotage, Bribery, and Coverup

    We note that the report does not recommend any specific actions involving safeguards.

12. Disincentives to Safety

    We have no comments to offer on this recommendation.

OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

Comments on the SIG Recommendations Involving AEOD

(1) AEOD recommendations for actions should be required to be rejected, modified, or imposed as recommended by the appropriate program office of NRC within a fixed period of time.

It is understood that AEOD is responsible to develop formal recommendations concerning action by other NRC offices. These AEOD recommendations would be specific with regards to what actions AEOD believed necessary and by whom the action must be implemented. The AEOD recommendations would, of course, be supported by a specific analysis or technical basis which would be provided to the responsible office for review with the formal request for action.

It is standard operating procedure that formal interoffice requests for action are tracked, considered, and resolved by the responsible office either through implementing the request or through a formal response providing a definitive basis for not proceeding. However, to assure that there are no misunderstandings and to obtain Commission level attention on the specifics of this procedure, the draft manual chapter covering the collection, assessment, and feedback of operating experience will be expanded to include provisions for handling AEOD and other NRC office recommendations for actions based upon their assessment of operating experience. These provisions will address the nature and timing of the response by the responsible office, and the process for resolving areas of disagreement.

(2) NRC staff functions devoted to performing quantitative risk assessment of reactors should be relocated to AEOD.

Quantitative risk assessment is an extremely valuable analytical tool to prioritize and gain a perspective on the safety significance of complex events, postulated sequences, and differing design approaches. Thus, there is no question that AEOD must have the capability to perform these types of analysis which are an integral and important part of the AEOD charter.

Four options to obtain this capability for AEOD have been discussed: (a) contract this work to a qualified contractor; (b) request the necessary assistance from other NRC offices, specifically PAS-RES; (c) recruit qualified individuals for the AEOD staff; and (d) recommend that AEOD obtain this capability by transferring the function and individuals from other NRC offices. Preliminary conclusions are that: (1) the use of quantitative risk assessment within the NRC is becoming more widespread; and (2) AEOD must have this capability in-house in order to effectively discharge its responsibilities and thus options (a) and (b) have been discounted. There are a number of studies underway, some involving inter-office cooperation, directed toward the applications of this important tool to improve the licensing bases and to assure the proper allocation of resources. Consequently, it may not be desirable to concentrate this capability in one organizational element. By frequent exchanges of information and close communications, there should be a maximum cross fertilization of thoughts and minimum duplication of activities. Based on these considerations, option (d) was discounted. Accordingly, AEOD has initiated steps to recruit individuals possessing this technical expertise for the AEOD staff and discussions have, in fact, been initiated towards hiring such individuals.

In sum, we believe that AEOD must be strong and self-sufficient with regard to developing quantitative risk assessments, but that our use of this analytical technique is not unique. We would expect and, in fact, encourage other offices to use this methodology in a planned and coordinated way. Therefore, we do not recommend that such capability be transferred to AEOD from the other staff offices.

(3) AEOD should be staffed, in part, on a rotational basis from all the other offices and branches of the NRC staff.

This recommendation recognizes that a tradeoff can be made between the benefits to be gained as a result of permanency, and the benefits obtained from involving a relatively large number of individuals in the direct assessment of operating experience. Thus, the recommendation to have both permanent and rotating staff members is a reasonable way to obtain the benefits of both approaches. It should be noted, however, that the integrated program within the NRC involving the offices of NRR, I&E, and PAS/RES already involves a large number of individuals in the direct assessment of operating experience without a defined system for rotating personnel.

In initially staffing AEOD, the priority has been placed on permanent personnel. This is a reflection of the need to expand the capabilities and activities of AEOD in a rapid manner with meaningful and lasting results. Also there is sound advice in Admiral Rickover's recent statement that "With permanence you gain experience, judgment, and a 'corporate memory' which are hard to replace." These latter characteristics are particularly important in the detailed analysis of operating experience. Further, it is recognized that if ongoing efforts are successful in obtaining individuals of outstanding technical capabilities, such individuals will, in time, leave to assume greater responsibilities. Thus, benefits similar to those obtained by rotation of personnel will be provided by the natural and healthy turnover of AEOD staff without the need for temporary assignments on detail.

After a permanent staff has been established and is working effectively, further consideration will be given to the benefits and disadvantages of additional personnel assignments to AEOD on a rotating basis. Should a decision be made to implement such a program, the Commission will be informed in advance regarding the particulars and bases for such a program.

(4) AEOD staff level should be no less than 35 to 40 professionals.

The AEOD staff level was established, via the FY80 supplement, at 18 positions. In addition, the EDO has combined the 3 positions associated with the Office of Technical Advisor to the EDO with AEOD, making a total of 21 authorized positions. Two additional positions are requested in the FY81 budget.

It is recognized that the depth and scope of the AEOD activities within its broad charter may be resource limited. Further, it seems quite clear that the need to: intensively analyze reactor operating experiences; thoroughly study non-reactor operating data; accommodate the increasing number of operating reactors; account for the anticipated increase in the scope of reporting requirements; and work closely with other NRC offices, the ACRS, licensees and industry organizations will eventually require greater resources.

Thus, we would agree that an increase in AEOD professional staffing will be warranted, but the proper level and timeframe are uncertain. Until the permanent staff is established at currently authorized levels and the associated technical activities are in progress, it is not possible to definitively estimate the ultimate resources required to adequately accomplish the full scope of AEOD responsibilities.

Another factor influencing required resources is the charter or scope of AEOD activities and should the present scope of AEOD be expanded, such as the SIG recommendation discussed above for NRC risk assessment functions to be relocated in AEOD, the number of professionals would have to expand commensurately.

OFFICE OF INTERNATIONAL PROGRAMS
COMMENTS ON THE ROGOVIN REPORT ON THE ACCIDENT AT THREE MILE ISLAND

The Rogovin report recommends restructuring NRC as an independent agency under the Executive Branch with a central goal of promoting the safety of nuclear reactors and handling of nuclear materials. NRC's jurisdiction over nuclear export licensing would be transferred to the Department of State or the Arms Control and Disarmament Agency, which would then consult with the NRC on safety-related matters.

The Energy Reorganization Act of 1974 and the Nuclear Non-Proliferation Act of 1978 gave NRC specific responsibilities to provide an independent check on the export recommendations of the Executive Branch agencies, largely as an outgrowth of Congressional concern about the adequacy, from the nonproliferation point of view, of the export reviews performed by the Executive Branch.

If these Rogovin recommendations were accepted, legislation would be required and this would present some serious problems. Congress would clearly want to look closely at the very serious question of whether the principle of independent export review could in fact still be carried out satisfactorily within an Executive Branch agency whose head reported to the President, even though the agency had an independent status (as in the case of ACDA). Such a restructuring should have the advantages associated with reducing export processing time.

If the recommended new agency is not formed and the present Commission (or a restructured NRC outside the Executive Branch) has the domestic safety responsibilities, the question arises of the extent to which Commissioner time devoted to such areas as export licensing can be minimized in order to allow more Commission time for consideration of domestic safety matters. One of the ways this could be done is to limit the amount of Commissioner time spent on export questions to what is required to address only major policy questions, with as much responsibility as possible delegated to the staff. A paper with proposed additional delegations of authority to the staff in the export area is pending Commission approval, and the suggestions in that paper, if implemented, would reduce the Commission workload in the export area while retaining Commission decision authority over the most significant issues. Greater reliance on the staff's expertise in the policy, legal and technical aspects of export licensing, which has been developed to perform NRC's independent export reviews, could lead to even further delegations in the future.

Now that many of the key issues associated with export licensing, particularly in the nonproliferation area, have been extensively addressed in recent years, there are increased opportunities to limit the Commission's time to focusing on significant new questions, while the staff uses guidance from previous decisions to apply to specific implementation of case-by-case export review. In recent months the staff, for example, has increasingly been able to process routine reload export cases at the staff level without the need to refer these to the Commission. These and other measures which are being taken help the U.S. Government improve its perception as a reliable nuclear supplier and thus contribute to U.S. nonproliferation policy.

As noted above, the Rogovin report recommends that Executive Branch agencies handling exports consult with NRC only on safety-related matters that arise in the export licensing context (insofar as they are judged relevant to the U.S. export process). This would be a reasonable function to be performed by the U.S. domestic nuclear safety authority; however, the effect on the licensing process of dispersed agency responsibilities for various aspects of export reviews would need to be carefully studied.

With regard to other, nonexport-related, international functions of NRC, such functions are presently focused on safety matters. Our regulatory information exchange arrangements, research project agreements and the bulk of our IAEA, NEA and technical assistance activities are concerned with reactor safety. The international cooperation activities of NRC, which are directly linked to improving U.S. public health and safety through acquisition and use of foreign reactor operating experience and research results, would appear to be largely unaffected by a reorganization such as that proposed by the Rogovin report, except for possibly intensified work in these areas.

In several places, the Report discusses the protection of foreign information given in confidence. The main theme of the implied criticism is that the foreign information is not made public. No note is made of the fact that action can be taken on the information, by official and directly involved U.S. parties, other than the public at large, despite this confidentiality. The discussion of the Beznau Incident (pp. 0196-7, Vol. II, Part I), could have explained (1) that the full report was available to the Inquiry Group from the time they first requested it, and (2) that the Swiss Government and the utility, NOK, despite strict Swiss laws providing for protecting this type of information, were most cooperative in allowing full disclosure, once the interest of the Rogovin and Kemeny investigators was made known. This full cooperation, by foreign regulatory authorities, is typical, even of those countries such as Switzerland, having the strictest laws protecting proprietary information.

Recognition of the value of such protected foreign information is indicated by the sentence on page 0046, Vol. II, Part 1: "Insisting that the constraints be removed may result in no information being received at all, however."

OFFICE OF PUBLIC AFFAIRS

DETAILED COMMENTS

The Special Inquiry Group, in Volume I, makes two specific recommendations related to public information (10-page 156). Volume II, Part III contains 10 additional recommendations pages 0406-0407). In addition, there are six applicable recommendations elsewhere in Part III (pages 0274-0279). Many of these recommendations are consistent with the Kemeny recommendations in the area of public information. However, in one instance, Rogovin and Kemeny recommendations are in direct conflict and in another Rogovin and the President's response to Kemeny conflict. The recommendations of the Special Inquiry Group go well beyond the task identified in the Action Plan (III-C).

Volume I Recommendations:

1. Provide for public information in NRC and utility emergency plans and coordinate with State and local plans. This effort, which is already underway, generally is consistent with the Kemeny recommendations and does not need to be identified as a separate Task in the Action Plan.

2. A senior NRC official should be the principal spokesman during an accident. This recommendation is in direct conflict with the Kemeny recommendation: "...the utility...should also be primarily responsible for information..." and "...a designated state agency should be charged with issuing all information on this subject" (protective action including evacuation). In reality, OPA believes this must be a coordinated effort involving each agency and the licensee. Resolution should not be achieved by an NRC-mandated task in the Action Plan. Rather NRC's role is largely dependent on a yet-to-be-made decision by the Commission on its responsibility in an accident. Information responsibilities will flow from such a decision.

Volume II Recommendations (pages 0274-0279):

1. The NRC should develop a policy about dealing with briefing requests from public officials, Congress, the media and others. A special onsite team should handle such requests. This goes beyond the Kemeny report and would unduly restrict senior officials of NRC from carrying out their duties to keep the White House and Congress informed. OPA does not agree with this approach.

2. (a) The NRC should advise all other response team members to defer to the special team with respect to media briefings; (b) a single location at or near, the site for all media briefings should be considered; and (c) the NRC should provide guidance on the types of information to be made available. This bears little resemblance to the Keme recommendations. The first part, as a practical matter, probably cannot be implemented--Congress, the Washington-based media and agency officials would not stand still for such a referral. The effort to upgrade utility emergency plans include an on-site press center. The third part has "news management" implications but warrants further consideration. As a whole, this recommendation does not warrant an additional Task in Part III.C. of the Action Plan.

3. The information policy should be issued, along with an implementing procedure, as part of the emergency response plan; the NRC should be prepared to request all officials to refrain from site visits and requests for hearings and briefings if they interfere with the emerge cy response. Since OPA objects to much of the recommendation in this part, its implementation is secondary. The suggestion that NRC seek to ban Congress and other officials from visiting the site or holding briefings or hearings is impractical.

4. The NRC should intensify its efforts to keep accident information on "recorded" telephone tapes. This is a normal post-accident function which we are discussing with FEMA.

5. Individuals should be properly trained to write understandable PNs. This recommendation was not addressed by Kemeny. While PNs were not originally intended to be used as a vehicle for conveying information to the lay public, they nevertheless have been used for that purpose, including Congress. As such, they need to be written much more clearly and simply.

6. The NRC should prepare appropriate documents to assist Government officials and others in understanding nuclear accident terminology. This is not inconsistent with the Kemeny recommendations and should be considered as a modification to an existing Task in Part III.C. of the Action Plan.

Volume II, Part III Recommendations (pages 0406-0407:

1. Utilities should designate a place equipped to serve as a communication center. This recommendation has the same thrust as 3(b) above, and already is under way.

2. A senior NRC official should be the principal spokesman during an accident. This is the same as recommendation 2 in Volume I. It directly contradicts Kemeny.

3. Each utility should hire a member of its staff who has extensive experience in dealing with the news media. This recommendation is not appropriate for NRC action.

4. Each utility should prepare a standard briefing package for each of its plants. We agree.

5. The NRC should establish requirements that will ensure prompt notification of the news media of nuclear accidents. This recommendation is consistent with Kemeny, and it conforms to our existing practice. Either the licensee or the NRC, often both, make prompt modifications.

6. The NRC response teams should include at least two technical individuals to communicate to the NRC Public Affairs staff, a team member to maintain open channels of communication and specific personnel to communicate with on-site personnel to exchange information. This recommendation is consistent with Kemeny but only the first part is applicable to Part III.C of the Action Plan. It should be included as an additional Task.

7. The NRC should choose and train members of the technical staff to be advisers to the news media. This recommendation also is consistent with Kemeny and the effort is being initiated. It could be identified as an additional Task in Part III.C. of the Action Plan.

8. The NRC should develop a standard format for press releases. The Kemeny Report did not address this matter, but "canned" press releases are not an effective public information tool.

9. The NRC should establish a clear policy of issuing prompt public announcements of accidents. This is consistent with Kemeny. Since such a policy has been in effect since the establishment of the NRC, it does not need to be included as an additional Task in Part III.C. of the Action Plan.

10. The NRC should take the lead in working with State agencies to develop a public information program to educate the public. This conflicts with the President's response. FEMA was charged with this general responsibility. Clarification of the two agencies' roles currently is underway outside of the Action Plan.

ENCLOSURE 2

PRELIMINARY ANALYSIS OF THE REPORT OF THE
NRC SPECIAL INQUIRY GROUP

TMI Action Plan Steering Group
February 6, 1980

PRELIMINARY ANALYSIS OF THE REPORT OF THE
NRC SPECIAL INQUIRY GROUP

Introduction

Our preliminary analysis is provided below in four sections that correspond to the four chapters of draft 2 of the TMI Action Plan, NUREG-0660. As a preliminary measure of the degree of compatibility between the plan and the SIG report, we estimate on the basis of reviews to date that approximately 10 to 20 new tasks will need to be added to the plan.

Chapter I - Operational Safety

The Special Inquiry Group (SIG) has made many recommendations that relate to the individual action items and general subject matter of Chapter I of the NRC Action Plan. This is reflective of the emphasis placed on human factors and control rooms by the SIG, consistent with other studies of the accident. The impact of specific program or control equipment recommendations by the SIG on the Action Plan is not expected to be significant. There are, however, many management, organization and policy recommendations relating to operational safety that will need to be factored into the plan, probably in Chapter IV.

Our preliminary analysis of the impact of the SIG report on draft 2 of the Action Plan indicates that Chapter I will be affected as follows, where the headings refer to items in the Task Action Plan, NUREG-0660:

I.A Operating Personnel

The recommendations made by the SIG regarding operating personnel are in basic agreement with this area of the Action Plan. In some cases, items are identified that should and will be incorporated into draft 3 as revisions of present tasks. At this stage of review of the SIG report we have not identified the need to add new tasks in this section of the plan.

I.B Support Personnel

The recommendations of the SIG report with respect to support personnel are in general agreement with other post-TMI recommendations and are, for the most part, appropriately included in Section 1.B of the Action Plan. The SIG report has properly focused considerable attention on broad personnel and organizational type activities. These basic policy issues, such as the chartering of a National Operating Company, will need to be studied further before decisions are made. Such studies can be addressed in Chapter IV of the Action Plan.

Although there is general agreement, many of the SIG's program type recommendations are more specific than task action plans or view the solution in a different way than stated staff positions. For example, several recommendations address the positions and qualifications of licensed operators and the

Shift Technical Advisor. The SIG recommends that the function of the Shift Technical Advisor be performed by supervisory management, rather than by advisors. This is in agreement with the original recommendation of the Lessons Learned Task Force, which proposed the STA concept as an interim approach. The staff's presently approved, but interim, requirement for Shift Technical Advisors, with flexibility afforded individual utilities to choose normal supervisory personnel if they are qualified to perform the function, is still believed to be the best short-term approach pending long-term, general upgrading of the engineering qualifications of operations management personnel and their training in reactor dynamic response and long-term upgrading of control rooms.

The SIG also recommends that the supervisory management on each crew should have an engineering degree. Such matters will be resolved in accordance with the Action Plan, which provides for studies and further staff analysis before long-term upgrading of operations and support personnel. There are other important items in the detailed recommendations of the SIG. We believe that the action items in draft 2, Chapter I, combined with the policy and organizational changes identified in Chapter IV, will assure appropriate recognition of SIG items that relate to this area of concern.

I.C Operating Procedures

The recommendations made by the SIG on operating procedures are in basic agreement with the Action Plan. In some cases, items identified are more specific, recommend a higher priority, or would have been developed in some form as routine staff followup tasks after the completion of already approved studies or studies recommended in the Action Plan. Examples of these include the following:

a. The studies and reviews scheduled in the Action Plan relating to control room designs, human factors, instrumentation, degraded core, and risk assessment will, upon completion, directly impact on orderly procedure development and meaningful training of operating personnel in such procedures.

b. The pilot review program for emergency procedures is an action item that will be implemented by multi-disciplined NRC review teams for all NTOL facilities. The program is also scheduled to be expanded to cover representative operating facilities of different designs. It is to be noted that implementation of the current action plan may affect the priority and the necessity to review emergency procedures for all facilities as recommended by the SIG, if the pilot program findings indicate this to be an appropriate action item. The current action plan does include the orderly review, within a specified time period, of all emergency procedures following the completion of various studies, some currently under way.

There is one SIG finding, without a specific recommendation, that implies that NRC should approve plant procedures. This matter is one the staff has considered in the past (e.g., Lessons Learned Task Force) but rejected, and it is not now included in the Action Plan. Apparently the SIG rejected the idea also in deciding not to elevate the concept from a finding to a recommendation.

I.D Control Room Design

With one exception, the Action Plan is basically unaffected by the SIG recommendations on control room design. The one exception, which has been adopted and included in the near-term operating license requirements list and will be folded into the Action Plan in draft 3, is the qualitative assessment of control room design before new plants go into operation. The control room design studies currently called for in the Action Plan will continue on a priority basis but, in the interim, a qualitative assessment should be required prior to licensing to catch significant problem areas.

I.E Dissemination of Operating Experience

The SIG report places heavy stress on dissemination of operating experience, and so does the Action Plan. We believe that no specific changes are necessary to the Action Plan; however, there are managerial and policy issues which must be resolved that relate to staffing and priorities. Their resolution will affect the performance of this operating experience evaluation function in the long term. Management programs to assure optimum resolution of these questions will be considered for inclusion in Chapter IV of draft 3 of the Action Plan.

I.F Quality Control

There are many recommendations in the SIG report that indirectly relate to the term "Quality Control" but directly relate to the need to improve the safety, quality, and classification of equipment and safety systems, including equipment not designated safety-related. The SIG recommendations are not in conflict with the Action Plan and the staff will give consideration to refinements of the plan in draft 3 to more clearly reflect the SIG recommendations.

I.G Training During Preoperational and Low-Power Testing

There are no recommendations made in the SIG report on training that conflict with the Action Plan. No changes are contemplated.

Chapter II - Siting and Design

Volumes I and II of the SIG report contain a large number of recommendations which relate to the subject matter of Chapter II of the TMI Action Plan. But the general impact of the SIG recommendations, with regard to possible changes or additions to the Action Plan, is relatively minor. The great majority of the SIG recommendations relating to siting and design concerns are already appropriately addressed in the Action Plan, either as specific hardware changes or additional analyses, or as studies which provide the basis for future Commission decisions, rulemaking, or revised regulatory requirements. The reason for this similarity is that the SIG report identified few additional siting and design concerns or conclusions not already addressed in the report of the President's Commission, the various Lessons Learned reports, the Bulletins and Orders activities, or the ACRS recommendations which provided the bases for the current draft 2 of the Action Plan. The few exceptions which would require either revision of existing action items or incorporation into the plan as new tasks in areas already addressed by the Plan are summarized below. One design area that is not treated in the Action Plan at all and that was given high attention by the SIG report is that of standardization of designs. It will be considered for inclusion in draft 3.

II.A Siting

The development of new siting criteria (interim for pending construction permits and long-term rulemaking) as presently described in Section II.A.1 of draft 2 of the Action Plan should consider the emergency planning distances based on maximum dose levels, probability factors, and associated time limits from various projected accidents, as recommended by the SIG.

II.B Consideration of Degraded or Melted Cores in Safety Reviews

The SIG report recommends a number of specific changes in plant design, equipment modifications, and operator training to improve plant response to an accident which may result in a degraded-core condition. The current Action Plan adequately addresses most of these recommendations. A number of these recommendations including a revised "design basis" for nuclear power plants, additional research on degraded- or melted-core behavior, and revised design requirements for support systems will be addressed in the proposed rulemaking on consideration of degraded or melted cores in safety reviews.

II.C Systems Engineering

The SIG report gives considerable attention to the need for improvement in the basis for safety review of reactor design and increased use of quantitative risk-assessment techniques. It calls for the use of a more sophisticated and comprehensive approach to "hazard control" that takes advantage of human-factors techniques as well as significant advances in quantitative risk analyses. The staff generally agrees with these goals and draft 2 of the Action Plan reflects those actions or studies we have presently conceived to accomplish those goals. But there is much offered by the SIG report in this area that needs to be factored into our thinking and planning for the long term. This area should be emphasized by NRR and the RES Probabilistic Analysis Staff in their consideration of changes for draft 3 of the Action Plan.

More specifically, our review of the SIG report to date indicates that the systems reliability and systems interaction studies described in Section II.C should be revised to specifically address the possible need for upgrading of "nonsafety" related systems to some level of "safety grade." The SIG recommendation for a graded scale of significance between safety grade and nonsafety grade is consistent with the Lessons Learned Task Force recommendations but has some practical problems in implementation that need study.

The SIG also calls for specific studies of loss of power to engineered safety features during a critical transient or accident sequence. This has been an outstanding question in reactor safety for 10 years or more for which there are good reasons for and against its inclusion in the design basis. It falls naturally into risk-assessment methods and can probably best be resolved in that context.

II.D Reactor Coolant System Relief and Safety Valves

The primary recommendation in the SIG report related to reactor coolant system relief and safety valves is the need to establish the capability of pressure-operated relief valves (PORVs) to discharge water or two-phase fluid. We believe that the industry test program described in Section II.D of the Action Plan, supported by NRC review and possible confirmatory tests by NRC, will meet the intent of this recommendation.

II.E System Design

In light of the SIG report, the staff will consider whether a new task needs to be added to Section II.E that would require future designs to provide piping configurations which would permit periodic testing of valves at system conditions expected during transients and accidents. This is a pre-TMI concern of long-standing controversy within the technical staff. All other relevant SIG recommendations seem to be adequately addressed in the Action Plan.

II.F Instruments and Controls

In development of draft 3 of the Action Plan, NRR should consider inclusion of a new task for the development of periodic testing criteria for control circuit components at degraded power supply conditions to ensure performance capability. This subject has been of concern since well before TMI, and it does not appear to the Steering Group that the accident itself attaches special new priority to its resolution, but the testing approach suggested by the SIG may be an efficient means of achieving early resolution of the concern. It deserves priority attention for that reason.

<u>Sections II.G, H, J and K</u>

Very few SIG recommendations relate to Sections II.G, H, J, and K of the Action Plan. Those identified are addressed in the Action Plan in a manner consistent with the SIG findings; such as, possible licensing of reactor NSSS vendors and architect-engineers.

<u>Chapter III - Emergency Preparedness, Public Information and Radiation Protection</u>

The report of the Special Inquiry Group has identified a large number of recommendations for improvement in emergency preparedness, public information, and radiation protection. About half of the recommendations in Volume II of the SIG report are in these areas.

Despite their large number, these recommendations will likely lead to minimal change in Chapter III of the present Action Plan. That is, there are no significant new findings not previously recognized and accounted for in the Action Plan. The SIG recommendations are generally more detailed and prescriptive than the action items in the Action Plan, and certainly those details should be considered by the organizational entities that will execute the plan. But, in most instances, the thrust of the SIG recommendations are consistent with the NRC staff thinking that led to the tasks in this area of the Action Plan. Any judgment differences that we perceive at this time are more in the area of how to do something than in whether to do it.

III.A NRC and Licensee Preparedness

There were relatively few recommendations in the area of licensee preparedness. Most have already been fully incorporated in the Action Plan. One new thought has not. The SIG recommends that an inoperable emergency plan should be treated in the same way as an inoperable engineered safety system. That is, if for any reason (e.g., flood, blizzard, civil disturbance, etc.), the emergency plan could not be executed over some period of time, the licensee should be required to notify the NRC, who would then determine whether the plant should be shut down. This thought has not been included in the proposed emergency plan rule and will be considered by the staff in developing the final rule.

The SIG report contains a large number of very specific recommendations that deal with the NRC role and organization in an emergency. It concludes that management of NRC's overall response should be from an on-site location, with a response team headed by the Regional Director, or his alternate. The role of the NRC Headquarters Emergency Management Team (EMT), according to the SIG report, should be one of providing support to the on-site group, when asked. The SIG concludes that the Commission "should not interject itself into the management's response to an emergency," but that the predesignated emergency response organization should be relied on. These recommendations are in stark contrast to the actual roles of the various NRC organizational elements that evolved during the TMI-2 accident. The NRC Office of Inspection and Enforcement is revising the NRC emergency response program to reflect lessons learned from TMI-2 and, as indicated in Action Item III.A.3.1, will be interacting with the Commission in the development of a clear statement of NRC's overall role in responding to emergencies. This item is also included on the staff's recommended list of requirements for ending the licensing pause. Insofar as the related items in the Plan, the staff will consider the specific recommendations of the SIG in developing draft 3. It appears likely, however, that a different concept than envisioned by the SIG of the relative roles to be played by the Commission, the EMT, the headquarters staff, and the Regional offices will emerge. As a result, many of the detailed recommendations in the SIG report may not be adopted. Such differences as develop, however, will be more the result of differences in judgment on the details of how to accomplish the overall NRC role than fundamental differences regarding what the overall NRC role should be.

II.B Emergency Preparedness of State and Local Governments

The SIG recommendations on emergency preparedness will not significantly affect the Action Plan. The ongoing rule-making action (amending Appendix E of 10 CFR Part 50) and the ongoing coordinating activities with FEMA being carried out under the recently executed memorandum of understanding are generally consistent with the SIG recommendations. Some exceptions follow:

a. The SIG report concludes that FEMA, rather than NRC, should approve State and local government emergency plans. Because of its responsibilities under the Atomic Energy Act of 1954, as amended, it still is necessary that NRC make the final decisions regarding the overall adequacy of emergency preparedness (i.e., the integration of emergency preparedness onsite under the control of a licensee and regulated by NRC and offsite as determined by FEMA and reviewed by NRC).

b. The SIG recommends that operating reactors that cannot meet criteria for minimum evacuation areas (to be developed) be shut down unless, among other things, the President determines that temporary continued operation of the plant is vital to the national interest. The Action Plan (Action Item II.A.2) includes a critical examination of plants located in areas of high population density. One possible outcome of this examination is that it could lead to a recommendation for shutting down facilities. We believe this is a proper function of NRC. The Commission will need to give consideration to whether it should initiate action, as implied in the SIG report, to establish a legislative mechanism that provides for a Presidential determination regarding the national interest in such cases.

III.C Public Information

The SIG recommendations regarding public information in some instances go beyond the measures described in Section III.C of the Action Plan. With three exceptions, we believe the recommendations are valid and the Action Plan will be modified accordingly. The following are the exceptions:

a. The SIG recommends that the senior NRC official onsite should be the principal spokesman at press conferences. The President's Commission recommended that this lead role should be played by the utility. The staff has earlier concluded that the handling of press conferences following an accident should be a coordinated effort involving the NRC, the utility, the State, and other supporting Federal agencies. Each of these entities would have certain prescribed responsibilities and lead interests related to the accident. Accordingly, it has been the staff position that the designation of any single spokesman is not necessary or useful.

b. The SIG recommends that NRC should develop a standard format for press releases to ensure inclusion of basic information concerning a nuclear accident. To the extent that this recommendation was meant to result in a regulatory requirement to be placed on licensees, it is doubtful that the staff effort to develop such a standard format is justified.

c. The SIG recommends that NRC take the lead in working with State agencies to develop public information programs on nuclear power and its consequences. The staff is presently negotiating with FEMA, either as an amendment to the existing memorandum of understanding or as a separate agreement, to define the public information responsibilities of the two agencies. It is not clear at this time that NRC will have the lead role envisioned by the SIG.

III.D Radiation Protection

No major new tasks have been identifed from our preliminary review of the radiation protection recommendations in the report of the Special Inquiry Group. In several instances, the SIG recommendations go farther than was envisioned in the related task description in the Action Plan. We will be considering these recommendations in more detail in preparing draft 3 of the Action Plan and expect that, for the most part, the action plans will be revised as necessary so that the detailed recommendations of the SIG will be specifically considered in the execution of the plan. At this point in time, we do not consider that these revisions would significantly increase NRC or industry resource requirements in this area.

Chapter IV - NRC Organization, Management, Practices and Procedures

The Special Inquiry Group concluded that "the one theme that runs through the conclusions we have reached is that the principal deficiencies in commercial reactor safety are not hardware problems, they are management problems." Whether one agrees or disagrees with this simplified summary of the "bottom line" (recall that the President's Commission didn't say "mismanagement," it said "bad attitude"), the SIG clearly had much to say about the management of and by NRC. Over the next few weeks, a comparable degree of attention will need to be directed by the Commission and its principal line officials to the recommendations of the SIG and to the prompt resolution of compelling management and organizational deficiencies of the agency in the wake of the accident and the quagmire of the licensing pause.

The relationship of the SIG recommendations in this area to draft 2 of the Action Plan is summarized below.

IV.A Overall Policy and Organization

The SIG recommendations impacting on Task IV.A of Chapter IV, which deals with overall NRC policy and organization, are for the most part found in Volume I of the SIG report.

The SIG recommendation that there be articulated a substantive risk objective for nuclear power plants for public discussion and Executive and Congressional consideration (Vol. I, p. 152) relates to Task IV.A.1. This recommendation goes beyond the identified Task and would require the inclusion of a risk objective in the Task.

The SIG recommendation that the NRC be headed by a single chief executive relates to Task IV.A.2 (Vol. I, p. 115). This recommendation conflicts with the Action Plan. However, no change in the Action Plan appears warranted in view of a prior Commission and Executive Branch decision not to pursue this course of action.

The SIG recommendation that a single director of the EMT be designated with exclusive authority (Vol. I, p. 134) relates to Task IV.A.3 but adds a different thought which warrants consideration of a modification of this Task.

The SIG recommendation (Vol. I, p. 117) that NRC give high priority to locating the agency in a single location and to promptly relocating the Commissioners and their personal staffs in Bethesda for the interim relates to Tasks IV.A.4 and IV.A.5. This recommendation generally parallels the Task Action Plan and no change appears warranted except for the inclusion of the interim proposal in the short-term plan.

The SIG recommendation (Vol. I, p. 141) that the NRC significantly limit the ex parte rule and apply it more rationally is encompassed within Task IV.A.6. No modification of the plan appears warranted. The SIG recommendation (Vol. I, p. 142) that the Appeal Board be abolished is encompassed within Task IV.A.6. No modification of the plan appears warranted. The SIG recommendation (Vol. I, pp. 120-121) that consideration be given to the transfer to other agencies of NRC functions which are not safety related is covered in Task IV.A.7. No modification of the plan appears warranted. Some of these transfer issues were treated in the letter to Dr. Frank Press of October 9 and the President's statement of December 7, 1979, and were apparently resolved at that time.

The SIG recommendation (Vol. I, p. 117) that NRC consolidate NRC resources relating to monitoring operating reactors in a single office relates to Task IV.A.9. Since this recommendation is more specific than the Task Action Plan, a modification would be required to incorporate it. The SIG recommendation (Vol. I, p. 140) that the ACRS be retained in a strengthened role is encompassed in Task IV.A.11. No modification of the plan appears warranted.

The SIG recommendations (Vol. I, pp. 142-144) that the NRC establish an Office of Public Counsel and adopt a program for intervenor funding relates to Task IV.A.13. Both recommendations appear to be encompassed within this task and, accordingly, no modification of the plan appears warranted.

The SIG recommendation (Vol. I, pp. 117-119) that an independent Nuclear Safety Board be established is not presently included in the Action Plan. This recommendation could be included as a new task which would call for a study as to the need for such a Board. The SIG recommendation (Vol. I, p. 141) that the NRC abolish the two-step licensing process relates to Task IV.A.15. Inclusion of this recommendation would require modification of the plan.

IV.B Staff Organization and Practices

The SIG recommendations impacting on Section IV.B.1, which deals with the increase of emphasis on human factors, are generally consistent with the Plan. Its recommendations concerning an interdisciplinary organizational unit for human factors (Vol. II, Pf2, p. 0387 and Vol. II, Pf2, p. 0669) are consistent with but an extension of Action Items IV.B.1 and IV.B.2, since these tasks are in terms of increased attention to human factors (and in IV.B.2 to other aspects of enforcement in individual offices). The assessment of whether to create a separate interdisciplinary unit should be included in Task IV.A.9, "Reexamine organizations and functions of NRC offices." The SIG recommendation that NRC Staff be given improved training in design and operation of actual plants (Vol. I, p. 120) would be an extension of Action Item IV.B.6. In-plant training probably should be extended to those having a reasonably direct effect on plant design and operating characteristics.

There are a number of other recommendations which touch upon the general subject area of Task IV.B but which go beyond the specifics of the present Task IV.B. The recommendations for rotation of managers (Vol. I, p. 120) and studying ways to reduce office insulation (Vol. II, Pf1, p. 0188) appear to be related to the studies identified in Action Item IV.A.9. The recommendation to modify organization to improve attention to radiation protection (Vol. II, Pf2, p. 0039) is really the gist of the combined effect of a number of the Task Action Plans in Part III. Similarly, the recommendation to establish a headquarters-based incident investigation team (Vol. II, Pf3, p. 0669) will be a part of Action Items IV.A.8, IV.A.9 and III.A.3.

The recommendation to establish and enforce maximum working days for steps in the Board notifications process (Vol. II, pf, p. 0190) is not in the plan and probably need not be included since action has already been taken to correct the problem. The recommendation to improve attention to utility fiscal incentives (Vol. II, Pf1, p. 0355) is one that will be added to the Plan. The recommendation to designate a unit to track and publicize the resolution of TMI-related issues (Vol. II, Pf3, p. 067) could be factored into Action Item IV.A.8.

The recommendation to exercise better management control over work priorities is the basic thrust of the effort to prepare an action plan in the first place. Upon its completion, after priority ordering of the tasks relative to non-TMI activities of the NRC, it will be a tool of great significance in enabling the agency management to exercise better control over agency work priorities.

IV.C Improve Followup on ACRS Advice

There appears to be nothing in the SIG report that directly impacts on or warrants any modification on the task of improving followup on ACRS advice.

IV.D Identification and Resolution of Safety Issues

The SIG recommendations that might impact on identification and resolution of safety issues (Task IV.D) appear to be somewhat diffuse and scattered throughout the SIG report. Within the limited time available for this initial review of the SIG report it can only be said that the incorporation of the SIG recommendations will likely require modification of Tasks IV.D.1 and IV.D.2. The present version of Task IV.D.3 appears to be compatible with the intent of the SIG report and will not likely require modification.

IV.E Improvement of Safety Rulemaking Procedures

The SIG report did not specifically address the question of rulemaking procedures, although it did call for increased use of rulemaking (Vol. I, p. 142) and improvement of the quality of regulations (Vol. I, p. 140; Vol. II, pp. 0023, 0036). Specifically, the SIG called for the designation of an organization "to have primary responsibility in the rulemaking area to assure that the quality of the regulations are adequate" (Vol. II, pp. 0041-42); this appears to be within the intent of Action Items IV.A.8, 9 and 10.

Task IV.E, as presently structured, recognizes the need for improvement in the quality (as well as the content) of the regulations and need not be changed to incorporate the observations of the SIG on this point. The recommendation of the SIG regarding the establishment of a new quality control group for rulemaking is not supported in the SIG report, and seems impractical since quality of regulations reflects technical, legal, policy and editorial considerations. The present process, which involves SD, OELD, one or more affected "program offices," public comment, Commission Staff offices, and the Commission itself, provides ample opportunity for quality control. Since there is no basis provided for the recommendation, it is not clear whether it's more a question of poor quality regulations issuing from NRC or poor use of rulemaking to resolve technical issues that is of concern.

IV.G New Section for Draft 3

Within Chapter IV we have identified the need to include a new Task IV.G to accommodate the SIG recommendations regarding conflicting responsibilities of regulatory authorities (Vol. II, Part 1, p. 0352 et. seq.). It would appear that the various elements of the recommendation will warrant substantial study, since they encompass the regulatory authorities at the state and Federal levels and involve a variety of financial and economic considerations.

ENCLOSURE 3

COMPARISON OF RECOMMENDATIONS IN VOLUME I OF SIG REPORT, NUREG/CR-1250, WITH DRAFT 2 OF TMI ACTION PLAN, NUREG-0660

COMPARISON OF RECOMMENDATIONS IN SIG REPORT
NUREG/CR-1250, VOLUME I, WITH DRAFT 2 OF
TMI ACTION PLAN, NUREG-0660

Key:

Impact on TMI Action Plan

a. Recommendation is adequately covered in Draft 2 of the Action Plan; no revision to the plan is necessary.

b. There is a related TMI Action Plan Task, but the SIG recommendation adds a new or different thought; Draft 3 of the Action Plan will include consideration of the SIG recommendation in the related task description.

c. There is no directly related TMI Action Plan Task and the recommendation merits consideration as a new Task; Draft 3 of the Action Plan will include a new task that responds to the recommendation.

d. Staff or Commission do not agree with recommendation; no action will be taken.

Note: When related TMI Action Plan is listed as "none," the proposed new task number is identified.

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
A. GENERAL		
1. How Safe Is Safe Enough		
a. The Administrator should formulate an ultimate safety objective for the regulatory program in the first instance for review and approval by the President and Congress, and then when a standard is approved should apply it (p. 116).	IV.A.1	a
b. Decisions about the ultimate safety objective in the regulated program and about the expansion or reduction of our country's reliance on nuclear power should be made by the Executive and Congress as part of our national energy strategy (p. 91, 116, 151).	IV.A.1	a
2. Oversight Over the NRC		
a. Firm commitment on the part of the President and the congressional oversight committees, and a commitment by the public--if what it wants is safer nuclear power plants--to keep the pressure on elected representatives for major,meaningful reform (p. 92).	None	-
b. Congressional oversight committees should hold the NRC accountable with respect to outstanding generic items (p. 93).	None	-
3. Public Education		
a. Renewed effort must be made to educate the public as to the actual risks of nuclear power and that the risks and benefits associated with nuclear power plants must be weighed against the very real health and environmental risks associated with other forms of power generation (p. 91).	III.C	a
b. Substantial efforts are necessary to provide information to the public about actual radioactive releases during the TMI accident and their actual hazards, as opposed to perceived hazards. The NRC should play an effective role in this task (p. 154).	III.C	a

SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
4. Moratorium or Suspension of Licensing Reviews		
a. The NRC should satisfy itself that every licensee for an operating reactor has evaluated: (1) the management and technical qualifications of its site crews and site management and their familarity with the plant; (2) the adequacy of emergency operating procedures; (3) possible significant human factors or instrumentation problems in the control room; (4) and their training program for operators (p. 146).	(1) I.B.1.1 (2) I.C.7 (3) I.D.1 (4) I.A.2.1, I.A.3.1, I.G, II.B.4	a a b a
b. The NRC would be wise to suspend processing of applications for Construction Permits and Limited Work Authorizations until it considers the various recommendations we have made for reforming the licensing process and for increased standardization (p. 92).	None (IV.A)	c
5. Statutory Base		
Changes will require new legislation, executive reorganization, and substantial overhaul of the way the NRC is organized and managed, at the very least (p. 92).	IV	a
B. EVALUATION OF OPERATING EXPERIENCE		
1. Basic Responsibility		
a. Operating information must be evaluated both by industry and the NRC to identify items of potential safety concerns, and these must then be investigated in depth (p. 97).	I.D, I.B.1.4	a
2. Office of Analysis and Evaluation of Operating Data (AEOD)		
a. The Office of Analysis and Evaluation of Operational Data (AEOD) should be given the task of developing recommendations as to where actions to meet operating problems ought to be required. These recommendations should, in each instance, be required to be rejected, modified, or imposed by the appropriate program office of the NRC within a fixed period of time. Unresolved disagreements between AEOD and a program office could be required to be reviewed by the Commission or Administrator (p. 99).	I.E.1, IV.A.9, IV.B.7	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
b. Present NRC staff functions devoted to performing quantitative risk assessment of reactors should be relocated in AEOD (p. 99).	I.E.1, IV.A.9 IV.B.7	b
c. AEOD should be staffed in part on a rotational basis from all the other offices and branches at a level of no less than 35-40 professionals (p. 99).	I.E.1, IV.A.9	b
d. To aid AEOD, consideration should be given to a revised comprehensive reporting system applicable to both utilities and vendors. This system should require more in-depth reporting and followup of significant events, provide for the reporting of minor incidents in a separate format amenable to statistical analysis, and include reporting requirements that are uniform so that the data have a common basis (p. 99).	I.E.5, I.E.6	a
3. Inspection of Plants		
a. More emphasis should be given to supplementing the resident inspector with a team or blitz approach in which a group of inspectors make unannounced visits to conduct in-depth inspections of the overall operation of a plant (p. 100).	I.B.2	b
b. More attention should be given to reactive inspections (responding to notifications, complaints, specific problems, or following up on previous difficulties) (p. 100).	I.B.2.1	b
c. I&E should develop new programs to monitor and evaluate utility management and technical competence on an on-going basis (p. 100).	I.B.2.3, I.B.2.4	a
d. More effort should be devoted by regional offices to evaluating each utility across the board and vis-a-vis other utilities in order to identify weak spots and problem areas. If the operation is not judged to be satisfactory, then the reactor should be required to be shut down (p. 100).	I.B.2.3	a
e. The proficiency of the IE staff and management should be increased by staff rotation (field and headquarters) and by conducting regular seminars, attended by both NRC management officials and inspectors, in order to identify problems the inspectors are encountering (p. 101).	IV.B.2	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
4. Institute of Nuclear Power Operation (INPO)		
a. We urge the rapid implementation of the industry-wide Institute of Nuclear Power Operation (INPO) to evaluate operating information and problems and police and upgrade the management and operating competence of its members (p. 110).	I.A.3.6	a
C. ONSITE PERSONNEL AND PROCEDURES		
1. Training		
a. There is a clear need for more operator training with emphasis on response to emergencies and on system diagnosis (p. 105).	I.A.2.1, I.A.2.6	a
b. The NRC should assume a direct role in the training of operators including certification of training facilities, establishment of a minimum curriculum, and certification of instructors (p. 105).	I.A.2.7	a
c. Operators must be trained as a team on the simulator with more emphasis on response to emergencies and on system diagnosis (p. 105).	I.A.2	b
2. Technical Expertise		
a. NRC should require every licensee to hire a cadre of graduate engineers knowledgeable in reactor engineering and physics. Each should be provided with training in the specific characteristics of the plant, with special emphasis on integrated plant response and transient behavior. The utility should be required to deploy at least one such engineer supervisor whose qualifications have been examined by the NRC as shift manager (not as an "advisor") on every shift (p. 106).	I.A.1.1	a
b. A substantially more detailed and upgraded set of requirements should be developed by the NRC for technically competent, NRC-certified, supervisory and management officials to be present on each shift to direct operations (p. 106).	I.A.2.1, I.A.2.6	a

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
3. Station Manning		
a. Minimum manning requirements for each shift need to be increased by the NRC (p. 106).	I.A.1.3, I.A.1.5	a
D. INDUSTRY-WIDE TECHNICAL RESOURCES		
1. Data and Analysis Centers		
a. One or more data centers should be established by the industry, manned 24-hours a day by nuclear experts, to which essential plant parameters would be telemetered automatically (p. 107).	II.A.3	b
b. NRR's "Lessons Learned" Report proposed that each utility be required to maintain a data center of its own where important plant parameters could be read. Additional stations such as these would be useful (p. 108).	III.A.3.4	a
2. Industry-Wide Consortium		
a. A number of existing plants now owned by different utilities could be owned an/or constructed or operated by an industry-wide consortium or a public corporation similar to COMSAT (p. 110).	None (IV.B.7)	c
b. Utilities not meeting safety requirements regarding technical or management competence could be placed into "receivership" by the NRC. Their operation (or construction) then would be undertaken by the consortium as a condition of the NRC license (p. 110).	None (IV.B.7)	c
E. NRC ORGANIZATION		
1. Single Chief Executive		
a. There is a central and overwhelming need for legislative and executive reorganization to establish a single chief executive with the clear authority to supervise and direct the entire NRC staff. We do not believe that the current administration's proposal to strengthen the NRC Chairman's executive authority goes far enough. (p. 115, 117).	IV.A	d

SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
b. The single administrator should be responsible directly to the Executive Branch, and to Congress through strong congressional oversight.	IV.A	d
2. Consolidation of Resources Devoted to Operating Reactors		
a. All the agency's resources related to monitoring operating reactors should be consolidated in a single office, probably the current Office of Inspection and Enforcement. The basic design approval function should remain in NRR. Mechanisms should be developed for better coordination between the licensing and operation monitoring offices of the NRC, including strong project management teams to monitor construction and testing of new reactors that have ties to both offices (p. 99, 117).	IV.A.9	b
3. Independent Nuclear Safety Board		
a. There is a clear and pressing need for an independent organizational entity within the agency that will be responsible for observing, evaluating and making recommendations to improve the quality of the overall performance of the regulatory staff. This need can best be satisfied by the establishment of an Independent Nuclear Safety Board.	IV.A.12	c
b. The Board should be composed of a number of persons who are trained in technical disciplines associated with nuclear safety and radiation protection and who are thoroughly experienced with the licensing and regulatory process (p. 118).	IV.A.12	c
c. The Board should not duplicate the functions of any office or provide another layer in the review process, but should instead: (1) exercise oversight in the effectiveness of the licensing review process and the regulation of existing plants; (2) advise the Commissioners on regulatory goals and important issues for rulemaking: (3) act as an Ombudsman group to receive complaints and technical dissents; (4) enhance reactor safety by monitoring the effectiveness of the Office of Analysis and Evaluation of Operational Data; (5) monitor the staff's use of the late analytical and design tools; (6) develop and maintain a capacity to investigate accidents and important safety-related incidents, independent of other offices and the Commission or Administrator; and (7) provide a quality assurance function for the agency's regulatory process as a whole (p. 118, 119).	IV.A.12	c

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
4. Project Management		
a. Strengthening of project management is necessary to obtain an overall balance in the staff's safety evaluations (p. 119).	IV.A.9	a
b. The need for overall plant and systems analysis has been clearly recognized and should be coordinated through a strong project management organization (p. 119).	IV.A.9	a
5. Periodic Manager Reassignments		
a. There should be an exchange or rotation of senior level managers on a more pre-planned basis (p. 120).	IV.A.9	a
6. Staff Training		
a. The agency should establish a policy that practical experience in the design, construction, and operation of nuclear power plants and in the problems of radiation protection is a requisite for key staff personnel and arrange an effective program to obtain this experience for the appropriate individuals (p. 120).	IV.B.6	b
7. Transfer of Non-Health and Safety Responsibilities		
a. Present NRC responsibilities that do not relate to radiological health and safety should be considered for possible transfer to appropriate agencies. Examples are antitrust responsibilities and jurisdiction over export licenses (p. 121).	IV.A.7	a
8. NRC Office Consolidation		
a. We recommend that high priority be given to locating the entire agency in a single location. In the interim, the offices of the Commissioners and their personal staff should be promptly relocated in Bethesda, Maryland (p. 117).	IV.A.4, IV.A.5	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
F. HUMAN FACTORS ENGINEERING		
1. Instrumentation		
a. The NRC should develop new standards for instrumentation, computers, print-out devices, CRTs, and other digital displays to facilitate information transfer (p. 127).	I.D.4	a
b. Every nuclear plant should be required to install the equivalent of a reactimeter that constantly monitors important plant parameters and is tied to an information and display computer that can call up these parameters on an instantaneous or trend basis. This information would also be telemetered to the offsite data center (p. 127).	II.F, I.D.1	b
c. Distrubance analysis systems should be developed to provide operators a clearer picture of reactor conditions (p. 127).	I.D.5	a
2. Control Room Design		
a. Using human factors engineering, the NRC should move forward to develop standardized criteria for control room design such as parameters to be displayed, fundamental grouping of instruments and controls, panel layout, and alarm systems. A deadline for implementation of these requirements related to control room design should be established and enforced (p. 128).	I.D.1, I.D.4	a
G. MORE REMOTE SITING AND IMPROVED EMERGENCY PLANNING		
1. More Remote Siting		
a. Future reactors should be located only at sites that are at least 10 miles, and perhaps more, from any significant center of population (p. 130).	II.A.1, II.A.2	a
b. Specific criteria for reactor siting should be developed promptly by the NRC in conjunction with other federal and state agencies with experience in emergency evacuation. Considerations should be given to the specific characteristics of the area that influence the effectiveness of evacuation, such as population density, population centers beyond 10 miles, and evacuation routes (p. 130, 131).	II.A.1, II.A.2	a

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
2. Emergency Planning		
a. Evacuation of citizens at least 10 miles around a nuclear power plant must be considered as an independent means of protection over and above the engineered safety features designed to mitigate an accident and prevent radiological releases. Distances should be regarded as the ultimate defense-in-depth barrier (p. 130).	III.A.1.1, III.A.2, III.B	a
b. Federal emergency planning functions for accidents at nuclear reactors should be consolidated into a single federal agency. The new Federal Emergency Management Agency (FEMA) appears to be the appropriate agency for such planning (p. 131).	III.B	a
c. FEMA and the NRC must coordinate closely on emergency planning and FEMA should make maximum use of the work that the NRC has already done and is presently doing (p. 131).	III.A.2.2, III.B	a
d. The specific details of the emergency plan must be worked out at county and local levels (p. 132).	III.A.1.1, III.A.2.2	a
e. Consideration must be given by the NRC and FEMA to the methods by which funds can be made available to local communities near nuclear plans for emergency planning. Two possible options are: (1) specific Federal grants could be provided for such activity and (2) the NRC could require utilities to pay for local planning efforts (p. 132).	III.B	a
f. Workable State emergency plans, approved by FEMA, should be a prerequisite to continued operation of existing and future reactors (p. 132).	III.B	a
g. Plant operation should not be made absolutely contingent on approved local plans since this would, in effect, give local municipal governments the power to close a plant (p. 131, 132).	III.A.2.2, III.B	a
h. The emergency plan should not be just an abstract document. It should make realistic provisions for such seasonal or other variations as snow storms and large summer populations; and it should provide that the plant may have to be shut down, if the plan becomes inoperable for more than a short period of time (p. 132, 133).	III.A.1.1, III.A.2.2	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
i. In establishing specific emergency planning distances, probably maximum radiation doses from various projected accidents at different distances from a plant should be carried forward by the Commission into specific criteria that incorporate maximum dose levels, probability factors, and associated time limits (p. 133).	III.A.2	c
j. Once criteria for minimum workable evacuation areas are established by the NRC, prudence dictates that plants that cannot meet these criteria should be shut down, unless: (1) additional safety systems for the mitigation of accidents can be installed either to reduce the area of likely accident consequences or to increase the permissible time for evacuation; or (2) there is a determination by the President that the temporary continued operation of the plant is vital to the national interest (p. 133).	II.B.6, III.A.2	a
3. **NRC Emergency Response**		
a. The Executive Management Team (EMT) should have a single director who should exercise the authority of the entire agency during an emergency (p. 134).	III.A.3.1	b
b. Any decision by NRC headquarters to recommend evacuation should be made by the director of the EMT and thereafter should be communicated to the State authorities by the highest official of the NRC available (p. 134).	III.A.3.1	b
c. FEMA and other federal agencies involved should have senior representatives present at the NRC's Incident Response Center (p. 134).	III.A.3.1, III.A.3.c(2)	b
d. The NRC emergency response plans should be revised to shift the management of the NRC's overall response to the site as quickly as possible (p. 135).	III.A.3.1	b
e. The onsite NRC official must have enough clout to assume control of the agency's overall response (normally at least the Regional Office Director level) (p. 135).	III.A.3.1	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
f. NRC must not be constrained by its own attitudes to take a passive role during an accident, if circumstances demand more direct intervention. If NRC's regulations should make clear the authority to demand information, and in the extreme case to impose its own decisions on a licensee (p. 94). However, the senior NRC official at the site and his technical team should not be authorized to assume command of the plant (p. 135-136).	III.A.3.1	b
4. Radiological Monitoring		
a. DOE should be formally designated by Executive Order as the lead coordinating federal agency, to organize the emergency resources of all other federal agencies in the case of an accident at a commercial nuclear plant requiring radiological monitoring (p. 137).	III.A.3.6(2)	a
b. EPA should be assigned long-term radiological monitoring responsibilities after an accident; and HEW should be given the lead responsibility for population dose assessments and calculation of health impacts (p. 137).	III.A.3.6(2)	b
c. Serious consideration should be given to installation of real time, on-line radiological monitoring devices around every nuclear plan in concentric circles at various distances. These instruments should be capable of being read from the plant control room or some other remote site (p. 137).	III.D.2.4	a
H. OVERHAUL OF THE LICENSING PROCESS		
1. Advisory Committee on Reactor Safeguards		
a. The ACRS has a distinct role to play in the regulation process and that role should be strengthened (p. 119).	IV.A.11	a
b. The Nuclear Safety Board might be composed of five full-time members who would also be members of the ACRS. The ACRS would be composed of these five members plus 10 part-time members (p. 119).	IV.A.12	a
c. Additional staff should be provided to the ACRS. This recommendation could be met by having the Nuclear Safety Board staff provide support for the ACRS (p. 119).	IV.A.11, IV.A.12	a

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
d. The ACRS's role should be strengthened by removing the requirement that it advise the Commission on every license application, encouraging it to play a more formal role as a party in licensing and rulemaking proceedings, and by upgrading its staff (p. 140).	IV.A.11	a
e. The ACRS's independence could be enhanced by decreasing the tremendous time commitment required, so that membership can be offered to individuals who cannot afford to devote half or more of their time to the ACRS (p. 140).	IV.A.11	a
2. Ex-parte Rule		
a. The ex-parte rule should be very significantly limited and applied more rationally. Commissioners should become involved in safety issues pending in particular cases, as long as their involvement is on the record (p. 141).	IV.A.6	a
3. One-Step Licensing Process		
a. The two-step licensing process should be abolished for nuclear plants of conventional design. Instead, a single licensing review should be held prior to construction (p. 141).	IV.A.15	b
b. Once a license is granted, jursidiction to oversee construction and confirm that the plant is constructed and consistent with the design plans should be placed in the NRC staff (p. 141).	IV.A.15	b
4. Atomic Safety and Licensing Appeal Board		
a. The Atomic Safety and Licensing Appeal Board should not, by default of the Commission, have to continue to interpret, "improve" and apply ambiguous standards (p. 140).	IV.A.6	a
b. If the Commission is replaced, the Atomic Safety and Licensing Appeal Board's decisions on granting a license should be final and any appeal from the Appeal Board should be directly to the Federal Court (p. 141).	IV.A.6	a

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
c If the Commission is retained, consideration should be given to abolishing the Licensing Appeal Board and requiring the Commission to consider and approve every new reactor license. Appeal Board members could be transferred to a support office to assist the Commission in this work (p. 142).	IV.A.6	a
5. Rulemaking		
a. Generic safety issues and other important policy issues should be handled by the agency or the Commission directly, through rulemaking and policy directives (p. 142).	IV.D.2.2	a
b. Important decisions that lead to the establishment of required safety levels should be promulgated by agency policy through a more open and definitive procedure (p. 142).	IV.B.7	a
c. Steps should be taken to eliminate possible protracted public hearings on individual rules and to ensure that the amount of public input is appropriate to the substantive issues involved. For example, rulemaking can often be carried out by consideration of written comments, rather than through public hearings (p. 142).	IV.E.4	a
6. Office of Public Counsel		
a. An Office of Public Counsel should be established reporting to the head of the agency. The primary functions of the office should be to: (1) provide a source of legal and technical counsel to potential or actual intervenors and to public interest groups; (2) intervene as a party directly in agency rulemaking or licensing proceedings when appropriate; (3) fund and monitor, where appropriate, independent technical peer reviews; and (4) handle details of intervenor financing (p. 143).	IV.A.13	a
7. Intervenor Funding		
a. A program of funding of individual intervenors or groups of intervenors should be adopted for both licensing and rulemaking proceedings, administered through the Office of Public Counsel (p. 143).	IV.A.13	a

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
b. Strict requirements should be established that funding be conditioned upon the intervenor propounding non-frivolous issues that are not being effectively advanced by others (p. 144).	IV.A.13	a
c. Funding should be appropriate to the effort necessary with the final decision on reimbursement being made by the Office of Public Counsel, the Licensing Board, or (in rulemaking proceedings) by the Commissioner or Administrator (p. 143, 144).	IV.A.13	a
8. Standardization		
a. Use of standardized designs should be required for all future applications, unless the Commission or administrator grants an exemption for good cause (p. 144).	IV.D.2	b
b. Once a standard model plant is under construction, it should be treated by the NRC as if it were already in operation for purposes of deciding whether new design changes should be required. If a design change were clearly needed to make operating reactors safe, then the change should also be made on those "standard models" under construction (p. 144).	IV.D.2	b
9. Regulatory Requirements Review Committee		
a. The Ratchet Committee's function is of sufficient importance to warrant its deliberations to be reported in some depth if not actually transcribed completely (p. 146).	IV.B.7	b
b. The voting members of the Ratchet Committee should be lower than Office or Division Director level (p. 146).	IV.B.7	b
c. Additional steps should be taken to increase the opportunity for industry, public, and ACRS involvement in the issues considered by the Ratchet Committee functioning (p. 146).	IV.B.7	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
10. Bases for Safety Reviews		
a. The present licensing review process, including design basis accidents, safety related systems, and the single failure criterion, should now be amalgamated with and ultimately supplanted by a more sophisticated and comprehensive approach to "hazard control" that takes advantage of human factors techniques as well as the significant advances in quantitative risk analyses (p. 148).	IV.B.7	a
b. The best way to improve on the existing design review process is to place increasing reliance upon quantitative risk analysis, emphasizing those accident sequences that contribute significantly to risk. We do not suggest that the existing safety review process be supplanted immediately by a more probabilistic review. This will be a long process, but the present review process should be augmented and quantitative methods used as the best available guide to which accidents are the important ones, and which approaches are best for reducing their probability or their consequences. (p. 150).	IV.B.7	a
c. A hybrid approach to the transition might be appropriate which includes the following: (1) an expanded spectrum of design basis accidents used for safety assessment purposes by using operational experience, research results, lessons from accidents, and advice from the ACRS, all studied through quantitative risk analysis; (2) the effects of multiple equipment and human failures, where the risk of occurrence is significantly high; (3) a risk related scheme for classification of equipment on the basis of safety significance; (4) human factors considerations and operational procedures in the review process; and (5) on a selective basis, a determination whether some design features to mitigate the effects of some Class 9 accidents should be required (p. 151).	IV.B.7	a
d. A thorough review should be made of loss of core cooling and the resultant core damage to determine if certain predictable consequences might be substantially mitigated by design improvements of less than staggering cost or complexity. Such improvements should be specifically evaluated in the normal design review process. Specific examples are: (1) expedited consideration should be given to the use of vented, filtered containment systems to guard against the high pressure rupture of existing containments; and (2) redesign should be undertaken of some of the waste gas and filtering systems that will inevitably be exposed to water and gas coming from the primary system during a major accident (p. 151).	II.B	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
I. OCCUPATIONAL AND PUBLIC HEALTH		
1. Occupational Health		
a. Radiation protection, which has always been secondary in importance to reactor operations and reactor safety, must be given a higher priority (p. 155).	I.B.1.3, III.D	a
b. The NRC should give a greater emphasis to radiation protection in both its safety review and inspections (p. 155).	I.B.1.3, III.D	a
c. At reactor sites, the radiation protection function should be made independent of operations and be elevated to equal importance (p. 155).	I.B.1.3	d
J. INFORMATION MADE AVAILABLE TO THE NEWS MEDIA		
1. Emergency Response Planning		
a. A provision for public information should be incorporated in the emergency response plans of both the NRC and the utility, and those plans should be coordinated with State, county or local plans (p. 157).	III.A.2	b
2. Principal Spokesperson		
a. A senior NRC official should be the principal spokesman at onsite or near-site press conferences during an accident at a nuclear power plant. A utility spokesman should be present at such press conferences to provide simultaneously any differing views or additional information the utility feels is necessary to keep the public fully informed (p. 157).	III.A.2	b
b. As appropriate, a State official should also be present at these press conferences and should have sole jurisdiction for public information concerning evaluation and related emergency planning (p. 157).	III.A.2	b
c. The utility should maintain responsibility for initial public statements until the NRC estimates an onsite or near-site capability. Press briefings should be held three times a day, or more frequently if dictated by the situation (p. 157).	III.A.2	b

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SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
K. DISINCENTIVES TO SAFETY		
1. NRC Evaluation of Utility Finances		
a. The NRC will have to become more aware of the relationship between the business and technical sides of the utility. Consideration should be given to an expanded financial analysis of the utility licensee so that the NRC might be alerted when financial pressures combine to impact on safety (p. 164).	None (IV.G)	c
2. Communication with Other Regulatory Bodies		
a. The agency needs better methods for making other regulatory bodies aware of the effect of their regulatory programs on the overall safety of nuclear plants (p. 164).	None (IV.G)	c

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ENCLOSURE 4

COMPARISON OF RECOMMENDATIONS IN VOLUME II OF SIG REPORT, NUREG/CR-1250, WITH DRAFT 2 OF TMI ACTION PLAN, NUREG-0660

COMPARISON OF RECOMMENDATIONS IN VOLUME II OF SIG REPORT, NUREG/CR-1250, VOLUME I, WITH DRAFT 2 OF TMI ACTION PLAN, NUREG-0660

Key:

Impact on TMI Action Plan

a. Recommendation is adequately covered in Draft 2 of the Action plan; no revision to the plan is necessary.

b. There is a related TMI Action Plan Task, but the SIG recommendation adds a new or different thought; Draft 3 of the Action Plan will include consideration of the SIG recommendation in the related task description.

c. There is no directly related TMI Action Plan Task and the recommendation merits consideration as a new Task; Draft 3 of the Action Plan will include a new task that responds to the recommendation.

d. Staff or Commission do not agree with recommendation; no action will be taken.

Note: When related TMI Action Plan is listed as "none," the proposed new task number is identified.

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
I.A.1.C (0041)	1.	A Nuclear Safety board should be established to exercise independent oversight of of the effectiveness of the system.	IV.A.12	c
	2.	A statement of regulatory objectives should be developed, including policy on risk objectives and methods, to better use risk assessment techniques.	IV.A.1, IV.B.7	a
	3.	WASH-1400 techniques should be emphasized through an expanded risk assessment program.	II.C.1, IV.B.7	a
	4.	The reactor system vendors and the architect-engineer should either be licensed or made accountable by some equivalent system.	I.J.1.3	a
	5.	An organization should be designated to assure that the quality of the regulations is adequate.	IV.A.9, IV.B.7, IV.E.2	a
(0042)	6.	The two-step licensing process should be abolished.	IV.D.2	c
	7.	Incentives should be established that would result in more information prior to construction, fewer unresolved issues, and less variety in the design of important systems.	IV.D.2	b
	8.	Important licensing areas should be examined and prompt action taken to publish applicable regulatory criteria.	IV.B.4, IV.B.7	b
	9.	The NRC should be relieved of its responsibilities under the Nuclear Nonproliferation Act of 1977.	IV.A.7	a
	10.	The NRC should be relieved of its prelicensing antitrust review responsibilities.	IV.A.7, IV.E.2	a
	11.	The U.S. Government should decide on a national policy on societal risks.	IV.A.1	b
I.B.1 (0148)	1.	Rational risk objectives should be established and approved by Congress.	IV.A.1	b
	2.	Current requirements should be reevaluated to meet specific risk objectives.	IV.A.1	c
	3.	An explicit rationale should be established for the evaluation of proposed new safety requirements against the criteria "substantial additional protection required for public health and safety."	IV.A.1	b

PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	4.	New requirements should be implemented in a staged, controlled process that provides for holding plant designs standard over significant periods of time.	IV.A.1, IV.D.2	b
	5.	An organizational element to carry out recommendations 2, 3, and 4 above should be established.	IV.B.7	b
	6.	The existing design basis accident concept should be enlarged.	II.B, II.C.1, IV.B.7	a
	7.	The hearing process should be modified.	IV.A.13	c
(0149)	8.	Operating experience should be applied to the development of new or modified regulatory requirements.	I.E	b
	9.	An internal quality assurance program should be established to ensure that the licensing process is conducted in accordance with Commission approved standards.	IV.B.7	b
	10.	The Standard Review Plan should be expanded.	None (IV.B.7)	c
I.C (0185)	1.	Administrative and physical prohibitions must be instituted to prevent all operator actions during an accident or assume the operator will act when he should not.	I.A.2, I.A.4.2, I.C.7, I.C.9, II.B.4	a
(0186)	2.	The entire industry and the NRC must broaden their review of operating experience.	I.B.1.4, I.E	a
(0187)	3.	The charter of AEOD should require that the recommendations of AEOD be followed unless the Commissioners or the Director of the applicable Program Office direct otherwise.	I.E	b
(0188)	4.	A program should be developed to reduce the insulation and lack of effective communications that currently exist by: a. Selection of management dedicated to the interchange of information. b. An incentive program for identification and exchange of safety information. c. Regulator interchange conferences with broad agendas including industry and NRC delegates. d. Interorganizational training on communications.	IV.A.9, IV.B.6	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
(0189)	5.	Each inspection should be an occasion to comment on the general state of affairs at the particular plant.	I.B.2.1, IV.B.2.1	b
	6.	A permanent unit with each Region reporing to IE headquarters, other NRC offices involved and AEOD should review and evaluate Licensee Event Reports.	I.B.2.3, I.E	b
(0190)	7.	Organizational separation between IE and NRR should be reduced by integrating IE and the Division of Operating Reactors into a single group.	IV.A.9	b
	8.	Requirements for the maximum number of working days that a Board Notification request can be held at each step in the process should be established and strictly enforced.	IV.B.4	b
(0190)	9.	Simplified event-tree and fault-tree analysis techniques should be used to evaluate each nuclear power plant.	II.C.1	a
	10.	Event-tree and fault-tree analysis techniques should be used for assignment of priorities and allocation of resources to various safety issues.	II.C.1, IV.B.7, IV.D.1	a
	11.	Better management control over the priority of assigned work should be implemented.	IV.A.9	b
I.D (0289)	1.	Systems controlling pressurizer level for anticipated operating transients should be distinctly and separately operated from systems designed to supply cooling water for loss-of-coolant accidents. Systems designed for loss-of-coolant accidents should be designed to actuate in response to breaks in the reactor coolant system and should be designed to operate unabated until their function is served.	II.C.1, II.E.2	c
	2.	The NRC should consider reviewing acceptance criteria for startup tests to determine whether similar component limitations exist.	I.G, II.C.1	c
	3.	Instrumentation should be installed to provide indication of water level in the reactor vessel.	II.F	a
	4.	The NRC should review the B&W pressurizer design to determine whether equipment modifications are needed.	None (II.E.5)	c
	5.	The NRC should review the reliability of secondary equipment.	II.C.1	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
I.E (0355)	1.	Establish an expanded Financial Analysis office to monitor situations in which business considerations may impact on nuclear safety.	None (IV.G)	c
(0356)	2.	Establish better communication and coordination with the "economic regulators."	None (IV.G)	c
	3.	Establish a better system at IE for balancing the pressures created by financial incentives.	None (IV.G)	c
(0357)	4.	Scrutinize the power ascension test program to prevent compromising safety.	I.B.2, I.G	b
(0358)	5.	Examine the status of the FSAR listing of the power ascension tests to be performed.	I.G.	c
(0359)	6.	PUC's must recognize the unique problems associated with challenging utility decision-making on a nuclear unit.	None (IV.G)	c
(0360)	7.	When nuclear units are involved, a truly future test year should be employed by PUC's.	None (IV.G)	c
	8.	CWIP in the rate base should be allowed for nuclear units by PUC's to reduce the "lump sum" that is otherwise accumulated.	None (IV.G)	c
(0361)	9.	PUC's should recognize a distinction between a nuclear plant in "commercial operation" and one that is "used and useful."	None (IV.G)	c
	10.	PUC's should consider the long-term effects on nuclear plant decision-making of disallowance arguments.	None (IV.G)	c
(0362)	11.	PUC's should improve a dialogue with the NRC and other PUC's to coordinate nuclear plant treatment.	None (IV.G)	c
	12.	FERC should improve its communication and coordination with the NRC.	None (VI.G)	c
(0363)	13.	FERC should eliminate the threat of disallowance of AFUDC that is implied in Electric Plant Instruction 9D.	None (IV.G)	c
	14.	The IRS should require the use of the qualified progress expenditures basis for recognizing ITC for nuclear units.	None (IV.G)	c

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	15.	The IRS should bring its standard for recognizing a nuclear unit into closer conformity with the standards used by other regulators.	None (IV.G)	c
(0364)	16.	A study should be conducted of the relative safety of privately owned vs. publicly owned nuclear units.	None (IV.G)	c
	17.	A study should be conducted of the conflicting, ambiguous responsibilities of the various regulatory agencies in this area.	None (IV.G)	c
II.B.1 (a)	1.	Radiation protection must be given greater emphasis.	III.D	a
(39)	2.	NRC must change its organizational structure to improve management effectiveness for ensuring that its mandate "to protect public health and safety" is fulfilled.	IV.A.8, IV.A.9	a
	3.	Radiation protection programs at existing reactors should be reexamined to ascertain whether they are adequate to cope with normal and emergency conditions.	III.D	a
	4.	The public must be fully informed of the manner by which nuclear power plants are designed, licensed, and operated and of the actual risks associated with radiation and radioactive materials.	III.C	a
II.B.2 (71)	1.	The design bases for radwaste and other related systems, such as the makeup and purification systems, should be reexamined to determine appropriate design criteria for the expected levels of activity and volumes that will be generated in both normal operation and accident situations.	II.B.8	a
	2.	Review of radwaste systems should include all related systems, such as the Industrial Waste Treatment System, to ensure that all potential releases are treated.	II.B.8, III.D.1.1	a
	3.	Radwaste system components should be periodically tested for leaks, and any leaks exceeding a minimum acceptance level should be repaired.	III.D.1.1, III.D.1.2, III.D.1.3	a
	4.	Consideration should be given to locating systems such as the makeup and purification system in an isolating building.	II.B.8, III.D.1.4	b
	5.	Consideration should be given to the installation of tie-lines back to containment from components outside containment having the potential to contain significant activity.	III.D.1.1, III.D.1.3	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	6.	Methods should be developed for inplace testing of ventilation systems to ascertain overall filter system performance when needed.	III.D.1.5, III.D.1.6	b
	7.	Procedures should be developed for the evaluation of spent carbons exposed to accident conditions and to consider the effect of high concentrations of noble gas and iodine	III.D.1.6	b
	8.	Certain filtration systems should be designated and designed for use only after an accident; separate filter systems should be provided for normal operation.	II.B.8, III.D.1.5, III.D.1.6	c
	9.	Dampers around filter systems should be eliminated or improved to minimize leakage.	III.D.1.5, III.D.1.6	b
	10.	To increase the radioiodine removal capabilities, consideration should be given to co-impregnating carbons with an amine, such as triethylenediamine, and to using deeper carbon beds.	III.D.1.6	b
II.B.3 (f) (104)	1.	NRC should reevaluate requirements for environmental radiological monitoring for released radioactive materials in both normal and accident conditions.	III.D.2, III.D.4, III.D.6	a
II.B.5 (b) (135)	1.	Licensees, in their design, and NRC, in its review, should assure that adequate consideration is given to radiation protection matters.	II.B.2, III.D.1	a
(141)	2.	The functions of radiation protection and chemistry should be separated and technicians should not be required to perform in both roles.	I.B.1.3, III.D.3.1	b
	3.	The duties of a radiation protection manager should be clearly specified and performed by a qualified individual.	I.B.1.3, III D.3.1	a
	4.	NRC should require minimum qualifications for the positions of Radiation Protection Foreman and Chemistry Foreman.	I.B.1.3, III.D.3.1	a
	5.	Technical Specifications should be amended to include the positions of Radiation Protection Foreman and Chemistry Foreman.	I.B.1.3, III.D.3.1	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	6.	Technicians should be given training adequate to meet FSAR requirements and to develop and maintain adequate job skills.	I.B.1.3, III.D.3.1	a
	7.	Met Ed should take appropriate steps to eliminate the serious communications problems in the radiation protection organization.	I.B.1.3, III.D.3.1	a
(143)	8.	Emergency Plans should provide for radiation protection staff response to inplant radiation hazards.	III.A.2, III.D.3	a
	9.	Radiation protection procedures should be followed during emergencies, and appropriate documentation should be maintained.	III.D.3	a
(147)	10.	NRC should require the implementation of an adequate radiation protection training program.	I.A.2, I.B.5, III.D.3	a
	11.	NRC should inspect for actual competence of the trainees and trainers.	I.B.1.3, III.D.3.1	b
	12.	NRC and the licensee should review radiation protection staffing and organization to assure that radiation protection functions are fulfilled by adequately trained personnel.	I.B.1.3, III.D.3.1	a
	13.	NRC should develop guidance regarding the specific use and training of "rent-a-techs" at licensed facilities.	I.B.1.3, III.D.3.1	b
II.B.5 (b) (147)	14.	NRC should examine the feasibility and advisability of licensing or certifying radiation protection personnel at commercial nuclear power reactors.	I.A.3.5	a
	15.	NRC should defer action on a petition (PRM-20-13) presently pending before the Commission, which requests that radiation protection personnel at all levels in licensed activities be certified by the Commission until the aforementioned study is completed.	I.A.3.5	b
(157)	16.	NRC should reassess the requirements for inplant fixed-radiation monitoring instruments.	II.F, III.D.3.3	a
	17.	NRC should evaluate and specify requirements for type, quality and quantity of operational portable radiation survey instruments for both normal and accident conditions.	III.D.3.2	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
(158)	18.	NRC should require that the responsibility for the respiratory protection program be vested in a single individual and that technicians be permanently assigned to perform the tasks of inspection, maintenance and decontamination of respiratory protection equipment.	III.D.3	b
	19.	NRC should specify the minimum number of functional respiratory protection devices required by type and size for both normal operations and for emergencies.	III.D.3	b
(161)	20.	Met Ed should establish an improved system for control, issuance and recovery of personnel dosimeters.	III.D.3	a
	21.	Met Ed should assure that their personnel dosimetry program is managed and implemented by competent personnel.	I.B.1.3	b
	22.	NRC should require licensees to have adequate personnel dosimetry services, including sufficient staff, and that personnel dosimetry records, evaluations and referrals for bioassay be maintained during emergencies	III.D.3.1	a
II.B.5 (c) (172).	1.	The radiation protection function at commercial nuclear power plants should be independent of operations and be elevated to equal importance with production.	I.B.1.3, III.D.3	a
	2.	NRC should give greater emphasis to radiation protection in its licensing review and inspection processes and should reassess the radiation protection programs at commercial nuclear power reactors.	III.D.3	a
	3.	NRC should give additional emphasis to radiation protection and radiological health.	III.D.3	a
	4.	NRC should develop a regulatory base for assuring that inplant radiological conditions resulting from an accident are considered in the planning of emergency procedures.	III.D.3	a
II.C.1 (181)	1.	The required "design basis" for nuclear power plants should be reconsidered, as well as the importance and impact of core melt and disruption accidents in the licensing process.	II.B.8, II.C.1, II.C.2	a
II.C.1 (b) (187)	1.	The frequency of PORV operation in B&W plants should be reduced.	II.K	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
(189)	2.	The capability of PORVs to discharge water or two-phase fluid should be established.	II.D	a
(191)	3.	Bounding thermohydraulic analyses should be reevaluated to determine their accuracy in predicting system variations.	II.E.2.2, II.E.2.3	a
	4.	Automatic reactor protection actions should be derived from independent process variables.	None (II.F)	c
	5.	Automatic actions through coincidence of independent process variables should be limited for nonreactor protection functions.	None (II.F)	c
	6.	Pressurizer level instruments should be designed to criteria applied for instrumentation systems important to safety, and emphasis should be placed on achieving diversity in the measured parameters.	I.D.5	b
(193)	7.	The need for immediate trip of reactor coolant pumps should be reevaluated.	II.K	a
	8.	ECCS capacity should be sufficient to preclude uncovery of the core when the reactor coolant pumps continue to run during any accident.	II.E.2, II.K	b
	9.	Control logics for all complex systems and components should be made available to the operators to assure their continued familiarity with all control permissives and inhibits.	I.A.2.1, I.A.3.1	b
(197)	10.	The use of the steam generators as a heat-removal mechanism during transient-initiated and small-break accidents should be a matter of careful discussion among the regulatory, vendor, and utility staffs.	II.B.4	a
	11.	The addition of remotely operable vent valves, or the modification of presently installed manual vents, appears to be a desirable change.	II.B.1	a
(199)	12.	Transient and LOCA reanalyses should be performed to confirm important parameters for actuation of reactor building isolation from direct measurements of such parameters.	II.E.4.2	a
	13.	Reevaluation should be made to determine the criteria for defeating an isolation signal for any component and system during an accident mitigation sequence.	II.E.4.2	a

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
(200)	14. The principal sources of hydrogen generation should be determined more accurately for the implementation of an appropriate hydrogen recombination system or consideration should be given to containment designs that would not require hydrogen recombination systems.	II.B.5, II.B.7	a
(203)	15. Engineered safety feature systems and components should be capable of performing intended function without operator intervention for at least 10 minutes following a real safety feature actuation signal initiation.	C.1	c
	16. A thorough evaluation should be performed to determine adequate response requirements for automatic or manual reinitiation of engineered safety features following inadvertent loss of power supply during a critical transient or accident mitigation sequence.	I.A.4.2, I.C	b
(204)	17. The engineered safety features actuation signals should automatically remove components and systems important to safety from off-normal position and place them back to normal alignment for safety actuation.	II.C.1.1	c
	18. Control circuit components should be designed and periodically tested at expected degraded power supply conditions to ensure that they are capable of performing their intended function.	II.F, II.G	c
(205)	19. An explicit assessment of the effects of the core barrel vent valves should be included as part of the small-break loss-of-coolant accident analyses begun since the TMI-2 accident (B&W designed plants only).	I.C.1, II.E.2.2, II.K	b
(208)	20. Analysis should be performed to determine the consequences of inadvertent interruption of engineered safety features from loss of power at any time during a transient or accident mitigation sequence.	II.C.2	b
(211)	21. Surveillance procedures should not permit the simultaneous defeat of redundant systems important to safety.	I.B.1.5, I.C.9, I.D.3, II.C.1	a
	22. The emergency feedwater system should be designed with a diverse and redundant automatic safety feature actuation of pumps, discharge valve alignment and emergency steam generator level. This automatic actuation should be independent of the integrated control system.	II.E.1.2, II.K	a
(215)	23. The distinction between "safety" and "nonsafety" related systems should be replaced by a graded scale of significance.	II.C.1	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	24.	System designs should consider implementation of piping configurations that can permit periodic testing of valves at system conditions expected during emergencies.	None (II.E)	c
II.C.1 (b) (215)	25.	Interconnections of control, process and safety systems should be limited unless suitable isolation can be provided to ensure that failures in the control or process systems do not cause unacceptable plant disturbances.	II.C.1.2	a
II.C.1 (e) (232)	1.	Utilities should install Malfunction Detection Analyzers in each plant to assist operators in controlling the plant.	I.D.5	b
(234)	2.	Design of cables and some sensors for operation after flooding should be considered, and their required time for postaccident operation should be lengthened.	II.F.3	a
	3.	Accident monitoring and safe shutdown systems should be qualified to function under full accident conditions.	II.F.3	a
	4.	A category for "systems required to maintain the plant in a stable condition" should be established. These systems should be qualified to operate in full accident conditions.	II.F.3	b
	5.	Careful review of instrument and control systems should be carried out to make sure that items such as pressurizer heaters do not get left out or get placed in improper categories.	II.C.1	a
	6.	Administrative review of instrument repair records is necessary so that unreliable systems will be upgraded.	II.C.1.3	a
	7.	Stricter control on strip-chart marking should be instituted.	I.D	b
	8.	Data presented to the operators should be reviewed to make sure that important data are continuously available.	I.D.2	a
	9.	Consideration should be given to layout so that important data can be readily assimilated without distraction by less important displays.	I.D.2	a
	10.	Recording devices meant to document data for historical reconstruction of accidents or off-normal incidents, such as control room voice recorders, magnetic tape, disk recording of important parameters, and dedicated strip charts, should be installed.	I.D	c

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
II.E (387)	1.	NRC should develop an interdisciplinary human factor capability.	IV.B.1	a
	2.	NRC should require the development and implementation of formal human factors programs by utilities, vendors, and architect-engineer organizations.	IV.B.1	b
	3.	NRC should promulgate detailed regulations for the design of new nuclear powerplant control rooms requiring the application of human factors principles to such designs.	I.D.4	a
	4.	NRC should initiate a program of control room enhancement. In the near term the NRC should conduct an onsite human factors evaluation of control rooms in operating plants and plants for which operating licenses are imminent. On a long term basis, the NRC should conduct an indepth evaluation of nuclear powerplant control rooms to determine the adequacy of the man-machine interface.	I.D.1	b
	5.	Additional diagnostic operational aids, such as logic trees or disturbance analysers, should be required in all control rooms.	I.D.5	a
	6.	NRC should certify and approve operator training facilities, training instructors, and training curricula.	I.A.2.7	a
	7.	NRC should require increased emphasis on diagnosis and accident response training of control room operators.	I.A.2.1	a
	8.	The operator selection and training criteria, manning levels, procedural format, and content should consider analysis and research performed to determine operator responsibilities and actions during normal and abnormal conditions.	I.A.1.3, I.A.2.1	a
	9.	Until recommendation 8 can be implemented, the NRC should require that all hot operations shifts be manned by a minimum of one SRO, two CROs and one additional individual with demonstrated and tested capabilities in abnormal system diagnosis. Two of these individuals should be required in the plant control room at all times.	I.A.1.3, I.B.1.1	a
	10.	NRC should require power plant operations supervisors and management personnel to be trained in investigation techniques and reporting methods for events involving human behavior.	I.A.2.6	b
	11.	NRC should conduct an immediate review of the emergency procedures of all operating plants to identify and correct problems associated with symptoms identification, technical accuracy, and systems compatibility.	I.C.1, I.C.7, I.C.8, I.C.9	a

PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	12.	NRC should develop improved methods for measuring operator performance and the effectiveness of training programs in meeting training objectives.	I.A.3	a
	13.	The NRC should consider the licensing of auxiliary operators and testing and maintenance personnel for specific plants.	I.A.3.5	a
II.F (424)	1.	The public in the vicinity of nuclear power reactors should be well-informed about reactor operations and malfunctions.	III.C	b
	2.	Timely, relevant, and understandable information about the status of an accident and likely offsite consequences should be made available to State, county, and local decision-makers responsible for recommending or implementing offsite protective action.	III.A.1, III.A.3, III.B	a
III.A.2 (66, 67)	1.	Prompt action should be taken to upgrade the diagnostic and emergency response capabilities of personnel licensed to operate reactor plants and their supervisors up to at least the level of unit superintendent. This recommended action should be assigned the highest priority.	I.A.1	a
	2.	On the same priority basis, on-shift manning levels should be increased to levels determined to be needed by the results of accident response task analyses.	I.A.1.3	a
	3.	Supervisors of licensed reactor operators, up to at least the level of unit superintendent, should be required to hold a senior reactor operator license on any unit to which they are assigned supervisory responsibilities for normal or emergency operations.	I.A.2	b
	4.	The shift manager or equivalent, who is assigned the responsibility for the safety of operation and in direct charge of the operators in the control room, should have a college degree in a technical discipline closely related to reactor plant design and operations, and at least 3 years of operating experience. This requirement should be met as soon as practicable but no later than July 1, 1983.	I.A.2, I.B.1.1	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	5.	The duties and responsibilities and the qualifications and training of all personnel assigned to support the unit operators and their supervisors in maintaining the unit should be reassessed and upgraded to be consistent with the upgraded levels of the reactor operators and their supervisors.	I.A.2, I.B.1.1	a
	6.	Revise emergency procedures for inplant response to accidents to have plans available for the organization and efficient use of off-duty operating staff personnel who would be expected to report to the site in the event of an accident.	III.A.1	a
	7.	Take immediate action to provide the operations staff with the means to acquire prompt expert advice from offsite sources. Immediate implementation of interim measures that will be developed into a final program should be approved by the NRC in accordance with its requirements no later than January 1, 1982.	I.B.1.1, III.A.1	a
	8.	Develop procedural guidance for the use of the operations staff in responding to situations beyond the normal design bases of the facility.	I.C, II.B.4	a
	9.	Develop plans for effective mobilization and use of industry resources for the mitigation of consequences and for recovery from reactor accidents.	III.A.3.6	a
III.A.3 (95)	1.	Plant procedures and personnel training requirements related to radiological emergency recognition and response should be reviewed and upgraded.	I.C.1, III.A.1	a
	2.	Real-time, on-line radiation monitoring equipment should be installed around all nuclear power plants.	III.D.2.4	a
	3.	Inplant and portable radiation monitoring instruments and trained personnel should be available at all nuclear power plants.	III.D.2	a
	4.	Emergency plans should include provisions for a prolonged radiological response effort and clear chain of command. In addition, guidance should be provided to assure that the emergency director is promptly informed of critical information, and that State and Federal agencies are kept accurately informed of plant status and radiological conditions.	III.A.1	a
	5.	Communications equipment should be provided at all nuclear power plants to assure unimpeded contact between inplant locations and all locations where offsite monitoring teams are likely to perform radiation dose rate measurements.	III.A.1	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	6.	Emergency plans should be suitably definitive to provide an adequate response to a realistically anticipated accident under adverse conditions, such as inclement weather, minimum allowable staff, and a rapidly developing accident.	III.A.1, III.A.2.2, III.B	b
III.A.4 (123)	1.	The NRC should require that the emergency plans for all nuclear power plants include provisions to assure prompt technical support to plant operations personnel coping with a reactor accident and its consequences. Also, the NRC should ensure that adequate technical and managerial personnel and resources will be requested and integrated into a preplanned emergency organization for response to and recovery from an accident.	III.A.1	a
	2.	The NRC should interact with nuclear industry organizations in defining the criteria and guidance for emergency planning.	III.A.2.1, III.A.3	a
III.A.5 (153, 154)	1.	The NRC should identify and qualify those NRC personnel relied upon to obtain or evaluate critical information during nuclear power plant or radiological emergencies.	III.A.3.5, IV.B.6	a
	2.	The NRC should provide training, equipment, and guidance that assure rapid, efficient, and comprehensive gathering of information by NRC personnel during nuclear power plant or radiological emergencies.	III.A.3	a
	3.	The NRC should determine the minimum staffing and composition of the initial NRC response teams, both onsite and offsite, for nuclear power plant accidents and other foreseeable radiological emergencies.	III.A.2.2, III.A.3.1	a
III.A.6 (168)	1.	Prompt action should be taken to upgrade the qualification and experience requirements for personnel managing and supervising activities at nuclear power plants. A suitable method of certification of the qualification and experience requirements should be established. These actions should be completed as soon as practicable but no later than January 1, 1982.	I.A.2	a
	2.	The NRC should require that each key management position at a nuclear power plant be staffed by a qualified person working full time in that position.	I.A.2, I.B.1	a
	3.	The NRC should perform a timely evaluation of personnel changes in key plant management positions and changes in the plant organizational structure to assure that adequate staffing is maintained.	I.A.2.4, I.B.1.1	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	4.	Offsite safety review committees, or equivalent, should include within the scope of their activities the evaluation of personnel changes in key management positions and the evaluation of changes in plant organizational structure.	I.B.1.1	a
	5.	Qualifications for personnel participating on safety review committees should be established by the NRC.	I.B.1.1	a
III.A.7 (183)	1.	Regulatory Guide 1.101 should be revised to include a requirement that each nuclear utility employee with an emergency response assignment receive appropriate training and participate in at least one emergency plan drill each year.	III.A.2	a
	2.	The NRC should expedite review and upgrading of existing emergency planning and preparation requirements.	III.A.1, III.B	a
III.B.3 (c) (274-279)	1.	In an emergency of predetermined severity, the NRC should send an emergency response team to the site. The team should be drawn principally from personnel in the appropriate Regional office.	III.A.3	b
	2.	Whenever this team is activated and sent to the site, its leader should be the Regional Director or the Regional official who, in the absence of the Director, would become the Acting Regional Director.	III.A.3	b
	3.	The onsite team leader should have the delegated authority to manage and direct the NRC's entire emergency response and to be the Agency's spokesman concerning the emergency response from the time of the team's arrival.	III.A.3	b
	4.	This authority should include the power to require the licensee to take such action as the onsite team leader deems appropriate to ensure adequate protection of the public's health and safety. Also included should be the authority to make a final recommendation to State and local officials on behalf of the the NRC about the appropriateness of various protective actions, including evacuation.	III.A.3	b
	5.	The onsite team leader's authority should be made known through preplanned notification procedures to all NRC officials; officers and employees of the licensee; and appropriate Federal, State and local officials.	III.A.3	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	6.	The functions of the onsite team should include, among others, the following: (1) observing, evaluating, and reporting on operational and radiological status and activities; (2) giving advice or orders to the licensee regarding accident recovery; and (3) advising State and local authorities on public protection actions. A program plan should be prepared in each region.	III.A.3	b
	7.	Regional project inspectors or, where applicable, resident inspectors should be part of the onsite team. These managers and inspectors must all have extensive exposure to the plant and good knowledge of its design, layout, operating procedures, and other essential information.	III.A.3	b
	8.	Procedures should be prepared that explain in detail the onsite team's role. Team members should be adequately instructed as to who is team leader, what they should do upon arrival at the site, what to look for and report, to whom to report, and from whom they will receive instructions.	III.A.3	b
	9.	The procedures should describe the emergency response structure that will be organized by the licensee during an emergency.	III.A.3	b
	10.	The procedures should describe State and local officials and offices that may play a role during the emergency.	III.A.3	b
	11.	Upon arrival at the site, the onsite team should set up an operations center at a pre-designated location, to which all available information concerning plant and offsite conditions will be transmitted. The licensee should set up a similar operations center at the same location.	III.A.3	b
	12.	Upon arrival at the site, the onsite team should immediately establish and maintain telephone contact with those individuals whom the licensee has designated to have direct supervisory authority.	III.A.3	b
	13.	Recognizing the onsite team leader's obligations as agency spokesman, the onsite team should be organized so that the team leader's deputies and principal managers in the normal organizational structure are designated and prepared to assume primary responsibility for supervising the work of all NRC personnel at the site.	III.A.3	b

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	14. When the NRC is first notified of an emergency requiring activation of an onsite team, NRC Headquarters officials should bear the responsibility for managing and directing the Agency's emergency response until the onsite team arrives at the site.	III.A.3	a
	15. A duty officer should be available at the Headquarters incident response center on a round-the-clock shift basis. When notified of an emergency requiring activation of an onsite tem, the duty officer should supervise activation of the Headquarters' center.	III.A.3	a
	16. Upon notification of an emergency requiring activation of an onsite team, the Headquarters duty officer should immediately establish telephone contact with individuals to whom the licensee has designated direct supervisory authority.	III.A.3	b
	17. Once the onsite team leader takes command, the function of the personnel at the Headquarters incident response center should be to provide support and advice to the onsite team when and as requested.	III.A.3	b
	18. The support and advisory functions of Headquarters should be provided as requested by the onsite team leader.	III.A.3	b
	19. The Headquarters Incident Response Plan should describe the support and advisory functions that may have to be performed in any given emergency, and should specify which component office at Headquarters will be responsible for providing each such function. The Plan should also describe the management structure each office will use in discharging an assigned function.	III.A.3.1	a
	20. Except for the command function at the incident response center just mentioned, the Commission should not interject itself into the management's response to an emergency. We expect that individual Commissioners will keep closely informed and act as spokesmen within the Government.	III.A.3.1	b
	21. Automatic data retrieval systems should be developed to telemeter important plant data to the onsite response team's operations center, as well as to the affected regional office and the Headquarters incident response center.	III.A.3.4	b

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	22. Permanently open communication pathways should be maintained between each site and the regional and Headquarters response centers. These communication lines should be backed up by alternative means of communication resistant to loss from possible environmental conditions.	III.A.3.3	b
	23. In an emergency, the oral communication of information among the onsite team members, the regional office, and Headquarters should be the responsibilities of individuals specifically assigned to only this task.	III.A.3.1	b
	24. In an emergency, separate pathways should be provided for the oral communication of operating and radiological information.	III.A.3.3	a
	25. The oral communication of information should be transmitted by the most direct means possible to the party having the principal need for the information. Thus, to the extent possible, emergency plans should establish communications priorities concerning the different categories of information.	III.A.3.1	b
	26. Each region should have available what has previously been determined to be the emergency equipment required to perform all necessary independent measurements, and to allow the NRC emergency response team to fulfill its mission.	III.A.3.1, III.D.2.6	a
	27. The regional and Headquarters incident response centers all should have duty officers available on a round-the-clock basis to immediately receive the licensee's notification.	III.A.3.1	b
	28. The NRC should prepare and publish a policy statement concerning its role in responses to nuclear accidents.	III.A.3.1	a
	29. The NRC's present policy referred to in paragraph 024 of Manual Chapter 0502, "NRC Incident Response Program," should be clarified. The NRC should prepare and publish a policy statement concerning whether and under what conditions the NRC will intervene to direct recovery actions following an accident. The statement should clarify the responsibilities of licensee management unless and until these are preempted by the NRC.	III.A.3.1	a

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	30. The NRC should consider in advance the assistance that will be needed by the State, licensee, the NRC, and other Federal agencies in any nuclear accident. Agreements should be established between the NRC, the DOE, and other agencies as to what each will do in an emergency, and how and by whom the activities will be coordinated.	III.A.3.6(2)	a
	31. The NRC should develop a policy about dealing with briefing requests from State and local officials, Congress, other Federal officials, the media, and others during emergencies.	III.C	b
	32. The NRC should advise all other response team members--at Headquarters, the Regional office, and at the site--to defer to the special team with respect to media briefings or discussions. A single location at or near the site for all media briefings should be considered.	III.A.3.1, III.C	b
	33. The information policy should be issued, along with an implementing procedure, as part of the emergency response plan. The NRC should inform the States, the Congress, the media, and the public of this policy, and request that they work only with this special information group.	III.A.3.1, III.C	b
	34. The NRC should intensify its efforts to keep up-to-date information on nuclear accidents available on a prerecorded tape accessible to the public by direct dial phone.	III.A.3.1, III.C	b
	35. Individuals who write preliminary notification documents (PN's) should be properly trained and instructed to prepare PN's for nontechnical readers.	III.A.3.1, III.C, IV.B.6	b
	36. The NRC should prepare and be able to provide to Government officials and others the documents appropriate to assist them in understanding technical explanations provided by the NRC staff during or after a nuclear accident.	III.C.1	b
	37. The Region's onsite team and the Headquarters support team should include a distinct group of officials whose assigned function is to evaluate contingencies.	III.A.3.1	b

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	38.	The NRC support team at Headquarters should be organized in advance to identify centers of expertise for different technical areas. Based on the bad experience in one particular area during the TMI response, the NRC should establish within the staff an organization with concentrated expertise in reactor chemistry matters.	III.A.3.1	b
	39.	The contingency group should appraise the need for public protective measures as soon as possible after responding to an emergency.	III.A.3.1	b
	40.	The NRC should have a clearly identified single spokesman for making recommendations on protective actions. There should be a clear advance knowledge on the part of State and local officials as to who this NRC spokesman is, with whom he will consult, and to whom he will make his recommendations. The spokesman for NRC should be the onsite team leader.	III.A.3.1	b
	41.	The NRC should prepare multiple plant accident and offsite hazard descriptions for each plant using realistic analyses and reference meteorology conditions. These descriptions should cover a wide range of serious accidents, including core melt sequences. They should be made a part of the emergency plan documentation.	II.A, III.A.2	b
III.C.3 (303-305)	1.	On September 16, 1978, Federal Reorganization Plan No. 3 established the Federal Emergency Management Agency (FEMA) as the agency responsible for centralized overall planning and coordination for Federal agency response to emergencies, including nuclear reactor accidents. We endorse this action.	III.A.3.6	a
	2.	To handle planning and coordination, FEMA must have sufficient authority to generate a timely response from other Federal, State, and local agencies. Such authority must recognize the responsibility of the NRC, State and local governments, and the utility.	III.A.3.6(2), III.B	a
	3.	FEMA must develop a comprehensive Federal response plan for peacetime nuclear emergencies.	III.A.3.6	a
	4.	The proposed NRC appropriations bill (S. 562) requires that an NRC emergency plan be developed that provides appropriate details for rapid agency response to reported incidents at nuclear facilities. Some of these provisions are in the area of NRC interface with other Federal agencies; hence, coordination with other agencies will be required.	III.A.3.6(2)	a

PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	5.	There could be a substantive conflict between elements of the State law that authorize FEMA to exert command control over local emergency actions and the portions of the law that charges the local jurisdictions with the responsibility for protection of the health and safety of their citizens. This possible problem should be carefully considered by the States.	III.B	b
III.C.4 (319, 320)	1.	Official channels for the transmittal of protective action recommendations to the responsible decision authority must be set up in advance and understood by all parties.	III.A.3.1, III.A.3.6	b
	2.	Procedures must be established in advance by the decision-making authority for verifying protective action recommendations and their bases. These procedures must provide for timely verification, according to the temporal nature of the public hazard.	III.A.3, III.B	a
	3.	The NRC, in cooperation with HEW and EPA, must develop clear and commonly acceptable protective action guidelines (PAGs) that are understood by decision-makers and can be applied in a relatively unambiguous manner.	III.A.3	a
	4.	The NRC, in cooperation with EPA, HEW, and FEMA, must evaluate the array of protective actions available in the event that PAGs may be exceeded and develop recommendations for action accordingly.	III.A.3	a
	5.	FEMA must study and, to the extent reasonable, lower possible economic barriers to protective actions, such as evacuation.	III.B	b
III.C.5 (335-338)	1.	Each Federal, State, county, and local organization involved in emergency response must develop complete, integrated emergency response plans which prescribe the organization's functions, its emergency organization, and its modus operandi and assure that proper information will be obtained and disseminated by the agency so it can discharge its responsibilities.	III.A.1.1, III.B	b
	2.	State, county, and local plans for response to nuclear plant accidents must include the following: a. It must be clearly stated that Federal agencies do not have the authority to order an evacuation.	III.A.1.1, III.A.2.2, III.B	b

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	b. The division of authorities and responsibilities must be clearly spelled out.		
	c. Federal, State, and local relationships must be clearly defined and the resources that each agency could provide must be predetermined.		
	d. Local jurisdictions must develop emergency plans in sufficient detail to assure that their responsibilities are understood. Matters requiring detailed planning include the size of planning zones; evacuation routes; designation of host areas; communications procedures for plant personnel to provide specific information concerning the extent of the hazard to State, county, and local government officials; coordination of public information releases; and tests and drills.		
	e. Funding is required for establishing and maintaining county and local emergency preparedness. We believe funding assistance to county and local governments for nuclear facility emergency planning is necessary. Such funds could come from the NRC, FEMA, the State, or the utility. We believe the utility should fund the county and local effort necessary for effective nuclear emergency planning.		
	f. Training of State, county, and local emergency response personnel must be provided by the utility in areas such as basic plant operations and the site emergency plans.		
	g. FEMA should offer assistance to the States in establishing and/or carrying out training programs for State, county, and local officials.		
	h. Plans must consider impacts on transportation, food, shelter, and communications.		
	3. FEMA and the NRC should study the Mississauga evacuation, as well as other evacuations of populated areas, to determine:	II.A.2, II.B.6, III.A.3	*
	a. The extent to which prior planning can improve the effectiveness of an evacuation.		
	b. The impact of population density and other factors on the effectiveness of evacuation.		
	4. FEMA should be required to certify the status of State emergency planning prior to the issuance of an NRC license.	III.B	*

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III.C.6 (349, 350)	1.	NRC, in cooperation with FEMA and HEW, must establish criteria for the storage and distribution of a thyroid-blocking agent, such as potassium iodide. Specifically, consistent guidance needs to be developed for the use of potassium iodide in the total context of nuclear hazards. Prompt attention should be given to the population at risk in the vicinity of nuclear plants.	III.A.1.3	a
	2.	The utility must fund the purchase and storage of potassium iodide, based on the same rationale that supports our recommendation to require utility funding of the development of local emergency plans.	III.A.1.3	b
	3.	Each State must develop specific criteria and procedures governing the storage, distribution, and use of potassium iodide that are consistent with Federal guidance and stockage requirements.	III.A.1.3	b
	4.	Unlike evacuation, which requires substantial time to implement, other protective actions should be strongly considered if radiation levels or doses are likely to approach protective action guides.	II.A, III.A.2.2	b
III.C.7	1.	DOE must be the lead agency for coordination and implementation of a prompt, large-scale emergency radiological monitoring response, since it is already operationally equipped for such a function. However, the EPA should be the lead agency for long-term, low-level, followup monitoring actions; and HEW should be the lead agency for determining the long-term health effects of the accident.	III.A.3.6(2)	b
	2.	FEMA must assure that personnel dosimetry equipment capable of measuring and indicating both low and high radiation exposures is available for those involved in conducting evacuations and securing the evacuated areas, such as the State police, fire personnel, and the National Guard, and that training is provided in the use of this equipment.	III.A.3.6(2), III.B.1	a
	3.	RAP and AMS/NEST teams must be promptly dispatched by DOE upon the occurrence of a potentially serious radiological incident without waiting for an invitation or request by the State or NRC.	III.A.3.1, III.A.6(2)	a
	4.	Radiological monitoring and radio relay positions must be preplanned by the utility in cooperation with the NRC, DOE, and the State and should be based on land use, terrain, accessibility, and other considerations.	III.A.2.3, III.D.2.4, III.D.2.6	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
III.C.8 (365, 366)	1.	FEMA must carefully evaluate communications systems to determine if the preassigned authorities and responsibilities of the various Federal, State, and local agencies and the utility can be carried out effectively during an emergency situation.	III.B	a
	2.	Necessary information regarding the status of the emergency must be transmitted routinely and consistently by all parties to all appropriate Federal, State, county, and local government agencies.	III.B	a
III.C.10 (371, 378)	1.	The NRC must adopt a policy that requires reasonable offsite emergency planning, and such planning must consider emergency response to low probability accidents having offsite consequences greater than those analyzed as "credible" in the design review.	III.A.3.1	a
	2.	The NRC must establish the areas for which evacuation planning is required and the maximum times within which evacuation of the areas must be conducted.	II.A.1, II.A.2.1, II.A.2.2, III.A.2.2, III.A.3.1	b
	3.	Clear and explicit Federal and State emergency response coordination and command roles must be established and understood by all parties.	III.A.2.2, III.A.3 III.B	a
	4.	Appropriate emergency plans must be developed and routinely tested at all levels of government and suitably meshed with the utility's plan. These plans must include sufficient detail to facilitate a reasonably prompt and effective 10-mile evacuation. The utility should in some manner provide the funding appropriate for the development and testing of local emergency plans.	III.A.3.5	a
	5.	Evacuation plans must be prepared in anticipation that the evacuation of selected persons will result in the voluntary evacuation of many more people than specified, and that many people living at least twice as far from the reactor as specified will also evacuate.	III.A.1, III.A.2.2, III.B	a

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PART (Page)		SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	6.	The NRC, in cooperation with HEW and the EPA, must establish uniform and agreed upon protective action guides. The NRC must also develop criteria for the storage and distribution of potassium iodide so that it can be reasonably available to the public if needed.	III.A.1.3, III.A.3	a
	7.	A DOE Radiological Assistance Team must automatically be dispatched whenever there is a clearly abnormal radiological situation at a nuclear power plant; formal procedures to this effect must be instituted. DOE should be the lead agency with regard to the collection and assessment of radiological monitoring data in any multi-agency emergency response. Also, arrangements must be made for the ready availability near every nuclear power plant of appropriate radiation monitoring equipment for all emergency response personnel and for training of the emergency personnel in its use.	III.A.3, III.A.3.6(2)	b
	8.	FEMA must carefully evaluate communications linking all participants in emergency response systems to assure that the systems are adequate for emergency communications. Such an evaluation should consider the availability of backup systems, as appropriate, communications from alternate command posts, and the use of automated data transmission.	III.A.3.6(2), III.B	a
	9.	All organizations involved in emergency response must assess their information needs to assure the effective and timely communication of all necessary information during an emergency.	III.A.1, III.A.3.6(2), III.B	b
III.D (403-408)	1.	All utilities operating nuclear power plants should designate a place equipped to serve as a communications center in the event of an accident that requires extensive interface with the news media. Such a facility must be near the site.	III.A.1.2, III.C	a
	2.	A senior NRC official should be the principal spokesman at onsite or near-site press conferences during an accident at a nuclear plant. A utility spokesman should be present at such press conferences to provide any differing views or additional information the utility feels is necessary to keep the public properly informed. A cognizant State official should be present at these press conferences and should have sole jurisdiction for public information concerning evacuation and related planning.	III.A.3.1, III.C	b
	3.	Each utility that operates nuclear power plants should ensure that a member of its public relations staff has extensive experience in dealing with the local media and that the staff member has a detailed understanding of the operating and radiological aspects of the utility operating plants.	III.A.1.1, III.C	a

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PART (Page)	SPECIAL INQUIRY GROUP RECOMMENDATION	RELATED TMI ACTION PLAN TASK	IMPACT ON TMI ACTION PLAN
	4. Each utility that operates nuclear power plants should prepare a standard briefing package for each plant which provides background information about the plant and which can be disseminated to the media as required. This briefing package should be approved by the NRC.	III.A.1.1, III.C	b
	5. The NRC should establish requirements that will ensure prompt notification of the news media when a nuclear facility experiences an event that could impact the public's health and safety.	III.A.3.1, III.C	a
	6. The NRC nuclear accident response teams should include at least two technical individuals, one with a background in health physics and the other in reactor design and operations. Another response team member should be designated to establish and maintain open channels of communication to offsite centers involved in media interface activities.	III.A.3.1, III.C	b
	7. The NRC should choose and train members of the technical staff to serve as technical advisors to the media following any future nuclear accident.	III.A.3.1, III.C	b
	8. The NRC should develop a standard format for press releases to ensure inclusion of basic information concerning a nuclear accident.	III.A.3.1, III.C	d
	9. The NRC should establish a clear policy of issuing prompt public announcements concerning nuclear accidents.	III.C	a
	10. The NRC should take the lead in working with responsible State agencies to develop a public information program to educate the general public on nuclear power and its consequences.	III.C	b

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ENCLOSURE 5

REVISED LIST OF REQUIREMENTS FOR
NEAR-TERM OPERATING LICENSE APPLICANTS

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TMI ACTION PLAN

NEAR-TERM OPERATING LICENSE REQUIREMENTS

PART 1 - NTOL REQUIREMENTS NOT PREVIOUSLY ISSUED[1]

REQUIREMENT	WHEN APPLICABLE[2]
(1) I.A.1.3 Shift Manning	
(a) SRO and RO in control room.	FL
(b) Restrictions on use of overtime.	FL
(2) I.A.3.1 Revised Scope and Criteria for Licensing Examinations	
Prepare applicants for new examinations.	FL
(3) I.B.1.1 Organization and Management Criteria	
Interoffice NRC review of licensee management to determine organizational and managerial capabilities, using internal NRC draft criteria pending development of formal criteria.	FL
- No immediate action required by OL applicant pending completion of NRC review of licensee management.	

[1]On September 27, 1979 and November 9, 1979, all pending operating license applicants were issued a letter containing a set of requirements resulting from staff investigations of the TMI-2 accident and approved by the Commission. The new requirements listed in this Part 1 are in addition to the previously issued requirements which are listed in Part 2, below. Of the 13 items in this Part 1, 3 have been previously approved for application to operating plants (2, 10, and 11) but have not been issued formally to operating license applicants. Five of the 13 are applicable to operating reactors and will be issued after approval by the Commission (1, 2, 5, 12, and 13).

[2]FL = Before fuel loading
FP = Before full-power operation

REQUIREMENT		WHEN APPLICABLE
(4)	I.B.1.2 Safety Engineering Group Provide onsite safety engineering group to provide supplemental engineering review and support. Interoffice NRC review of the adequacy of this groups, using internal NRC draft criteria pending development of formal criteria.	FL
(5)	I.C.5 Licensee Dissemination of Operating Experiences Procedures that assure feedback of operating experiences to operators and other personnel.	FL
(6)	I.C.7 Vendor Review of Procedures NSSS vendor review of licensee procedures.	
	(a) Emergency Procedures	FP
	(b) Low Power Test Procedures	FL
	(c) Power Ascension Procedures	FP
(7)	I.C.8 Pilot Program for Review of Selected Emergency Procedures NRC conduct in-depth review of development and use of selected emergency procedures on NTOL plants.	FP
(8)	I.G Training During Low Power Testing Conduct "hands on" training in selected plant evolutions and off-normal events for shift personnel.	
	- Define training plan	FL
	- Conduct training	FP

REQUIREMENT	WHEN APPLICABLE
(9) II.B.4 Degraded Core - Training	
(a) Establish training program for all operating personnel in the mitigation of severe core damage using existing equipment.	FL
(b) Complete initial training.	FP
(10) II.E.1.1 Auxiliary Feedwater System Reliability Perform simplified reliability analysis of AFW system and modify as necessary.	FP
(11) II.K.1 IE Bulletins on Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents Implement, as applicable, TMI-2 related IE bulletins. - Bulletins were issued to ORs.	FL
(12) II.K.3 Generic Review Matters - Small Break LOCA's and Loss of Feedwater Accidents Implement Bulletin and Orders Task Force recommendations on a schedule to be determined by NRR on a case-by-case basis.	As required by NRR
(13) III.D.3.4 Control Room Habitability Confirm compliance with existing Regulatory Guides and Standard Review Plan or establish schedule for necessary modifications to achieve compliance.	FP

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PART 2 - NTOL REQUIREMENTS ALREADY ISSUED[1]

REQUIREMENT	WHEN APPLICABLE[1,2]
(1) I.A.1.1 Shift Technical Advisor (STA) Provide technical advisors with engineering expertise on each shift.	
- STA on duty	FL
- STA training complete	1/1/81
- See NUREG-0578, Section 2.2.1b and September 27, 1979 and November 9, 1979 letters to all pending OL applicants for criteria.	
(2) I.A.1.2 Shift Supervisor Administrative Duties Minimize administrative duties.	FL
- See subitem 4 of Section 2.2.1a of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	

[1]On September 27, 1979, all pending operating license applicants received a letter which defined a set of requirements resulting from NRC staff investigations of the TMI accident and approved by the Commission. On November 9, 1979, a followup letter was sent to all pending operating license applicants further clarifying the requirements of the September 27, 1979 letter. Enclosures 6 and 8 of the September 27, 1979 letter provided implementation schedules for the short term requirements. The schedules have been refined here to reflect a difference between fuel load and full power dates.

[2]FL = Before fuel loading

FP = Before full-power operation

REQUIREMENT	WHEN APPLICABLE
(3) I.B.1.4 Licensee Onsite Operating Experience Evaluation Capability	
Capability for evaluation of operating experiences at nuclear power plants.	FL
- See NUREG-0578, Section 2.2.1b and September 27, 1979 and November 9, 1979 letters to all pending OL applicants for criteria.	
- See also Task Action Plan Sections I.B.1.1 and I.B.1.2.	
(4) I.C.1 Short-term Accident Analysis and Procedure Revision	
(a) Small break LOCAs.	FL
(b) Inadequate core cooling.	FL
(c) Transients anu accidents.	Same schedule as OR
- See Section 2.1.9 and 2.1.3b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(5) I.C.2 Shift Relief and Turnover Procedures	
Plant procedures for shift relief and turnover.	FL
- See Section 2.2.1c of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(6) I.C.3 Shift Personnel Responsibilities	
Plant procedures specifying responsibilities of shift personnel for safe operation of the plant.	FL

REQUIREMENT	WHEN APPLICABLE
(6) (Continued)	
- See Items 1, 2, and 3 of Staff Position of Section 2.2.1a to NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(7) I.C.4 Control Room Access	
Plant procedures for limiting access to the control room.	
- See Section 2.2.2a of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	FL
(8) II.B.1 Degraded Core - Primary System Vent	
Provide design of remotely operable high-point reactor coolant system vents.	FP
- Installation complete.	1/1/81
- See Enclosure 4 to September 27, 1979 and November 9, 1979 letter to OL applicants for criteria.	
(9) II.B.2 Degraded Core - Shielding	
Provide design of additional shielding required to provide access to vital areas and protect safety equipment.	FP
- Plant modifications complete.	1/1/81
- See Section 2.1.6b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	

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REQUIREMENT	WHEN APPLICABLE
(10) II.B.3 Degraded Core-Sampling Provide interim procedures and final system design for sampling and analyzing reactor coolant and containment atmosphere.	FP
- Plant modifications complete.	1/1/81
- See Section 2.1.8a of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(11) II.D.1 and II.D.2 Relief and Safety Valve Test Commit to performance testing of RCS relief and safety valves under the full range of normal and accident conditions.	FL
Test program complete	7/1/81
- See Section 2.1.2 of NUREG 0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(12) II.D.5 Relief and Safety Valve Position Install direct indication of relief and safety valve position.	FL
- See Section 2.1.3a of NUREG 0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(13) II.E.1.2 Auxiliary Feedwater Initiation and Indication Install control grade automatic start of AFW and control grade flow indicators.	FL
Complete implementation of safety grade equipment.	1/1/81

REQUIREMENT	WHEN APPLICABLE
(13) (Continued)	
- See Section 2.1.7a and b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(14) II.E.3.1 Emergency Power for Pressurizer Heaters	
Install capability to supply some pressurizer heaters and controls from emergency power supply and implement necessary training and procedures.	FP
- See Section 2.1.1 of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
- This item complements II.G.	
(15) II.E.4.1 Containment Penetrations	
Provide design of redundant dedicated containment penetrations for external hydrogen recombiner, if applicable.	FL
Complete installation.	1/1/81
Review procedures and bases for recombiner use.	FL
- See Section 2.1.5a and 2.1.5c of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(16) II.E.4.2 Containment Isolation	
Install diverse containment isolation signals.	FP
- See Section 2.1.4 of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	

REQUIREMENT	WHEN APPLICABLE
(17) II.F.1 Additional Accident Monitoring Instrumentation	
(a) Interim Procedures for Quantifying High Level Accidental Radioactivity Releases	FL
(b) Containment Pressure Monitor	1/1/81
(c) Containment Water Level Monitor	1/1/81
(d) Containment Hydrogen Monitor	1/1/81
(e) Containment High Range Radiation Monitors	1/1/81
(f) High Range Noble Gas Effluent Monitors	1/1/81
- See Section 2.1.8b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(18) II.F.2 Inadequate Core Cooling Instruments	
(a) Procedure development for use of existing instrumentation.	FL
(b) Install subcooling meter.	FL
(c) Submit analysis of capability to detect inadequate core cooling and vessel level indicator design, if new instrumentation desirable.	FL
(d) Install vessel level indicator, if required.	1/1/81
- See Section 2.1.3b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(19) II.G Emergency Power for Pressurizer Equipment Modify power supplies for the pressurizer relief valves, block valves, and level indicators to be from emergency power sources.	FL

REQUIREMENT	WHEN APPLICABLE
(19) (Continued) - See Section 2.1.1 of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria. - This item complements II.E.3.1.	
(20) III.A.1.1 Upgrade Em rgency Preparedness Implement provis of SECY 79-450. - See Enclosures 7 and 8 of September 27, 1979 letter to OL applicants for requirements. - See all Item III.A.1.2 below.	Phased implementation. - As specified in Enclosure 8 of September 27, 1979 letter to OL applicants
(21) III.A.1.2 Upgrade Emergency Support Facilities (a) Establish onsite technical support center and provide plans, procedures, staffing, communications, and radiation monitoring equipment.	FL
Upgrade technical support center. - See Section 2.2.2b of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	1/1/81
(b) Establish an operational support center. - See Section 2.2.2c of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	FL

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REQUIREMENT	WHEN APPLICABLE
(21) (Continued)	
(c) Establish an emergency operations center as a base for coordinating onsite and offsite activities and interface with State, local, and Federal agencies.	FL
Upgrade emergency operations center.	1/1/81
- See Item 3 of Enclosures 7 and 8 to September 27, 1979 letter to pending OL applicants for description.	
- Items (a), (b), and (c) above complement III.A.1.1 of Action Plan.	
(22) III.D.1.1 Sources Outside Containment	
Evaluate leakage from systems outside containment likely to present radiological hazards in the event of an accident and reduce leakage to the extent practical.	FP
- See Section 2.1.6a of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	
(23) III.D.3.3 In-plant Radiation Monitoring (Partial)	
Provide instrumentation to determine in-plant airborne radioiodine concentrations.	FL
- See Section 2.1.8c of NUREG-0578 and letters of September 27, 1979 and November 9, 1979 to pending OL applicants for criteria.	

PART 3 - NRC ACTIONS RECOMMENDED FOR COMPLETION BEFORE RESUMPTION OF LICENSING

REQUIREMENT	WHEN APPLICABLE
(1) I.B.2.2 Resident Inspector NRC Resident Inspector at each site for new OL.	FL
(2) II.B.7 Containment Inerting Reach decision on need for interim hydrogen control requirements for small containments and apply, as appropriate, to near-term plants.	FP
(3) II.B.8 Degraded Core - Rulemaking Issue notice of intent to conduct rulemaking on requirements for design features for accident involving severely damaged cores.	FP
(4) III.A.3.1 Role of NRC More detailed definition of role of NRC in emergencies.	FP
(5) III.A.3.3 Communications Install direct dedicated telephone lines between plant and NRC.	FL
(6) III.B.1 NRC Approval of Overall Emergency Preparedness Approve overall state of emergency preparedness, including integration of emergency preparedness onsite and offsite pursuant to the Memorandum of Understanding with FEMA.	FL
(7) III.D.2.4 Offsite Dose Measurements NRC establish TLD surveillance network around site.	FP

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PART 4 - NEW REQUIREMENTS PROPOSED BASED ON CONSIDERATION OF NRC SPECIAL INQUIRY GROUP REPORT RECOMMENDATIONS

REQUIREMENT	WHEN APPLICABLE
(1) Control Room Design Review OL applicant examine control room to identify outstanding human factors deficiencies and any instrumentation problems Interoffice NRC review to determine whether the applicant's self-examination was adequate	FL
(2) Power Ascension Test Schedule Increased IE scrutiny of the power ascension test program to prevent any compromising of safety in view of the proposed expansion of startup test programs and the economic incentives to achieve the already delayed commercial operation of new plants.	FL - until completion of program

PART 5 - REQUIREMENTS OF DRAFT 2 OF NUREG-0660 RECOMMENDED FOR DELETION FROM THE NTOL LIST BUT REMAINING IN THE ACTION PLAN FOR FURTHER DEVELOPMENT AS OPERATING REACTOR REQUIREMENTS

REQUIREMENT	REASON FOR DELETION
(1) I.A.1.3 Shift Manning Administrative aide to shift supervisor on each shift.	Section 2.2.1a of NUREG-0578 concerning need to minimize shift supervisor administrative duties is being implemented by NTOLs (see Part 2, Item 2). Implementation of NUREG-0578 adequately addresses problem in the interim. Shift manning will be addressed in a comprehensive manner in the longer term.
(2) I.E.4 Coordination of Operational Evaluation Program Establish mechanism to assure licensee evaluation program is coordinated with other evaluation programs; e.g., between reactors and between utilities and NSAC and INPO.	Coordination of operational evaluation programs should be pursued on a broad basis (NRC-industry-licensee joint effort) and not tied to an individual licensee.
(3) II.C.1.1 Mini-IREP Perform an interim study similar in principle to NRC IREP studies, but smaller	Requirements for interim study are not well defined. Program could place a heavy demand on NTOLs in terms of engineering manpower resources with limited near-term benefits. The benefits are limited because it is a scaled down version of IREP which is already scaled down relative

REQUIREMENT	REASON FOR DELETION
(3) II.C.1.1 continued	to the Reactor Safety Study and the increment in reliability is likely to be small. Study should be performed later as part of a more comprehensive program for operating reactors.
(4) II.C.1.3 <u>Reliability Assurance</u> Establish interim reliability assurance program.	Requirements for interim program are not well defined. NTOLs should be required to implement a reliability assurance program on the same schedule as operating reactors. Something narrow and special for NTOLs would only give an extra few reactor months of data compared to an overall program for all operating reactors. Interim program could place a heavy manpower demand on NTOLs.
(5) II.D.4 <u>Auto Close Block Valve</u> Install controls to auto matically close PORV block valve upon low RCS pressure.	This item is considered part of Item II.K.3 which is included on the NTOL list (Part 1, Item 12).
(6) II.E.4.2 <u>Containment Isolation</u> (Partial) Provide administrative controls for "sealed closed" valves.	Being done as part of normal NRR review of OLs. Will be retained in action plan for implementation on operating reactors.

REQUIREMENT	REASON FOR DELETION
(7) II.E.4.4 <u>Containment Purge</u> Restrict containment purge operation and demonstrate purge valve operability. - See NRR letters of 11/28/78, 9/27/79 and 10/15/79 to licensees.	Being done as part of normal NRR review of OLs. Will be retained in action plan for implementation on operating reactors.
(8) II.K.2 <u>Commission Orders on B&W Plants</u> Implement, as applicable, requirements in Commission orders.	Applicable only to B&W plants. No B&W plants are in NTOL category.
(9) III.D.1.2 <u>Improved Vent Gas System</u> Review vent gas and leak detection systems against new design criteria and provide schedule for modification.	Interim measures already being applied as an extension of item 22 in Part 2, above; i.e., action plan item III.D.1.1 This item (III.D.1.2) will stay in action plan for development of criteria for longer term application to other plants under construction.
(10) III.D.1.3 <u>Secondary Systems</u> Applicants review secondary systems for radiation hazards and recommended modifications.	Interim measures already being applied as an extension of item 22 in Part 2, above; i.e., action plan item III.D.1.1. This item (III.D.1.2) will stay in action plan for development of criteria for longer term application to other plants under construction.

REQUIREMENT	REASON FOR DELETION
(11) III.D.1.5 Auxiliary and Radwaste Building Vent:lation Identify improvements to control radioactive leakage from auxiliary and radwaste buildings, including requirements for building exhaust filtration where it doesn't already exist, and provide schedule for modifications.	Items of moderate priority and criteria are not well defined. Implement on same schedule as ORs once criteria are developed. In the interim, operating reactors are being required by NRR to improve control of ventilation equipment in context of implementation of short-term lessons learned.
(12) III.D.1.6 Surveillance Testing (Filtration Systems) Upgrade charcoal adsorbers and implement surveillance testing of non-ESF filtration systems.	Item of low priority and should be implemented on same schedule as ORs once criteria are developed. In the interim, operating reactors are being required by NRR to improve surveillance testing in context of implementation of short-term lessons learned.
(13) Installation of Reactimeter (Office Director Recommendations) Install reactimeter.	Criteria are not well defined. Installation of recording equipment should be on same schedule as ORs. Will be considered for inclusion in action plan draft 3 as a recommendation of the NRC Special Inquiry Group.

CROSS REFERENCE LISTING OF NTOL REQUIREMENTS

January 5, 1980 Listing (Based on Draft 1 of Action Plan)	Present Listing (Based on Draft 2 of Action Plan)
(1) I.A.1.1	Part 2 Items 1 and 3
(2) I.A.1.2	Part 2 Item 2
(3) I.A.1.3	Part 1 Item 1 and Part 5 Item 1
(4) I.B.1.1	Part 1 Item 3
(5) I.B.3.1	Part 1 Item 4
(6) I.B.3.4	Part 3 Item 1
(7) I.C.1.1	Part 2 Item 4 (Transients and Accidents Added to List)
(8) I.C.1.2	Part 2 Item 5
(9) I.C.1.3	Part 2 Item 6
(10) I.C.1.4	Part 2 Item 7
(11) I.C.2	Part 1 Item 6
(12) I.C.3	Part 1 Item 7
(13) I.E.1	Part 2 Item 3 Part 1 Items 3 and 4
(14) I.E.2	Part 1 Item 5
(15) I.G	Part 1 Item 8
(16) II.B.1	Part 2 Item 8
(17) II.B.2	Part 2 Item 9
(18) II.B.3	Part 2 Item 10
(19) II.B.4	Part 1 Item 9
(20) II.B.8	Part 3 Item 3
(21) II.B.9	Part 3 Item 2
(22) II.C.1.1	Part 5 Item 3
(23) II.C.1.8	Part 5 Item 4
(24) II.D.1.1	Part 2 Item 11
(25) II.D.1.5	Part 2 Item 12
(26) II.E.1	Part 1 Item 10
(27) II.E.1.3	Part 2 Item 13
(28) II.E.3	Part 2 Item 14
(29) II.E.4.1	Part 2 Item 15

CROSS REFERENCE LISTING OF NTOL REQUIREMENTS (Cont'd)

January 5, 1980 Listing (Based on Draft 1 of Action Plan)	Present Listing (Based on Draft 2 of Action Plan)
(30) II.E.4.3	Part 2 Item 16
(31) II.E.4.5	Part 5 Item 7
(32) II.F.2	Part 2 Item 18
(33) II.G	Part 2 Item 19
(34) III.A.1.1	Part 3 Item 4
(35) III.A.1.5	Part 3 Item 5
(36) III.A.2.1	Part 2 Item 21
(37) III.A.2.2	Part 2 Item 21
(38) III.A.2.3	Part 2 Item 21
(39) III.A.3	Part 2 Item 20
(40) III.B.3.2	Part 3 Item 6
(41) III.D.1.3a	Part 2 Item 23
(42) III.D.2.1	Part 1 Item 13
(43) III.D.2.2b	Part 2 Item 22
(44) III.D.2.2c	Part 5 Item 11
(45) III.E.1.1	Part 5 Item 9
(46) III.E.1.2a	Part 5 Item 12
(47) III.E.2.1b	Part 3 Item 7

Additional NTOL requirements not reflected in January 5, 1980 listing:

(1) Part 1, Items 2, 11, and 12.

(2) Part 2, Item 17

(3) Part 3, none

(4) Part 4, Items 1 and 2

(5) Part 5, Items 2, 5, 6, 8, 10, and 13 (Items in Part 5 are proposed for deletion as NTOL requirements.)

ENCLOSURE 6

NTOL OPERATOR FEEDBACK REPORT

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

FEB - 1980

MEMORANDUM FOR: R. J. Mattson, Chairman, TMI Action Plan Steering Group

FROM: E. J. Brunner, RONS Branch Chief, Region I
R. F. Heishman, RONS Branch Chief, Region III
G. L. Madsen, RONS Branch Chief, Region IV
R. L. Lewis, Acting RONS Branch Chief, Region II
D. M. Sternberg, RONS Section Chief, Region V

SUBJECT: NTOL OPERATOR FEEDBACK REVIEW TEAM REPORT

The enclosed report of our findings and recommendations is transmitted in accordance with your request made during our meeting on January 30, 1980.

E. J. Brunner
E. J. Brunner, Chief
Reactor Operations and Nuclear
Support Branch, RI

R. F. Heishman
R. F. Heishman, Chief
Reactor Operations and Nuclear
Support Branch, RIII

G. L. Madsen
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R. L. Lewis
R. L. Lewis, Acting Chief
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Support Branch, RII

D. M. Sternberg
D. M. Sternberg, Section Chief
Reactor Operations and Nuclear
Support Branch, RV

cc: V. Stello, IE
Steering Group Members
Regional Directors
IE Division Directors
Operator Feedback Review
Team Members
Action Plan Task Managers

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February 1, 1980

REPORT

OF THE

OPERATOR FEEDBACK REVIEW

TEAM'S

MEETINGS WITH UTILITIES

REGARDING

ACTION PLANS FOR IMPLEMENTING

RECOMMENDATION OF THE PRESIDENT'S COMMISSION

AND OTHER STUDIES OF

TMI-2 ACCIDENT (ACTION PLAN) NUREG-0660, DRAFT 1

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INTRODUCTION

The January 5, 1980 letter from L. V. Gossick, to the NRC Commissioners, "TMI Action Plan - Prerequisites for Resumption of Licensing" identified approximately 50 items from the 250 topics in NUREG-0660, Draft 1, as those applicable to Near Term Operating License (NTOL) facilities. These items were proposed to provide a basis for the issuance of licenses to the four NTOL facilities (Salem 2, Sequoyah 1, Diablo Canyon 1, and North Anna 2) relative to the TMI-2 lessons learned.

The Commissioners directed the Steering Group to obtain an industry assessment of the impact of the 50 NTOL topics on plant safety. The Steering Group created a number of interdisciplinary teams to conduct onsite meetings with operating personnel and utility management. These teams were composed of IE Regional Branch Chiefs, who served as the team leader, the licensing project manager, the resident inspectors, and various senior NRC managers and directors. In addition to the meetings held with the four NTOL facilities listed above, the Steering Group decided that meetings should be held with four operating facilities (Quad Cities, Brunswick, Arkansas Nuclear One and Connecticut Yankee) to ensure that operational experience would be reflected in the overall safety assessment.

The teams met in Bethesda on January 9 and 10, 1980, to be briefed by the Steering Group. Following this briefing, detailed schedules, agendas, and formats were developed. Telephone calls to the utilities were made by the team leaders to discuss the purpose and details of the meetings. A comprehensive package of pertinent reference materials was sent to the utilities on January 12, 1980 with a letter confirming the specific details of the meetings.

Approximately one week was allowed for the utility to receive the reference materials and become familiar with the topics. The meetings were held onsite and were conducted in two sessions. During the first session, the NRC team met separately with the licensed operators or license candidates. The second session was held with the site and corporate managers (see Table 1 for the meeting schedule and team members). The meetings were conducted in an informal manner to encourage frank and candid comments from the operating personnel relative to the safety impact of the NTOL topics. It was emphasized that the NRCs primary objective was to identify items which if implemented might result in a less safe, rather than more safe operation. Cost and schedule impact comments were not solicited, but the utilities provided some general thoughts on these subjects. The consensus of the team members is that the comments received during the meetings satisfied the goal of candor and frankness.

The meeting agendas were organized by broad functional areas such as Shift Manning, Procedures and Engineering changes. Each specific topic from the NTOL list was included in one of more of these broad topic areas to enhance the organization of the meetings.

Table 2 provides a summary list of the NTOL topics discussed during the utility meetings. Table 2 is organized by broad topic and item title to aid the reader in locating items of interest. Where potential negative safety impact comments were made by personnel at one or more utilities, a discussion of their concerns is provided as specific comments. In order to provide a more balanced view relative to these topics, differing comments and additional viewpoints from other utilities and the review team leaders are also provided. Items which received no substantive comments from utility personnel have no specific comments included, although these items are included in Table 2.

CONCLUSION

It is the concensus of the review team members, based on the meetings conducted, that no single NTOL requirement, will of itself produce a negative safety or quality impact if implemented at the NTOL facilities. However, the review team members believe that specific NTOL topics should be removed from the NTOL list and rescheduled by the Steering Group. The basis for this conclusion is the perception that the well established utility engineering and technical support staffs will be unduly diverted from their necessary and ongoing routine safety related tasks, and overall safety will be diminished.

RECOMMENDATIONS

The review team members recommend that the requirements as listed below be removed from the NTOL list and rescheduled by the Steering Group.

The candidates for removal from the NTOL list are those topics which currently lack specific definition or detailed acceptance criteria. These items have a heavy front end demand for technical resources but have a lesser contribution to near term safety improvement. It seems appropriate for the Steering Group to establish the schedule for the deferred topics to achieve a more orderly and expeditious accomplishment consistent with the evolving task definition.

The recommendations for deletion are based on the comments made at the utility meetings and the combined technical judgment of the team members. It is acknowledged that no quantitative basis for the proposed deferral can be offered, but it appears consideration of such deferrals is nonetheless warranted.

This list and the associated rationale is not intended to be an inclusive mandatory cut list. Clearly, other considerations will dictate additions or deletions from this listing.

RECOMMENDED NTOL DEFERRAL LIST

Topic	Discussion
II.B.2 - Degraded Core - Shielding	Criteria will not be available until March 1, 1980
II.B.3 - Degraded Core - Sampling	Criteria will not be available until March 1, 1980

RECOMMENDED NTOL DEFERRAL LIST(Cont'd)

Topic	Discussion
II.C.1.1 - Mini-IREP	Current vague requirements imply heavy demand for scoping and program definition with limited near term benefits
II.E.1.1 - Auxiliary Feedwater System Reliability	Current surveillance testing and failure investigation requirements provide interim assurance of system availability

GENERAL COMMENTS

The utilities made a variety of comments about the specific NTOL topics as well as comments of a general nature. The following general comments are based on the results of the meetings and comments received from the utilities.

1. There were no significant difference in the overall reaction to the NTOL topics by the operating plants compared to the NTOL facilities. (Review Team Comment)

2. The licensed operators and utility management generally shared similar views on the safety impact of the NTOL topics. (Review Team Comment)

3. The utilities felt that the cost and resource estimates of NUREG-0660, Draft 1, were low in some cases by a factor of 20. (Utility Comment)

4. Current implementation schedules for some of the NTOL topics are not realistic due to the present lack of specific acceptance criteria. (Utility Comment)

5. No single topic as presently perceived by the utilities would by itself produce an unsafe condition at the facilities. (Review Team Comment)

6. A negative safety and quality impact may result from the immediate implementation of all of the NTOL actions. This is due to the dilution of engineering and technical resources caused by the large demands resulting from the post TMI activities such as bulletins and orders. The feeling was that the organizations which have developed over the years to accomplish the many diverse safety related functions may become diluted and the routine functions shortchanged until an orderly increase in resources can be realized. Technical and engineering resource limitations throughout the industry were cited as a limiting consideration. (Utility Comment)

Specific Comments

The following specific comments are based on the results of the meetings conducted with utility operators and management personnel. Comments of the review team leaders are also provided to give an additional viewpoint on these topics.

1. Shift Technical Advisor (I.A.1.1)

Utility Comment

Generally, the addition of the Shift Technical Advisor (STA) function is considered to be beneficial in that additional technical expertise is provided on shift. Questions were raised regarding the long term value to safe operation of the operating shift organization perceived by the NRC Action Plan. The filling of these positions with degreed individuals, in an advisory capacity, was the point of strong discussion. The consensus opinion would indicate that the STA should be a licensed individual, perhaps from the operations ranks, who is provided additional specific technical education. The STA function should be part of the operating team. This approach is envisioned to provide individuals with a broader perspective, would be more readily acceptable to the shift crews and would afford a more stable condition because of minimization of personnel turnover.

Concern was also expressed by some operations personnel that the existence of STAs on shift with limited operations experience could detract from safety by the STA providing erroneous advice. Also that the STAs on shift would cause some confusion regarding shift responsibilities and decision making.

Shift Technical Advisors Comment

The STAs believe that Shift Supervisors or another licensed individual should be trained to the STA level in that they would be easier to train to that level than to train a STA with a degree to systems operation.

Review Team Comment

The mandated implementation date resulted in a detrimental impact on available engineering personnel. In most instances, the STAs were selected from the existing staff. One utility with a NTOL facility and operating units has reassigned twenty-one nuclear engineers from nuclear engineering functions to STA positions. Additional training has not been provided to those selected as STAs and replacement engineers are not yet available to perform the work previously performed by these persons.

The requirement for the Shift Technical Advisor to have an engineering degree should be reevaluated in that this function may be better served by an individual selected from licensed operations personnel and provided with NRC specified additional training. Additionally, the individual in charge of the shift should have received the additional training.

2. Shift Manning (I.A.1.3)

Utility Comment

Some of the utilities felt that requiring a licensed senior reactor operator to be present in the control room at all times was potentially unsafe in that he would be prohibited from responding to a problem outside the control room. The current practice of the utilities allowing or requiring the SRO to conduct plant tours is believed to have several beneficial advantages. The SRO must have and maintain an in-depth knowledge of the plant status, layout, state of repair and readiness and access restrictions. The SRO will lose a great deal of his familiarity with the plant if restricted to the relatively sterile and insulated control room environment. There is also no substitute for an on the scene review of a problem by the SRO. Valuable expertise would be lost or diluted by an overly restrictive limit on the mobility of the SRO. Utilities that had increased the number of SROs on duty did not have this problem. Shift personnel from one utility believes the requirement to have a SRO in the control room at all times, enhances safety.

The subject of overall shift crew strength was raised. Most operators believe the current crew size specified by Technical Specifications is too small to cope with potential problems. This problem was specifically cited for dual unit sites where as few as eight individuals may be involved in routine plant operations. Other operators felt that additional SROs and ROs in the shift complement would allow for plant tours and still maintain sufficient expertise in the control room in the event of an accident. Several operators expressed the fact that the highest workload shifts were in dual unit facilities where one unit was operating and the other one was in an outage. This shift requires additional help and is being provided at some sites through additional licensed personnel.

The desirability of having an Administrative Aide for the Shift Supervisor was questioned by most of the utilities, however, one utility manager felt an Administrative Aide was generally needed and aided safety. This position being filled by an operator trainee was felt to detract from proper operator training. It is believed that most utilities have unburdened the Shift Supervisor by establishing a day shift supervisor, assigning clerical help to the day shift or procedurally assigning purely administrative duties to other shift personnel. There was concern expressed that the definition of administrative duties might be to literally interpreted and activities such as equipment tag out review and clearances would be delegated away from the individual with the overall responsibility for the plant, namely the Shift Supervisor. One utility has routinely assigned a clerk typist to each shift for the past three years to take care of transportation activities, overtime duties, and communications. Comments were also made that the individual assigned as an Administrative Aide should have some operational knowledge.

Formal controls for restriction of overtime were considered desirable; however, those interviewed expressed a need for flexibility to provide emergency coverage. The thought was expressed that adequate shift manning would alleviate the overtime problem.

Review Team Comment

Shift manning must include sufficient SRO and RO personnel to provide continuous coverage by at least one SRO and one RO in the control room at all times and not preclude the continued practice of plant tours by the licensed operators. Manning of high workload shifts should also be considered in the development of shift manning requirements.

An NRC position on overtime must be promptly provided to the industry. The position being developed by the NRC staff and the ANS-3 Committee should be expedited.

A requirement that the shift supervisor be free of routine of administrative duties should be established, without mandating the assignment of an Administrative Aide on shift.

3. Organization and Management Criteria (I.B.1.1)

Utility Comment

The utility representatives stated that this item should not be included as a NTOL item, in that the existing requirements appear to be ambiguous and vague.

The methods to be used for evaluating corporate managements capabilities was questioned. It is felt that the NRC contractor that was developing criteria for utility organization and management evaluations did not have requisite experience in this area. Additionally, it is believed that NRC is becoming too involved in the internal affairs of the utility organization. The utilities requested that they be permitted to comment on draft acceptance criteria when developed.

Several operating license holders feel they are now meeting the intent of this item as they understand the criteria.

Review Team Comment

Development of specific Organizational and Management Criteria should be expedited by NRC with input from licensees. Case by case "on site" type determinations of the adequacy of the organization and management should be made by qualified interoffice NRC personnel.

4. Safety Engineering Group (SEG) (I.B.1.2)

Utility Comment

Concern was expressed that the requirement to establish another engineering group would result in an unnecessary layering of engineering functions that are already provided for within the utility. The comment was expressed that each utility should be permitted to establish how they can best accomplish the SEG requirements. For instance, to require that an SEG be located onsite (7 sites for one utility) would require an additional engineering staff of about 35 engineers who would, to some degree, be shuffling the same pieces of paper. The multiple SEG requirement for utilities with multiple sites was stated to have an adverse impact on safety due to further dilution of available engineering expertise within a utility. One utility thought the onsite SEG was a good idea if the onsite and offsite safety review committees were relieved of some of their review responsibilities by the SEG. Also, if the SEG is required to be onsite, the group should report to the Plant Manager.

A Plant Manager felt that location of the SEG onsite would detract from safety, in that the group would require meetings and discussions with him and the plant staff, which would further impact on their already limited time.

Concerns were expressed that the vague existing criteria for this new issue requiring a SEG be established onsite before fuel loading at NTOL facilities would be difficult to implement by fuel load.

Review Team Comment

The NRC should establish the functions to be performed by the SEG. However, the need to establish an SEG, located onsite, should be reevaluated in that there may be more efficient and effective means of performing this engineering function.

Draft 2 of NUREG-0660 states that NRC will develop the criteria for the SEG. However, the proposed location of the SEG is still required to be onsite.

5. Analysis and Procedure Modifications (I.C.1)

Utility Comment

The analysis and procedure modification activity, recently completed for the Small Break LOCA and Inadequate Cooling Event, was considered to be a beneficial endeavor and resulted in a better understanding of the required followup actions. One licensee commented that the NRC requirements reflected the vendor criteria, but had little input from the licensee regarding operational or human factors. Additionally, the requirements were established too fast to permit adequate evaluation and resulted in "false" starts. Responses were divided regarding the need for NTOL's going through a similar developmental activity. The resulting procedures were thought to be so detailed and lengthy that they might detract from safety.

Several operators indicated that minimum hours of simulator training should be specified and new procedures should be tested on a plant unique simulator prior to implementation.

Review Team Comment

The analysis, procedures, preparation, and training, at the NTOL facilities, should include the Small Break LOCA and Inadequate Cooling events similar to that recently accomplished at operating power reactors.

6. Shift Relief and Turnover (I.C.2)

Utility Comment

Most utilities felt that they have adequate control of the shift turnover activity and it does enhance safety. Some felt that excessive rechecking of one crew's activities by the relieving crew would result in unnecessary exposure and excessive demands on the crew's time. Some operators felt the NRC should require a minimum crew overlap time of approximately 30 minutes to ensure an effective turnover and provide a chance for each crew to exchange data and answer questions which might be overlooked during a briefer turnover. Reviews have resulted in added administrative procedures to document what operators had been doing.

Review Team Comment

The requirement for shift relief and turnover procedures described in NUREG-0578 is considered to be sufficient for this matter.

7. Vendor Review of Procedures (I.C.7)

Utility Comment

Vendors supply generic procedures which are incorporated into station specific procedures. Usually a NSSS representative is included in the review chain by participation in the onsite Joint Test Review Group. Doubt was expressed that a quality review of procedures could be provided by the NSSS individuals that were not at the site and involved in the checkout and startup of the unit. Additionally, the availability of qualified NSSS reviewers was questioned.

Vendor review of emergency operating, low power and power ascension test procedures best serves to enhance safety for a utilities first of a kind nuclear plant. The vendors review are of lesser value for the same NSSS plants located at different sites within an experienced utility and the review would be of little value at duplicate plants located on the same site (Procedures were reviewed or proven on the first unit).

Most individuals interviewed felt this item did enhance safety and should be included in NTOL review.

Review Team Comment

Review of facility emergency operating, low power, and power ascension test procedures will enhance safety and should be required as a condition for NTOL facilities. Consideration of procedure reviews by other than the NSSS vendor (AE) should be considered in the development of future consideration, especially for first of a kind plants in a utility.

8. Licensee Operating Experience Evaluation and Dissemination of Information (I.B.1.2 - Safety Engineering Group)

Utility Comment

Too many LERs without substance are being generated. Licensees staff reviews are performed to differing degrees by the various utilities. Some utilities have formal training on selected issues, while others only provide the NRC printout to the operators for reading. Operators were uniform in commenting that the existing program is ineffective. The reasons given include: too many LERs; even when sorting is provided; the computerized version is too brief; the significant issue is not alway apparent; and many include plant specific terms or plant component numbers.

A method for the dissemination of operating experience information, properly sorted and evaluated is needed and is beneficial to plant operational safety. As an interim, operators stated a strong preference for timely distribution of a sorted group of LERs (not the computerized version). They also requested a copy of the final event report when the full analysis and corrective actions are known.

Licensees state a concern about the technical resources required to perform an adequate review and sorting of LERs. Operators and management expressed an opinion that a single organization, AIF, INPO, or NRC should provide the sorting and distribution for the industry. One utility thought LERs should be distributed on microfische.

Review Team Comment

Draft 2 of NUREG 0660, dated January 23, 1980, includes NRC staffing dates. Until these systems are fully implemented, licensees should be provided a monthly sorted copy of LERs by plant type.

9. Training During Low Power Testing (I.G)

Utility Comment

The utilities are about equally divided concerning the desirability of using the actual power plant to perform the special tests for training of operators. Some shift personnel commented that all shifts should perform the tests to get hands on training. However, comments were made that the proposed special tests may not be the optimum tests to perform, as the results of the identified tests may be too similar. All agreed that the test data must be used to improve the simulator computer model for retraining and training of new and replacement operators. Those commenting against the benefits to be gained by actual plant training was based on the fact that throughout plant life, there will be a turnover in personnel and that at some point in time, people will be on shift who haven't participated in these tests. Additionally, the purpose of the testing is to demonstrate that the plant is built correctly and responds to transients as expected.

Review Team Comment

The team believes unnecessary testing for training of operators at facilities where exact plant simulators exist should be reexamined. The need to establish a requirement that each utility provide a plant specific simulator should also be examined with the goal of early implementation.

10. Degraded Core (II.B)

Utility Comment

The utilities indicated that the availability of a primary system vent, shielding, sampling and training as outlined in NUREG-0660 are good ideas, however, the high priority assigned is questioned. One utility questions the current source terms being used by NRR in design evaluation.

The requirement to draw samples within the time period specified was questioned, as it pertains to both unnecessary overexposure to personnel and data requirement. Sampling within 24-hours was stated to be more realistic.

Additionally, the utilities indicated that these items should not be NTOL requirements.

Review Team Comment

Because of the heavy impact on available engineering resources, these items should not be included as NTOL requirements.

11. Mini IREP (II.C.1.1)

Utility Comments

Most utilities stated that limited engineering resources are available to do this task that is estimated to take 40-50 highly qualified engineers six months to complete one system. Dilution of already limited resources to complete the Mini-IREP within the time frame identified is considered not to be the best utilization of available manpower. Criteria has not been developed as to what is required to satisfy this action item.

Utilities with NTOL facilities question the need to complete vague Mini-IREP requirements prior to a full power license.

Experience information for the Mini-IREP data base is not available.

Review Team Comment

NRC should expedite the issuance of criteria and a system selection that are to be included as part of the Mini-IREP review. Reevaluation of the action item implementation date for Mini-IREP should be performed to preclude diverting engineering talent from the other requirements for NTOL facilities. This would also permit the Mini-IREP pilot programs to be completed and evaluated.

12. Reliability Assurance Program for ESF (II.C.1.8)

Utility Comment

Utilities commented that the Standard Technical Specifications adequately address this item. However, if this is to be a prerequisite for fuel load, criteria is urgently needed.

Review Team Comment

The criteria for this item is covered in NUREG-0578 and will be examined during NTOL readiness review.

13. Containment Isolation Dependability (II.E.4.2)

Utility Comment

One utility questioned the safety impact of a design change at their facility which initiates a containment isolation signal with the SIS. One facility commented that the present design isolates component cooling water and let down flow. Safety concerns were expressed: that operators have approximately eight minutes before the primary system goes solid; the vessel is subjected to a rapid cooldown due to full feedwater flow; and reactor coolant pump seals may not be adequately cooled during pump coastdown.

Review Team Comment

This item is facility specific and is being evaluated by NRR.

14. Reactor Vessel Level Indication (II.F.2(2))

Utility personnel stated that the Westinghouse Owners Group review will not be complete until March 1980. Therefore, implementation of the completed design package is impractical for NTOL facilities.

Review Team Comment

The Steering Committee should reexamine the requirement for completion of this change for NTOL facilities.

15. Technical Support Center (III.A.1.2)

Utility Comment

Some enhancement in safety will result from the establishment of the TSC, in that it will reduce emergency actions required to be handled in the control room. One utility felt that the TSC could detract from safety through dilution of talent at the site.

A better location for the TSC may be in an area that is not in close proximity to the control room, i.e., outside the protected area near the document storage vault which increases access to needed documents and lessens impact on plant security during an accident condition.

Habitability requirements established for the TSC are considered unnecessary if the TSC location is onsite but remote from the control room.

Review Team Comment

Criteria which require the TSC location to be in close proximity to the control room should be reevaluated by the Steering Committee to allow site specific locations.

16. Near-Site Emergency Operations Center (NSEOC)(III.A.2.3)

Utility Comment

The utilities questioned the need for or the desirability of direct communications between the plant computer (data display terminal) or the control room and the NSEOC. Instead, the information for the NSEOC should come from the technical support center to minimize the possibility of giving out erroneous information and reduce the confusion in the Control Room. One utility with multiple sites believes that multiple centers is not the best way to meet the functions required of an emergency operations center. This utility is establishing a centralized emergency center near their corporate offices with multiple communications capability to all of their sites and the NRC (red telephone, radio, etc.).

Review Team Comment

This action should be reevaluated by the Steering Committee relative to NTOL facilities. The reevaluation should consider the advisability of centralized locations for emergency centers for utilities with multiple sites.

17. Control Room Habitability (III.D.3.4)

Utility Comment

Utilities with older plant designs expressed a need for sufficient time to perform good engineering design, procurement and installation of modifications.

Comments were made that this requirement imposed on NTOL facilities would not allow an adequate review and design to be completed.

Review Team Comment

The plant modification completion date should be established on a plant specific basis as required by Draft 2, NUREG-0660.

18. Secondary Side Leakage and R/A Hazards From Major Accidents (III.D.1.3) AND Auxiliary Building R/A Leakage Reduction (Building Exhaust Filtration) (III.D.1.5)

Utility Comment

Several utilities stated that the criteria needs to be more definitive to include examples of items that should be included in the review and that the review schedules were unrealistic.

Review Team Comment

Criteria has been developed and is included in Draft 2, NUREG-0660.

19. Improved Vent Gas System (III.D.1.2)

Utility Comment

Some utilities were uncertain as to the criteria to be used during their design review. Because of the lack of specificity of this item, a completion date by fuel load for NTOL facilities should be revised.

Review Team Comment

Draft 2 of NUREG-0660, lists April 1, 1980 as the proposed date for completion of the acceptance criteria. Licensee evaluation and submittal of system descriptions by full-power operation appears reasonable.

Table 1

TMI-2 ACTION PLAN (NTOL SUBSET)

OPERATOR FEEDBACK MEETING SUMMARY

Arkansas Nuclear (OL), RIV
January 22, 1980

*G. Madsen (OIE)

R. Heishman (OIE)
D. DiIanni (NRR)
W. Johnson (OIE)
C. Long (NRR)
J. Callan (OIE)

Connecticut Yankee (OL), RI
January 21, 1980

*E. Brunner (OIE)

T. Wambach (NRR)
D. Johnson (OIE)
P. Collins (NRR)

North Anna-2 (NTOL), RII
January 28, 1980

*R. Lewis (OIE)

J. O'Reilly, (OIE)
E. Brunner (OIE)
O. Parr (NRR)
A. Tattersal (OIE)

Sequoyah (NTOL), RII
January 22, 1980

*R. Lewis (OIE)

D. Hood (NRR)
W. Cottle (OIE)
T. Donat (OIE)

Brunswick (OL), RII
January 24, 1980

*R. Lewis (OIE)

G. Madsen (OIE)
J. Hannon (NRR)
M. Davis (OIE)
T. Donat (OIE)

Diablo Canyon (NTOL), RV
January 22, 1980

*D. Sternberg (OIE)

R. Engelken (OIE)
T. Young (OIE)
M. Bagaglio (OIE)
B. Buckley (NRR)

Salem 2 (NTOL), RI
January 23, 1980

*E. Brunner (OIE)

H. Rood (NRR)
L. Norrholm (OIE)
D. Skovholt (NRR)
M. Williams (NRR)
J. Milhoan (NRR)
W. Hill (OIE)
R. Kiemig (OIE)

Quad Cities (OL), RIII
January 24, 1980

*R. Heishman (OIE)

J. Keppler (OIE)
D. Verrelli (NRR)
N. Chrissotinos (OIE)
P. Collins (NRR)
W. Minners (NRR)
S. Dupont (OIE)
F. Reimen (OIE)

* Lead Reviewer

TABLE 2

NEAR TERM TASK LIST

Topic/Requirement	Action Plan* Reference Number	When Applicable**	Specific Comments Included
Shift Technical Advisor	I.A.1.1	FL	Yes
Shift Supervisor Duties	I.A.1.2	FL	Yes; included in Specific Comment 2
Shift Manning	I.A.1.3	FL	Yes
Organization and Management Criteria	I.B.1.1	FL	Yes
Safety Engineering Group	I.P.1.2	FL	Yes
Resident Inspector	I.B.2.2	FL	No
Analysis and Procedure Modifications	I.C.1	FL	Yes
Shift Relief and Turnover Procedures	I.C.2	FL	Yes
Shift Personnel Responsibilities	I.C.3	FL	No
Control Room Access	I.C.4	FL	No
Vendor Review of Procedures	I.C.7	FP	Yes
Pilot Program for Review of Selected Emergency Procedures	I.C.8	FP	No
Licensee Operating Experience Evaluation Capability	I.B.1.4	FL	Yes
Licensee Dissemination of Operating Experiences	I.B.1.4	FL	Yes; included in Specific Comment 8
Training During Low Power Testing	I.G	FP	Yes
Degraded Core - Primary System Vent	II.B.1	FP	Yes

* Taken from Draft 2
** FL - before Fuel Load
FP - before Full Power

Topic/Requirement	Action Plan* Reference Number	When Applicable**	Specific Comments Included
Degraded Core - Shielding	II.B.2	FP	Yes; included in Specific Comment 10
Degraded Core - Sampling	II.B.3	FP	Yes; included in Specific Comment 10
Degraded Core - Training	II.B.4	FL Partial FP Partial	Yes; included in Specific Comment 10
Degraded Core - Rulemaking	II.B.8	FP	No
Interim Hydrogen Control Requirements for Small Containments	II.B.7	FP	No
Mini-IREP	II.C.1.1	FP	Yes
Reliability Assurance	II.C.1.3	FP	Yes
Relief and Safety Valve Test	II.D.2	FL	No
Relief and Safety Valve Position	II.D.5	FL	No
Auxiliary Feedwater System Reliability	II.E.1.1	FP	No
Auxiliary Feedwater Automatic Initiation	II.E.1.2	FP	No
Emergency Power for Decay Heat Removal	II.E.3.1	FP	No
Containment Penetrations	II.E.4.1	FL	No
Containment Isolation	II.E.4.2	FP	Yes
Containment Purge	II.E.4.4	FP	No
Inadequate Core Cooling Instruments	II.F.2	FL	Yes
Emergency Power for Pressurizer Equipment	II.G.1	FL	No
Role of the NRC	III.A.3.1	FP	No

Topic/Requirement	Action Plan* Reference Number	When Applicable**	Specific Comments Included
Communications	III.A.3.3	FL	No
Technical Support Center	III.A.1.2	FL	Yes
Onsite Operational Support Center	III.A.1.2	FL	No
Near-Site Emergency Operations Center	III.A.1.2	FL	Yes
Upgrade Licensee Emergency Preparedness	III.A.1.1	FL	No
FEMA-NRC Concurrence in State and Local RERP	III.A.2	FL	No
Area Radiation Monitors (Partial)	III.D.3.3	FL	No
Control Room Habitability	III.D.3.4	FP	Yes
Evaluation of Secondary Site Hazards	III.D.1.3	FP	Yes
Improve Auxiliary Building	III.D.1.5	FP	Yes
Improved Vent Gas System	III.D.1.2	FP	Yes
Surveillance Testing (Filtration Systems)	III.D.1.6	FL	No
NRC Monitoring	III.D.2.6	FL	No

ENCLOSURE 7

STUDY OF RULEMAKING ACTIONS FOR IMPLEMENTING

NEAR TERM OPERATING LICENSE REQUIREMENTS

IN TMI ACTION PLAN

A-194

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

.-. : 1980

MEMORANDUM FOR: Roger J. Mattson, Director, DSS, NRR

FROM: Guy A. Arlotto, Director, DES, SD

SUBJECT: ACTIONS NECESSARY TO IMPLEMENT THE TMI ACTION PLAN NEAR-TERM OPERATING LICENSE REQUIREMENTS

An ad hoc subgroup met on January 11, 15, and 22, 1980, to consider the need for changes in NRC's regulations in order to implement the Near-Term Operating License Requirements. Those participating were:

Warren Minners, NRR
Jim Stone, IE
Jim Wolf, ELD
A. L. Eiss, SD
G. A. Arlotto, SD (Chairman)

Our recommendations are enclosed. Also enclosed is a memorandum from ELD stating "no legal objections to the recommendations" and providing additional views regarding implementation of these recommendations. This completes our assignment.

A L Eiss for
Guy A. Arlotto, Director
Division of Engineering Standards
Office of Standards Development

Enclosures:
1. Actions Necessary to Implement the TMI Action Plan, Near-Term Operating License Requirements
2. Memo from GCunningham, ELD, to GAArlotto, SD dtd 1/25/80

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KEY R - Rule Change Required
D - Rule Change Desirable
U - Rule Change Unnecessary

Actions Necessary to Implement the TMI Action Plan, Near-Term Operating License Requirements

Requirement	Applicable Rule	Interim Change	Final Change
I.A.1.1 Shift Technical Advisor	50.34(b)(6)(i) 50.34(b)(7) 50.57(a)(3)	Letter to licensees and applicants	U, Revise R.G. 1.8, SRP 13.1.1
I.A.1.2 Shift Supervisor Duties	50.54(1) 50.34(b)(6)(i)	Letter	U, Revise R.G. 1.8
I.A.1.3 Shift Manning			
(1) SRO and RO in Control Room	50.54(k) 50.54(m)	Tech Specs	D, Revise 50.54(k) and 50.54(m)
(2) Administrative Aide to Shift Supervisor	50.34(b)(6)(i) 50.57(a)(3)	Letter	U, Revise R.G. 1.8
(3) Restrictions on Use of Overtime	50.34(b)(6)(i) 50.34(b)(7) 50.57(a)(3)	Letter	U, Revise R.G. 1.8
I.B.1.1 Organization and Management Criteria	50.34(b)(6)(i) 50.57 (a) (4)	Letter	U, Revise R.G. 1.8
I.B.3.1 Safety Engineering Group	50.34(b)(6)(i) 50.57(a)(4)	Letter	U, Revise R.G. 1.8 or 1.33
I.B.3.4 Resident Inspector	50.70(a) 50.70(b)		U, No change needed
I.C.1.1 Analysis and Procedure Modification			
(1) Small Break LOCA	50.46 50.34(b)(6)(v) 50.57(a)(3)	Letter	U, SRP

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ENCLOSURE 1

Requirement	Applicable Rule	Interim Change	Final Change
I.C.1.1 Analysis and Procedure Modification (Continued)			
(2) Inadequate Core Cooling	50.34(b)(6)(v) 50.57(a)(3)	Letter	U, SRP or R.G. 1.33
I.C.1.2 Shift Relief and Turnover Procedures	50.34(b)(6)(iv) 50.57(a)(3)	Letter	U, Revise R.G. 1.33
I.C.1.3 Shift Personnel Responsibilities	50.34(b)(6)(i) 50.57 (a)(3)	Letter	U, Revise R.G. 1.8
I.C.1.4 Control Room Access	50.34(b)(6)(v) 50.34(b)(6)(iv) 50.57(a)(3)	Letter	U, Revise R.G. 1.33
I.C.2 Vendor Review of Procedures	Part 50, Appendix B, Criteria I & III	Letter	U, None
I.C.3 Pilot Program for Review of Selected Emergency Procedures	50.70(a)	None	U, None
I.E.1 Licensee Operating Experience Evaluation Capability	50.34(b)(6)(i) 50.57(a)(4)	Letter	U, Revise R.G. 1.8, 1.33, 1.70
I.E.2 Licensee Dissemination of Operating Experiences	Part 55, Appendix A 50.34(b)(6)(iv) 55.10(a)(6) 55.33(a)(4) Part 50, Appendix E, IV.H	Letter	U, Revise R.G. 1.8
I.G. Training During Low Power Testing	55.10(a)(6)	Letter	U, Revise R.G. 1.8
II.B.1 Degraded Core - Primary System Vent	50.34(b)(2)	Letter	D, new rule
II.B.2 Degraded Core - Shielding	Part 100 / Part 50 50.34(b)(2) / App. A, GDC 61	Letter	D, new rule
II.B.3 Degraded Core - Sampling	Part 100 / Part 50 50.34 (b)(2) / App. A, GDC 61	Letter	D, new rule

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Requirement	Applicable Rule	Interim Change	Final Change
II.B.4 Degraded Core -Training	50.34(b)(6)IV Part 50, App. E.IV.H	Letter	D, new rule
II.B.8 Degraded Core - Rulemaking	50.34(b)(2) Part 100	D, Advance Notice of Rulemaking in Process	D, New Rule
II.B.10 Interim Hydrogen Control Requirements for Small Containments	50.44	R, revision to 50.44	R, revision to 50.44
II.C.1.1 - Mini IREP	50.34(b)(2) 55.57(a)(3)	Letter	U, None
II.C.1.8 Reliability Assurance	Part 50, App. B	Letter	U, Revise R.G. 1.33
II.D.1.1 Relief & Safety Valve Test	Part 50, App. A GDC 1	Letter	U, Revise SRP 3.9.3
II.D.1.5 Relief & Safety Valve Position	Part 50, App. A GDC 13	Letter	U, Revise SRP
II.E.1 Auxiliary Feedwater System Reliability	Part 50, App. A GDC 34	Letter	U, Revise SRP 10.4.9
II.E.1.3 Auxiliary Feedwater Initiation	Part 50, App. A GDC 34	Letter	U, Revise SRP 10.4.9
II.E.3 Emergency Power for Decay Heat Removal	Part 50, App. A GDC 34 Part 50, App. E, IV.H	Letter	U, Revise SRP 5.4.7
II.E.4.1 Containment Penetrations	Part 50, App. A. GDC 41	Letter	U, already in SRP 6.5.1
II.E.4.3 Containment Isolation	Part 50, App. A GDC 22	Letter	U, Already in SRP
II.E.4.5 Containment Purge	Part 50, App. A GDC 54	Letter	U, already in SRP
II.F.2 Inadequate Core Cooling Instruments	Part 50, App. A GDC 13	Letter	U, May require change to R.G. 1.97
II.G Emergency Power for Pressurizer	Part 50, App. A GDC 17	Letter	U, may require R.G. revisi

Requirement	Applicable Rule	Interim Change	Final Change
III.A.1.1 Role of NRC	-------	--- --	--- --
III.A.1.5 Communications	Part 50, App. E. IV.D	Letter	U, Revise R.G. 1.101
III.A.2.1 Technical Support Center	Part 50, App. E, IV.A	Letter	U, Revise R.G. 1.101 Review SRP 6.4, 9.4.1, 9.5.2, 12.2, 12.3, 12.5 for possible revisions
III.A.2.2 Onsite Operational Support Center	Part 50, App. E, IV.A	Letter	U, Revise R.G. 1.101, SRP 13.3
III.A.2.3 Near-Site Emergency Operations Center	Part 50, App. E, IV.A	Letter	U, Revise R.G. 1.107
III.A.3 Upgrade Licensee Emergency Preparedness	Part 50, App. E	Letter	U, No change needed
III.B.3.2 FEMA-NRC Concurrence in State and Local RERP	Part 50, App. E, IV.D	Letter	D, Revise App. E also R.G. 1.101
III.D.1.3.a Area Radiation Monitors (Partial)	Part 50, App. A GDC 64	Letter	U, Revise SRP 12.5
III.D.2.1 Control Room Habitability	Part 50, App. A GDC 4	Letter	U, Revise R.G. 1.78 & 1.9
III.D.2.2.b Evaluation of Secondary Side Hazards	Part 50, App. A GDC 61	Letter	U, Revise SRP 9.3.4, 11.3
III.D.2.2.c Improve Auxiliary Building	Part 50, App. A GDC 61	Letter	U, Revise SRP 9.3.4, 11.3
III.E.1.1 Improve Vent Gas Systems	Part 50, App. A. GDC 60	Letter	U, Revise SRP 11.3
III.E.1.2.a Surveillance Testing (Filtration Systems)(Partial)	Part 50, App. I	Letter	U, ALARA Tech Specs
III.E.2.1.b NRC Monitoring	Part 50, App. A, GDC 64	Letter	U

ENCLOSURE 2

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

JAN 2 5 1980

MEMORANDUM FOR: Guy A. Arlotto, Director
Division of Engineering Standards
Office of Standards Development

FROM: Guy H. Cunningham, III
Chief Regulations Counsel
Office of the Executive Legal Director

SUBJECT: ACTIONS NECESSARY TO IMPLEMENT THE TMI ACTION PLAN NEAR-TERM OPERATING LICENSE REQUIREMENTS

OELD has no legal objections to the recommendations of the ad-hoc subgroup regarding actions necessary to implement the TMI Action Plan near-term operating license requirements. We wish to emphasize, however, certain considerations which should be borne in mind by those who must act on these recommendations.

The informal method of implementation chosen in the vast majority of instances is a letter to licensees followed by revisions to various regulatory guides. Because the current regulations are sufficiently broad and general, they provide room for the staff to adopt the interpretation that they may be satisfied only by meeting the Action Plan's near-term operating licensing requirements. Such informal interpretations, however, will not be binding on applicants, licensees, intervenors, Boards, or the Commission, and may be litigated in individual cases. For example, an applicant would be free to argue to a licensing board that he has demonstrated "technical competence" without providing for a shift technical advisor, since no rule calls for a shift technical advisor. Without such a rule, there can be no guarantee that the staff would prevail against all such challenges. Rulemaking, though it has other drawbacks, would eliminate such litigation by imposing binding requirements. As stated above, we have no legal objection to the informal methods selected, provided that the risks of repetitive litigation in individual adjudications are understood and accepted.

Where the option of implementation by technical specification changes is selected, pending a rule change (e.g., for SRO and RO in the control room), there is the additional factor that opportunities for hearing will be created, since technical specification changes are license amendments. (Of course, these amendments would not have to be pre-noticed if a finding of no significant hazards consideration was made.) Since a rule change is contemplated as the final method of implementation, additional consideration should be given to eliminating the interim stage of amending technical specifications and proceeding directly to an expedited rulemaking.

Guy H. Cunningham, III
Chief Regulations Counsel
Office of the Executive
Legal Director

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Appendix V

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

February 7, 1980

APPENDIX V
TMI-1 RESTART: BACKGROUND MATERIAL

ACRS MEMBERS

STATUS REPORT ON TMI-1 RESTART REVIEW

The TMI Subcommittee met in Middletown on January 31 and February 1. The Subcommittee felt too many things were incomplete to give Met. Ed. much hope that the ACRS could write a report and Met. Ed. was so advised. It is not clear that the Licensing Board requires an ACRS report. This is a restart review not a CP or OL application. The Committee should note that the Staff submitted a "Status Report" rather than a "Safety Evaluation Report" (SER). The Staff is planning to issue a Supplement to the "Status Report" by around the end of March.

Attached hereto are the following items:

1. A Schedule for the Full Committee Meeting. (This is rather extensive and may take a little more time than scheduled).

2. Written comments prepared by Subcommittee members and consultants after the February 1 meeti... - suggesting items for full committee review.

3. A highlights summary of the meeting prepared by Peter Tam.

4. A summary of the items contained in the Commission's restart order.

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Attachments:
As noted

cc: R. F. Fraley
M. W. Libarkin
T. G. McCreless
Tech. Staff

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DELETION 1

Pages A-202 thru A-208

ATTACHMENT 3

MEETING HIGHLIGHTS, REQUEST, AND AGREEMENTS

1. The NRC Staff gave a status report of the Staff review of the licensee's compliance with Order 10 NRC 141 (1979). Details of Mr. Silver's presentation are contained in the handout he gave the Subcommittee.

2. A total of 12 items were discussed by different individuals. These are summarized as follows:

   (1) Emergency feedwater (EFW) changes -- A number of changes pertaining to the EFW have been proposed by the licensee. The Staff indicated its comments on the proposed changes are as shown in the "Status Report on the Evaluation of Licensee's Compliance With the NRC Order, dated August 9, 1979" (the Status Report). No new comments were added by the Staff. The licensee said that the final method of cooling is not the EFW, but use of the ECCS in a feed-and-bleed mode.

   (2) Separation of Units 1 and 2 -- It was ordered that the licensee shall ensure that decontamination and restoration operations at Unit 2 will not affect safe operation of Unit 1. The licensee described his proposed physical separation of the fuel handling building, currently shared by both units, into portions belonging to Unit 1 and Unit 2, respectively. The solid waste disposal system, previously shared by both units and located at Unit 1, now serves only Unit 1; Unit 2 was given its own system. The Staff did not have additional comments.

   (3) Provision for Obtaining RCS Samples and Radwaste Systems -- The licensee explained the liquid waste and miscellaneous radwaste systems (see handout diagrams by E. Fuhrer) as revised. Also being revised is the RCS sampling procedure.

   (4) Training -- The licensee has formed two external committees to provide feedback to his training program. One of these committees will concern itself with human factors. Currently, the revised training program calls for a large portion of training time being alloted to basic subjects such as thermodynamics and thermohydraulics. The licensee also described specific training programs for operators, auxiliary operators, shift supervisors, and shift technical advisors.

(5) Operating Procedures -- Revised operating procedures will include requirements in the Lessons Learned reports and Bulletins and Orders. Mr. Ebersole voiced his concern on operating procedures about the futility of instruction to verify status without further instruction on further actions if the status is not as expected.

(6) Emergency Procedures -- Emergency procedures to deal with a stuck open safety valve, small LOCA, stuck open PORV, loss of offsite power, and loss of pressurizer void were discussed by the licensee. Mr. Ebersole pointed out the problem of loss of DC power should be dealt with since the situation may lead to an extended period of total station blackout. Upon request of the Subcommittee, the licensee has agreed to address the full ACRS on his compliance with Reg. Guide 1.97.

(7) Emergency Preparedness -- The licensee has submitted a final emergency plan for NRC review on November 29, 1979. The State of Pennsylvania is ultimately responsible for evacuation, and it has accepted the responsibility.

(8) Long-Term Actions Listed in NUREG-0578 -- The licensee indicated his intention to comply with most of these recommendations. There are items that he does not agree with the Staff. Notable among these is the need for vent valve at the top of the reactor vessel: the licensee argues that such is not necessary.

(9) Radiation Protection Plan -- The licensee has reorganized his radiation protection staff. In addition, outside consultants will be periodically employed to revised the program. The licensee emphasized his desire to maintain the ALARA criterion.

(10) Pipe Cracking -- The Staff said that about half a dozen plants, including TMI-1, have problems with stress corrosion cracking (SSC). A bulletin has been issued requesting these licensees to examine pipes containing stagnant boric acid. Dr. Dillon stated that he believes SSC may be caused by the synergism between boric acid and some undefined substance. Better information is needed on the chemistry of the storage water. Investigation of SSC at TMI is ongoing.

(11) Separation of Unit 1 and 2 Security -- The licensee has installed a separate security system at each unit; persons cleared for access to one unit may not necessarily have access to the other. The Staff indicated its satisfaction with the arrangement.

(12) Organizational Changes -- The licensee submitted copies of a Letter (to his employees) indicating a forthcoming reorganization. No additional details were available.

3. Mr. D. DiIanni briefed the Subcommittee on the backlog items for TMI-1. There are 38 of such items and are summarized in Mr. DiIanni's handout.

4. Mr. Etherington asked members and consultants to submit their additional views in writing.

5. Members and consultants indicated that the full ACRS should hear presentations on:

   (1) Items in dispute - especially the reasons for dispute.

   (2) General status of Lessons Learned backlog items.

   (3) Schedule for completion of a restart program.

   (4) Met. Ed.'s position with regard to implementation of Reg. Guide 1.97.

   (5) Licensee's organizational changes, especially in technical support.

   (6) Plant security (closed session).

   (7) Training of operating and maintenance personnel. (Dr. Lawroski advised tha' an applicable representation from the operating personnel be present at the full ACRS meeting.)

   (8) Appendices to emergency procedures. (This refers to actions to be taken after each "verify that,","confirm that," etc., type of instruction in the operating procedures.)

   (9) DC power failure.

   (10) Control room design.

   (11) In-containment H_2 monitoring.

   (12) Use of process computer.

   (13) Containment isolation capability during operational purge.

   (14) Reactor vessel vent valve (reasons for need or no need).

(The Subcommittee indicated that all the items discussed in the 1 1/2-day meeting should be presented to the full ACRS, with much less detail. The Subcommittee advised the Staff and licensee to place special emphasis on the 14 items mentioned above.)

FUTURE MEETINGS:

There will be need for another or other meetings but no dates have been established.

Feedwater (EFW) system ... Enclosure 1 of the licensee's June 28, 1979 letter. Changes in design will be submitted to the NRC staff for review.

(b) Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System (ICS) control.

(c) Install a hard-wired control grade reactor trip on less of main feedwater and/or on turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Augment the retraining of all Reactor Operators and Senior Reactor Operators assigned to the control room including training in the areas of natural circulation and small break loss of coolant accidents including revised procedures and the TMI-2 accident. All operators will also receive training at the B&W simulator on the TMI-2 accident and the licensee will conduct a 100 percent reexamination of all operators in these areas. NRC will administer complete examinations to all licensed personnel in accordance with 10 CFR 55.20-23..

2. The licensee shall provide for NRC review and approval of all applicable actions specified in IE Bulletins 79-05A, 79-05B, and 79-05C.

3. The licensee shall improve his emergency preparedness in accordance with the following:

(a) Upgrade emergency plans to satisfy Regulatory Guide 1.101 with special attention to action level critieria based on plant parameters.

(b) Establish an Emergency Operations Center for Federal, State, and Local Officials and designate a location and an alternate location and provide communications to plant.

(c) Upgrade offsite monitoring capability, including additional thermo-luminescent dosimeters or equivalent.

(d) Assess the relationship of State Local plans to the liacensee plans so as to assure the capability to take emergency actions.

(e) Conduct a test exercise of its emergency plan.

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4. The licensee shall demonstrate that decontamination and/or restoration operations at TMI-2 will not affect safe operations at TMI-1. The licensee shall provide separation and/or isolation of TMI 1/2 radioactive liquid transfer lines, fuel handling areas, ventilation systems, and sampling lines. Effluent monitoring instruments shall have the capability of discriminating between effluents resulting from Unit 1 or Unit 2 operations.
5. The licensee shall demonstrate that the waste management capability, including storage and processing, for solid, liquid, and gaseous wastes is adequate to assure safe operation of TMI-1, and that TMI-1 waste handling capability is not relied on by operations at TMI-2.
6. The licensee shall demonstrate his managerial capability and resources to operate Unit 1 while maintaining Unit 2 in a safe configuration and carrying out planned decontamination and/or restoration activities. Issues to be addressed include the adequacy of groups providing safety review and operational advice, the management and technical capability and training of operations staff, the adequacy of the operational Quality Assurance program and the facility procedures, and the capability of important support organizations such as Health Physics and Plant Maintenance.
7. The licensee shall demonstrate his financial qualifications to the extent relevant to his ability to operate TMI-1 safely.
8. The licensee shall comply with the Category A recommendations as specified in Table B-1 of NUREG-0578.

The Commission has additional concerns, which, though they need not be resolved prior to resumption of operation at Three Mile Island Unit 1, must be satisfactorily addressed in a timely manner. The Commission's Director of Nuclear Reactor Regulation (NRR) has recommended that the following actions (the "long-term actions") be required of the licensee to resolve these concerns and permit a finding of reasonable assurance of the safety of long-term operation. These are:

1. submit a failure mode and effects analysis of the ICS to the NRC staff as soon as practicable;
2. give continued attention to transient analysis and procedures for management of small breaks by a formal program set up to assure timely action of these matters;
3. comply with the Category B recommendations as specified in Table B-1 of NUREG-0578; and,
4. improve emergency preparedness in accordance with the following:
   (a) modify emergency plans to address changing capabilities of plant instrumentation,
   (b) extend the capability to take appropriate emergency actions for the population around the site to a distance of ten miles.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations in 10 CFR, it is hereby ordered that:

(1) the licensee shall maintain TMI-1 in a cold shutdown condition until further order of the Commission which will be issued following satisfactory completion of the required short-term actions and reasonable progress toward satisfactory completion of those required long-term actions referred to in section IV (such short-term and long-term actions to be considered "required" for purposes of this clause which are determined by the Commission, after review of the Licensing Board's decision, to be necessary and sufficient to provide adequate protection of the public health and safety); and

(2) the licensee shall satisfactorily complete the long-term actions listed in Table B-1 of NUREG-0578 on the schedule set out in such table and such other long-term actions listed above as promptly as practicable.

POOR ORIGINAL

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TABLE B-1. IMPLEMENTATION OF SHORT-TERM RECOMMENDATIONS FOR OPERATING PLANTS AND PLANTS IN OL REVIEW

Sect. No.	Position: Abbreviated Title	Position: Position Description	Implementation Category[a]
2.1.1	Emergency Power Supply Requirement	Complete implementation.	A
2.1.2	Relief and Safety Valve Testing	Submit program description and schedule.	A
		Complete test program.	By July 1981[b]
2.1.3.a	Direct Indication of Valve Position	Complete implementation.	A
2.1.3.b	Instrumentation for Inadequate Core Cooling	Develop procedures and describe existing instr.	A
		New instr. design, subcooling meter installation, and implementation schedule.	A
		Complete new instr. installation.	B
2.1.4	Diverse Containment Isolation	Complete implementation.	A
2.1.5.a	Dedicated H_2 Control Penetrations	Description and implementation schedule.	A
		Complete installation.	B

[a]Category A: Implementation complete by January 1, 1980, or prior to OL
Category B: Implementation complete by January 1, 1981.

[b]Relief and safety valve testing shall be satisfactorily completed for all plants prior to receiving an operating license after July 1, 1982.

TABLE B-1 (Continued)

Sect. No.	Position: Abbreviated Title	Position: Position Description	Implementation Category[a]
2.1.5.b	Rulemaking to Require Inerting BWR Containments	Inert Vermont Yankee and Hatch 2.	*
		Design and equipment to inert new Mark I and II containments.	*
		Inert new Mark I and II containments.	*
2.1.5.c	Combustible Gas Control Recombiner	Rulemaking to require capability of installing recombiners.	*
		Review procedures and bases for recombiner use.	B
2.1.6.a	Systems Integrity for High Radioactivity	Immediate leak reduction program.	A
		Preventive maintenance program.	A
2.1.6.b	Plant Shielding Review	Complete the design review.	A
		Implement plant modifications.	B

[a]Category A: Implementation complete by January 1, 1980, or prior to OL
Category B: Implementation complete by January 1, 1981.

*Implementation schedules will be established by the Commission in the course of the immediately effective rulemaking. The Task Force recommends that the rulemaking process be initiated promptly.

TABLE B-1 (Continued)

Sect. No.	Position: Abbreviated Title	Position: Position Description	Implementation Category[a]
2.1.7.a	Auto Initiation of Auxiliary Feed	Complete implementation of control grade.	A
		Complete implementation for safety grade.	B
2.1.7.b	Auxiliary Feed Flow Indication	Complete implementation.	A
2.1.8.a	Post-Accident Sampling	Design review complete.	A
		Preparation of revised procedures.	A
		Implement plant modifications.	B
		Description of proposed modification.	A
2.1.8.b	High Range Effluent Monitor	Installation complete.	B
2.1.8.c	Improved Iodine Instrumentation	Complete implementation.	A
2.1.9	Transient & Accident Analysis	Complete analyses, procedures & training.	**

[a]Category A: Implementation complete by January 1, 1980, or prior to OL
Category B: Implementation complete by January 1, 1981.

**Analyses, procedural changes, and operating training shall be provided by all operating plant licensees and applicants for operating licenses following the schedule in Table B-2.

TABLE B-1 (Continued)

Sect. No.	Position: Abbreviated Title	Position: Position Description	Implementation Category[a]
2.2.1.a	Shift Supervisor Responsibilities	Complete implementation.	A
2.2.1.b	Shift Safety Engineer	Shift technical advisor on duty.	A
		Complete training.	B
2.2.1.c.	Shift Turnover Procedures	Complete implementation.	A
2.2.2.a	Control Room Access Control	Complete implementation.	A
2.2.2.b	Onsite Technical Support Center	Establish center.	A
		Upgrade to meet all requirements.	B
2.2.2.c	Onsite Operational Support Center	Complete implementation.	A
2.2.3	Rulemaking to Revise LCOs for Safety System Availability	Tech. Spec. change.	*

[a]Category A: Implementation complete by January 1, 1980, or prior to OL
Category B: Implementation complete by January 1, 1981.

*Implementation schedules will be established by the Commission in the course of the immediately effective rulemaking. The Task Force recommends that the rulemaking process be initiated promptly.

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Appendix VI

STATUS SUMMARY UPDATE

LEGEND FOR MARKUP

	Resolved as of today
	Expected to be resolved by first supplement
2/15	Information scheduled for receipt
*	Considered significant open item
**	Item in dispute with licensee

STATUS SUMMARY

Order Item	Item Description	Comply or Reference	Description of Open Items: Group A	Group B	Group C	Group D
Short Term						
1a-1	Auto initiation of AFW					Functional test
-2	AFW valves fail open			Supporting analyses		Procedures & training
-3	Auto AFW load on diesels					Test
-4	AFW tech specs			3/15 Supporting analyses		Tech Specs
-5	AFW flow indication					Test
-6	AFW procedures					Procedures & retrain
-7	AFW aligament					Retraining
-8	AFW auto start annunciate			Detail drawings		
1b	EFW independent of ICS	Comply				Procedures
1c	Reactor trip on feed trip	(See 79-05B-5)				
1d	Small breaks analysis		PORV auto isolation	HPI flow rates		Tech Specs LOCA procedures
1e	Operator retraining				Operator exam - NRC	Audit exam
IE Bulletins						
79-05A-1	Accident understanding	Comply				
-2	Plant transient review	Comply				
-3	Transient procedures					Procedures
-4	Operating procedures					Procedures & retrain
-5	Valve position review					Procedures
-6	Containment isolation	(See 8-2.1.4)				

C-2-2-A

Order Item	Item Description	Comply or Reference	Description of Open Items Group A	Group B	Group C	Group D
-7	EFW valve procedures					Procedures
-8	EFW operability					Procedures & Tech Specs
-9	Transfer of containment liquids	(See B-2.1.4 & 2.1.6.a)				
-10	Safety system operability					Procedures
-11	Personnel actions - TMI-2					Retraining
-12	Prompt reporting	Comply				
79-05B-1	Natural circulation			Analysis of anticipatory fill		Procedures
-2	Vessel integrity					Procedures
-3	PORV setpoint	Comply				
-4	Manual reactor trip	Comply				Procedures
-5	Anticipatory trips			2/15 Detail design info		
-6	Prompt reporting	Comply				
-7	Tech Spec changes					Tech Specs
79-05C-1	RCP trips	Comply				Procedures
-2	Small LOCA analysis			2/15 Verify analysis		
-3	Operator action-RCP trips			2/15 Verify guidelines	~~Review guidelines~~	
-4	Reactor trip training					Retraining & procedures
-5	Inadequate core cooling			2/29 Design info & schedule		Procedures
3	Emergency Preparedness		1st proposed rule (long term)			
3a	Emergency plan update	Comply				
3b	Emergency operations center	Comply				

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Order Item	Item Description	Comply or Reference	Description of Open Items: Group A	Group B	Group C	Group D
3c	Offsite monitoring	Comply				
3d	State/local plans	Comply				
3e	Test exercise					Test exercise
4	Separation of TMI-1 & 2					
	Liquid radwaste	Comply				
	Gaseous radwaste			3/29 FHB mods detail design		
	Solid radwaste	Comply				
	Monitoring system	Comply				
	Sampling systems	Comply				
5	Waste management					
	Liquid & gaseous systems	Comply				
	Solid radwaste system		3/15 Plan for low activity storage	Permanent system [illegible] 5/80		
6	Managerial Capability					
	Management & technical		* New organization	~~Clarify inconsistencies~~	* Develop criteria	
	Safety review				* Revise R.G. 1.33 & ANSI N 8.7	
	Plant maintenance					
	Operations training				* Develop criteria	
	Operational QA			QA program		
	Facility procedures			QA program	~~Procedure audit~~	Procedures [illegible]
	Health physics		[illegible]		Radiation Protection Plan	
7	Financial Qualifications		* Complete State hearings			

Order Item	Item Description	Comply or Reference	Description of Open Items: Group A	Group B	Group C	Group D
8	Lessons Learned - Short-term					
2.1.1	Emergency Power Supply					
	Pressurizer heaters					Procedures & training
	Pressurizer level & block valves					Procedures
2.1.2	Relief valve testing	Comply				
2.1.3.a	Valve position indication			2/15 Flow sensor info 2/15 Qualification of equipment 2/15 Disch. temp study		Procedures
2.1.3.b	Inadequate core cooling					
	Existing instrumentation					Procedures & training
	Saturation meter			2/15 Instrument uncertainty		Procedures & training
	New instrumentation		* 2/29 Analysis, commitment, schedule, description			
2.1.4	Containment isolation			2/15 Letdown requirements, air sample lines, detail design drawings, 2/15 ICW isolation 2/15		Procedures
2.1.5.a	Dedicated recombiner penetrations	Comply				
2.1.5.b	Inerting BWR containments	Not Applicable				
2.1.5.c	Install recombiners	Compliance not required				
2.1.6.a	System integrity		* 2/29 Describe program			
2.1.6.b	Plant shielding		* 2/29 Design review			
2.1.7.a	AFW auto initiation		2/15 Single failure	Justify control-grade components		Tech Specs

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Order Item	Item Description	Comply or Reference	Description of Open Items: Group A	Group B	Group C	Group D
2.1.7.b	AFW flow indication			Flow indicator qualification 2/15		Tech Spec & procedures
2.1.8.a	Post-accident sampling		* 2/15 Description **			
2.1.8.b	Radiation monitor range			Interim high-level method 2/15		Procedures
2.1.8.c	Iodine instrumentation					Procedures & training
2.1.9	Transient & accident analyses		3/80 Long term analysis	Applicability to TMI-1 3/80		Procedures & training
2.2.1.a	Shift supervisor responsibilities			Clarify authority		Training
2.2.1.b	Shift Technical Advisor	Comply				
2.2.1.c	Shift Turnover	Comply		Shift checklist contents, auxiliary operator checklists		
2.2.2.a	Control room access	Comply				
2.2.2.b	Onsite Tech Support Center		2/15 Clarification items b, d, f, g			
2.2.2.c	Onsite Operations Support Center		2/15 Communication & management			
Add. 4	RCS Venting		* ** 2/29 RV head vent	Detail design & analysis * 2/15		
Long-Term						
1	ICS FMEA			Address recommendations 2/15		
2	Small Break Analysis		2/15 Address item			
3	Lessons Learned					
	Containment Pressure	Comply				
	Containment Water Level	Comply				
	Containment H_2 Level	Comply				
4	Emergency Preparedness	Comply				

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ADDITIONAL ITEMS

Order Item	Item Description	Comply or Reference	Group A	Group B	Group C	Group D
1a-Add'l	Reliability Analysis		* Submit 3/15			
1	Redundant instrumentation	Comply				
2	Endurance Test					Test
3	Transfer of EFW Supply					Procedures
4	EFW to intact OTSG	Comply				
5	Auto EFW protection on loss of water source	Comply				
6	EFW initiation independent of AC		3/15 Address item			
7	EFW operability in steam environment			Verify for EFW valves 3/15		
8	Cross-tie break		3/15 Address item			
79-05C Long term 1	Auto RCP trip			Address conditions in SMUD letter; document current/void fraction relationship		

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STATUS SUMMARY REPORT
BACKLOG ITEMS FOR TMI-1

TOTAL NUMBER OF OPEN BACKLOG ITEMS AS OF JANUARY 30, 1980.....37

CATEGORY

A NUMBER OF ITEMS SCHEDULED FOR COMPLETION IF TMI-1 IS PERMITTED TO RESTART DURING THE 4TH QUARTER 1980...32

B NUMBER OF ITEMS NOT SCHEDULED FOR COMPLETION BY RESTART...05

C NUMBER OF ITEMS EXPECTED TO BE IMPACTED BY THE HEARING, CONTENTIONS, INTERROGATORIES, ETC.........07

D NUMBER OF ITEMS GENERIC IN NATURE WHICH ARE ALSO APPLICABLE TO OTHER REACTORS SIMILAR TO TMI-1..... 24

E ITEMS UNIQUE TO TMI-1......................................13

BACKLOG OPEN ITEMS FOR TMI-1

ITEM NO.	DESCRIPTION	ACTION REQUIRED	COMMENTS*
1	River water pipes	PM to write off after resolving the conditions of the pipe near the aux. bldg.	A, E
2	Review vent gas header system problem	Write off after licensee submits revised FSAR fig.	A, E
3	Radiation protection manager	TS change after licensee agrees to the requirements of R. G. 1.8	A, C, E
4	Ring Girder Surveillance Schedule	Prepare SER & approve surveillance schedule after resolving the use of R. G. 1.35	A, E
5	Small Break ECCS Modification	No action required except to follow to assure modification is implemented	A, C, D
6	Conversion to STS	Review the proposed STS to include the latest requirements	B, D
7	Filter Tech, Spec.	Process the TS change after agreement with licensee	A, C, D
8	Eliminate non-radiological environmental from TS	Coordinate existing non-radiological requirements with those to be handled by EPA	A, D
9	Delete from the TS; Snubbers that are on non-safety related systems	TS change required from licensee	A, E
10	Remove Reactor Internal Vent Surveillance Program while TMI is shutdown	Waiting for licensee's input	A, E
11	Quality Assurance for Diesel Generator Fuel Oil	Staff reviewing licensee's input	A, D

(A-22)

ITEM NO.	DESCRIPTION	ACTION REQUIRED	COMMENTS*
12	Manually isolate steam generator in the event of a steamline break	Licensee is evaluating	A, E MB ques.?
13	Feedwater line break	Licensee is evaluating	A, E
14	Automatic shift to SUMP recirculation of ECCS	Licensee is evaluating	A, E
15	Reactor vessel beltline Mat'l surveillance as result of TMI-2	Waiting on revised program from licensee	A, E
16	PWR pump & S/G supports - lamellar tearing & fracture toughness	Review licensee responses	A, D
17	Feedwater Pipe Cracks	Review licensee's response 6-26-79 GQL-0807 & prepare a SER	A, D
18	Defective welds in safety related systems	Review magnitude of defects & repair method, prepare a SER	A, E
19	IST	Review for acceptability IST	A, D
20	ISI	Review for acceptability ISI	A, D
21	Review asymmetric LOCA loads submittal D-10	Complete review & prepare a SER	A, D
22	TS for Hydraulic Snubber B-17	Complete review & prepare a SER	B, D
23	Containment Purge	Interim Position issued; waiting on licensee's valve study	A, C, D
24	Containment leak testing - App. J	Licensee change request No. 22 is to be reviewed for compliance with App. J and determine need for exemption	A, D

ITEM NO.	DESCRIPTION	ACTION REQUIRED	COMMENTS*
25	Degraded Grid Voltage	Licensee's change request No. 60 is to be reviewed	A, D
26	Control of heavy loads near spent fuel pool	Licensee's submittal is to be reviewed & SER prepared	A, D
27	Fire Protection SER Supplement B-41	SER's are to be prepared on asterisked items FPSER 9-19-78	A, D
28	Spent Fuel cask drop	Licensee would reactivate this task after restart	B, D
29	Containment leakage due to seat deterioration	Staff to evaluate further	B, D
30	Loss of 125V DC Bus voltage w/loss of ann. system B-21	Review & SER to be prepared	A, D — DO.?
31	Appendix I Tech, Spec.	Licensee's draft radiological effluent TS under review	A, C, D
32	Uneven drawdown of reactor bldg. spray system	Determine if I doses following a LOCA are affected	A, E
33	Contingency Plan	Review and Approve	A, C, D
34	Guard Training	Review and Approve	A, C, D
35	Vital Area	Review and Approve	A, D
36	Nuclear instrumentation calbration	Review licensee's submittal regarding questions raised by the staff	A, E
37	Control Rod Guide Tube Wear	Complete Review & prepare SER	B, D

* Letters in this column correspond to the Category Letters in Status Summary Report.

Prepared by D. C. DiIanni
1-29-80 492 7435

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GPU NUCLEAR CORPORATION

Objective:

- Full-time, single-minded dedication, uniform policies, maximum availability of technical resources to safely operate all GPU nuclear stations.

Results:

- Triples professional technical staff supporting current TMI activities.

- Combines into a single organization the technical and management skills in:

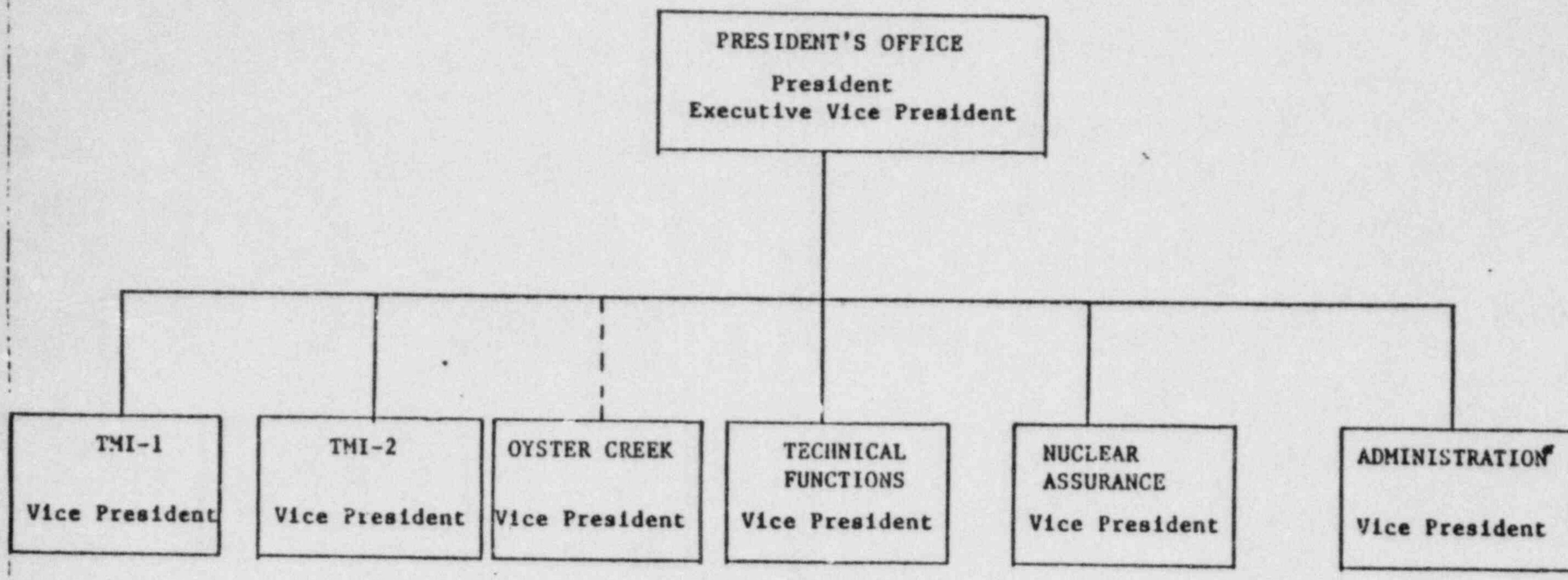
  - plant design criteria development
  - systems, analytical and design engineering
  - project, construction and procurement management
  - licensing

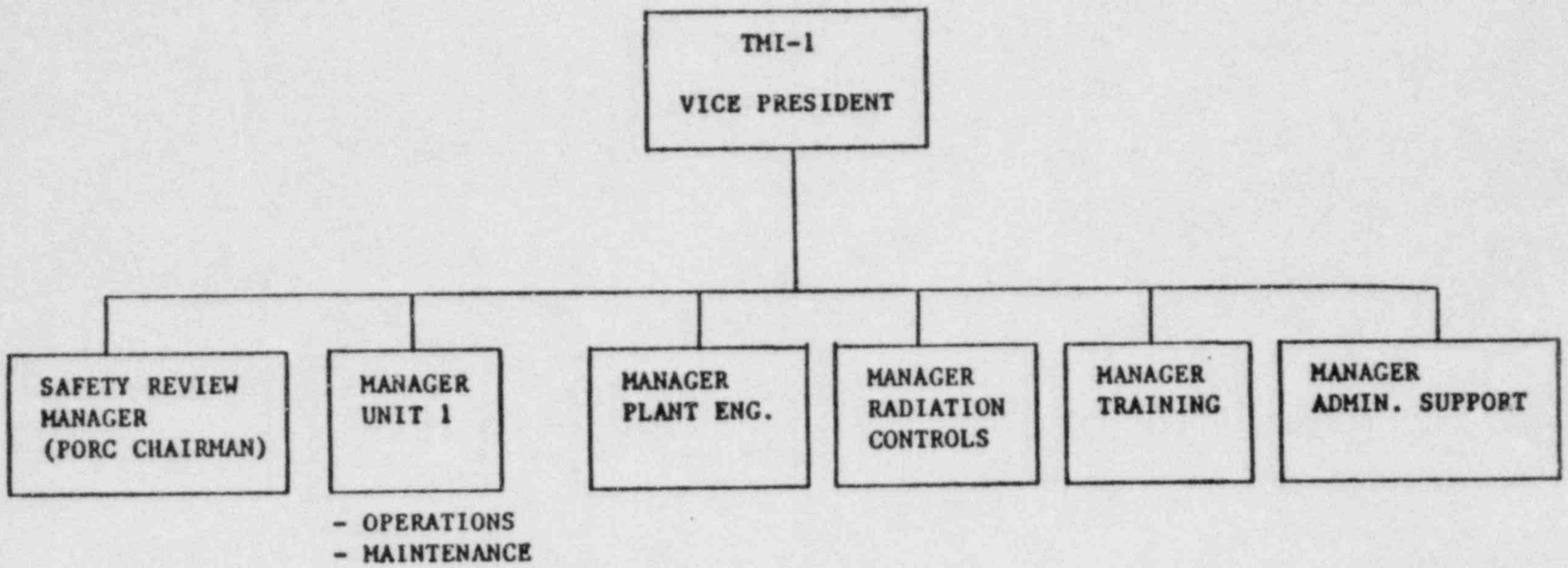
  associated with new generating stations with the, hands-on, operations-and-maintenance experience in conducting plant operations.

- Enhances customization of various vital administrative and support functions to the unique requirements of nuclear generating stations, including:

  - procurement
  - personnel
  - labor relations
  - security
  - facilities

GPU NUCLEAR CORPORATION




TMI-1
VICE PRESIDENT
SAFETY REVIEW MANAGER (PORC CHAIRMAN)
MANAGER UNIT 1
- OPERATIONS
- MAINTENANCE
MANAGER PLANT ENG.
MANAGER RADIATION CONTROLS
MANAGER TRAINING
MANAGER ADMIN. SUPPORT

TECHNICAL FUNCTIONS

Systems Engineering

Process Control

Nuclear Fuels Engineering and Management

Control and Safety Analysis

Performance Analysis

Human Factors

Engineering and Design

Fluid Systems

Mechanical Components

Plant Electrical Systems

Stress Analysis

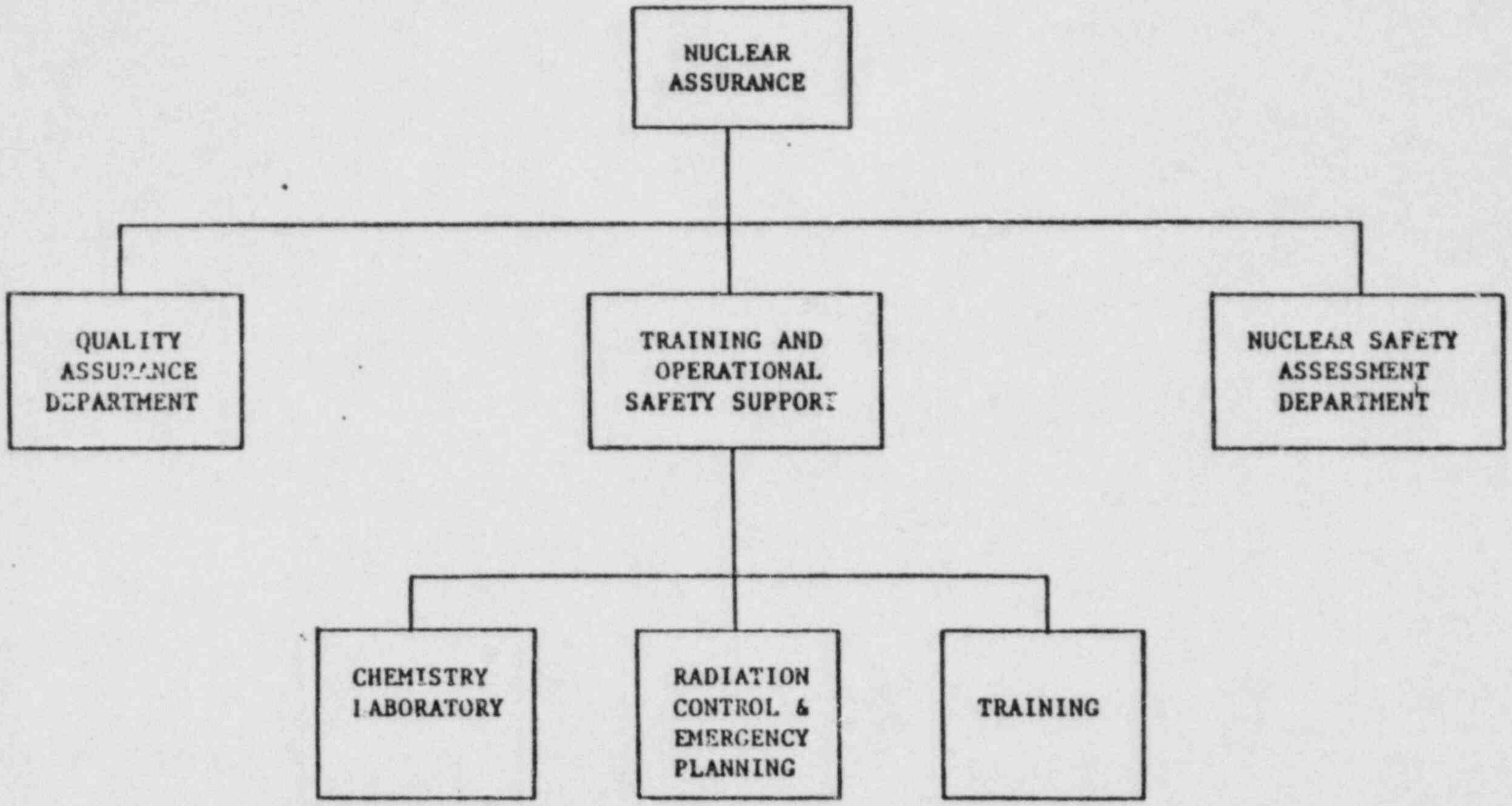
TMI-1 and TMI-2 Project Engineering Management

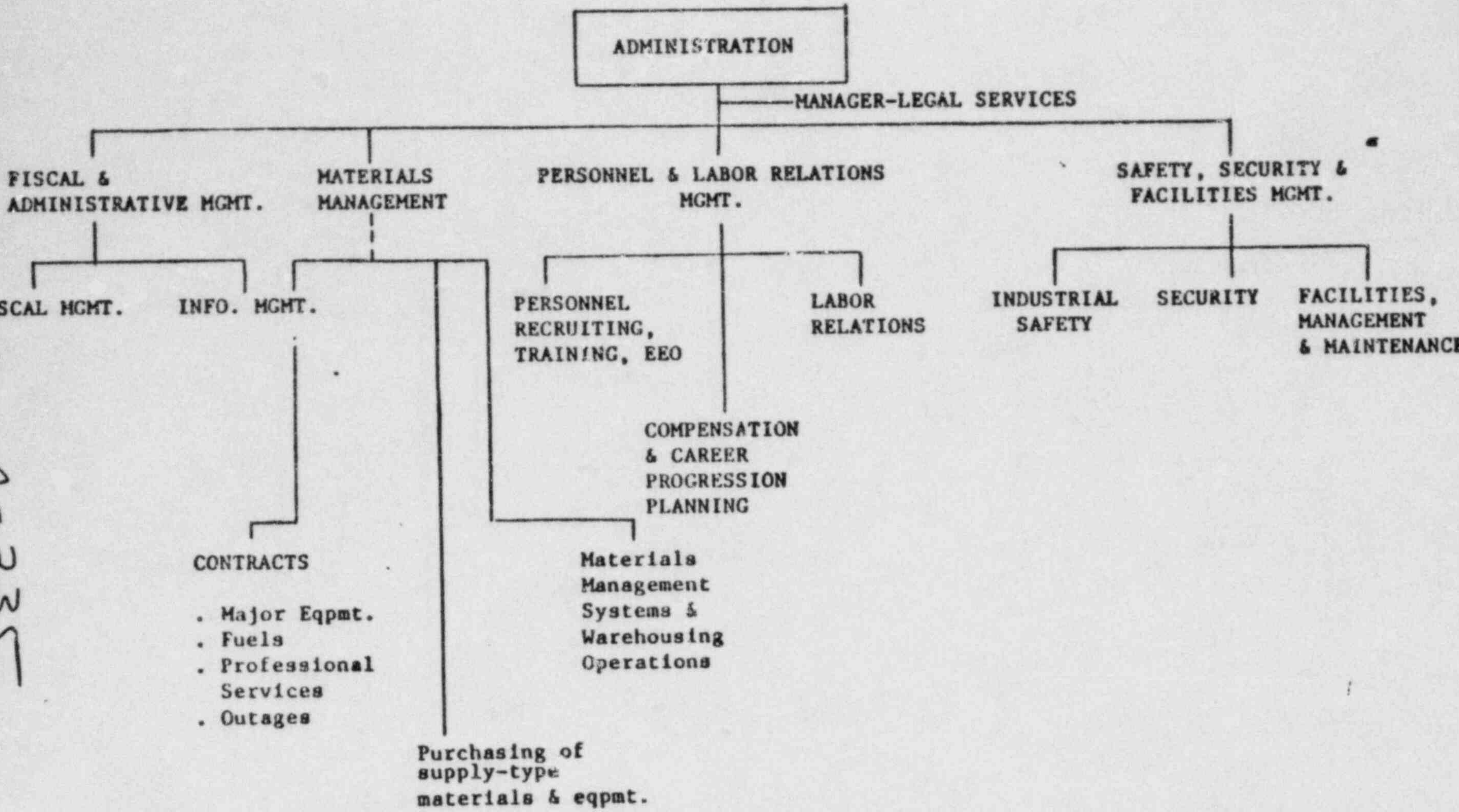
Licensing

Environmental Engineering

Independent Technical Reviews

NUCLEAR ASSURANCE - ORGANIZATION




ADMINISTRATION
MANAGER-LEGAL SERVICES
FISCAL & ADMINISTRATIVE MGMT.
MATERIALS MANAGEMENT
PERSONNEL & LABOR RELATIONS MGMT.
SAFETY, SECURITY & FACILITIES MGMT.
FISCAL MGMT.
INFO. MGMT.
PERSONNEL RECRUITING, TRAINING, EEO
LABOR RELATIONS
INDUSTRIAL SAFETY
SECURITY
FACILITIES, MANAGEMENT & MAINTENANCE
COMPENSATION & CAREER PROGRESSION PLANNING
CONTRACTS
. Major Eqpmt.
. Fuels
. Professional Services
. Outages
Materials Management Systems & Warehousing Operations
Purchasing of supply-type materials & eqpmt.

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Appendix IX

APPENDIX IX
TMI-1 RESTART: HANDOUTS PROVIDED BUT NOT DISCUSSED

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LOCA RESPONSE WITH AUX FEEDWATER

EVENT	PLANT RESPONSE	OPERATOR ACTIONS	INSTRUMENTS REQUIRED
Large Break LOCA	Rapid depressurization HPI CFT LPI	Trip RC Pumps Monitor Temperatures Balance LPI Flow Align LPI to Sump Terminate HPI	HPI Initiation Signal HPI Flow RCS Pressure Hot & Cold Leg RTD's LPI Flow BWST Level
Small Break LOCA - 1	Depressurization to Secondary Pressure HPI	Trip RC Pumps Raise S/G Level Check for Nat. Circulation Monitor Temperatures Bump pump, if necessary Depressurize with S/G Transfer to piggy-back mode if necessary	HPI Initiation Signal HPI Flow S/G Level RCS Pressure Hot & Cold Leg RTD's S/G Pressure Pressurizer Level BWST Level

LOCA RESPONSE WITH AUX FEEDWATER (CONT.)

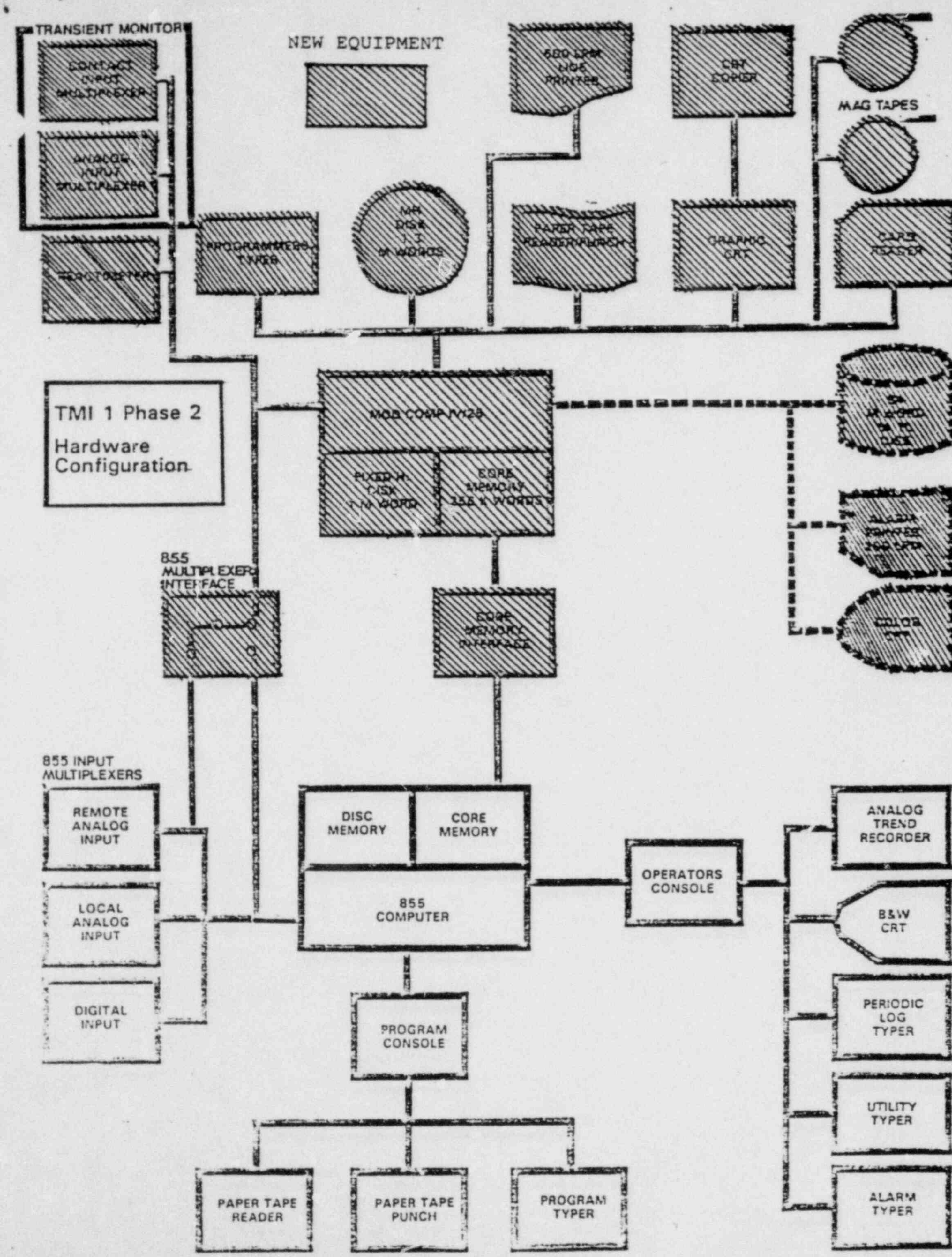
EVENT	PLANT RESPONSE	OPERATOR ACTIONS	INSTRUMENTS REQUIRED
SMALL BREAK LOCA -2	INITIAL DEPRESSURIZATION HPI REPRESSURIZATION WHILE SATURATED	TRIP RCS PUMPS RAISE S/G LEVEL CHECK FOR NAT. CIRCULATION OPEN PORV, IF NEEDED BUMP PUMP, IF NEEDED MONITOR TEMPERATURES DEPRESSURIZE WITH S/G	HPI INITIATION SIGNAL HPI FLOW S/G LEVEL RCS PRESSURE S/G PRESSURE HOT & COLD LEG RTD
SMALL BREAK LOCA - 3	INITIAL DEPRESSURIZATION HPI PRESSURE STABILIZES OR RISES SYSTEM REMAINS SUBCOOLED	TRIP RCS PUMPS RAISE S/G LEVEL CHECK FOR NAT. CIRCULATION MONITOR TEMPERATURES THROTTLE HPI, IF NEEDED RESTART RCS PUMP	HPI INITIATION SIGNAL HPI FLOW S/G LEVEL RCS PRESSURE S/G PRESSURE HOT & COLD LEG RTD PRESSURIZER LEVEL

LOCA RESPONSE W/O AUX FEEDWATER

EVENT	ADDED OR DIFFERENT OPERATOR ACTIONS	ADDED INSTRUMENT NEEDS
LOCA which Depressurizes RCS	None	None
LOCA which Stabilizes	Increase makeup if possible Depressurize with PORV	None
LOCA without HPI - S/G Boils Dry - RCS Repressurizes	Initiate HPI Control RCS Pressure with PORV	None

RESPONSE TO INADEQUATE CORE COOLING

INDICATION	OPERATOR ACTION
Incore Temp >Curve 1 (T_{CLAD} > 1400°F)	Start one RCS Pump per loop Depressurize S/G to 400 psig Open PORV to Reduce RCS Pressure Continue cooldown at 100°F/hr until RCS pressure <150 psig
Incore Temp >Curve 2 (T_{CLAD}> 1800°F)	Depressurize S/G to Atmospheric Pressure Start Remaining RCS Pumps Open PORV and Leave Open


TRANSIENT MONITOR
CONTACT INPUT MULTIPLEXER
ANALOG INPUT MULTIPLEXER
NEW EQUIPMENT
CRT COPIER
MAG TAPES
PAPER TAPE READER/PUNCH
GRAPHIC CRT
CARD READER
TMI 1 Phase 2
Hardware Configuration
855 MULTIPLEXER INTERFACE
855 INPUT MULTIPLEXERS
REMOTE ANALOG INPUT
LOCAL ANALOG INPUT
DIGITAL INPUT
DISC MEMORY
CORE MEMORY
855 COMPUTER
OPERATORS CONSOLE
ANALOG TREND RECORDER
B&W CRT
PERIODIC LOG TYPER
UTILITY TYPER
ALARM TYPER
PROGRAM CONSOLE
PAPER TAPE READER
PAPER TAPE PUNCH
PROGRAM TYPER

PLANT COMPUT... SPECIFICATIONS

	BAILEY 855		MOD COMP IV/25
	PRE TMI-2	POST TMI-2	
MEMORY CYCLE	2 μSEC	2 μSEC	0.5 μSEC
CORE MEMORY	32K WORDS	32K WORDS	256K WORDS
FLOATING POINT	SOFTWARE	SOFTWARE	HARDWARE
BULK MEMORY	512K WORDS	512K WORDS	86M WORDS
<u>ALARM PRINTER</u> SPEED	12 CPS	150 CPS (24 CPS COMPUTER LIMITATION)	480 CPS
RELIABILITY	POOR	VERY GOOD	VERY GOOD
OTHER I/O CAPABILITY	PAPER TAPE	PAPER TAPE	MAGNETIC TAPE (ACCESSIBLE BY 855) CARD READER TEKTRONIX 4014 GRAPHICS CRT AYDIN 5215 COLOR DISPLAY GEN. HIGH SPEED PRINTER (600 LPM)

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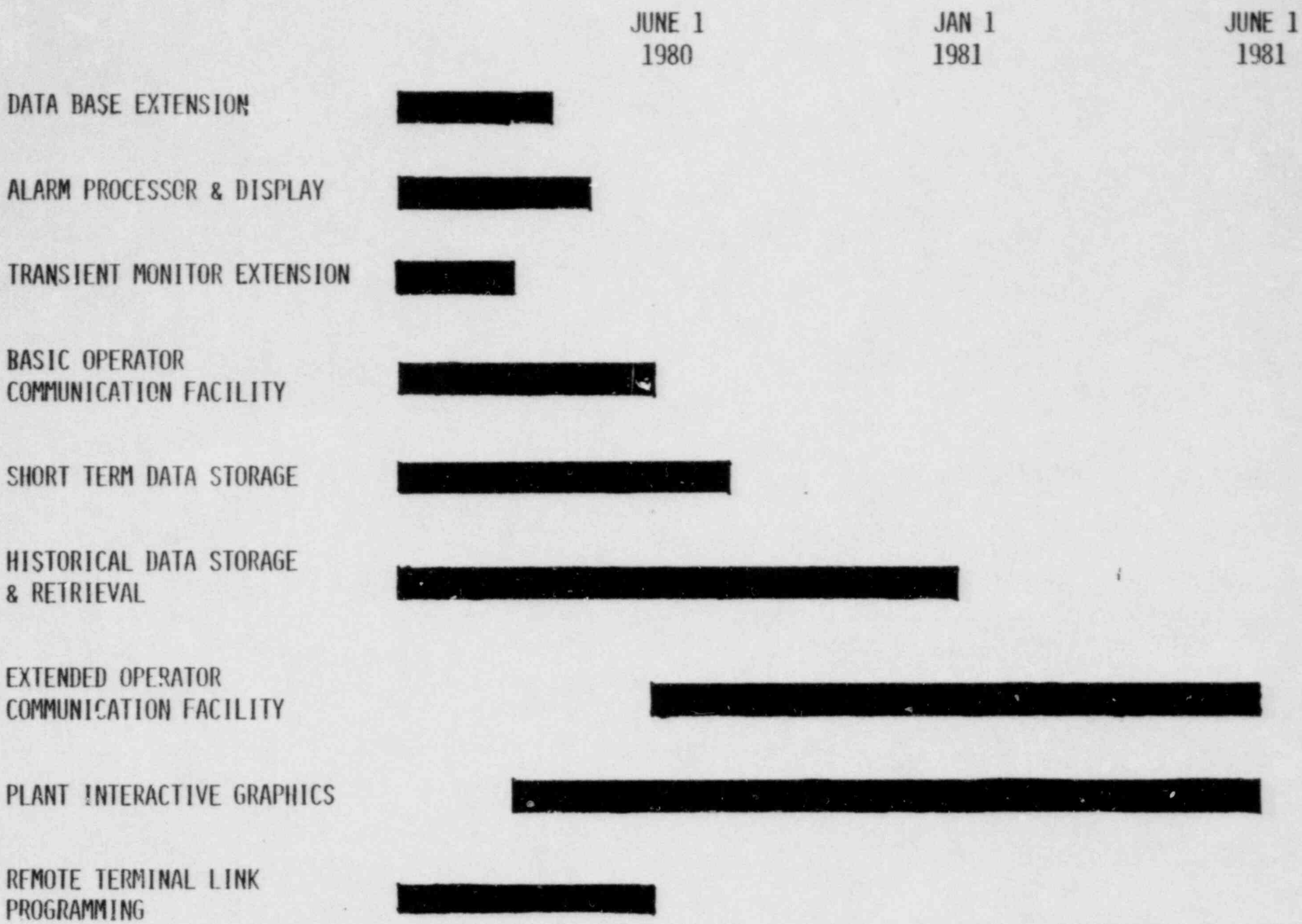
TMI-1 PLANT COMPUTER FUNCTIONS

FUNCTION	BAILEY 855 PRE TMI-2	BAILEY 855 POST TMI-2	MOD COMP IV/25 SHORT RANGE	MOD COMP IV/25 LONG RANGE
ANALOGS:	1166 POINTS	1227 POINTS	1227 POINTS	2048 POINTS
DIGITALS:	880	888	888	1024 POINTS
NSS CALCULATIONS	10 MINUTE	10 MINUTE	6 MINUTE, IMPROVED	6 MINUTE, IMPROVED
HUMAN COMMUNICATIONS	LIMITED 10 KEY INPUT, DIGITAL READOUT, PRINTER OUTPUT	SAME	SAME PLUS BASIC COLOR CRT DISPLAYS	EXPANDED, INCLUDING CRITICAL FUNCTIONS MONITOR, P&IDs WITH BEHAVIOR
HISTORICAL STORAGE AND RETRIEVAL	NO	NO	SHORT TERM STORAGE LIMITED RETRIEVAL	LONG TERM STORAGE, ENHANCED RETRIEVAL
TRANSIENT MONITOR	(ON MOD COMP) 112 ANALOG 112 DIGITAL 0.5 SECOND CYCLE ON LINE STORAGE ABOUT 5 MINUTES	SAME	INCREASE ON LINE STORAGE & RETRIEVAL TIME TO 24 HOURS	EXPANDED, (REQUIRES ADDITIONAL HARDWARE)

2Y2-4

MOD COMP COMPUTER SYSTEM DEVELOPMENT



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RESTART STATUS SUMMARY

90 Individual Item Addressed

65 Conceptually Resolved

- 28 Resolved Outright
- 25 Training Or Procedure Review Required
- 5 Test Performance Required
- 1 Technical Specification Required
- 6 Detailed Design Review Required

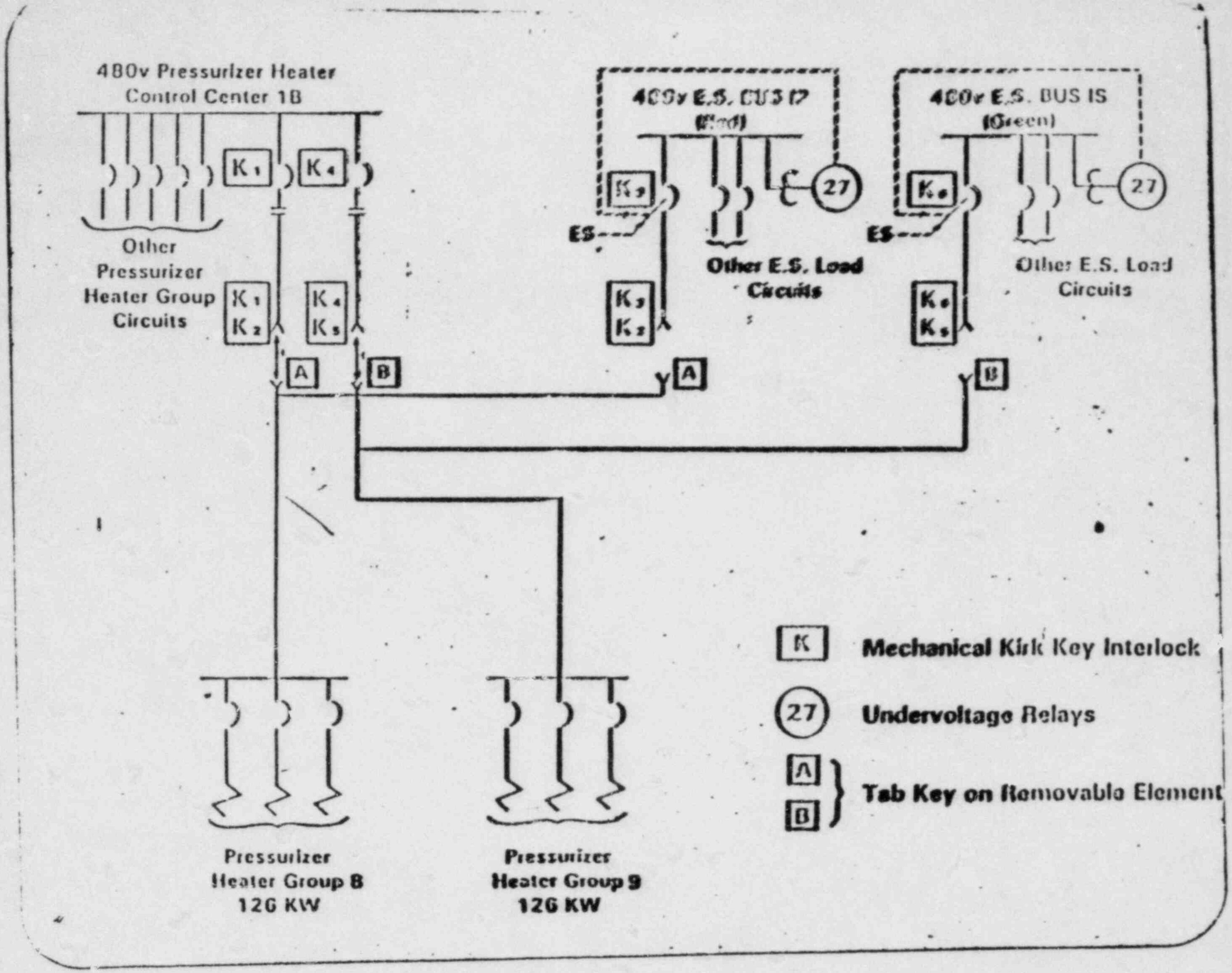
25 Open

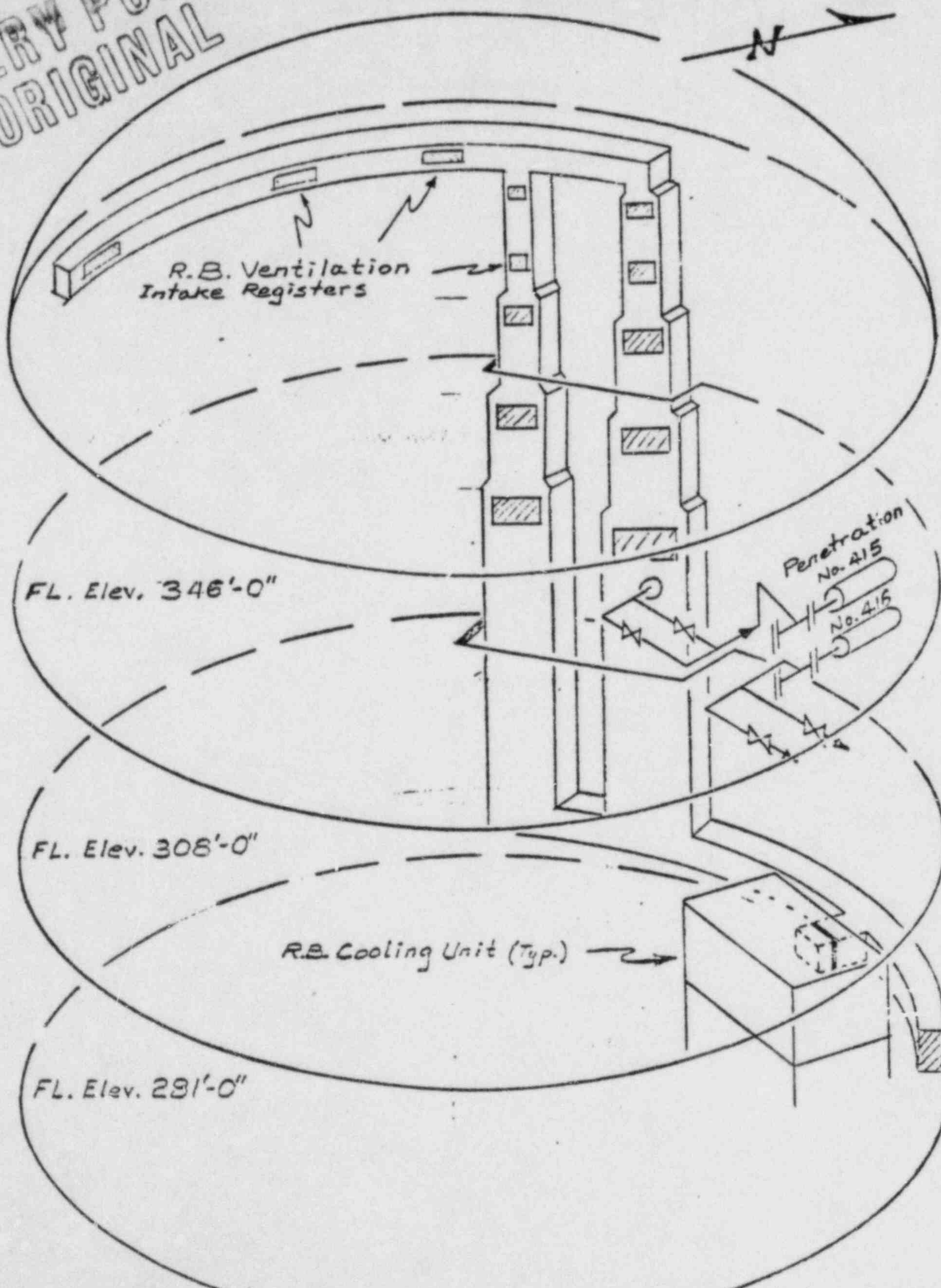
- 15 Potentially Resolved/Resolvable By Planned Submittals
- 10 Open And Unresolved

UNRESOLVED OPEN ITEMS

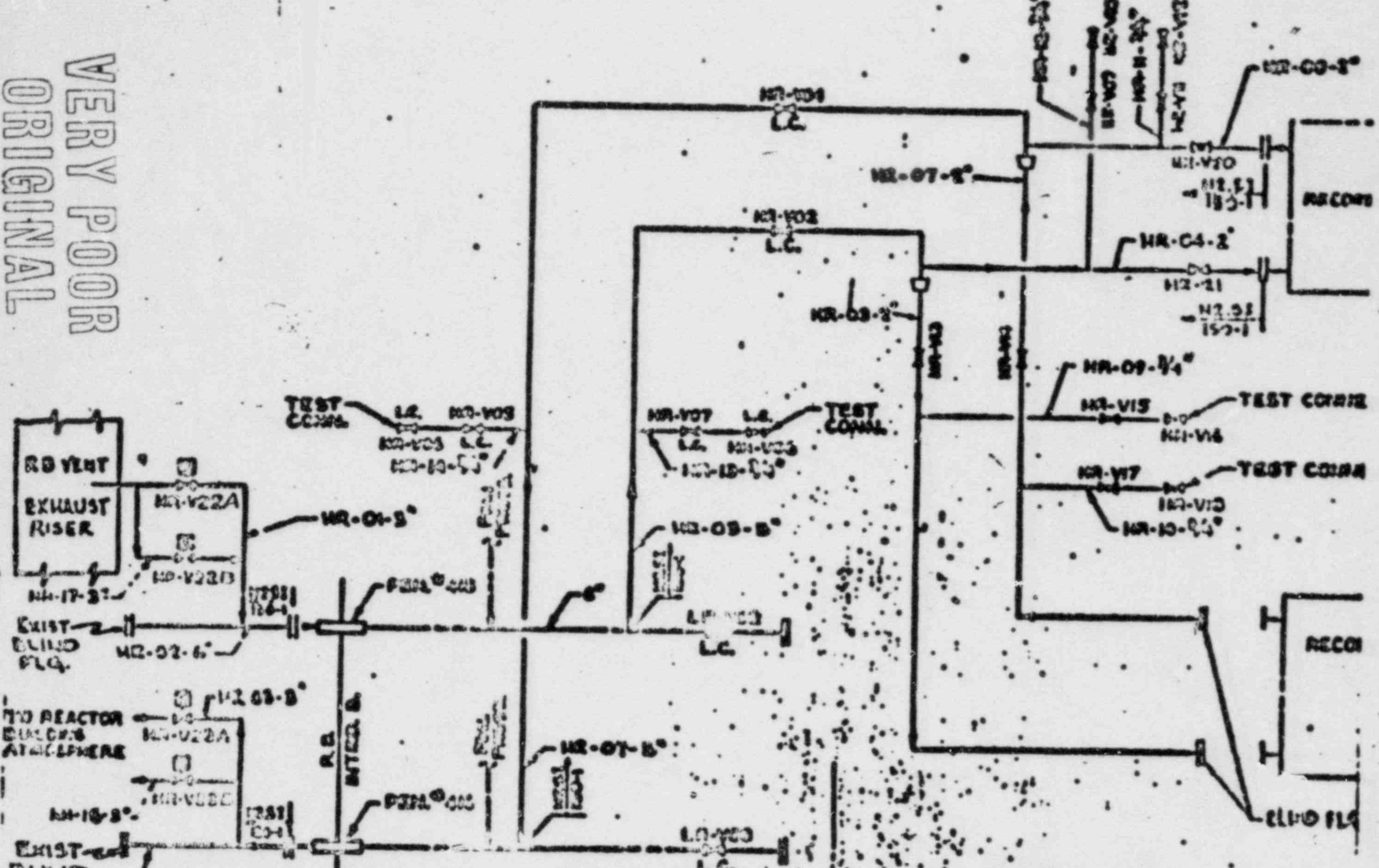
	Ref. Pg.
PORV Automatic Isolation	C1-13
EFW Discharge Line Rupture	C1-10
Reactor Vessel Water Level	C8-20
Containment Isolation of Letdown	C8-23
Quality Grade For Auto EFW Actuation on Main Feed Pump Trip	C8-34
RCS Venting	C8-58
Reactor Coolant Post-Accident Sampling	C8-38
Automatic RCP Trip Prior To Restart	C2-19
Management Capability	C6-10
Financial Capability	C7-15

SUBJECT Pressurizer Heater A-246 Emerg. Pwr.
NO.





TMI-1 Reactor Building



TMI-1
HYDROGEN RECOMBINER
SYSTEM DIAGRAM

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Appendix X

Status Report on the Mark I Long Term Containment Program

PURPOSE:

The NRC Staff has asked that the ACRS review the NRC Acceptance Criteria for the Mark I Containment Long Term Program. It is expected that the Committee would provide its comments in a letter to the Commission.

The Mark I Acceptance Criteria, issued in October, 1979 identify and define the loads to be applied to the Mark I containment system. Use of these generic loads in Plant Unique Analysis will then be reviewed by the NRC on a case-by-case basis. The loads under consideration include the dynamic forces associated with LOCAs and Safety Relief Valve (SRV) discharges to the Mark I torus. Figure 1 identifies the individual loads associated with a LOCA event. The SRV load sequence is similar, but generally localized, which eliminates some of the bulk pool response phenomena.

HISTORY:

I. Problem Identification

During large scale testing of the Mark III containment system in 1972-1974, new supression pool hydrodynamic loads were identified. Previous tests of the Mark I containment system at Bodega Bay and Humboldt Bay during 1958-1962 had not identified these loads due to inadequate instrumentation (pressure response too slow and insufficient analysis to understand all of the phenomena involved in the complex fluid structure interactions). In February and April of 1975 the NRC sent letters to all of the Mark I Owners requesting that they review their plant designs to determine whether the newly identified loads would affect the structural adequacy of their containments. As a result of the letters and the identified problem the Mark I Owners Group was formed to coordinate the resolution of the problem with the NRC Staff.

II. Short Term Program (1975-1977)

The Short Term Program was initiated to verify the integrity and functional capability of the Mark I containment when subjected to the most probable LOCA loads. The immediate effort was directed at verifying that the licensed Mark I's could operate safely while a long term program was conducted. At this point in time a safety factor of two was considered acceptable.

In December 1977, NUREG-0408 was issued in which the NRC Staff conclusions were reported. The Staff also granted exemptions, to operating Mark I facilities, relating to the structural factor of safety requirements contained in 10 CFR 50.55(e). The exemptions were granted for a period of approximately two years. In a number of plants modifications were made at this time. These generally involved tie downs to prevent uplift of the torus during a LOCA.

III. Long Term Program

The Long Term Program was intended to establish conservative design basis loads that would be appropriate for the anticipated 40 year life of each Mark I BWR and would restore the original intended design safety margins.

The Long Term Program includes an extensive program of tests at 1/12, 1/4, 1/5, and full scale in two and three dimensional geometries. Tests have been conducted by GE, EPRI, and NRC as well as several foreign countries. The Mark I Owners submitted the "Mark I Containment Program Load Definition Report" in December 1978 and the "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide" in July 1979. Additional supporting documents have also been submitted. The NRC Staff reviewed the Owners submittals as well as the information available from NRC research programs and foreign programs and issued the "NRC Mark I Acceptance Criteria" in October 1979. The Acceptance Criteria contain several modifications and additions to the Owners proposals in order to provide adequate conservations in the load definitions. The NRC bases for the Acceptance Criteria are contained in a Safety Evaluation Report which is available in draft form.

IV. ACRS Review

ACRS Fluid Dynamics Subcommittee meetings have been held periodically (May 23, 1978, Nov 28-30, 1978, Sep13-14, 1979 and Nov 16, 1979) to discuss the information available on the dynamic loads in the Mark I containment system.

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At the time of the last Subcommittee meeting several areas remained in disagreement between the Owners and the NRC Staff. These included an NRC required 21.5% margin when calculating torus uplift loads from the 2D experimental data, an NRC requirement to use fluid drag coefficients for the vent header deflectors that are a factor of 3 greater than the Owners proposed values, and an NRC requirement to run at least two additional large mass flux break tests in the GE Full Scale Test Facility (FSTF).

These problem areas have been resolved to the extent that the Owners have agreed to do the additional tests, the fluid velocities which correspond to the drag coefficients are being recalculated in a realistic manner (near field velocities rather than free field velocities will be used), and the margin on the torus uplift forces appears to have been accepted by the Owners.

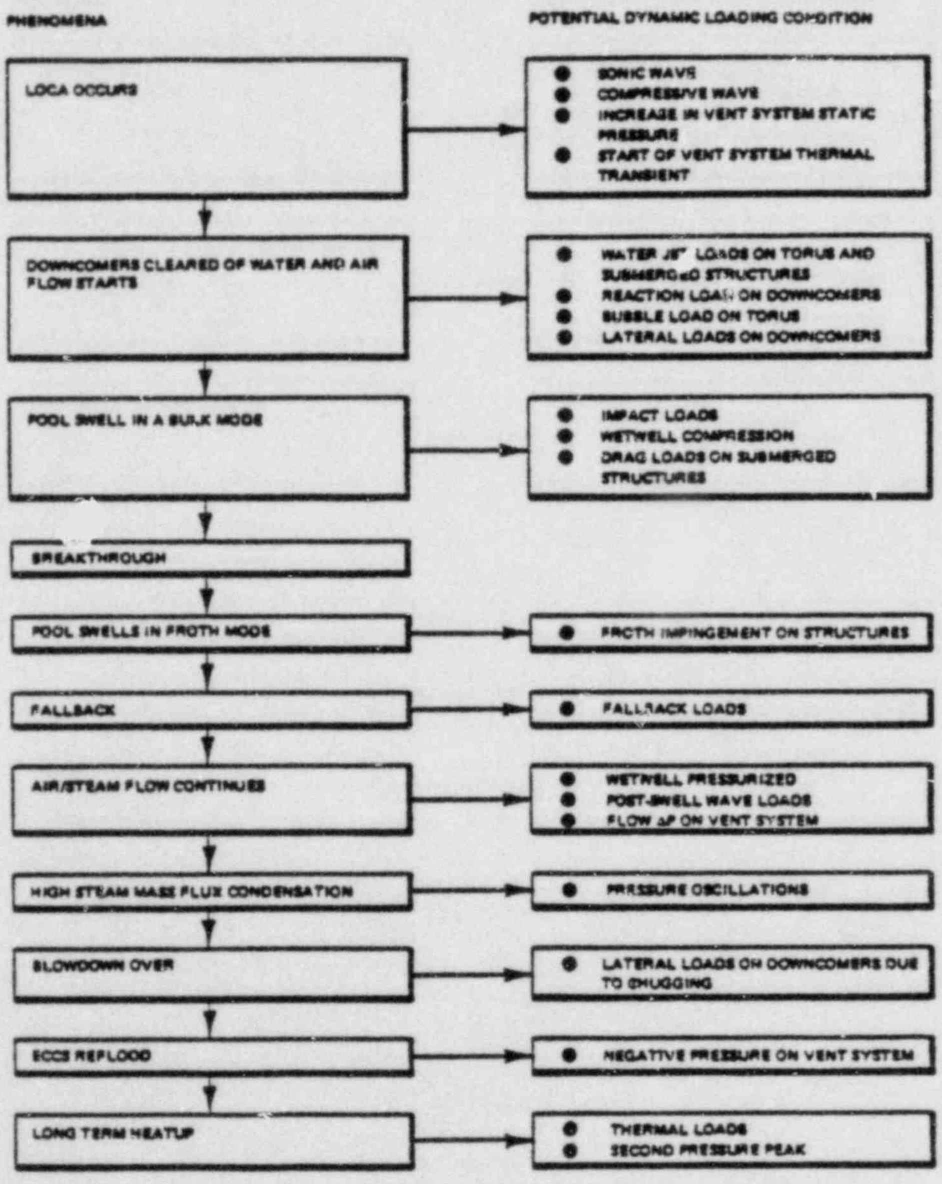
Subsequent to the November 16, 1979 ACRS Subcommittee Meeting a new problem arose. The Mark I Owners determined that several problems existed with regard to the calculated plant structural responses when the acceptable SRV produced loads were applied. It had been thought that the use of quencher devices on the SRV discharge lines would reduce the loads (as confirmed by tests at Monticello) to a level of no serious consequence. However; in a meeting with the NRC Staff on December 20, 1979 the Owners indicated that when the applied loads (from the acceptance criteria) were used in plant specific structural models the calculated plant responses were larger by a factor of 10 to 20 over measured responses to tests. The Owners have indicated that the problem is not so much in the load definition itself, but is in the conservative assumptions used in modeling the plant structural responses given an applied load. (See attached letter dated January 7, 1980, L. J. Sobon to D. Eisenhut).

The NRC Staff has been making a number of minor modifications to the loads defined in the Acceptance Criteria which will help to alleviate the Owners' difficulties. These modifications generally include additions to the present Acceptance Criteria which will allow individual utilities the option of using in-plant test data to verify acceptable plant structural responses. The procedure will allow a reduction to the analytically determined plant response based upon experimental responses at Monticello and which would then be verified at each plant utilizing the procedure. The Staff will also add a specification that will separate the local SRV loads from the global loads; this will reduce the calculated loads on the torus support columns.

The Staff expects to have these additions to the Acceptance Criteria completed and approved at the time of their presentation to the Full Committee and will discuss the additions during the meeting.

Figure

Sequence of Events and Potential Loading Conditions Following a Postulated LOCA --



A-253

GENERAL ELECTRIC

NUCLEAR ENER

PROJECTS DIVI

[GENER]AL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
Mail Code 905, Telephone (408) 925-3495

MFN-004-80

VERY POOR ORIGINAL

January 7, 1980

U. S. Nuclear-Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. D. G. Eisenhut, Acting Director
Division of Operating Reactors

Gentlemen:

SUBJECT: MARK I CONTAINMENT PROGRAM
IMPLEMENTATION OF THE MARK I LONG-TERM PROGRAM

Reference: Letter, D. G. Eisenhut to all Mark I Utilities, dated October 31, 1979, "Acceptance Criteria for the Mark I Containment Long-Term Program"

The NRC staff review of the Mark I Containment Program Load Definition Report (LDR) has resulted in NRC acceptance criteria for implementation of the Mark I Long-Term Program (LTP). These criteria were transmitted to all Mark I Utilities via the reference letter. During a December 20, 1979 meeting with NRC staff management, the Mark I Owners explained that initial structural analyses using the load definitions in the Mark I LDR in accordance with the NRC acceptance criteria are resulting in some unrealistic calculated structural responses. Current plant unique structural analysis techniques consist of using idealized predictions of hydrodynamic loads as input to conservative analytical models which then predict structural response. This method results in structural response predictions much greater than responses measured in full scale testing.

Certain of these analytically derived structural responses are of such a nature that the feasibility of practical structural modifications is questionable. Therefore, the Mark I Owners have approved continuing Mark I Program efforts to address this issue.

In response to a verbal NRC staff request made subsequent to the December 20, 1979 meeting, this correspondence is provided on behalf of the Mark I Owners Group to further describe the activities underway to develop more realistic load definition, load application and structural analysis techniques. The objective of these activities is to provide a basis for early decisions regarding plant modifications which conform to the NRC acceptance criteria for LTP implementation.

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Most of these current Mark I Program activities relate to the load application and structural analysis techniques. Therefore these efforts may not require revision to the LDR or the NRC acceptance criteria. These current activities involve three primary areas. The following is a description of these three main areas of emphasis and the planned activities in each area.

1. <u>Safety/Relief Valve (SRV) Shell Stresses</u>

   The typical SRV load definition wave form presented in the LDR is an idealized pressure load which does not account for the actual pressure decay or frequency variation with time that was observed in in-plant SRV tests. Activities underway are aimed at providing a wave form with pressure decay and frequency variation closer to the in-plant observations. Empirical factors to reduce current calculated structural responses to levels in closer agreement with actual in-plant test structural responses are also under development. These reduction factors would be based on comparing actual measured test structural responses to test structural responses calculated using current analytical techniques. Plant unique in-plant tests are being considered, if necessary, to confirm these empirical reduction factors derived on a generic basis from Monticello test data.

2. <u>SRV Column Loads</u>

   This SRV load definition in the LDR is based on bounding the peak torus pressures observed at in-plant tests. The method used is appropriate for peak local pressure determination but results in an overly conservative method for calculating the total load applied to the entire torus bay and thus the support columns. Current activities in this area are directed at determining a revised bounding factor to be used in evaluating the total load applied to the torus bay and support columns. This new factor would be based on a comparison of test results and analytical predictions as in item 1 above. The current bounding factor will be retained for use in evaluating local shell stresses. The additional in-plant tests mentioned in item 1 above are also being considered for confirmation of the revised torus support columns load application.

3. <u>Condensation Oscillation (CO) Load for the Design Basis Accident (DBA)</u>

   The LDR CO load definition for the DBA is comprised of varying pressure amplitudes over the 0-50 Hz range. The work underway in this area has demonstrated that most of the load frequencies are randomly phased. This approach will provide justification for taking credit for the random time phasing of most of the loading frequencies observed in the Full Scale Test Facility testing.

VERY POOR ORIGINAL

Some of the initial efforts involved in the above activities are expected to be completed in the next several weeks. Representatives of the Mark I Owners Group will be available to meet with the NRC staff in early February 1980 to discuss the status and details of the work in progress. Final completion of the engineering aspects of the above work is tentatively scheduled for April 1980.

Preliminary evaluations indicate that several items, in addition to the three identified above, may also require similar Mark I Program efforts. Evaluation of submerged structures is currently underway. Specific additional submerged structures activities, if any, are to be identified in the near future and completed by about June 1980. Torus attached piping cannot be evaluated until dynamic analysis of the torus is completed. However, scoping evaluations have indicated that additional activities may be forthcoming in this area. For some plants, additional efforts to provide a more realistic response to DBA CO and chugging may also be required.

Full scale CO and in-plant SRV tests resulted in typical maximum measured free shell stresses of 4Ksi or less. Even increasing these test results to account for suitable design loading conditions still provides a large margin of safety when applying the Short-Term Program (STP) criteria. Measured test responses other than free shell stresses show similar margins. Such full scale test results verify the conclusions of the STP and are the basis for the continuing Mark I Program activities described above.

L. Sobon

L. J. Sobon, Manager
BWR Containment Licensing
Containment Improvement Programs

LJS/d

cc: C. I. Grimes (NRC)
G. C. Lainas (NRC)
L. S. Gifford (GE-Bethesda)

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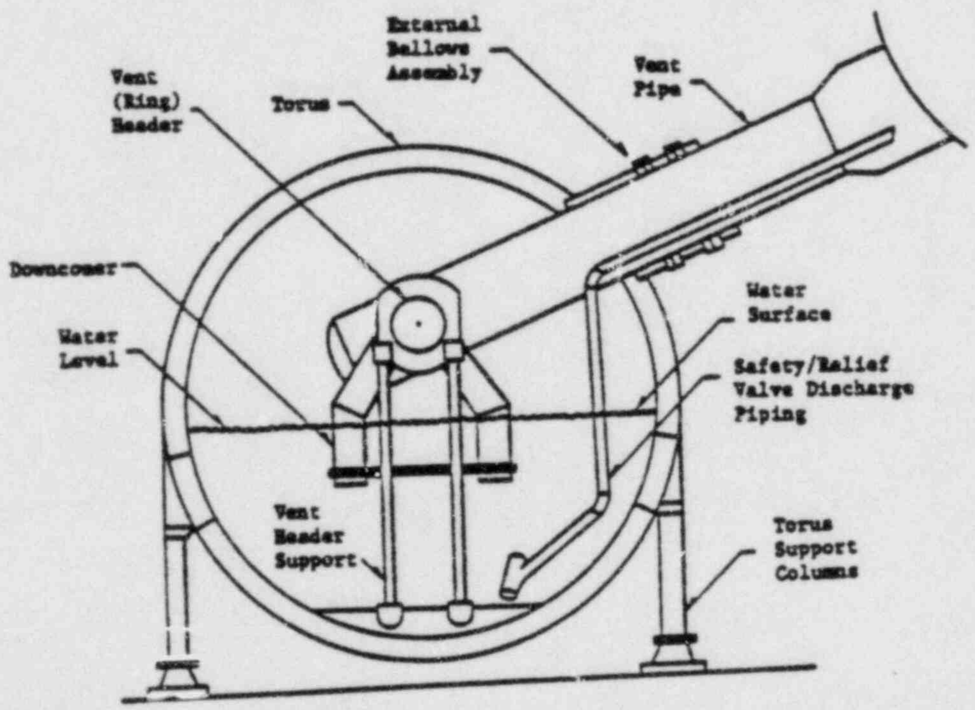
Figure II-2

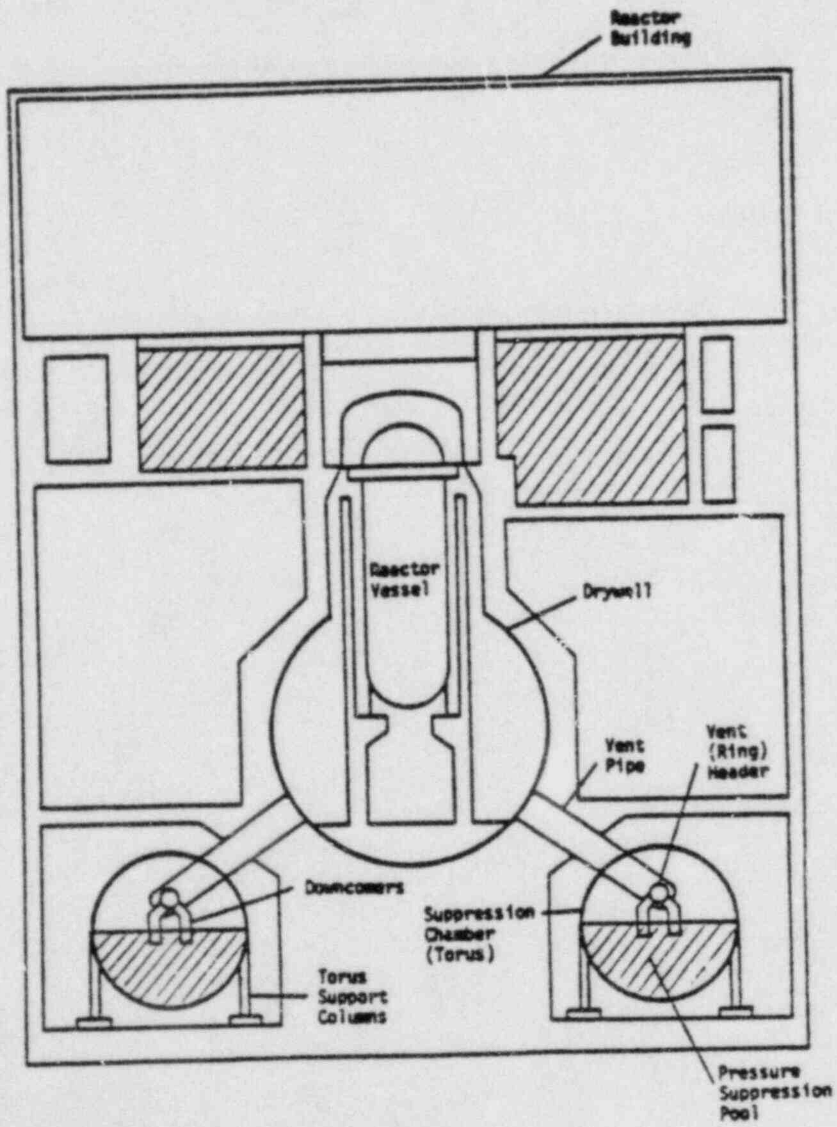
Composite Section Through Suppression Chamber



VERY POOR ORIGINAL

Figure II-?

Mark I Containment System



VERY POOR ORIGINAL

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EVENT COMBINATIONS		SRV	SRV + EQ		SBA IBA		SBA + EQ IBA + EQ				SBA + SRV IBA + SRV		SBA + SRV + EQ IBA + SRV + EQ				DBA		DBA + EQ				DBA + SRV		DBA + EQ + SRV			
						CO, CH			CO,CH			CO, CH			CO,CH		PS (1)	CO, CH	PS		CO,CH		PS	CO, CH	PS		CO,CH	
TYPE OF EARTHQUAKE			O	S			O	S	O	S			O	S	O	S			O	S	O	S			O	S	O	S
COMBINATION NUMBER		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
LOADS Normal (2)	N	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Earthquake	EQ		X	X			X	X	X	X			X	X	X	X			X	X	X	X			X	X	X	X
SRV Discharge	SRV	X	X	X							X	X	X	X	X	X							X	X	X	X	X	X
LOCA Thermal	T_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Reactions	R_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Quasi-Static Pressure	P_A				X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
LOCA Pool Swell	P_{PS}																X		X	X			X		X	X		
LOCA Condensation Oscillation	P_{CO}					X			X	X		X			X	X		X			X	X		X			X	X
LOCA Chugging	P_{CH}					X			X	X		X			X	X		X			X	X		X			X	X
STRUCTURAL ELEMENT	ROW																											
External Class MC Torus, External Vent Pipe, Bellows, Drywell (at Vent), Attachment Welds, Torus Supports, Seismic Restraints	1	A	B	C	A	A	B	C	B	C	A	A	B	C	B	C	A (3,6)	A	B (2,6)	C	B	C	O	O	O	C	O	C
Internal Vent Pipe General and Attachment Welds	2	A	B	C	A	A	B	C	B	C	A	A	B	C	B	C	A (3,5)	A	B (3,5)	C	B	C	O	O	O	C	O	C
At Penetrations (e.g., Header)	3	A	B	C	A	A	B	C	B	C	A	A	B	C	B	C	A (3)	A	B (3)	C	B	C	O	O	O	C	O	C
Vent Header General and Attachment Welds	4	A	B	C	A	A	B	C	B	C	A	A	B	C	B	C	A (3,6)	A	B (3,5)	C	B	C	O	O	O	C	O	C
At Penetrations (e.g., Downcomers)	5	A	B	C	A	A (4)	[illegible]	C	B (4)	C	A	A (4)	B	C	B (4)	C	A (3, 4,5)	A (4)	B (3, 4,5)	C	B (4)	C	O	O	O	C	O	C
Downcomers General and Attachment Welds	6	A	B	C	A	A	B	C	B	C	A	A	B	C	O	C	A (3,5)	A	B (3,5)	C	B	C	O	O	O	C	O	C
Internal Supports	7	A	B	C	A	A	B	C	B	C	A	A	B	C	B	C	A	A	B	C	B	C	O	O	O	C	O	C
Internal Structures General	8	A	B	C	A	A	C	D	C	D	C	C	D	E	D	E	E	E	E	E	E	E	E	E	E	E	E	E
[illegible]	9	A	B	C	A	A	C	D	C	B	C	C	D	D	D	D	D	D	D	D	D	D	D	D	D	D	D	D

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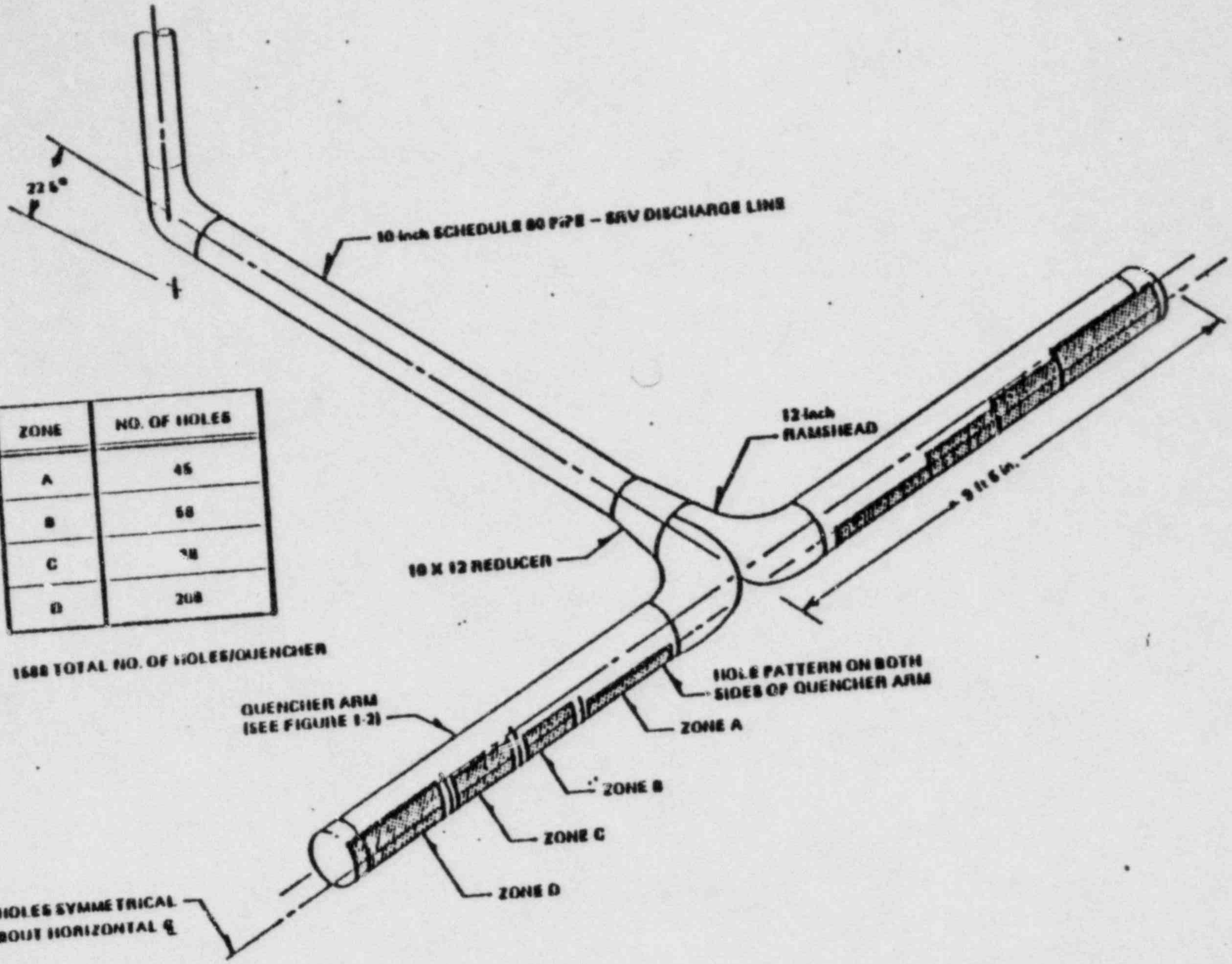
ATTACHMENT N

DETAILS OF MONTIC. T-QUENCHER



ZONE	NO. OF HOLES
A	45
B	58
C	[illegible]
D	208

1588 TOTAL NO. OF HOLES/QUENCHER

M.8

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9/14/79

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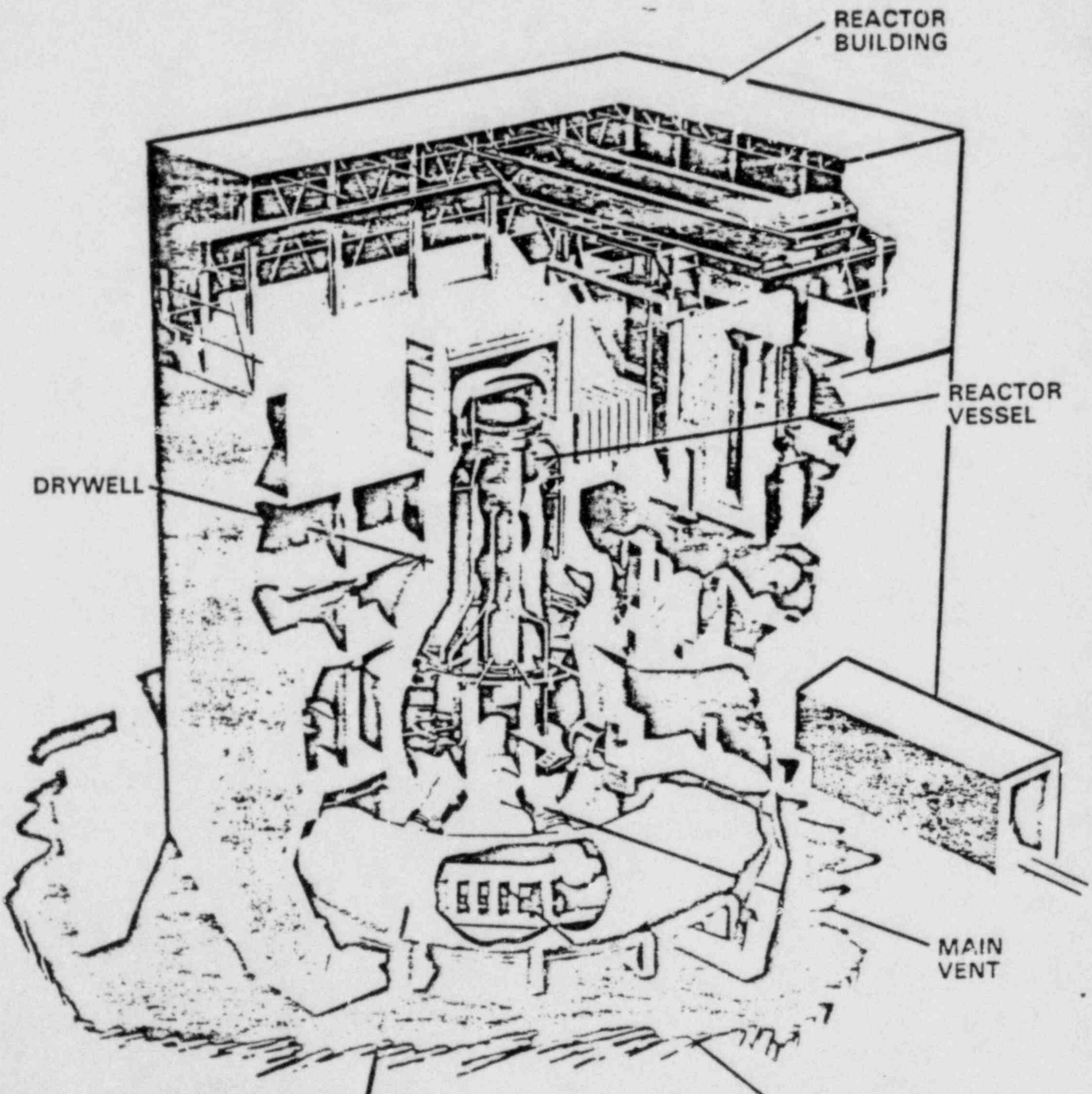
Appendix XI

LISTING OF DOMESTIC BWR FACILITIES WITH THE MARK I CONTAINMENT SYSTEM

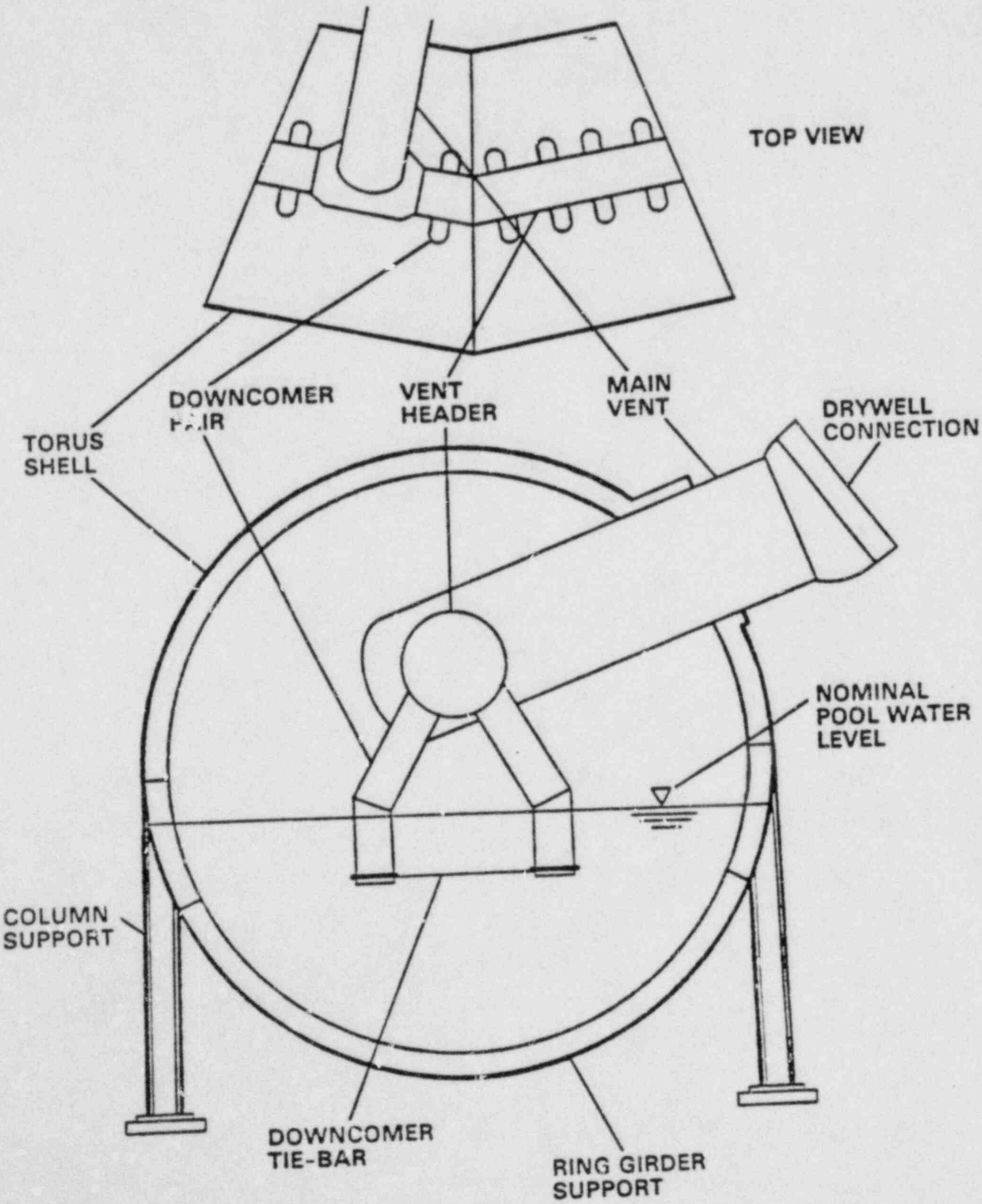
Plants Licensed for Power Operation	Licensee
Browns Ferry Unit Nos. 1, 2, and 3	Tennessee Valley Authority
Brunswick Unit Nos. 1 and 2	Carolina Power & Light
Cooper Station	Nebraska Public Power District
Dresden Unit Nos. 2 and 3	Commonwealth Edison Company
Duane Arnold	Iowa Electric Light & Power
FitzPatrick	Power Authority State of New York
Hatch Unit Nos. 1 and 2	Georgia Power Company
Millstone Unit No. 1	Northeast Nuclear Energy Company
Monticello	Northern States Power Company
Nine Mile Point Unit No. 1	Niagara Mohawk Power Corporation
Oyster Creek	Jersey Central Power & Light
Peach Bottom Unit Nos. 2 and 3	Philadelphia Electric Company
Pilgrim Unit No. 1	Boston Edison Company
Quad Cities Unit Nos. 1 and 2	Commonwealth Edison Company
Vermont Yankee	Yankee Atomic Electric Company

Plants Under Construction	Applicant
Fermi Unit No. 2	Detroit Edison Company
Hope Creek Unit Nos. 1 and 2	Public Service Electric & Gas

MARK I CONTAINMENT SYSTEM



MARK I SUPPRESSION CHAMBER



MARK I CHRONOLOGY

1958 - 1962	Humboldt Bay & Bodega Bay
1972 - 1974	Mark III PSTF Testing
February 1975 & April 1975	Staff Requests Assessment of Effects of Pool Swell & SRV
May 1975	Mark I Owners Group Proposes Short Term & Long Term Programs
January 1976	Vermont Yankee Preliminary Results Predict Torus Uplift
July 1976 to September 1976	STP Plant-Unique Analyses
December 1977	Staff Issues NUREG 0408
February 1978	Mark I Plants Issued Exemptions from GDC 50 (Containment Design Basis)

MARK I CHRONOLOGY

October 1976	Mark I Owners Submit LTP "Program Action Plan"
December 1978	Mark I Owners Submit Load Definition Report - Part A
March 1979	Mark I Owners Submit Load Definition Report - Part B
August 1979	Draft NRC Criteria Issued for Comments
September 1979	ACRS Subcommittee Meeting
October 1979	NRC Criteria Issued
November 1979	ACRS Subcommittee Meeting
December 1979	Draft SER Issued for Staff Comments

Appendix XII

MARK I CONTAINMENT PROGRAM - ORGANIZATION

- 16 UTILITY COMPANIES
- 25 MARK I PLANTS
- PROJECT-TYPE ORGANIZATIONS
  - OWNER COMMITTEES
    - GE PROGRAM OFFICE
    - GE TECHNICAL
    - SUBCONTRACTORS
    - ARCHITECT-ENGINEERS

Figure II-3. Mark I Owners Group Organization - SSE Phase

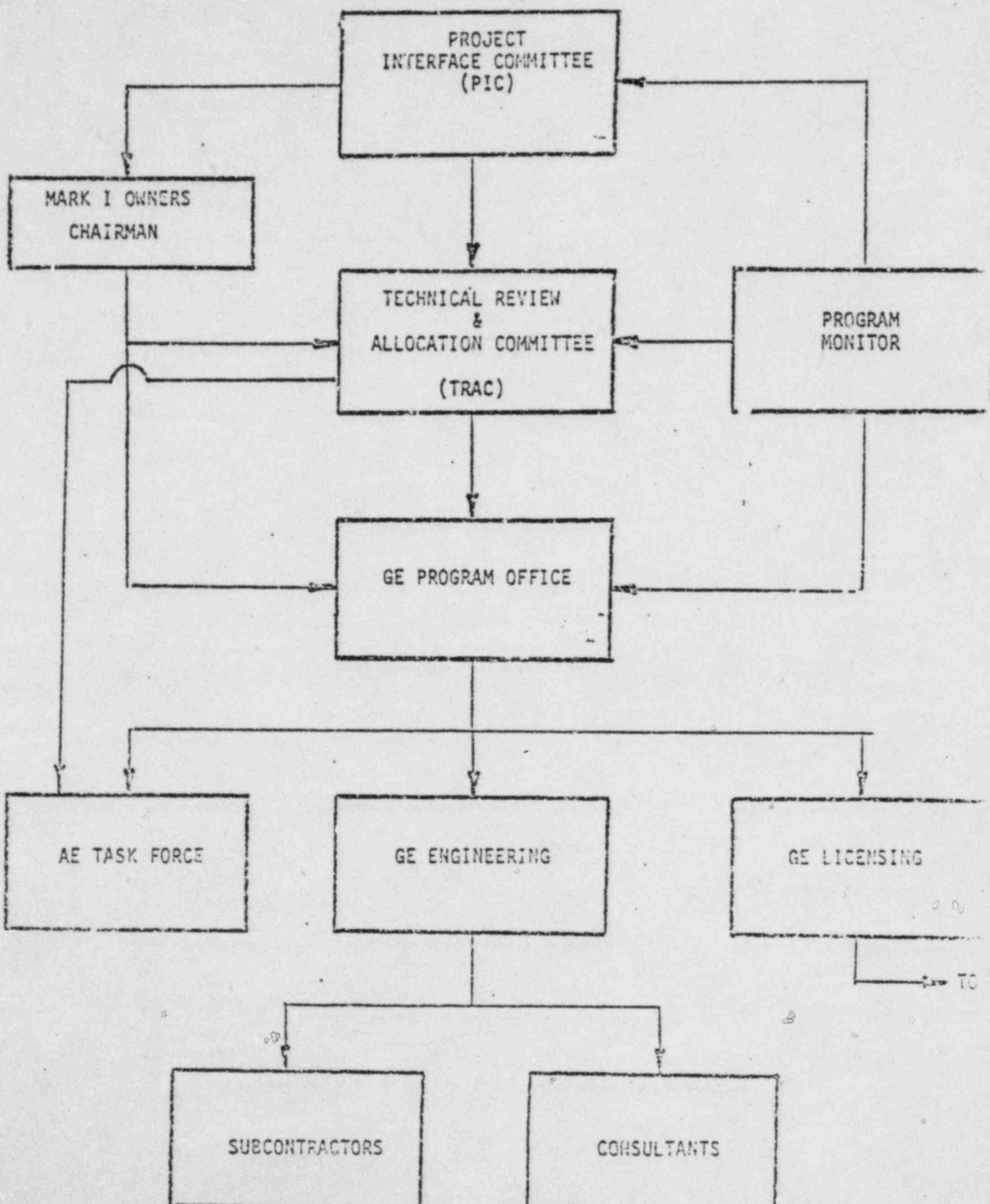


TABLE II-1

MARK I UTILITIES AND PLANTS

UTILITY NAME	PLANT NAME
Boston Edison Company Boston, Massachusetts	Pilgrim
Carolina Power & Light Company Raleigh, North Carolina	Brunswick 2,3
Commonwealth Edison Company Chicago, Illinois	Dresden 2,3 Quad Cities 1,2
Detroit Edison Company Detroit, Michigan	Fermi 2
Georgia Power Company Atlanta, Georgia	Hatch 1,2
Iowa Electric Light & Power Company Cedar Rapids, Iowa	Duane Arnold
Jersey Central Power & Light Company Morristown, New Jersey	Oyster Creek
Nebraska Public Power District Columbus, Nebraska	Cooper
Niagara Mohawk Power Corporation Syracuse, New York	Nine Mile Point
Northeast Utilities Service Company Berlin, Connecticut	Millstone
Northern States Power Company Minneapolis, Minnesota	Monticello
Philadelphia Electric Company Philadelphia, Pennsylvania	Peach Bottom 2,3
Power Authority of the State of New York New York, New York	Fitzpatrick
Public Service Electric and Gas Newark, New Jersey	Hope Creek
Tennessee Valley Authority Knoxville, Tennessee	Browns Ferry 1,2,3
Yankee Atomic Electric Company Westboro, Massachusetts	Vermont Yankee

MARK I CONTAINMENT PROGRAM

PROGRAM MILESTONES

• HIGH CONTAINMENT LOADS IDENTIFIED BY MARK III TESTING	1974-1975
• NRC REQUIRED MARK I RE-EVALUATION - OWNERS GROUP FORMED - SHORT TERM PROGRAM INITIATED - GE SELECTED AS PROGRAM MANAGER	APRIL 1975
• LONG TERM PROGRAM INITIATED	JULY 1976
• LONG TERM PROGRAM TEST WORK COMPLETED	SEPTEMBER 1978
• LOAD DEFINITION REPORT SUBMITTED TO NRC	MARCH 1979
• NRC LOAD ACCEPTANCE CRITERIA ISSUED	OCTOBER 1979
• REVISED LOAD DEFINITION REPORT TO BE SUBMITTED TO NRC	MAY 1980*
• PLANT UNIQUE ANALYSES TO BE SUBMITTED TO NRC	PLANT SPECIFIC
• PLANT MODIFICATIONS TO BE COMPLETED - SINGLE PLANT UTILITIES - THREE OR FOUR PLANT UTILITIES	PLANT SPECIFIC

———————— PROGRAM CLOSURE ———— ————

*ESTIMATED DATE

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Appendix XIII

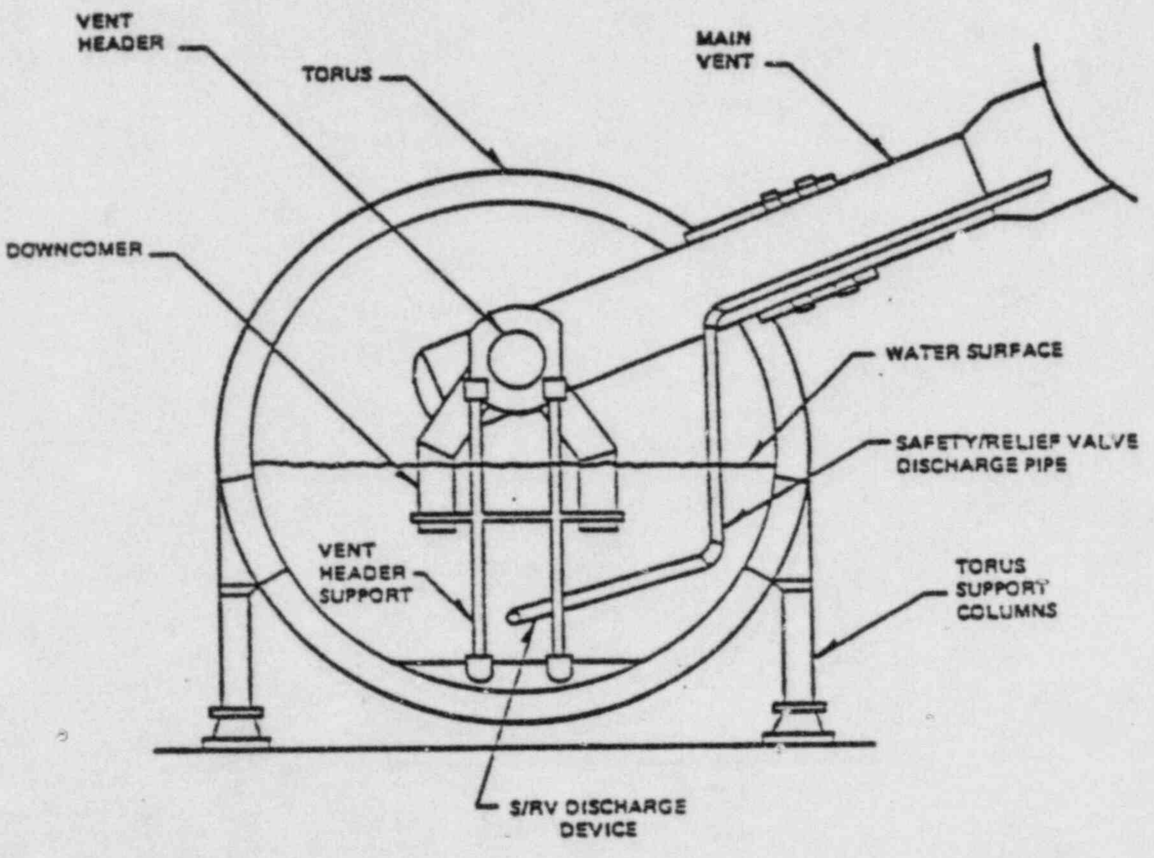

Figure 1.1-2. Typical Composite Section Through Suppression Chamber

LONG TERM PROGRAM (LTP)

- MAJOR EFFORTS

  PROGRAM ACTION PLAN

  PRELIMINARY LOAD EVALUATION

  STRUCTURAL ACCEPTANCE CRITERIA

  GENERIC STRUCTURAL EVALUATION

  LOAD EVALUATION TESTS & ANALYSIS

  LOAD MITIGATION TESTS

  LOAD DEFINITION REPORT (LDR)

LOAD DEFINITION - GENERAL

THE PROGRAM LOAD DEFINITION ACTIVITIES CAN BE CLASSIFIED AS . . .

- POOL SWELL
- CONDENSATION
- SAFETY RELIEF VALVE DISCHARGE

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MARK I CONTAINMENT PROGRAM
STATUS OF TEST PROGRAMS

Task No.	Description	Performing Agency/Facility	Scale	Phenomena Being Tested	Testing Fluid	Date for Completion of Testing	Comments
3.2.1	Column Buckling Test	TES/TES	N/A	Dynamic Load Capacity	N/A	February 1977 (Complete)	---
3.2.2	Ring Header/Vent Pipe Intersection Test	Bechtel/Anamet	N/A	Load Capacity	N/A	Indefinite	Task put on hold on April 25, 1978. Reactivation of task will depend upon identification of need.
5.1.1	Monticello S/RV Ramshead Test	GE/NSP	Full	S/RV Discharge Loads	Air/Steam	July 1976 (Complete)	---
5.1.2	Monticello S/RV Quencher Test	GE/NSP	Full	S/RV Discharge Loads	Air/Steam	December 1977 (Complete)	---
5.2	4T High Temperature Tests	GE/GE	Full	Chugging Wall & Vent Loads	Steam	July 1976 (Complete)	Mark II configuration.
5.3.2	Flexible Cylinder Tests	EPRI/DSI	1/6 & 1/3	Fluid/Structure Interaction-Vent Header	Water	July 1977 (Complete)	---
5.3.3	Flexible Cylinder Tests	GE/NSC	1/4	Fluid/Structure Interaction-Vent Header	Air/Water	November 1977 (Complete)	---
5.4	Seismic Slosh	GE/SWRI	1/30	Seismic Slosh Loads/Vent Uncovering	Water	July 1977 (Complete)	---
5.5.1	1/4-Scale 2D Test	GE/NSC	1/4	Pool Swell Scaling Laws	Air	November 1976 (Complete)	---
5.5.2	1/4-Scale 2D Test	GE/NSC	1/4	Download Oscillations	Air	October 1977 (Complete)	---
5.5.3	1/4-Scale 2D Test	GE/NSC	1/4	LDR Loads	Air	October 1978 (Complete)	Additional plant unique test series in progress; to be completed by March 1979.

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MARK I CONTAINMENT PROGRAM
STATUS OF TEST PROGRAMS

Task No.	Description	Performing Agency/Facility	Scale	Phenomena Being Tested	Testing Fluid	Date for Completion of Testing	Comments
5.6.1	1/12-Scale 3D Test	EPRI/SRI	1/12	Pool Swell Loads	Air	July 1978 (Complete)	Split orifice tests in progress; to be completed by March 1979
5.6.2	1/30-Scale 3D Test	GE/SRI	1/30	Torus/Cylinder Geometry	Air	September 1977 (Complete)	Qualitative supplement to 5.6.1.
5/8	1/12-Scale 2D Test	GE/GE	1/12	Pool Swell Scaling Laws	Air	October 1976 (Complete)	---
5.11	Full Scale 3D Test	GE/Braun	Full	Chugging	Steam	August 1978 (Complete)	---
5.13	1/12-Scale 3D Test	GE/NUTECH	1/12	Chugging	Steam	September 1977 (Complete)	Qualitative multivent effects.
5.14	Submerged Structures	GE/Wyle	1/3	Steady State & Transient Drag Loads	Air/ Steam	June 1977 (Complete)	---
		GE/NSC	1/4	Submerged Loads	Air	January 1978 (Complete)	---
		GE/SRI	N/A	Components of Drag	Water	February 1978 (Complete)	---
5.15.2	Structural/Hydro-dynamic Interactions	GE/Acurex	1/12	Fluid/Structure	Steam	April 1978 (Complete)	Flat plate only. Design level QC implemented.
5.16.1	Reduced Submergence	GE/GE Licensee	Full	Chugging	Steam	April 1977 (Complete)	Testing at Mark I submergence levels.
5.16.2	Chugging Mitigation	GE/GE Licensee	Full	Chugging	Steam	May 1977 (Complete)	Testing mitigator at Mark I submergence.
5.17	Condensation Oscillation	GE/ARAP	1/12	Condensation Oscillation	Steam	August 1978 (Complete)	Parametric testing.

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MARK I CONTAINMENT PROGRAM
STATUS OF TEST PROGRAMS

Task No.	Description	Performing Agency/Facility	Scale	Phenomena Being Tested	Testing Fluid	Date for Completion of Testing	Comments
5.6.1	1/12-Scale 3D Test	EPRI/SRI	1/12	Pool Swell Loads	Air	July 1978 (Complete)	Split orifice tests in progress; to be completed by March 1979
5.6.2	1/30-Scale 3D Test	GE/SWRI	1/30	Torus/Cylinder Geometry	Air	September 1977 (Complete)	Qualitative supplement to 5.6.1.
5.[illegible]	1/12-Scale 2D Test	GE/GE	1/12	Pool Swell Scaling Laws	Air	October 1976 (Complete)	---
5.11	Full Scale 3D Test	GE/Braun	Full	Chugging	Steam	August 1978 (Complete)	---
5.13	1/12-Scale 3D Test	GE/NUTECH	1/12	Chugging	Steam	September 1977 (Complete)	Qualitative multivent effects.
5.14	Submerged Structures	GE/Wyle	1/3	Steady State & Transient Drag Loads	Air/ Steam	June 1977 (Complete)	---
		GE/NSC	1/4	Submerged Loads	Air	January 1978 (Complete)	---
		GE/SWRI	N/A	Components of Drag	Water	February 1978 (Complete)	---
5.15.2	Structural/Hydrodynamic Interactions	GE/Aerotherm	1/12	Fluid/Structure	Steam	April 1978 (Complete)	Flat plate only. Design level QC implemented.
5.16.1	Reduced Submergence	GE/GE Licensee	Full	Chugging	Steam	April 1977 (Complete)	Testing at Mark I submergence levels.
5.16.2	Chugging Mitigation	GE/GE Licensee	Full	Chugging	Steam	May 1977 (Complete)	Testing mitigator at Mark I submergence.
5.17	Condensation Oscillation	GE/ARAP	1/12	Condensation Oscillation	Steam	August 1978 (Complete)	Parametric testing.

LOCA POOL SWELL LOADS

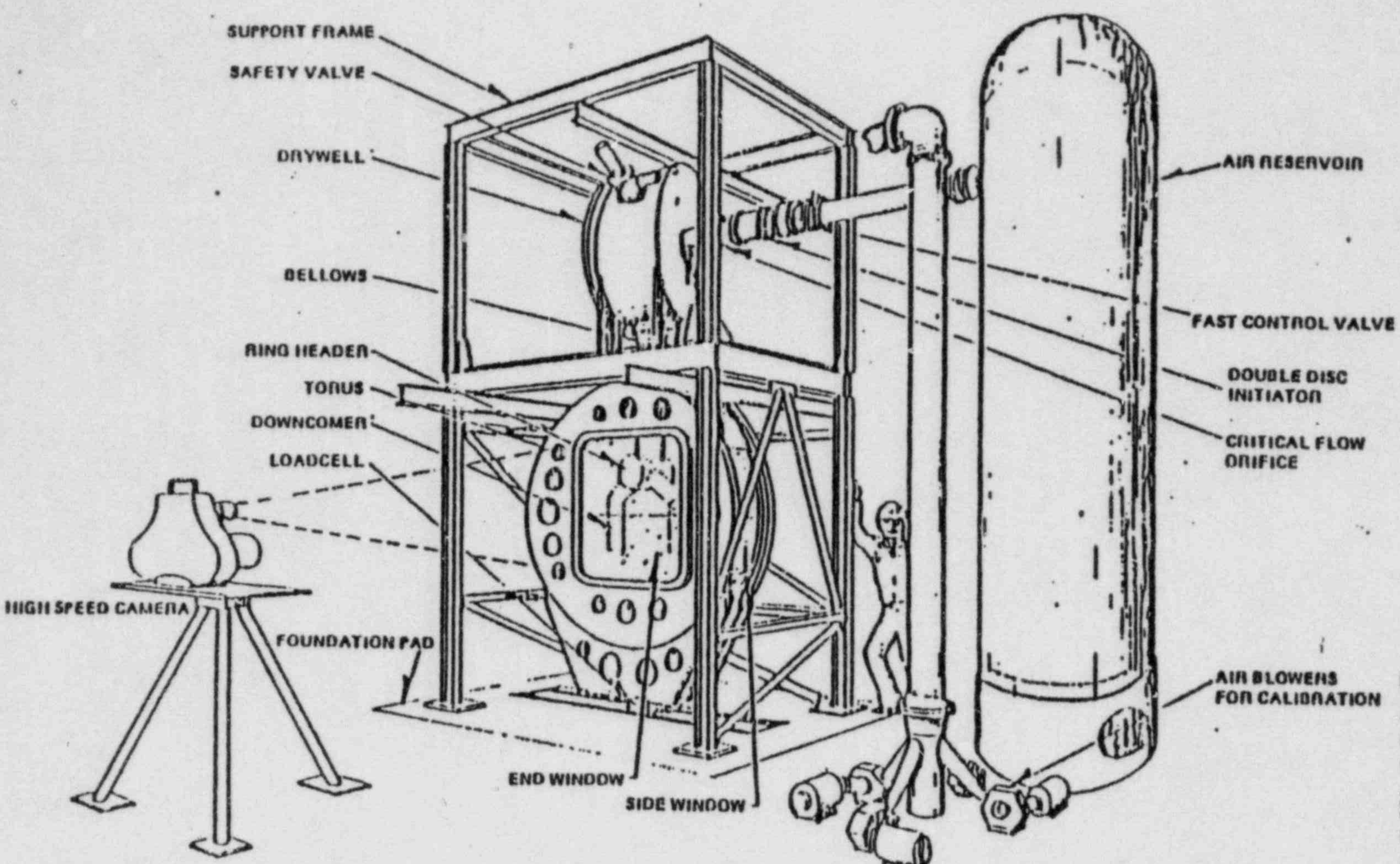


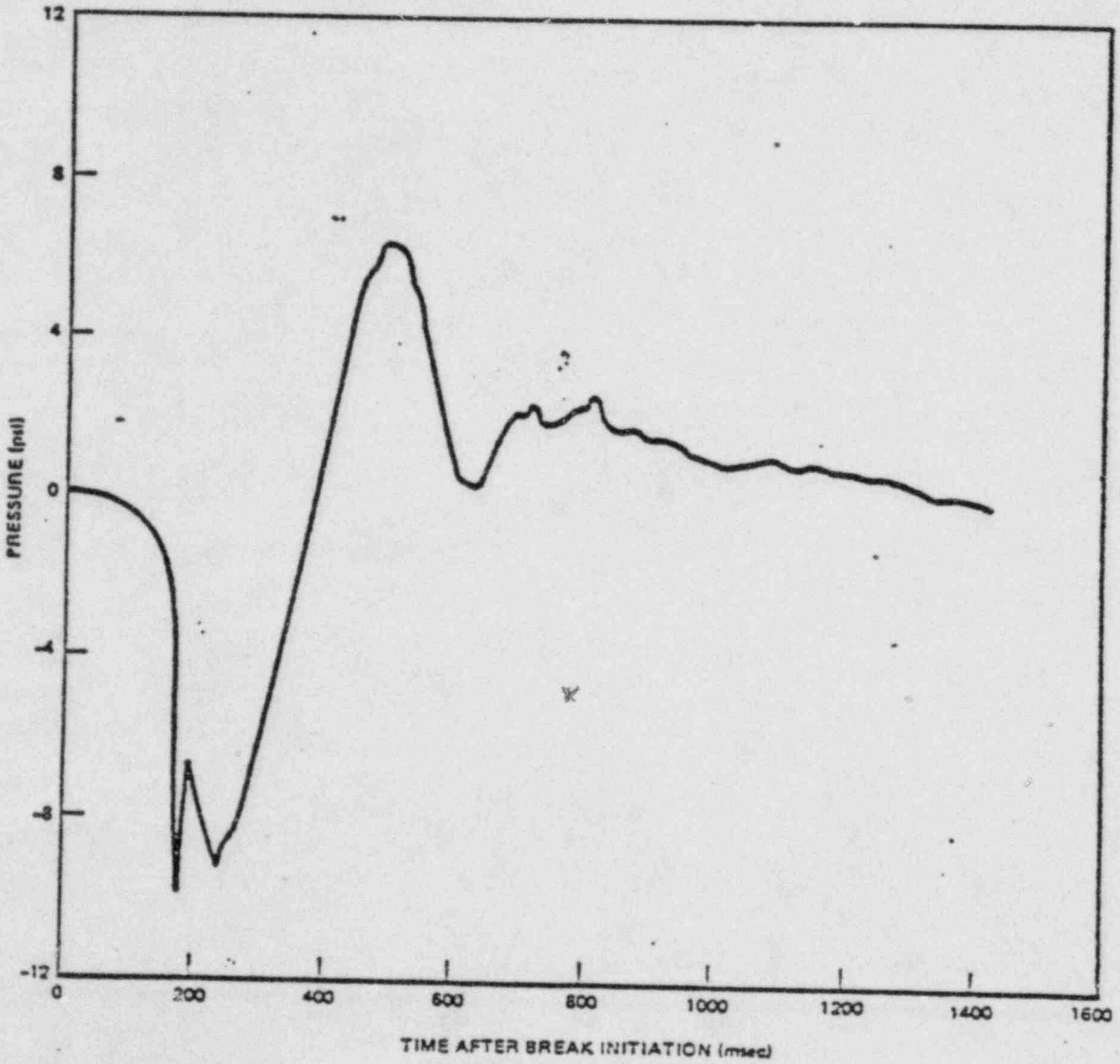
Figure 2-1. Mark I BWR Containment Pool Swell 1/4 Scale Test Facility - Perspective

POOL SWELL - DYNAMIC EFFECTS OF DRYWELL AND VENT SYSTEM AIR FORCED IN TO WETWELL

- DBA GUILLOTINE BREAK
- DRYWELL PRESSURE AND TEMPERATURE INCREASE
- DOWNCOMER WATER CLEARS; DRYWELL AIR IS EXPOSED TO WETWELL
- BUBBLE EXPANSION IN WETWELL
- POOL WATER COMPRESSES WETWELL AIR
- POOL WATER IMPACT ON VENT HEADER
- POOL BUBBLE BREAKTHROUGH

TJM-10

TYPICAL NET TORUS VERTICAL LOADING HISTORY



TJM-11

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POOL SWELL - CONCLUSIONS

- DEVELOPED HYDRODYNAMIC SCALING LAWS

- EXPERIMENTALLY VERIFIED SCALING LAWS IN 1/4 AND 1/12 SCALE TESTS

- PERFORMED EXTENSIVE FACILITY CHARACTERISTICS AND PARAMETER SENSITIVITY TESTS

- TESTED <u>ALL</u> PLANTS ON PLANT UNIQUE BASIS

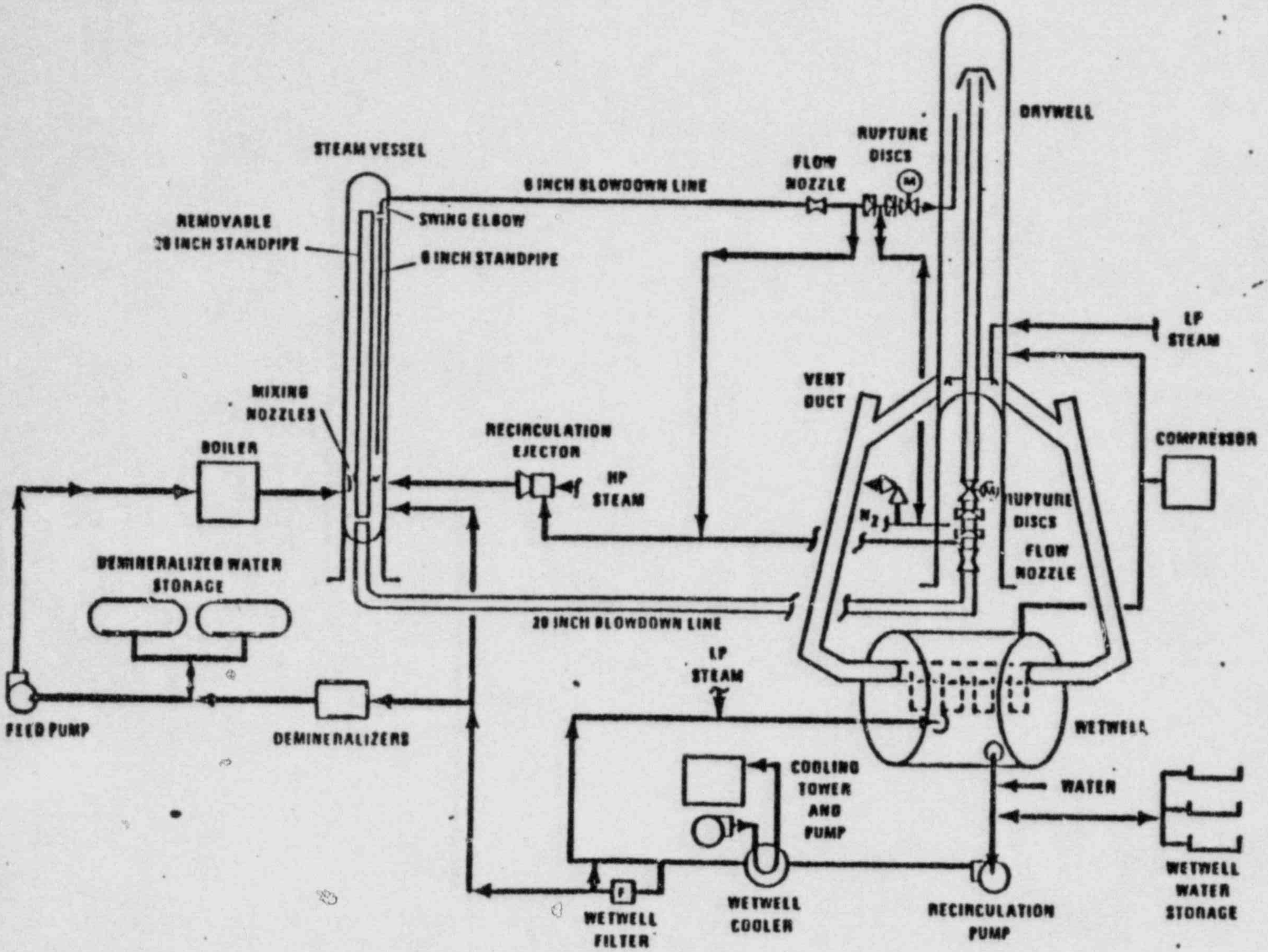
- DETERMINED LOADS FROM CONSERVATIVE 1/4 SCALE 2D PLANT UNIQUE POOL SWELL TESTS

TJM-12

LOCA CONDENSATION LOADS

FULL SCALE TEST FAC'' ITY

SIMPLIFIED FLOW DIAGRAM



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MARK I FULL-SCALE TEST FACILITY

PROGRAM OBJECTIVES & AFFROACH

OBJECTIVE:

Obtain data to define Hydrodynamic Loads and Dynamic Structural Response resulting from Steam Condensation Phenomena on a representative Topus Sector in a full Scale Test Facility.

FACILITY APPROACH

- Full-Scale 22-1/2° Sector or Wetwell (8 Downcomers)
- Scaled Drywell, Vents, Flash Boiler
- Typical Structural Response
- Hydrodynamic and Structural Instrumentation
- High Speed DAS

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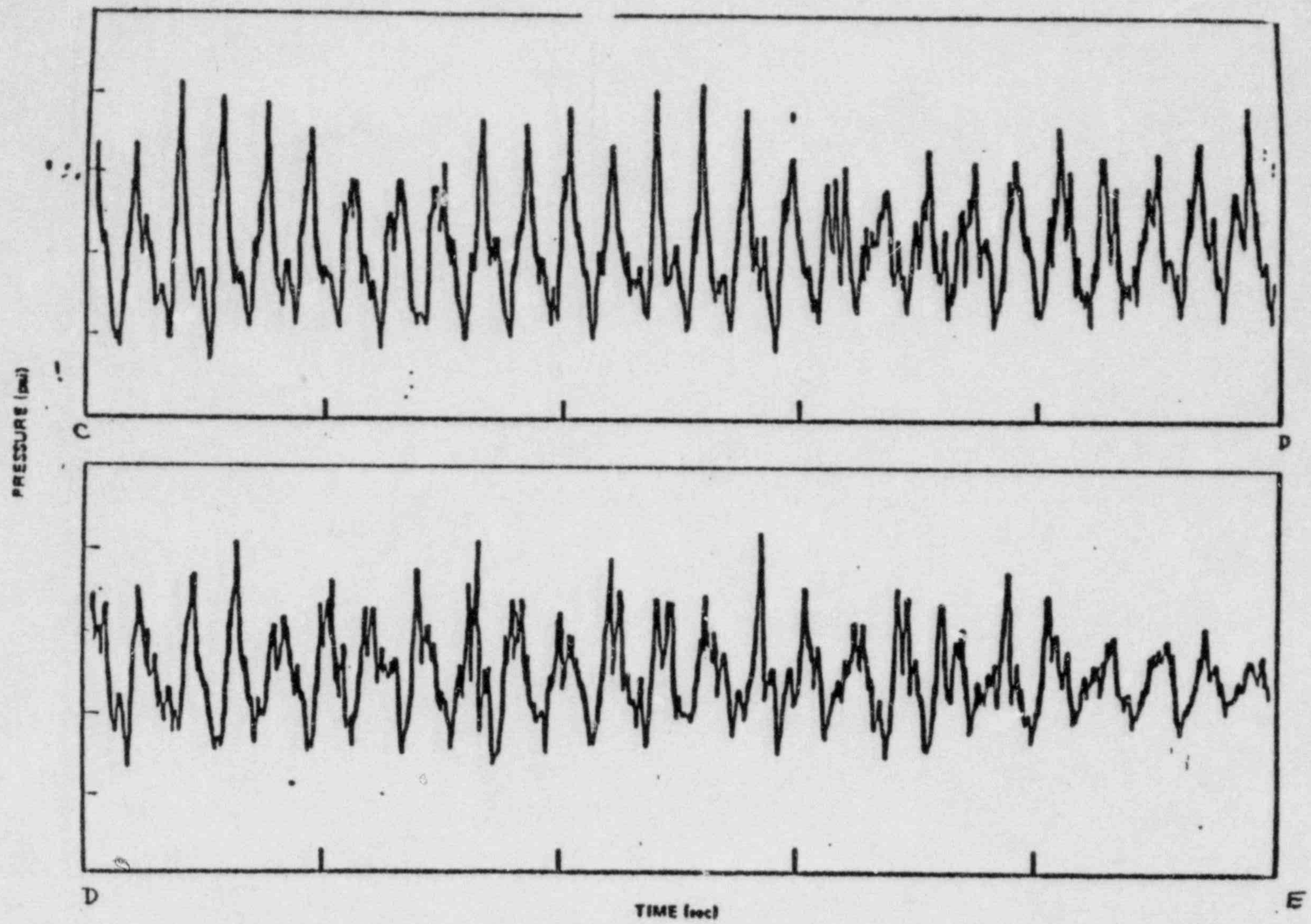


Dynamic Portion of Wetwell
Bottom Dead Center.

MARK I CONDENSATION OSCILLATION

LOAD DEFINITION

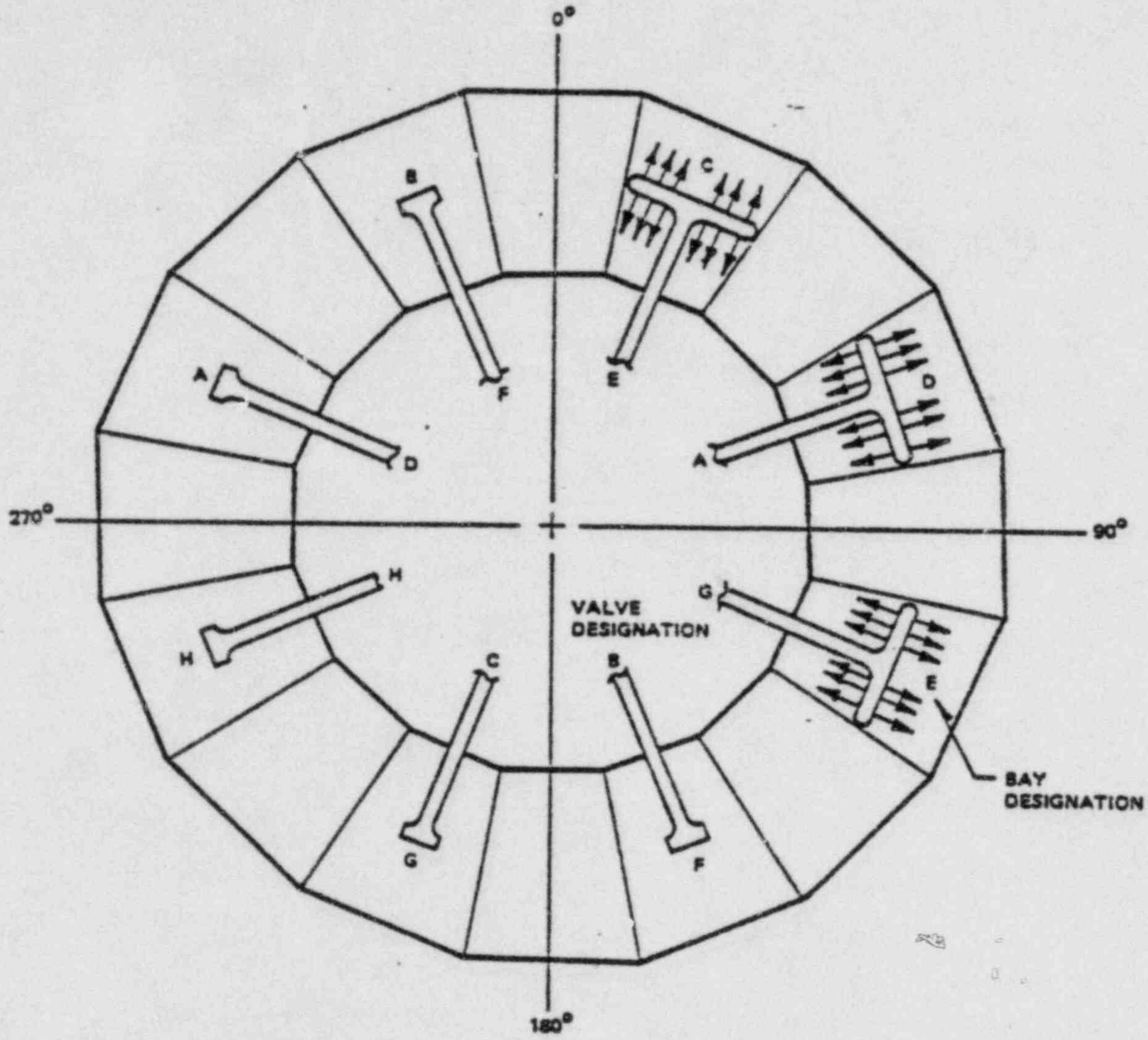
TORUS - THREE DATA BANKS/FSI REMOVED

DOMINANT FREQUENCY	PEAK AVERAGE AMPLITUDES	TEST
4-5 Hz	± 1.8 PSI	LARGE LIQUID
5-6 Hz	± 2.7 PSI	LARGE LIQUID
6-7 Hz	± 1.0 PSI	LARGE STEAM

TJM-17

S/RV LOADS

MONTICELLO T-QUENCHER TEST ORIENTATIONS



BAY	SR VALVE DESIGNATION	CATEGORY	AZIMUTH (DEGREES)	ACCESS LOCATION
A	RV2-71D		292-1/2	
B	RV2-71F		337-1/2	
C	RV2-71E		22-1/2	
D	RV2-71A	ADS*	67-1/2	48-in. MANWAY
E	RV2-71G		112-1/2	
F	RV2-71B	ADS	157-1/2	
G	RV2-71C	ADS	202-1/2	
H	RV2-71H		247-1/2	48-in. MANWAY

*ADS = AUTOMATIC DEPRESSURIZATION SYSTEM

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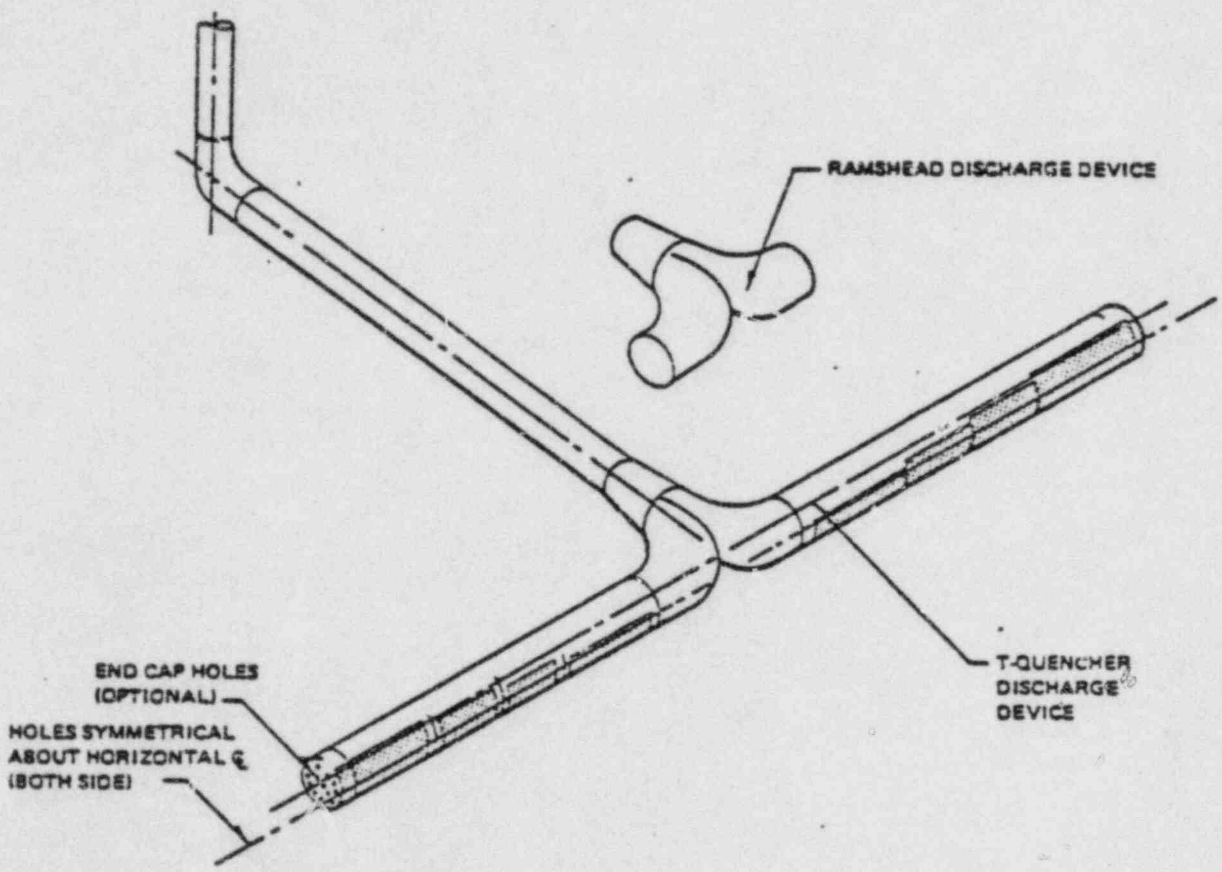
MONTICELLO TESTS DISCHARGE DEVICES



TJM-20

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MONTICELLO T-QUENCHER TEST

- OBJECTIVE - TO DEVELOP AND TEST SRV LOAD MITIGATION DEVICE THAT WOULD:

  - REDUCE AIR-CLEARING LOADS FOR FIRST AND SUBSEQUENT ACTUATIONS OF SINGLE AND MULTIPLE S/RV'S

  - DISCHARGE WITH STABLE CONDENSATION UP TO THE SATURATION TEMPERATURE (LICENSEE DATA ALREADY AVAILABLE)

- OVERALL PROGRAM

  - DEVELOPED MITIGATOR BASED ON GE LICENSEE DESIGN AND TESTS

  - PERFORMED SUCCESSFUL CONFIRMATORY TEST OF THE MITIGATOR (T-QUENCHER) IN THE MONTICELLO PLANT, DECEMBER, 1977.

- TEST RESULTS

  - POSITIVE LOAD DOWN BY A FACTOR OF 4 (~ 5.5 PSI)

  - NEGATIVE LOAD DOWN BY A FACTOR OF 2 (~ 4.6 PSI)

  - NO PEAK SHELL PRESSURE INCREASE WITH MULTIPLE VALVES

  - LOWER PEAK SHELL PRESSURES

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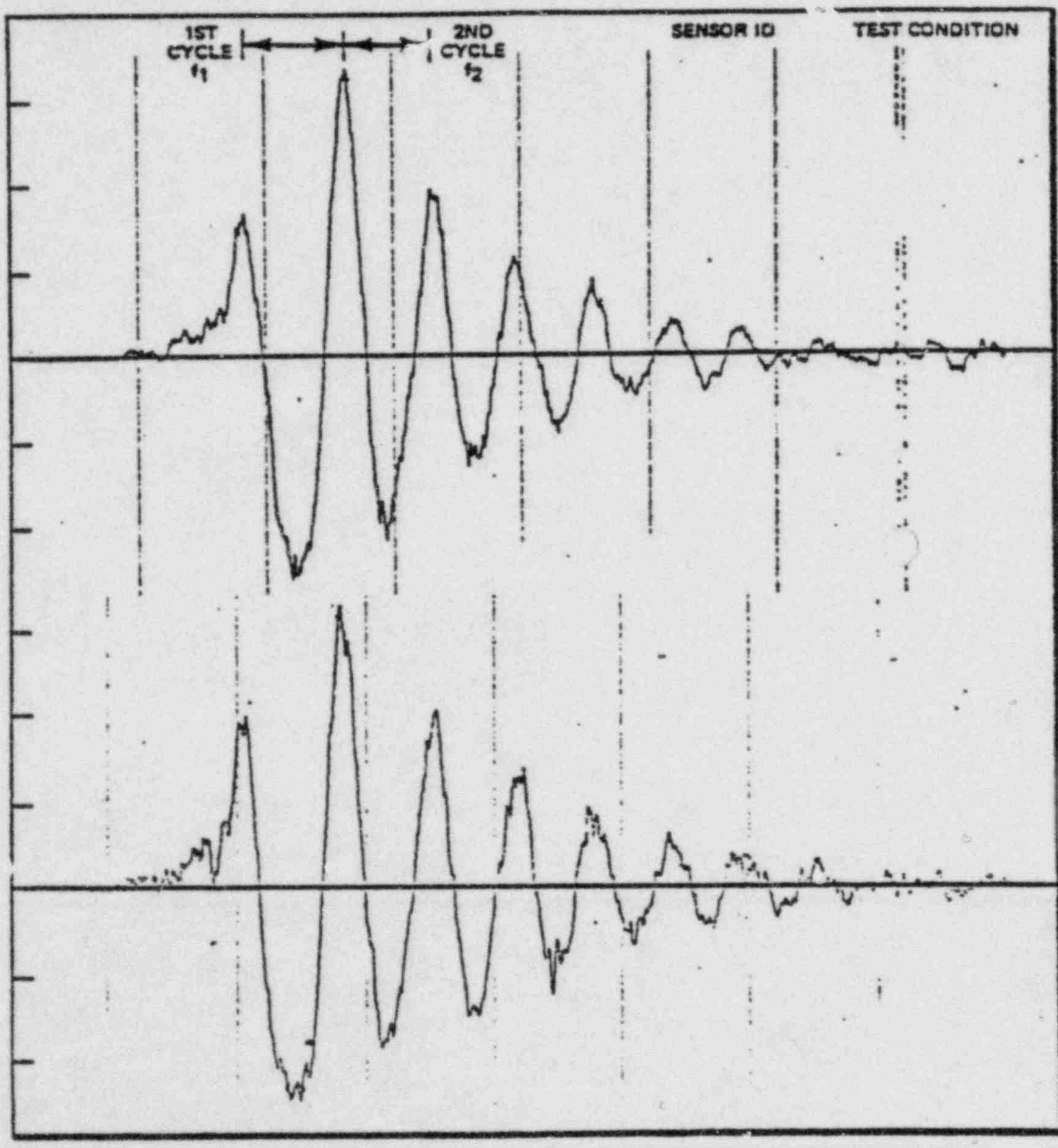
HIGHEST MEASURED SHELL PRESSURES (BAY D) FOR VARIOUS TEST CONDITIONS



CONCLUSIONS - S/RV LOADS

- T-QUENCHER DEVICE TESTED FULL SCALE
- ANALYTICAL MODEL DEVELOPED
- 1/4-SCALE TESTS FOR PARAMETRIC SENSITIVITY
- METHOD PROVIDED FOR PLANT UNIQUE LOAD CALCULATIONS

TJM-23

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

LOAD DEFINITION REPORT (LDR) OBJECTIVE

TO PROVIDE LOAD DEFINITION PROCEDURES FOR POSTULATED LOCA AND S/RV ACTUATION EVENTS FOR USE IN THE STRUCTURAL REEVALUATION OF MARK I PLANT CONTAINMENT COMPONENTS

SUMMARIZED LDR CONTENT

SECTION 1.0 - INTRODUCTION

SECTION 2.0 - REVIEW OF PHENOMENA

SECTION 3.0 - LOAD COMBINATIONS

SECTION 4.0 - LOCA RELATED LOADS

SECTION 5.0 - SAFETY RELIEF VALVE LOADS

SECTION 6.0 - OTHER CONSIDERATIONS

POOR ORIGINAL

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

POOR ORIGINAL

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PUAAG OBJECTIVE

TO ENSURE THAT THE MARK I PROGRAM STRUCTURAL ACCEPTANCE CRITERIA ARE APPLIED CONSISTENTLY BY THOSE EVALUATING EACH OF THE MARK I PLANTS

SUMMARIZED PUAAG CONTENT

SECTION 1.0 - INTRODUCTION

SECTION 2.0 - CLASSIFICATION OF STRUCTURAL ELEMENTS

SECTION 3.0 - LOADINGS

SECTION 4.0 - DESIGN AND SERVICE LIMITS

SECTION 5.0 - COMPONENT LOADINGS - SERVICE LIMIT ASSIGNMENTS

SECTION 6.0 - ANALYSIS GUIDELINES

SECTION 7.0 - REFERENCES

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MARK I CONTAINMENT PROGRAM

PROGRAM OVERVIEW - CONCLUSIONS

- PROGRAM LOAD DEFINITION REPORT COMPLETE
- AE STRUCTURAL EVALUATIONS IN PROGRESS
- UTILITIES PLANNING (AND INSTALLING) MODIFICATIONS
- FUTURE TECHNICAL ISSUES ANTICIPATED WITH NRC

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Appendix XIV

MARK I LONG TERM PROGRAM

ACCEPTANCE CRITERIA

- Pool Swell Loads
- Condensation Loads
- SRV Discharge Loads
- Related Loads & Criteria
- Structural Acceptance Criteria
- Criteria Revisions

POOL SWELL LOADS

TORUS [illegible]SURE LOADS

- 3-D POOL SWELL EFFECTS
- UNCERTAINTY MARGIN
- PRESSURE DISTRIBUTION

UPLOAD - 15% + 6.5%

DOWNLOAD - 2×10^{-5} DN^2

VENT SYST[illegible] PRESSURIZATION & THRUST

- Δ P EFFECTS

NO ΔP UNTIL DOWNCOMER CLEARING

IMPACT AND DRAG LOADS

- LONGITUDINAL SWEEP TIME — "VENT ORIFICE" TESTS
- "OTHER" STRUCTURES
  - A) CYLINDRICAL
  - B) FLAT-SURFACED
  - C) GRATINGS
  - D) IMPULSIVE

  LIMITED GEOMETRIES
  CORRECT PRESSURE PROFILE
  CONSERVATIVE DRAG
- VENT HEADER DEFLECTOR
  - A) QSTF FORCE HISTORIES — IMPACT SPIKE
  - B) ANALYTICAL FORCE HISTORIES — THEORETICALLY CONSISTENT

POOL SWELL LOADS
(CONTINUED)

Froth Impingement and Fallback	
• SOURCE VELOCITY	2.5 IMPACT VELOCITY
• DENSITY	PROPORTIONAL TO SIZE
• DIRECTION OF LOAD APPLICATION	90 DEGREE SECTOR
Pool Fallback Loads	
• DRAG EVALUATION	CONSISTENT WITH SUBMERGED DRAG
Water Jet Submerged Drag	
• INDUCED POOL MOTION	
LOCA Bubble Submerged Drag	
• DRAG COEFFICIENT	$C_D = 1.2$ $D_{EQ} = \sqrt{2}\, L_{MAX}$
• NODALIZATION	
• INTERFERENCE EFFECTS	CORRECTION FACTOR

CONDENSATION LOADS

CONDENSATION OSCILLATIONS

- CONFIRMATORY TESTS — ESTABLISH UNCERTAINTY
- "TIED" DOWNCOMER LOADS — UNACCEPTABLE
- DOWNCOMER DLF SCALING — ESTABLISH FSTF DLF

CHUGGING

- MAXIMUM DOWNCOMER LOAD
- STATISTICAL DOWNCOMER LOAD
- DOWNCOMER TIE-BAR LOAD — TRIANGULAR PULSE

SAFETY-RELIEF VALVE DISCHARGE

1. DISCHARGE DEVICE
2. CLEARING TRANSIENT - PIPING
3. CLEARING TRANSIENT - SHELL PRESSURES
4. REFLOOD TRANSIENT
5. AIR AND WATER THRUST LOADS
6. PIPE TEMPERATURE TRANSIENT
7. SRV EVENT CASES
8. SUPPRESSION POOL TEMPERATURE LIMITS
9. WATER JET LOADS
10. BUBBLE DRAG LOADS

TORUS SRV PRESSURE LOADS

1.	DISCHARGE LINE WATER LEG TREND	13.5 FEET
2.	DISCHARGE LINE VOLUME TREND	65 CUBIC FEET
3.	SUBSEQUENT ACTUATION PRESSURE	SAME AS FIRST ACTUATION

4. MULTIPLE VALVE SUPERPOSITION
   - ABSOLUTE SUM

. DISCHARGE LOAD FREQUENCY
   - FIRST ACTUATION ± 25%
   - SUBSEQUENT ACTUATION ± 40%

RELATED LOADS & CRITERIA

CONTAINMENT PRESSURE & TEMPERATURE

SECONDARY LOADS

- SEISMIC SLOSH
- POST-SWELL WAVES
- ASYMMETRIC VENT FLOW
- SONIC AND COMPRESSION WAVES
- DOWNCOMER AIR CLEARING

DIFFERENTIAL PRESSURE CONTROL

- DESIGN REQUIREMENTS
- TECHNICAL SPECIFICATIONS

ALL OTHER LOADS - FSAR

STRUCTURAL ACCEPTANCE CRITERIA

STRUCTURES AND COMPONENTS

- TORUS AND SUPPORTS
- TORUS INTERNALS
- TORUS ATTACHED PIPING
- SRV DISCHARGE PIPING

SERVICE LEVEL ASSIGNMENTS - ASME CODE

- DYNAMIC TO STATIC COLLAPSE LOAD (N-197)
- LOSS OF ΔP - LEVEL D
- DBA + SRV - LEVEL C

DYNAMIC LOAD COMBINATIONS

- ABSOLUTE SUM
- ALTERNATE - CDF 84%

ACCEPTANCE CRITERIA REVISIONS

1. CLARIFY VELOCITY FOR DRAG FOLLOWING IMPACT

2. CRITERIA FOR FROTH LOADS FROM QSTF MOVIES

3. CORRECT CYLINDRICAL DRAG COEFFICIENT

4. CORRECT TYPE 2 & 3 DEFLECTOR FORCE HISTORIES

5. ADD CRITERIA FOR QUENCHER LOADS FROM IN-PLANT TESTS:
   - 4 COLD SINGLE VALVE TESTS
   - "TUNE" LOAD-STRUCTURE MODEL
   - DETERMINE MAXIMUM AMPLIFICATION FROM RESPONSE SPECTRUM ANALYSES
   - APPLY CORRECTED MODEL AT DESIGN BASIS CONDITIONS

6. SET LIMIT FOR SRV DISCHARGE LINE VOLUME TREND

7. ALLOW SEPARATE BOUND FOR SRV GLOBAL TORUS PRESSURE LOADS

8. ADD CRITERIA FOR SRV BULK-TO-LOCAL ΔT FROM IN-PLANT TESTS

9. ALLOW ALTERNATE MONITORING FOR LOCAL POOL TEMPERATURE

10. SPECIFY THAT PROCEDURES OR EQUIPMENT MINIMIZE ACTIONS TO DETERMINE BULK POOL TEMPERATURE

Appendix XV

MARK I CONTAINMENT PROGRAM

CURRENT PROGRAM ACTIVITIES

CURRENT PROGRAM ACTIVITIES

- REVISE LDR ACCEPTANCE CRITERIA PER NRC
- CONDUCT A/E STRUCTURAL ASSESSMENTS
- RETEST IN FSTF FOR C/O STATISTICAL DATA
- DEVELOP LOAD DEFINITION APPROACH UTILIZING IN-PLANT S/RV TESTS
- COMPLETE PLANT UNIQUE STRUCTURAL ANALYSIS
- CONTINUE INSTALLATION OF PLANT MODIFICATIONS

RNS-2

ARCHITECT ENGINEER (A/E) ACTIVITIES

- STRUCTURAL EVALUATIONS ON PLANT UNIQUE BASIS
- SUPPORT OF GENERIC A/E ACTIVITIES
  - COMPARE LDR METHODOLOGY TO TEST DATA
  - IDENTIFY OVERLY CONSERVATIVE LOADS
  - REFINE STRUCTURAL ANALYSIS TECHNIQUES
- DEFINE REQUIRED PLANT MODIFICATIONS
- ASSESS ADVANTAGES OF IN-PLANT TESTS (S/RV)

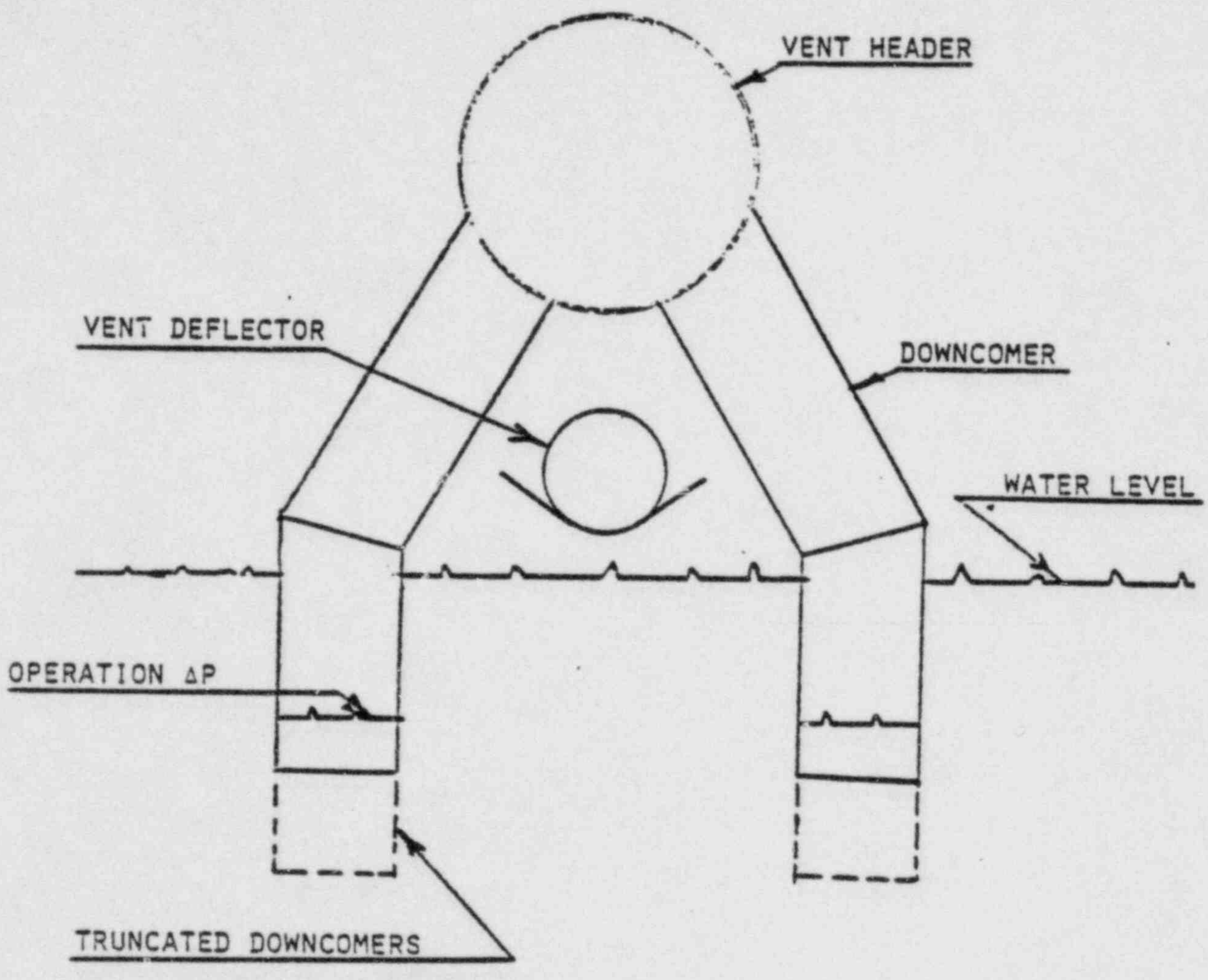
RNS-3

TYPICAL MODIFICATIONS

LOAD	LOAD MITIGATION TECHNIQUE	LOAD REDUCTION
POOL SWELL		
- VERTICAL	ΔP	~40%
	REDUCED SUBMERGENCE	~15%
- IMPACT	VENT HEADER DEFLECTOR	~90%
CONDENSATION		
- CONDENSATION OSCILLATION - CHUGGING	MITIGATION GENERALLY NOT REQUIRED	LOW STRESSES OBSERVED IN FSTF
S/RV	T-QUENCHER	~50% TO 75%

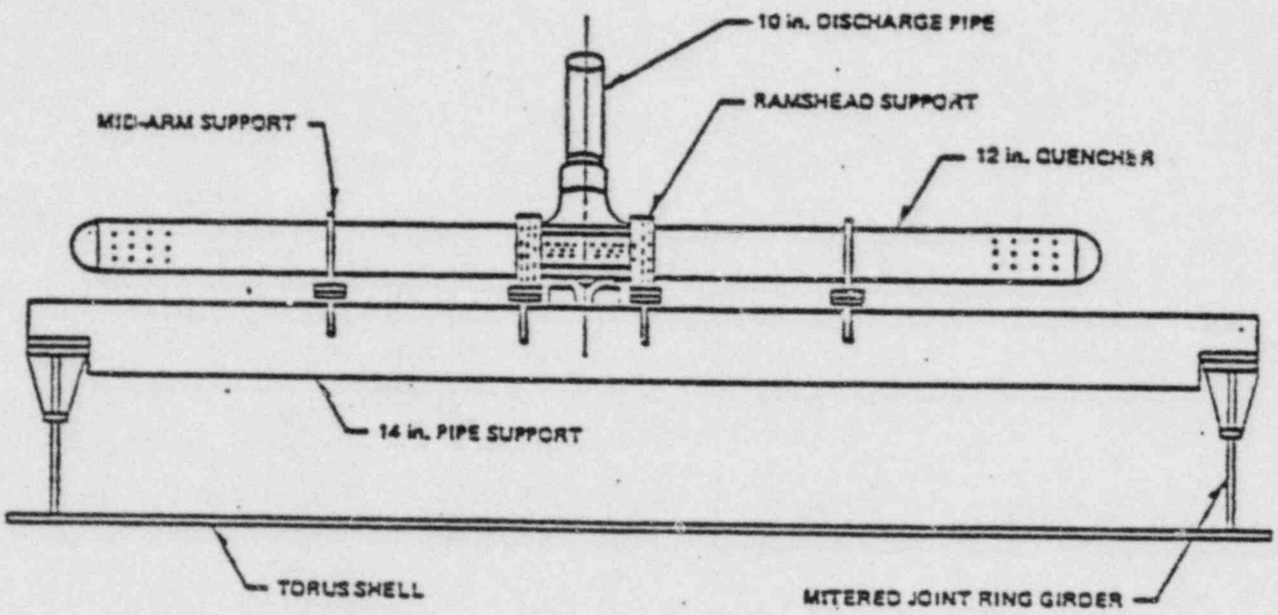
RNS-4

PLANT MODIFICATIONS RESULTING FROM POOL SWELL LOADS



RNS-5

TYPICAL MID-BAY T-QUENCHER INSTALLATION



TYPES OF PLANT MODIFICATIONS

SADDLES

TORUS COLUMN REINFORCEMENT

COLUMN TO TORUS REINFORCEMENT

RING GIRDER BRACING

TORUS TIE DOWN

VENT HEADER DEFLECTOR

DOWNCOMER TRUNCATION

DOWNCOMER BRACING

TORUS ATTACHED PIPE SUPPORT

QUENCHER DISCHARGE DEVICE

S/RV VACUUM BREAKERS

INTERNAL MISCELLANEOUS SUPPORT

RNS-7

MARK I CONTAINMENT PROGRAM

UTILITY COMMITMENT

- SUBSTANTIAL RESOURCES DIRECTED TOWARD RESOLVING CONCERNS
  - UTILITY GENERIC PROGRAM FUNDING EXCEEDED $50 MILLION
  - MOST LOADS BASED ON FULL SCALE PROTOTYPICAL TESTING
  - COSTS FOR PLANT MODIFICATIONS ON ORDER OF $15-$20 MILLION PER UNIT
- UTILITIES ARE IMPLEMENTING MODIFICATIONS
  - SOME MODIFICATIONS COMPLETE
  - ENGINEERING/PROCUREMENT COMPLETE OR IN PROGRESS
  - FINAL PLANT UNIQUE ANALYSIS BEING ACCOMPLISHED IN PARALLEL WITH MODIFICATIONS
- DISCUSSIONS WITH NRC ANTICIPATED TO RESOLVE SOME TECHNICAL ISSUES

RNS-8

Appendix XVI

ACRS REVIEW OF
NRC BULLETINS AND ORDERS TASK FORCE EFFORTS
238th MEETING
FEBRUARY 8, 1980
WASHINGTON, D.C.

PROJECT STATUS REPORT

PURPOSE:

The purpose of the meeting is to continue ACRS review of the efforts of the NRC Bulletins and Orders (B&O) Task Force begun at the January 1980 meeting. NRC is requesting ACRS comment on the B&O effort.

BACKGROUND:

Following the issuance of bulletins to all operating plants by I&E immediately after the TMI accident, and the NRC Orders issued to B&W plants, the NRC organized the B&O Task Force to develop and implement the requirements of these bulletins and orders. The Task Force objective was expanded to confirm the bases for continued safe operation of operating plants. The scope of review encompassed the loss of feedwater and small-break LOCA events in the areas of systems reliability, analyses and operator guidelines, plant procedures, and operator training. The principal work products of the Task Force are 4 generic reports (one for each vendor s plants) plus a summary report (NUREG-0645). Draft copies of 0645 were provided to the ACRS during the January meeting.

MEETING TOPICS:

To date NRC has issued "camera-ready" versions of 4 of the 5 reports noted above, the exception is the GE generic report which is scheduled to be issued around the first of February. Copies of the Summary Report (NUREG-0645), the Westinghouse Report (NUREG-0611) and the B&W Report (NUREG-0565) have been mailed to you. Copies of the CE report, and if it is available, the GE report will be distributed to you prior to the February meeting.

Several ACRS Members have raised the broad question of the impact of all the NRC mandated changes on overall plant risk. Dr. Kerr in particular has written a memo (Attachment I) which urges that "...careful attention should be given to making certain that the proposed changes do reduce risk!" At the urging of Mr. Mathis, and following consultation with Dr. Plesset, it has been decided that the NRC B&O Presentation will be limited to 2 hours instead of the 4 hours originally scheduled. At this time it is not expected that the Committee will provide comments on the B&O Task Force efforts during the February meeting.

One hour of discussion time has been scheduled with the Director, NRR on Thursday, February 7, from 3-4 p.m. I suggest that the Committee raise the topic of the impact of the NRC-mandated changes on overall plant risk with Mr. Denton at this time.

For the B&O Presentation, the Staff will highlight the activities of the Task Force since the January ACRS Meeting and provide an overview of the B&O Recommedations. The Staff will again note the differing technical opinions between Drs. Ross and Rosztoczy. These differences were noted at the January meeting. The attached memos (Attachments II-IV) detail this item.

Dr. Okrent requested that the NRC provide a rough estimate of the probability of certain accidents stated by the NRC as leading to possible core melt, given loss of all feedwater or loss of natural circulation (Attachment V). NRC will address this item under the topic of "Impact Assessment of the B&O Recommendations on Reducing Risks".

TASK FORCE ACTIVITIES:

The review and upgrade of plant auxiliary feedwater systems resulted in a number of short and long-term requirements being imposed on all operating plants. Among the short-term requirements include: automatic actuation of auxiliary feedwater, technical specifications limiting the outage time of one train of auxiliary feed, and procedures to assure valves necessary for auxiliary feed are locked open. Long-term requirements include: redundant piping and valves to assure at least two flowpaths for auxiliary feed, elimination of AC dependency for one train of auxiliary feed for at least 2 hours, and installation of safety-grade auxiliary feed automatic

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start signals. It is noted that some Utilities appear to have some serious objections to the NRC requirements that auxiliary feedwater systems should be automatic. Some exploration of the advantages and disadvantages of automatic AFW initiation may be appropriate at the meeting.

The analysis of design and off-normal accidents and transients includes NRC review of the vendors small-break LOCA models, and inadequate core cooling analyses. Two issues that arose during the review of vendor's LOCA models was the questio whether the reactor coolant pumps (RCPs) should be tripped or all to run during a small-break LOCA, and when HPI should be terminate. The NRC has published a report (NUREG-0623) that states their position on RCP trip. HPI termination requirements have been documented in the vendor emergency guidelines. One question of particular interest is the effect of tripping the RCP's on other non-LOCA transients that initially react like a small-break LOCA (decreasing RCS pressure). Tripping of the pumps for these events which are more common may increase the difficulties associated with the ultimate cool-down of the reactor system.

The work on revision of vendor emergency guidelines, implementation of operator procedures, and operator training is presently underway. The revised vendor emergency guidelines for small-break LOCA have been approved by the NRC. B&O Task Force members have performed audits at selected plants to check the implementation of the vendor's guidelines into the plant procedures as well as the degree of operator training in the area of small-break LOCA accomplished to date.

MEETING OUTCOME:

As noted above, NRC desires Committee comment in the form of a letter on the efforts of the B&O Task Force. Committee actions taken at the February meeting will depend on the outcome of the scheduled Staff discussion. It may be necessary for the Task Force to meet again with the Committee, or another Subcommittee meeting(s) may be required.

DELETION 1

Pages A-321 thru A-232

ESCT COMM

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

DEC 10 1979

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DEC 0 7 1979

DISTRIBUTED TO ACRS MEMBERS

MEMORANDUM FOR: H. Denton, Director, Office of Nuclear Reactor Regulation
D. Ross, Jr., Director, Bulletins & Orders Task Force

FROM: Z.R. Rosztoczy, Chief, Analysis Group, Bulletins & Orders Task Force

SUBJECT: APPENDIX VIII OF NUREG-0611, "GENERIC EVALUATION OF FEEDWATER TRANSIENTS AND SMALL BREAK LOSS-OF-COOLANT ACCIDENTS IN WESTINGHOUSE DESIGNED OPERATING PLANTS".

Reference: Memo from D. Ross to H. Denton on "B&O Report on W Plants", dated December 4, 1979.

When the B&O Task Force was formed, it was organized in three groups: Projects, Systems and Analysis. I was asked to head the Analysis Group. The goal of the analysis group was to gain an assurance that the operators and designers of nuclear plants: (1) have a thorough understanding of plant response in case of a small break LOCA; (2) have trained the reactor operators and prepared the plant emergency procedures based on their knowledge; and (3) modified their plant designs, if needed, so continued operation of these plants will not represent an undue risk to public safety in the event of a small LOCA.

The Analysis Group has completed its work, and is now in the process of documenting its findings. The Analysis Groups safety evaluation relative to operating Westinghouse plants is scheduled to be published as Appendix VIII of NUREG-0611. Section 2 of this Appendix, entitled "Conclusions and Recommendations" is attached for your convenience. The most important recommendations are as follows:

- The Westinghouse small LOCA analyses relied on equipment which have not previously been characterized as part of the reactor protection system or part of the engineered safety features... The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to provide appropriate protection. These systems should also be qualified for post LOCA environment.

ECCS-1

ATTACHMENT II

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- Unless the capability of the relief valves to provide sufficient depressurization in case of loss of heat removal through the steam generators can be demonstrated with due account of input uncertainties and calculational uncertainties, continued operation of the effected plants should be conditioned on a timely design change. Replacement of the high pressure inject:on pumps with high cut off head pumps or increased relief valve capacity are possible, acceptable design changes.

- The various modes of two-phase flow natural circulation which are expected to play a significant role in plant response following a small LOCA should be demonstrated experimentally. The results of the tests should be available for NRC review not later than December 31, 1980.

- Appropriate means, including additional instrumentation if necessary, should be provided in the control room to facilitate checking whether natural circulation has set in.

- Frequent overpressure transients should not result in opening of the relief valves. Licensees of Westinghouse plants should (1) install an anticipatory reactor trip on turbine trip if it is not already present; (2) show that overpressure feedwater transients challenge the relief valves only in exceptional cases (less than 5% of transients). If the later point cannot be supported by the 150 years of operating experience presently available, appropriate design changes should be introduced. Changes in relief valve and in high pressure reactor trip setpoints and the installation of additional anticipatory reactor trips should be considered. The licensees should comply with this requirement by March 1, 1980.

- Each licensee should be required to submit a report on safety valve challenge rate and safety valve failure rate based on past history of the plant. These reports should be submitted to NRC prior to February 1, 1980.

Various members of the Analysis Group worked on the Westinghouse review. Appendix VIII is a summary of the Groups work and the reason for its recommendations. All members of the Group support the recommendations.

A draft copy of Appendix VIII was circulated for review. Dr. Ross comments indicated that he disagrees with some of the recommendations. He requested that the recommendations he disagrees with be removed from Appendix VIII together with the supporting material that lead us to these recommendations. His views are expressed in the above Reference 1. I have carefully considered the deletions and modifications recommended by Dr. Ross. While I find many of them helpful and constructive, full compliance with his request would prevent an important portion of the information we gathered and an important portion of our deliberations from reaching the readers of this report, namely ACRS, NRC Commissioners, and the public. Therefore, I would like to suggest the following approach. The report has a main part, and a number of appendices. The Analysis Groups safety evaluation in its entirety including the recommendation should be published as Appendix VIII. The main part of the report discusses each of the appendices and brings forward the most important recommendations. This would be an appropriate place for Dr. Ross to express his views and recommend to you and the NRC commissioners that only portions of the Analysis Group recommendations be implemented, if that is his position.

Finally, I would like to urge both of you to consider carefully each of the attached recommendations. They are the results of long, hard work, and represent a significant improvement in public safety. If these recommendations are enforced, the public risk associated with Westinghouse plants relative to small LOCA's, would be reduced to approximately the same level as it was required from B&W plants last May. Furthermore, these recommendations are responsive and provide resolutions to most concerns we received from other groups, like ACRS.

Zoltan R Rosztoczy
Z.R. Rosztoczy, Chief
Analysis Group
Bulletins & Orders Task Force

Attachment:
As stated

cc: All B&O Task Force Members
R. Mattson
S. Hanauer
ACRS (16)
E. Case

:ion 2. Conclusion and Recommendations

1. The small break analysis methods used by Westinghouse are satisfactory for the purpose of predicting trends in plant behavior following a small LOCA. The results of the analyses can be used to develop improved emergency procedures, and for training of reactor operators. However, several individual models have been identified in Section 4.2.1 as requiring improvement, or further confirmation. In addition, comparison of the total analysis method with available small break integral test data (Semiscale Test S-02-6) has indicated large uncertainties in the calculations. The analysis methods should be revised and verified before they can be considered for NRC approval under 10 CFR 50.46.

Recommendations:

(a) The analysis methods used for small break LOCA analysis for compliance with Appendix K should be revised, documented, and submitted for NRC approval within 6 months from the issuance of this report.

(b) Plant-specific calculations using the NRC approved model for small breaks to show compliance with 10 CFR 50.46 should be submitted by all licensees prior to December 31, 1980.

(c) The NRC review of the conservatisms in LOCA calculations should be accelerated. An NRC position on required conservatisms in small LOCA analysis should be issued by June 30, 1980.

2. Westinghouse has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses provide an adequate basis for determining improved operator guidelines, and demonstrate that operator action and a combination of heat removal by the steam generators, high pressure injection system, and the break ensure adequate core cooling. The required operator action is: tripping the reactor coolant

pumps shortly after initiations of a LOCA. This action is required because Westinghouse calculations show for a narrow range of small breaks that the 10 CFR 50.46 limits could be exceeded if the pumps are not tripped. According to Westinghouse estimates, at least 10 minutes are available for the operator to perform this action. Our evaluation of the Westinghouse analysis indicates that the times available for the operators could be shorter than 10 minutes may be as short as 3 minutes, indicating a need for automatic actuation.

If in addition to the small LOCA, feedwater flow (both main feedwater and auxiliary feedwater) is lost or if for any reason natural circulation fails to take place, there will be no heat removal through the steam generators. In this case operator action is required to restore steam generator feedwater flow or to open the pressurizer relief valves and block valves (if closed). According to Westinghouse, in case of a loss of feedwater flow, either action will serve to depressurize the primary system so that sufficient injection flow can be established. If natural circulation fails, the operator must open the pressurizer relief valves. Westinghouse indicated that approximately 1 hour is available for the operators to reinitiate feedwater flow. Opening of the relief valves must be accomplished within 40 minutes in order to maintain the consequences of the event within acceptable limits. The staff review of the Westinghouse calculations reveals that Westinghouse overestimated the relief valve flow rate used in the calculations. Considering the above bias in the calculations, the large uncertainties of the calculations and the fact that Westinghouse was unable to provide test data on valve discharge flow, we cannot agree with the Westinghouse conclusion. The information presently available does not provide an assurance that the operator can actually achieve the needed depressurization.

Recommendations:

(a) Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. The licensees should be required to consider other solutions to the small break problem, for example an increase in safety injection flow rate. In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small LOCA. The signals designated to intiate the pumps trip should be carefully selected in order to differentiate between a small LOCA and other events which do not require reactor coolant pump trip.

(b) The Westinghouse small LOCA analyses relied on equipment which have not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide reactor coolant pump trip, the pressurizer relief valves, the pressurizer relief block valves, equipment used to automatically actuate the pressurizer relief valves, and equipment used to remotely control the pressurizer relief and block valves fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to provide appropriate protection. These systems should also be qualified for post LOCA environment.

(c) Unless the capability of the relief valves to provide sufficient depressurization in case of loss of heat removal through the steam generators can be demonstrated with due account of input uncertainties and calculational uncertainties, continued operation of the effected plants should be conditioned on a timely design change. Replacement of the high pressure injection pumps with high cut off head pumps or increased relief valve capacity are possible, acceptable design changes.

(d) Plant simulators used for operator training should offer, as a minimum, the following small LOCA events:

- continuous depressurization
- pressure stabilized at a value close to secondary pressure
- repressurization
- stuck open pressurizer relief valve(s)
- stuck open letdown valve

Each of these cases should be simulated with reactor coolant pumps running as well as reactor coolant pumps turned off. The first three events should be simulated for both cold and hot leg breaks. In addition to the usual single failures in the ECCS and feedwater systems, complete loss of feedwater system should also be simulated in conjunction with the above events.

3. A number of concerns related to decay heat removal following a very small break LOCA, and other related items, were identified by M. C. Michelson of TVA (see Section 4.1). These concerns were identified for PWRs designed by Babcock & Wilcox and Combustion Engineering. Westinghouse has reviewed these concerns and provided an analysis of those items that relate to plants of their design. Postulated modes of two-phase flow natural circulation play an important role in the Westinghouse analysis. The analysis provides an adequate assessment of these concerns; however, experimental results are not available to support the analytical predictions.

Recommendations:

(a) The various modes of two-phase flow natural circulation which are expected to play a significant role in plant response following a small LOCA should be demonstrated experimentally. The results of these tests should be available for NRC review not later than December 31, 1980.

(b) Appropriate means, including additional instrumentation if necessary, should be provided in the control room to facilitate checking whether natural circulation has set in.

4. The record of relief valve failures for all PWRs, 13 in approximately 200 reactor years, have demonstrated relief valve failure to be the most likely cause of a small LOCA. The high failure rate is the result of a high relief valve challenge rate. The most frequent overpressure transients, feedwater transient and turbine trip, will open the relief valves unless an early reactor trip limits the pressure excursion to a value less than the relief valve setpoint. Thus, the selection of reactor trips and relief valve setpoint has a strong effect on relief valve challenge rate. For example, during the course of 12 reactor transients experienced by B&W plants during the summer of 1979, which would have opened the relief valve with the pre TMI-2 reactor trips and setpoints, no actuation of the valve occurred with the new reactor trips and revised setpoints.

The transient analysis provided by Westinghouse indicates that (1) opening of the relief valves in case of a feedwater transient is likely; and (2) the relief valves will not open in case of a turbine trip provided the plant is equipped with a reactor trip on turbine trip. Westinghouse also presented information on relief valve openings based on operating experience However, this information did not list a single feedwater transient as the cause of relief valve opening. Unfortunately, the information available on operating experience is incomplete at the present time.

Recommendations:

a. Frequent overpressure transients should not result in opening of the relief valves. Licensees of Westinghouse plants should: (1) install an anticipatory reactor trip on turbine trip if it is not already present ; (2) show that overpressure feedwater transients challenge the relief valves only in exceptional cases (less than 5% of transients). If the later point cannot be supported by the 150 years of operating experience presently available, appropriate design changes should be introduced. Changes in relief valve and in high pressure reactor trip setpoints and the installation of additional anticipatory reactor trips should be considered. The licensees should comply with this requirement by March 1, 1980.

b. Licensees of Westinghouse plants up to date reported approximately 300 feedwater transients. The peak reactor system pressure reached during these transients and possible indication of relief valve opening during these transients should be reported to NRC prior to March 1, 1980.

c. All future relief valve challenges should be recorded and reported to NRC.

5. One possible way to completely eliminate the risk associated with the failure of relief valves is to operator the plants with the block valves closed. This mode of operation, however, could result in some increase in the lift frequency of one safety valve. The licensees so far have filed to provide information on the observed failure rate of safety valves. Consequently, neither the desireability nor the acceptability of this mode of operation can be evaluated at this time.

Recommendations:

a. Each licensee should be required to submit a report on safety valve challenge rate and safety valve failure rate based on past history of the plant. These reports should be submitted to NRC prior to February 1, 1980.

b. All future safety valve challenges should be recorded and reported to NRC.

6. The derivative control system, installed on at least one relief valve in most Westinghouse plants, has caused spurious valve actuations. This resulted in a considerable number of relief valve challenges. Westinghouse recently recommended the elimination of derivative control.

Recommendation:

The derivative control, if it is still in use, should be replaced with a constant pressure setpoint system. This change should be accomplished within 30 days from the issuance of this report.

7. Relief valves supplied by Control Components, Inc. were used, at the first time, on the McGuire plant, owned by Duke Power. One of these valves failed during hot functional testing. Following failure the manufacturer of the valves recommended modifications to the valves.

Recommendation:

The McGuire relief valves should either be replaced with valves which have an operational data base, or should be sufficiently tested under design conditions prior to start-up of the plant to assure their reliability.

D. Ross

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

DEC 4 1979

MEMORANDUM FOR: H. Denton, Director, Office of Nuclear Reactor Regulation

FROM: D. Ross, Jr., Director, Bulletins & Orders Task Force

SUBJECT: B&O REPORT ON W PLANTS

I have completed my review of the subject report. During my review I concluded that several recommendations would not be included in the final version of the report, for the reasons stated below:

1. Conservatism in SBLOCA

   It was recommended that the NRC review of the conservatism in LOCA calculations be accelerated, with a position on SBLOCA by 6/80.

   I took this out on grounds of relevance. It belongs, if anywhere, in the NRC action plan, since it involves prioritization of resources.

   I intend to delete this from the other reports also.

2. Loss of Feedwater (All)

   It was recommended that either the capacity of the relief valves to provide depressurization in a case of a complete loss of feedwater be demonstrated (with due account of input uncertainties and calculational uncertainties) or continued operation of the affected plants be conditioned on timely design changes.

   I deleted this because:

   (i) we have already required improvements in AFW; and,

   (ii) I believe it needs more careful attention under USI. S. Hanauer agrees with this position.

   I intend to delete this from CE and B&W reports. The ability of relief valves to pass vapor or 2-Ø mixtures is the subject of STLL.

3. PORV Operation

   It was recommended that the likelihood of stuck-open PORV be reduced by a combination of items which are attached here as Appendix A.

   I decided not to embrace these recommendations because:

   (i) Through STLL there is emergency power for PORV and block valves, and performance testing for PORV, and direct position indication for PORV,

ATTACHMENT III

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(ii) Through B&O the plant procedures have been upgraded to account for stuck-open PORV;

(iii) One opening mechanism (see Appendix B) is being eliminated;

(iv) Safety classifications and qualifications are being upgraded through point 9 of LTLL; and,

(v) I have decided to include a new item which calls for auto-closure of block valve on low pressure (for all PWRs).

It is my judgment that these five items are sufficient and superior to Appendix A. I embraced Appendix B.

I also deleted a related recommendation to do some detailed comparison of plant experience vs. analytical predictions. If this is to be done at all it should be in an orderly manner under the topical report program for LOFTRAN.

4. Small Break Methodology

There were many deficiencies noted in the small-break methodology. Most of these stayed in; in context, they will need attention during the 1980 Appendix K phase.

5. Loads

There was a concern about loads due to injection of cold water. I asked for a staff judgment, absent conclusive data.

6. Pump Seals

The recommendation was that the worst failure of RCP seals be assumed with the worst small break. I asked for an "or" clause on the ability of the pump seals to survive during SBLOCA.

I believe this represents the important elements.

For historical purposes there will be available, in due course, a complete manuscript of Appendix VIII (Analysis) as I receive it which may be compared with the final as-printed version.

All of the changes that I made were discussed with the review team and there was no disagreement, except possibly from Dr. Rosztoczy, who may ultimately have a differing viewpoint.

There is, however, unanimity on the conclusion that:

The small break analysis methods used by W are satisfactory for the purpose of predicting trends in plant behavior following a small LOCA. The results of the analyses can be used to develop improved emergency procedures and for training of reactor operators.

As that was the principal charter of B&O, I believe that we can confidently close our task force effort on analysis. Further wo-k can proceed under a different organization unit.

D. F. Ross, Jr., Director
Bulletins & Orders Task Force

cc: All B&O Task Force Members
R. Mattson
S. Hanauer
ACRS (16)
E. Case

Attachments:
Appendix A & B

Appendix A

(Not Adopted)

Recommendations:

a. Frequent overpressure transients should not result in opening of the relief valves. Licensees of Westinghouse plants should: (1) install an anticipatory reactor trip on turbine trip if it is not already present; (2) show that overpressure feedwater transients challenge the relief valves only in exceptional cases (less than 5% of transients). If the latter point cannot be supported by the 150 years of operating experience presently available, appropriate design changes should be introduced. Changes in relief valve and in high pressure reactor trip setpoints and the installation of additional anticipatory reactor trips should be considered. The licensees must comply with this requirement by March 1, 1980.

b. Licensees of Westinghouse plants up to date reported approximately 300 feedwater transients. The peak reactor system pressure reached during these transients and possible indication of relief valve opening during these transients should be reported to NRC prior to March 1, 1980.

c. All future relief valve challenges should be recorded and reported to NRC.

One possible way to completely eliminate the risk associated with the failure of relief valves is to operate the plants with the block valves closed. This mode of operation, however, could result in some increase in the lift frequency of one safety valve. The licensees so far have failed to provide information on the observed failure rate of safety valves. Consequently, neither the desirability nor the acceptability of this mode of operation can be evaluated at this time.

Recommendations:

(a) Each licensee should be required to submit a report on safety valve challenge rate and safety valve failure rate based on past history of the plant. These reports should be submitted to NRC prior to February 1, 1980.

(b) All future safety valve challenges should be recorded and reported to NRC.

Appendix B
(Adopted)

The derivative control system, installed on at least one relief valve in most Westinghouse plants, has caused spurious valve actuations. This resulted in a considerable number of relief valve challenges. Westinghouse recently recommended the elimination of derivative control.

Recommendation:

The derivative control, if it is still in use, should be replaced with a constant pressure setpoint system. This change should be accomplished within 30 days from the issuance of this report.

Relief valves supplied by Control Components, Inc. were used, at the first time, on the McGuire plant, owned by Duke Power. One of these valves failed during hot functional testing. Following failure the manufacturer of the valves recommended modifications to the valves.

Recommendation:

The McGuire relief valves should either be replaced with valves which have an operational data base, or should be tested under design conditions prior to startup of the plant to assure their reliability.

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
January 7, 1980

MEMORANDUM FOR: Zoltan Rosztoczy, Chief, Analysis Branch
Division of Systems Safety
Office of Nuclear Reactor Regulation

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: APPENDIX VIII OF NUREG-0611

I have reviewed your memorandum dated December 7, 1979 and your memorandum addressed to Dr. Ross dated December 18, 1979 which transmitted your recommended draft of Appendix VIII of NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants." Based on my review of these memoranda, my discussion with you on December 17, 1979, and discussions with members of the staff, I have reached the following conclusions on four items in your recommended draft of Appendix VIII.

1. NRC Review of the Conservatisms in LOCA Calculations

Your recommendation states that the NRC should accelerate its review of the conservatisms in LOCA calculations and issue a position on required conservatisms in small LOCA analysis by June 30, 1980.

I believe that the conservatisms in LOCA calculations is a topic that should be investigated but the priority on which this is to be done has to be assessed. An important consideration in this regard is the need to await the results of the LOFT small-break tests to assess the adequacy of analysis methods. Further, I do not believe that it is appropriate for this recommendation to be incorporated in each of the reports on the evaluation of vendor designed plants to be issued by the Bulletins and Orders Task Force. Rather, I believe that it is more appropriate to review this recommendation at the Office level and determine the schedule of work based on overall Office priorities. I understand that Dr. Ross will include this topic in the final report of the Bulletins and Orders Task Force to assure that it will be considered.

ATTACHMENT IV

2. Loss of Heat Sink

Your recommendation states that there should be a diverse heat removal path in case heat cannot be removed through the steam generators. The NRC Action Plan is to include a number of considerations with regard to the potential loss of heat sink. For example, consideration of a high pressure RHR system will be part of the Action Plan. It is my understanding that the Action Plan is currently being revised to include these considerations.

My opinion is that the impact of this recommendation needs to be considered as an integral portion of the NRC Action Plan. I therefore conclude that pending the resolution of the Action Plan, it is not appropriate to require the industry to make modifications at this time. I understand that Dr. Ross will also document this issue in the final report of the Bulletins and Orders Task Force.

3. PORV

Your recommendations state that the licensees of Westinghouse plants should assure that overpressure transients do not result in system blowdown due to stuck open relief valves. The recommendations require that the licensees demonstrate that there are challenges to the relief valves from overpressure feedwater transients in less than 5% of the transients and/or that design changes be made.

This recommendation is applicable to Combustion Engineering as well as Westinghouse designed plants. As you know, the following improvements have been required or are being recommended for implementation on these plants.

a. PORV and Block Valve on Emergency Power
b. Direct PORV Position Indication
c. Qualifications and Safety Classifications per Item 9 of the LTLL
d. Operational Procedures and Training
e. Replacement of Derivative Control
f. Recommended Replacement of the McGuire Relief Valves or Testing of the Valves Under Design Conditions Prior to Startup
g. Failure to Close Emergency Action Level
h. Relief and Safety Valve Qualification Testing per Section 2.1.2 of the STLL
i. Automatic Isolation of the Block Valves

POOR ORIGINAL

I have also instructed Dr. Ross that I want the ability of the block valve to be isolated under dynamic conditions to be verified at each PWR.

There are other pertinent considerations including the apparent low challenge rate for PORVs in Westinghouse plants based on the data supplied to the NRC to date.

I have concluded that the proper approach is to have Westinghouse and Combustion Engineering perform a study on their plants to be completed by October 1, 1980 that will assess the performance of the PORV and demonstrate that the above modifications will reduce the likelihood of a stuck open PORV to an acceptable level. Pending these studies, I believe the above constitute a sufficient set of changes.

4. Technical Specifications on Allowable ECCS Outage Times

The recommendation to restrict the allowable outage times of the ECCS was only recently added to the draft. I have concluded that your recc endation provides an appropriate solution; however, I have decided that the best way of handling this recommendation is to refer it to DOR for action. I have requested Dr. Ross to forward this recommendation in a memorandum to Darrell Eisenhut.

In summary, your viewpoints regarding Items 1 and 2 above, will be documented in the final Bulletins and Orders Task Force Report, your recommendations on Item 3 will be reflected in other reports, and your recommendation on Item 4 will be forwarded to DOR for action. While I disagree with your recommendation to publish your version of the report, your views are appreciated and have received careful management review.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: R. Mattson
D. Ross
D. Eisenhut

POOR ORIGINAL

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	B&W before 5/1/79	after 5	--	High head HPI	Low Head HPI	
CHALLENGE RATE OF RELIEF VALVES (NO. OF VALVES OPENED PER REACTOR YEAR)	3	0.2	1	2*		15
FAILURE RATE OF RELIEF VALVES (NO. OF FAILURES TO CLOSE PER OPENING) failures to reclose	1/20		1/20	1/20*		1/20
SIZE OF RELIEF VALVE (IN^2)	1.1		1.4	1.4		14
CONSEQUENCE OF A STUCK OPEN RELIEF VALVE WITH SINGLE FAILURE ASSUMPTIONS	NO CORE UNCOVERY		NO CORE UNCOVERY	NO CORE UNCOVERY		PARTIAL CORE UNCOVERY
LIKELIHOOD OF EXTENDED LOSS OF NATURAL CIRCULATION	UNLIKELY		HIGHLY UNLIKELY	HIGHLY UNLIKELY		N/A
CONSEQUENCE OF A STUCK OPEN RELIEF VALVE WITH EXTENDED LOSS OF NATURAL CIRCULATION	** 40 min. available to initiate HPI		Possible Core Melt	60 min available to initiate HPI	Possible Core Melt	N/A
CONSEQUENCE OF TEMPORARY LOSS OF ALL FEEDWATER (FEEDWATER AND RCIC FOR BWRs)	*** 20 min. available to initiate HPI or AFW		30 min. available to initiate AFW	30 min. available to initiate HPI or AFW	30 min. available to initiate AFW	NO CORE UNCOVER
CONSEQUENCE OF STUCK OPEN RELIEF VALVE WITH TEMPORARY LOSS OF ALL FEEDWATER	*** 40 min. available to initiate HPI or AFW		60 min. available to initiate AFW	60 min. available to initiate HPI or AFW	60 min. available to initiate AFW	7 min. availabl to initiate ADS
CONSEQUENCE OF EXTENDED LOSS OF ALL FEEDWATER	** 20 min. available to initiate HPI		Possible Core Melt	30 min. available to initiate HPI	Possible Core Melt	20 min. availabl to initiate ADS
CONSEQUENCE OF STUCK OPEN RELIEF VALVE WITH EXTENDED LOSS OF ALL FEEDWATER	** 40 min. available to initiate HPI or AFW		Possible Core Melt	60 min. available to initiate HPI or AFW	Possible Core Melt	4 min. availabl to initiate AD

1. PWR calculations assumed prompt tripping of the reactor coolant pumps
2. Results apply only if accumulative outage time of ECCS is very small
3. Time available for operator action is based on no core uncovery

*Information incomplete, actual values might be more favorable

**In case of the Davis Besse plant the consequence is "possible core melt"

***In case of the Davis Besse plant AFW has to be initial

(1)

Appendix XVII

STAFF PRESENTATIONS ON B&OTF ACTIVITIES
FOR FEBRUARY 8, 1980 ACRS MEETING

- OVERVIEW (W. KANE)
- ACTIVITIES COMPLETED SINCE JANUARY 1980 ACRS MEETING
- B&OTF RECOMMENDATIONS RESULTING FROM GENERIC REVIEWS
    - B&W PLANTS (R. CAPRA)
    - GE PLANTS (C. THOMAS)
    - W PLANTS (P. O'REILLY)
    - CE PLANTS (I. VILLALVA)
- EFFECTIVENESS OF B&OTF RECOMMENDATIONS IN STRENGTHENING RELIABILITY OF W AND CE AFW SYSTEMS (M. TAYLOR)
- EFFECTIVENESS OF B&OTF RECOMMENDATIONS IN REDUCING THE PROBABILITY OF STUCK-OPEN PORVs (I. VILLALVA)
- RECOMMENDATIONS NOT INCLUDED IN B&OTF GENERIC REPORTS (W. KANE)

ACTIVITIES COMPLETED SINCE JANUARY ACRS MEETING

- NUREG-0645 (FINAL REPORT) - ISSUED TO ACRS 1/24/80
- NUREG-0611 (W REPORT) - ISSUED TO ACRS 1/24/80
- NUREG-0565 (B&W REPORT) - ISSUED TO ACRS 1/25/80
- NUREG-0635 (CE REPORT) - ISSUED TO ACRS 1/30/80
- NUREG-0626 (GE REPORT) - ISSUED TO ACRS 2/1/80
- B&O RECOMMENDATIONS INCORPORATED INTO TMI ACTION PLANS (NUREG-0660) AS ACTION PLAN II.K

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Appendix XVIII

VUGRAPHS PRESENTED AT THE ACRS MEETING

FEBRUARY 8, 1980

BY

M. TAYLOR, NRC (RES/PAS)

SUBJECT: EFFECTIVENESS OF THE B&O RECOMMENDATIONS FOR STRENGTHENING RELIABILITY OF AUXILIARY FEEDWATER SYSTEMS (W & CE NSSS)

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APPENDIX XVIII
BULLETINS AND ORDERS IMPLEMENTATION:
EFFECTIVENESS OF B&O RECOMMENDATIONS
FOR IMPROVING RELIABILITY OF AUXILIARY
FEEDWATER SYSTEMS

SUBJECT: EFFECTIVENESS OF B&O RECOMMENDATIONS FOR STRENGTHENING AUXILIARY FEEDWATER SYSTEMS RELIABILITY (W & CE NSSS)

BACKGROUND

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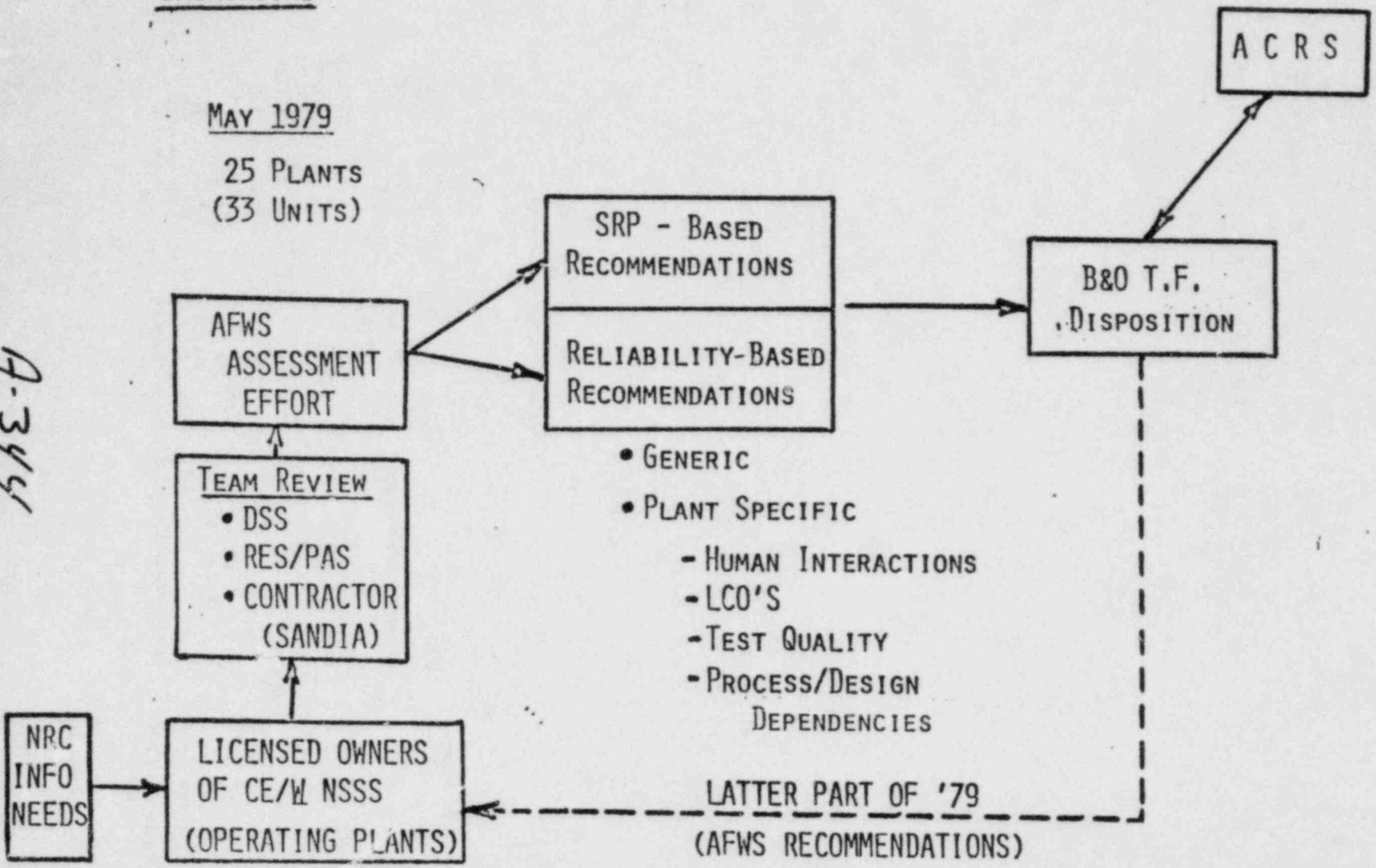

Results of Review

Generic Recommendations - Short Term

GS-1 Tech Spec LCO - Time limit on outage of 1 train

GS-2 Tech Spec Administrative Controls on manual valves - lock and verify valve position
- single suction lines and valves

GS-3 Re-evaluate AFWS flow limits to reduce AFWS water hammer occurrence

GS-4 Emergency procedure for connecting backup water source to AFWS pump suction

GS-5 Emergency procedures to assure necessary operator actions are taken to assure AFWS availability in event of AC blackout

GS-6 Tech Spec - AFWS flow verification to steam generator following maintenance outage which affects AFWS flow capability

GS-7 Non-Safety Grade AFWS Automatic Start Signals

GS-8 Automatic Actuation of AFWS

Generic Recommendations - Long Term

GL-1 AFWS systems should have automatic initiation. Retain manual start and stop capability with manual start as backup to automatic initiation

GL-2 Install redundant path (piping and valves) where primary and alternate water sources pass through single pipe and valve.

GL-3 Evaluate AFWS design to eliminate A-C dependency for one AFWS

GL-4 Evaluate AFWS design to prevent multiple pump damage due to dry pump operation resulting from natural phenomena damage (earthquake, tornado) to unprotected primary water supply concurrent with automatic pump start

GL-5 Provide Safety Grade AFWS Automatic Start Signals

Assessment Approach

- Engineering/Reliability Insights From Other Studies
- Simplified Event and Fault Tree Logic Models/Techniques
- Specific Data Base Compiled
- Short Term (Demand) Availability Considered in Quantitative Manner
- Point Value Estimates Made To <u>Roughly</u> Assess Dominance and Relative Contribution of Various Faults Identified
- Overall Categorization of AFWS Designs Into Various Availability Categories Based on Above
- Generic and Plant Specific Recommendations Based on Above (All AFW Systems)

Analysis Conditions

Loss of Main Feedwater (LMFW) Transients

Given:

- Case 1 — Offsite Power Available
- Case 2 — Loss of Offsite Power
- Case 3 — Loss of Offsite and Onsite AC Power

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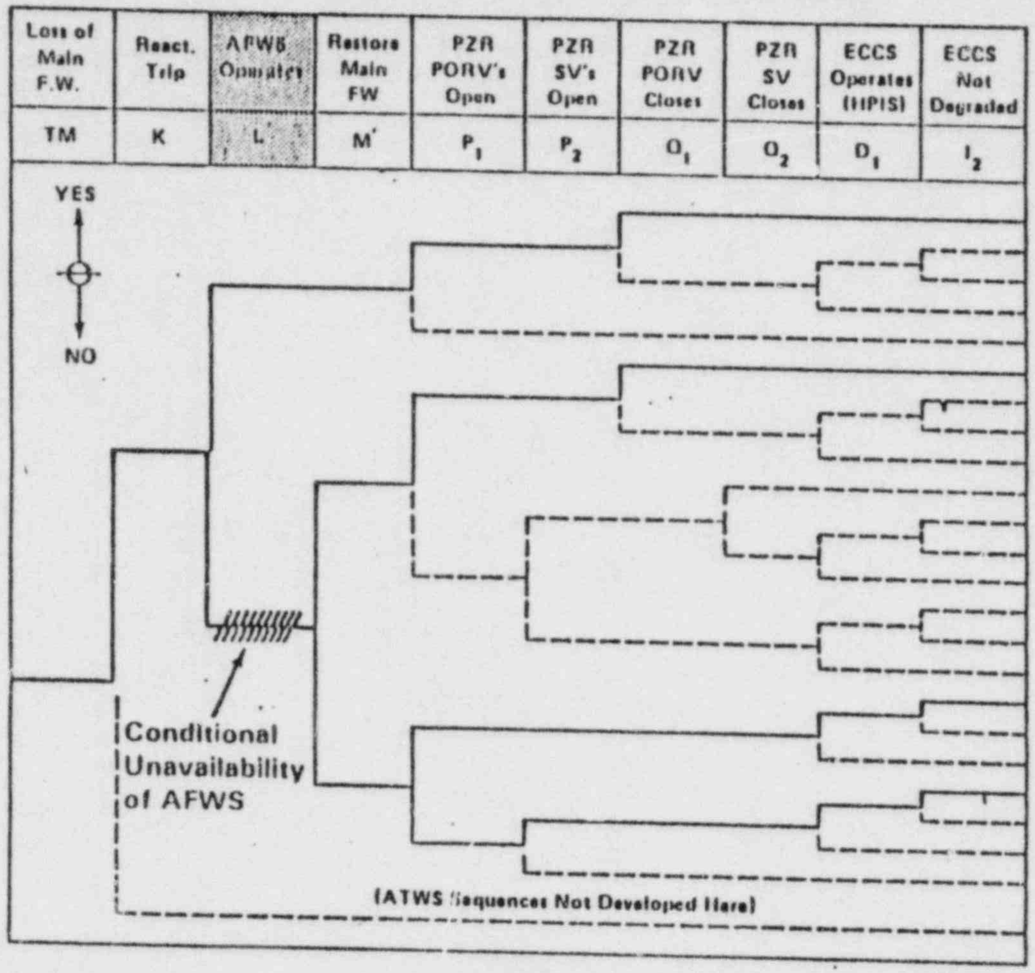
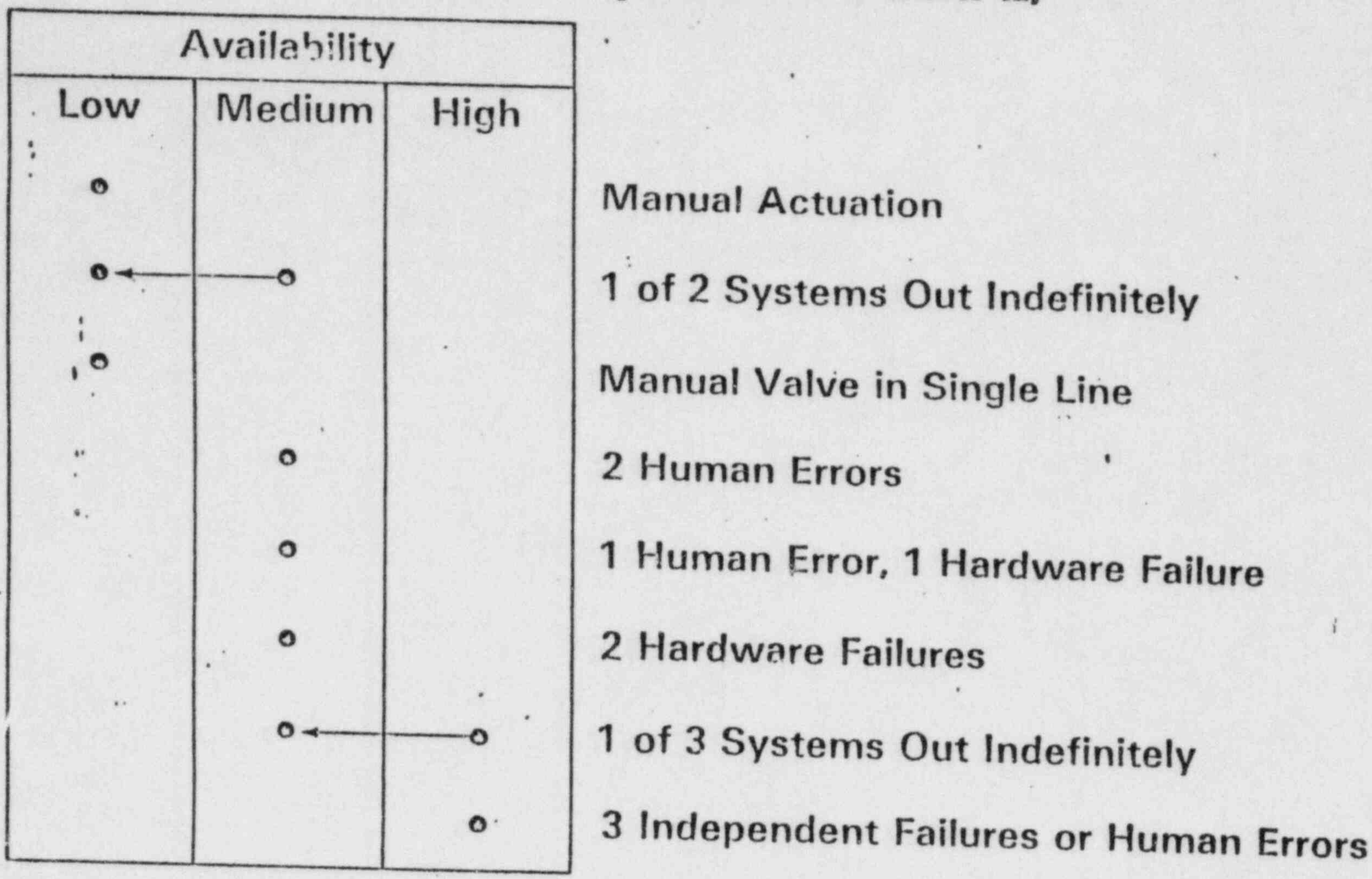
Example Event Tree: Transient Involving Loss of Main Feedwater — Offsite AC Power Available (Case of LMFW)



POOR ORIGINAL

Effect of Various Faults on AFWS Availability (Cases 1 and 2)



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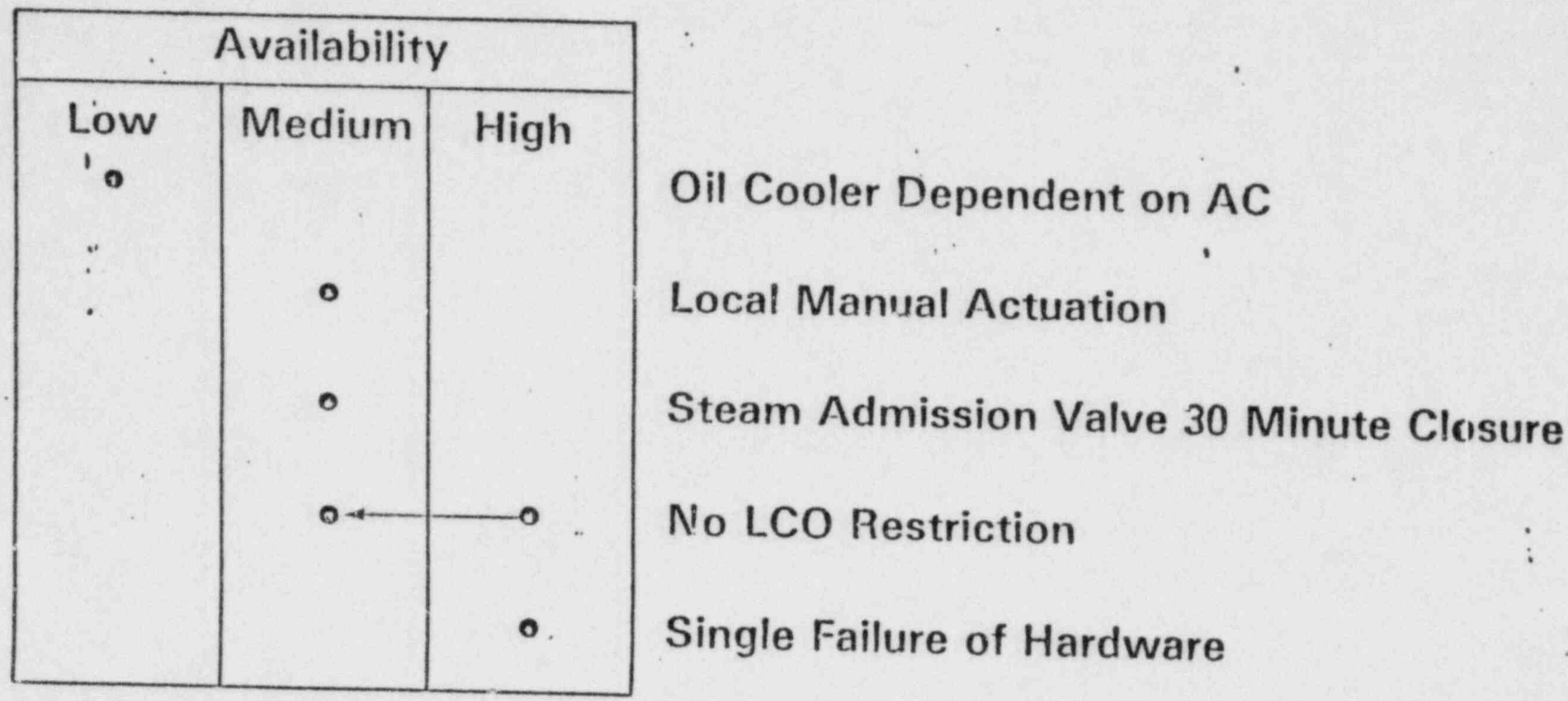
Effect of Various Faults on AFWS Availability (Case 3, Loss of All AC)



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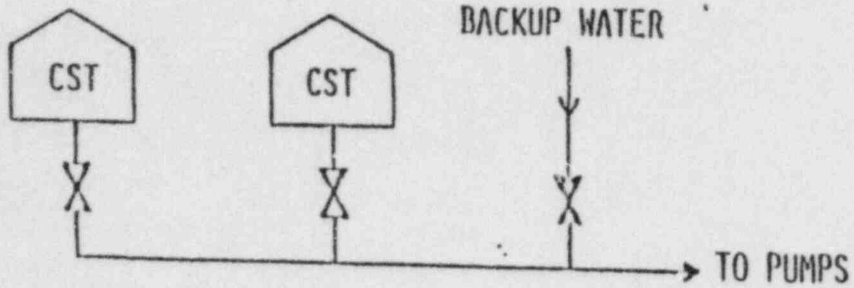
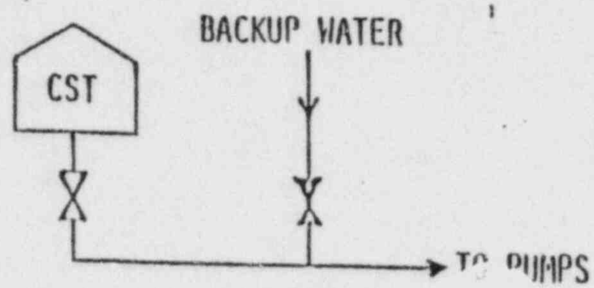
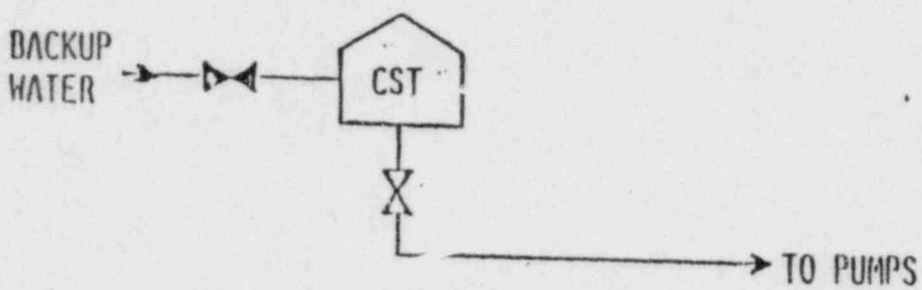
Types of Supply



REDUNDANT TANKS
WITH
BACKUP WATER
DOWNSTREAM



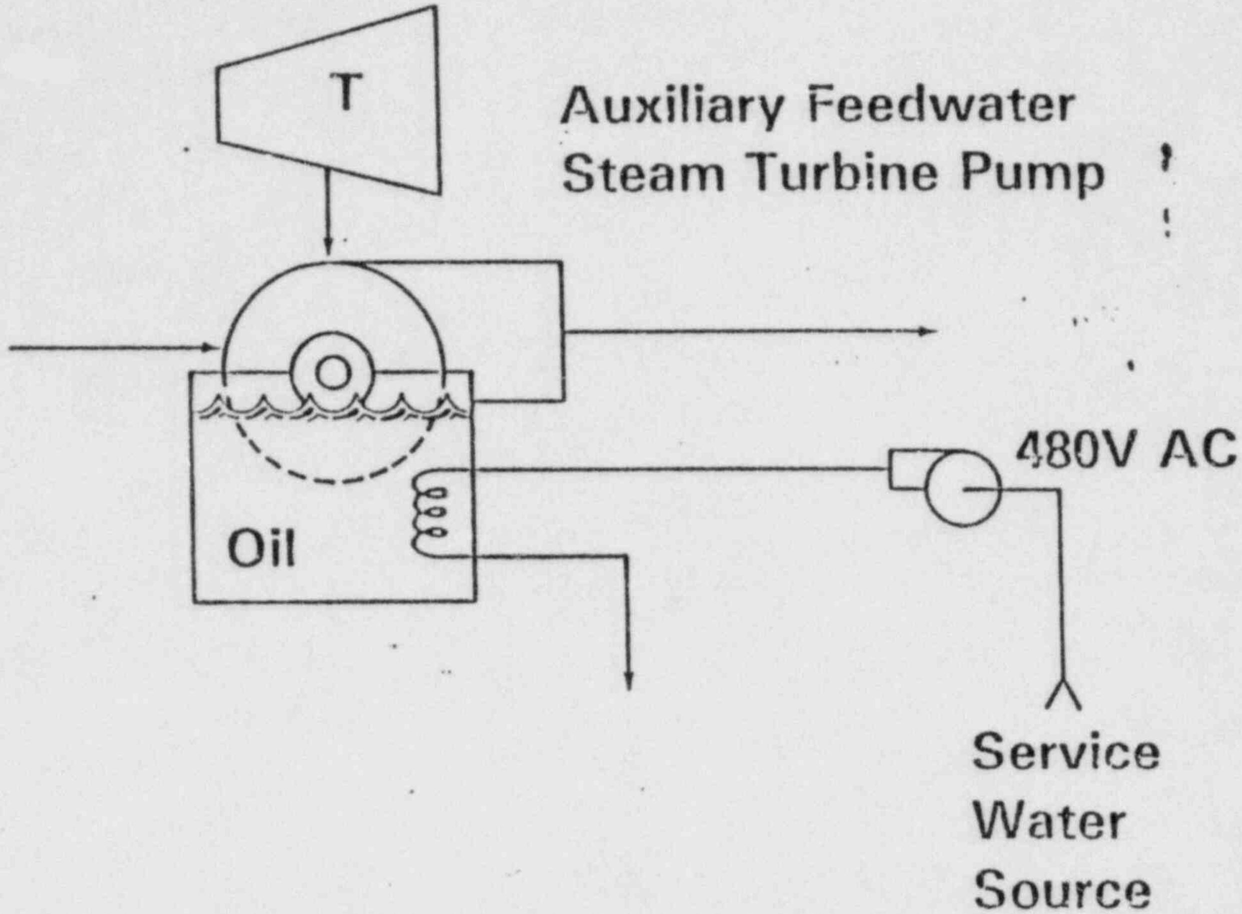
SINGLE TANK
WITH
BACKUP WATER
DOWNSTREAM



ALL SUPPLY
THROUGH
SINGLE TANK

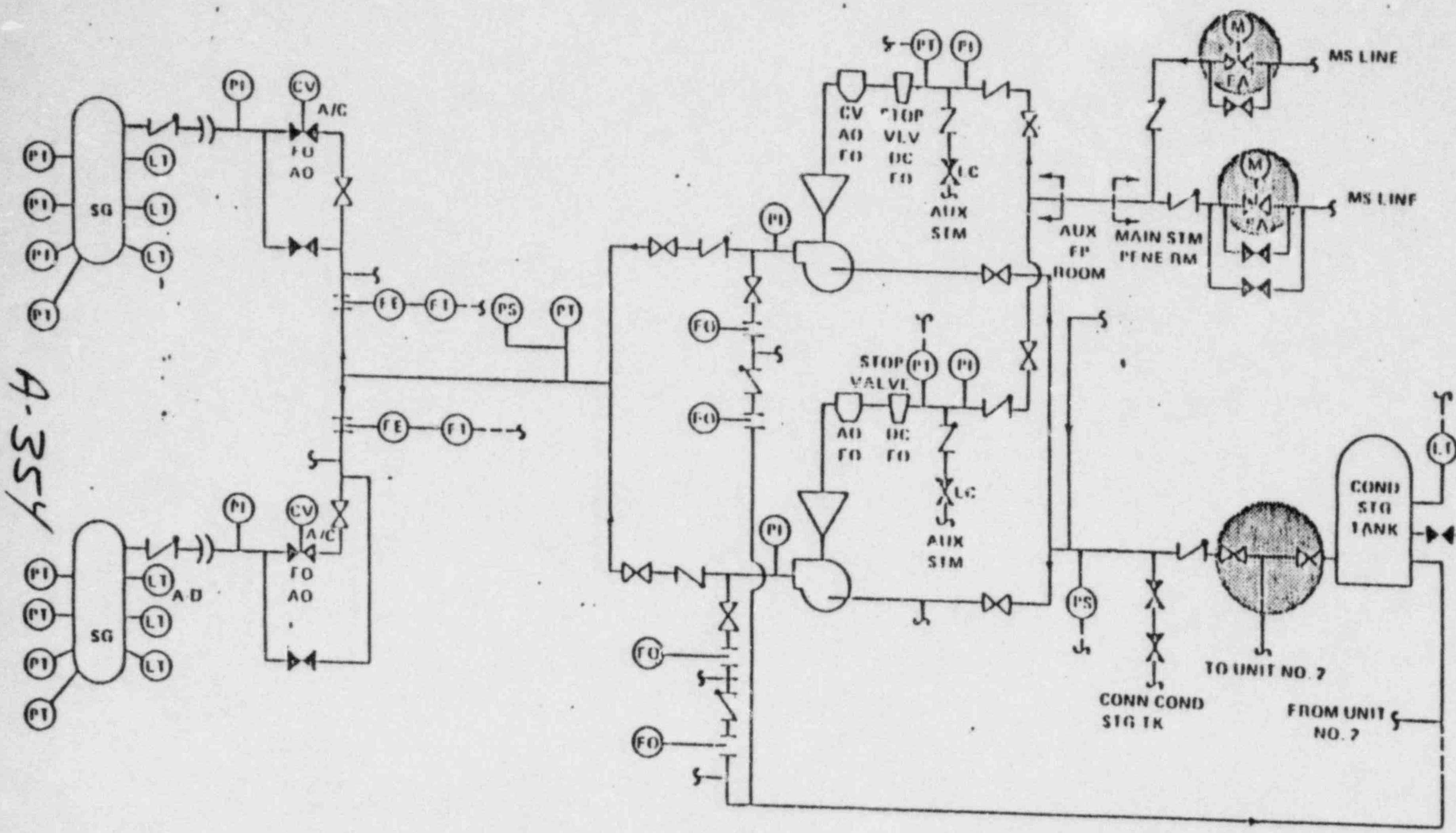
A-35-2

Oil Cooler AC Dependency

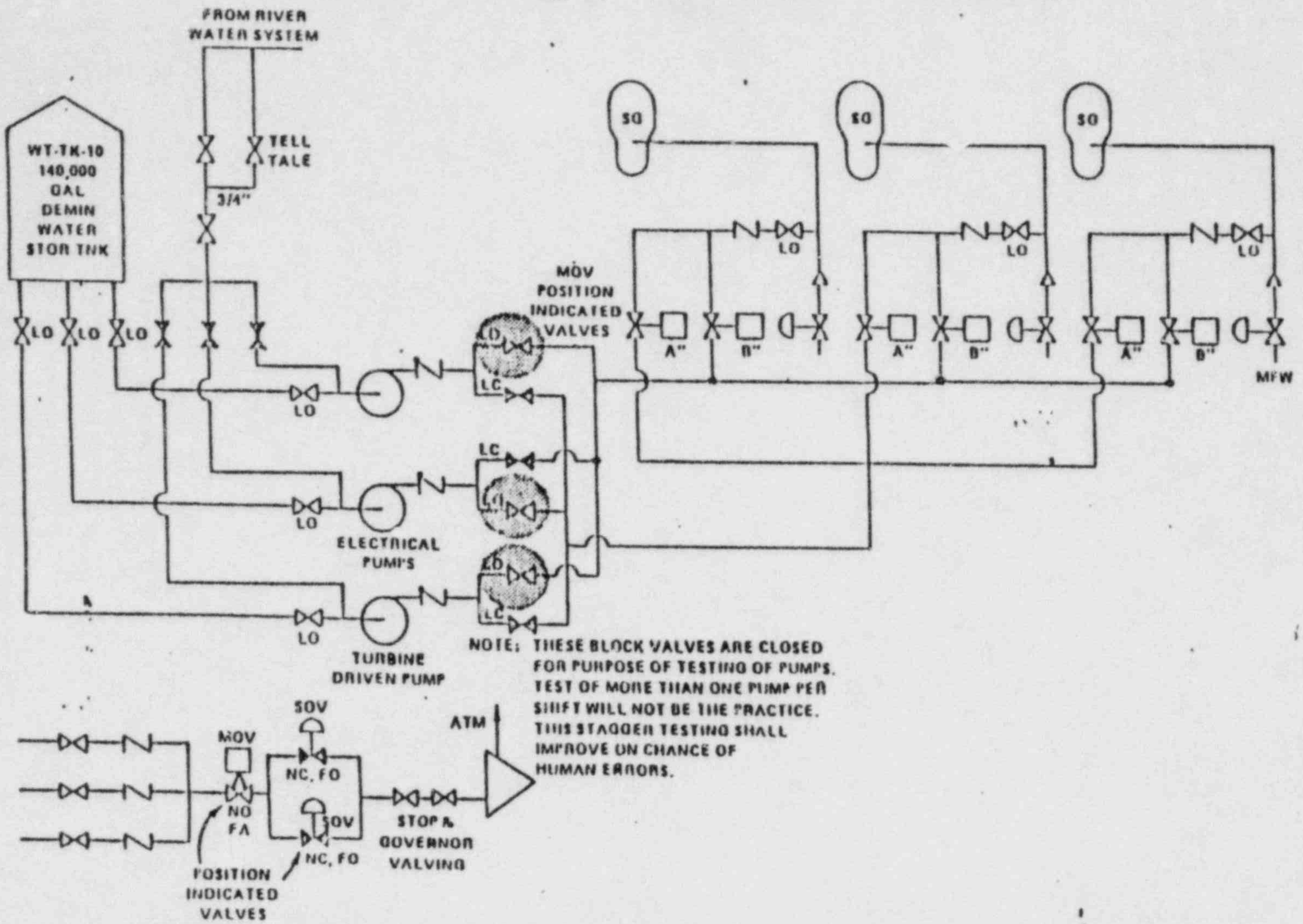


A-35-3

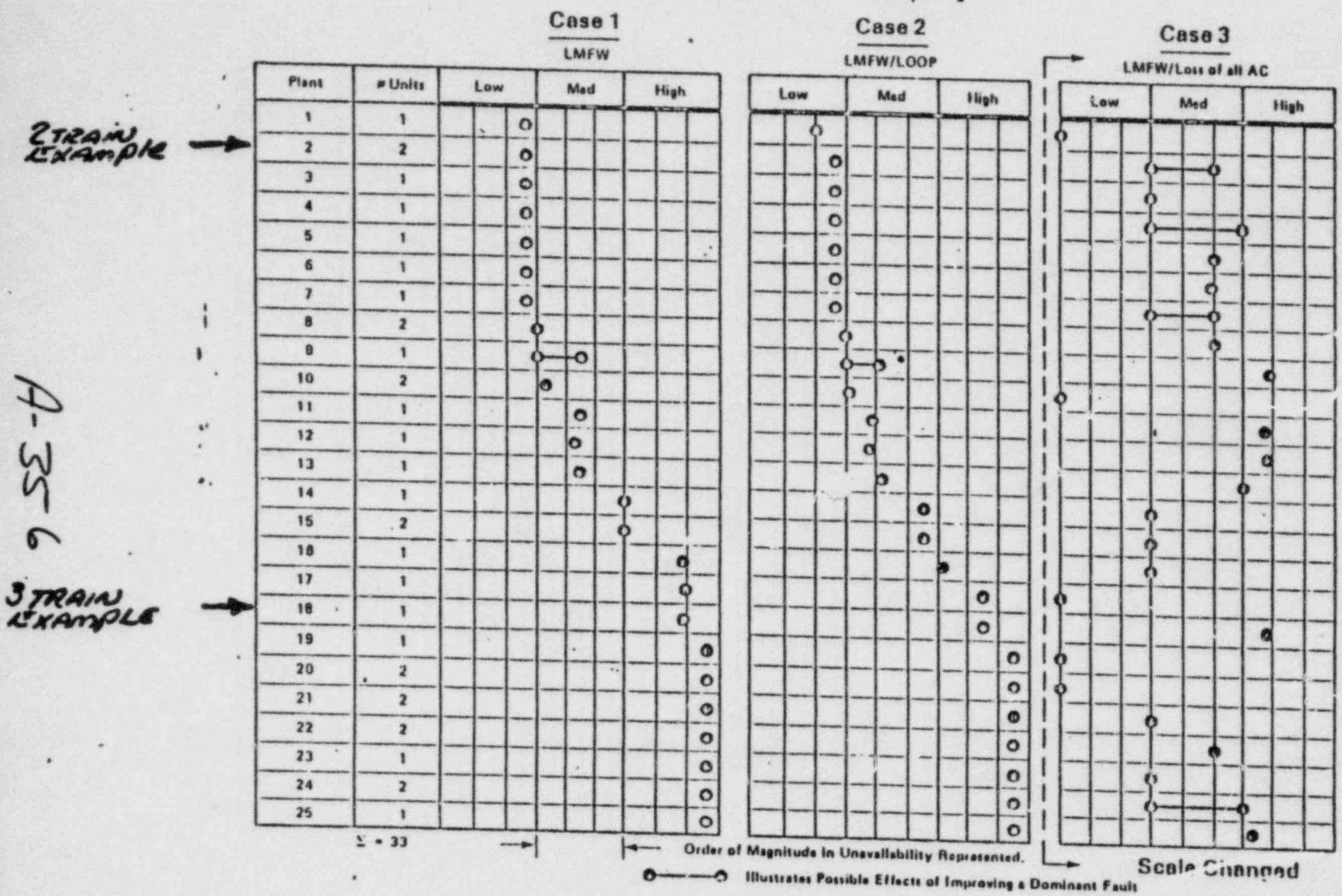
Example Auxiliary Feedwater System
("2 Train" Design, Manually Actuated)



Example AFWS
("3 Train" Design, Automatic Actuation)



A Generic Perspective: Comparisons of AFWS Reliability



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SOME CONCLUDING OBSERVATIONS

- WHEN IMPLEMENTED, THE RELIABILITY-BASED B&O RECOMMENDATIONS SHOULD SERVE OVERALL TO STRENGTHEN RELIABILITY OF AFWS ACROSS A LARGE POPULATION OF EXISTING DESIGNS

- EVENTUAL INCORPORATION OF THESE RECOMMENDATIONS INTO SRP FRAMEWORK SHOULD ALSO SERVE TO STRENGTHEN FUTURE NRC SAFETY REVIEWS AND RELIABILITY OF AFWS DESIGNS REVIEWED

- VARIABILITY IN THE EXISTING AFWS DESIGNS, IN THEIR OPERATION, AND THE PLANT TO PLANT VARIABILITY BEING OBSERVED IN COMPONENT FAILURE DATA PRECLUDE ANY HIGH DEGREE OF PRECISION BEING ATTACHED TO RELIABILITY IMPROVEMENTS TO BE GAINED THROUGH THE B&O RECOMMENDATIONS

- IF ONE DESIRES AN OVERALL QUANTITATIVE CHARACTERIZATION ON THE IMPROVEMENT POTENTIAL OF RELIABILITY-BASED B&O RECOMMENDATIONS, THEN REDUCTIONS IN AFWS UNAVAILABILITY OF 2 TO 10 MIGHT BE AN APPROPRIATE CHARACTERIZATION. (<u>NOT ALL AFWS DESIGNS WILL REALIZE VERY LARGE IMPROVEMENT</u>)

- THE QUESTION "HOW RELIABLE IS ENOUGH?" FOR THE AFWS IS YET TO BE ANSWERED

Appendix XIX

APPENDIX XIX
BULLETINS AND ORDERS IMPLEMENTATION:
HANDOUT PROVIDED BY NRC STAFF BUT NOT
DISCUSSED

ON THE EFFECTIVENESS OF THE B&O RECOMMENDATIONS IN REDUCING THE LIKELIHOOD OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS DUE TO STUCK-OPEN PORV'S

SLIDES PRESENTED AT
THE 8 FEBRUARY 1980 ACRS MEETING
BY

I. VILLALVA

SUMMARY OF RECOMMENDATIONS

THE RECOMMENDATIONS MADE BY THE BULLETINS AND ORDERS TASK FORCE INCLUDE BOTH SHORT-TERM (E.G., THOSE TO BE IMPLEMENTED BY JANUARY 1980) AS WELL AS LONG-TERM (E.G., THOSE TO BE IMPLEMENTED BY APPROXIMATELY JANUARY 1981) RECOMMENDATIONS. THUS, THE ASSESSMENT OF IMPROVEMENTS INCLUDES THE EFFECTIVENESS OF BOTH THE SHORT-TERM AND THE LONG-TERM RECOMMENDATIONS.

IN ADDITION, THE RECOMMENDATIONS ARE OF TWO TYPES:

I. HARDWARE-TYPE RECOMMENDATIONS,

AND

II. SOFTWARE-TYPE RECOMMENDATIONS.

HARDWARE-TYPE RECOMMENDATIONS

THE HARDWARE-TYPE RECOMMENDATIONS INCLUDE:

(A) <u>PLACING THE PORV'S AND BLOCK VALVES ON EMERGENCY POWER</u> — (THIS RECOMMENDATION IS SIMILAR TO RECOMMENDATION NO. 2.1.1 OF NUREG-0578);

(B) <u>DIRECT POSITION INDICATION OF PORV'S</u> — (THIS RECOMMENDATION IS SIMILAR TO RECOMMENDATION NO. 2.1.3.a OF NUREG-0578);

(C) <u>AUTOMATICALLY ISOLATING THE PORV'S ON LOW REACTOR SYSTEM PRESSURE</u> AND

(D) <u>DERIVATIVE "FIX"</u> — (THIS RECOMMENDATION IS APPLICABLE TO <u>W</u>-DESIGNED PLANTS ONLY. IN BRIEF, IT INVOLVES RAISING THE TRIP PRESSURE SETPOINT ON THE PID CONTROLLER WHICH IS USED ON MOST <u>W</u>-DESIGNED PLANTS. SINCE THIS RECOMMENDATION TENDS TO BE PLANT-SPECIFIC, AND SINCE IT HAS BEEN IMPLEMENTED IN THE AFFECTED PLANTS PER A SIMILAR RECOMMENDATION MADE BY WESTINGHOUSE, ITS EFFECTIVENESS WAS NOT ASSESSED.)

PLUS

(E) <u>CHANGING PORV SETPOINT</u> — THE PORV SETPOINT WAS RAISED ON THE B&W PLANTS ONLY.

SOFTWARE-TYPE RECOMMENDATIONS

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THE SOFTWARE-TYPE RECOMMENDATIONS INCLUDE:

(A) QUALIFICATIONS – (THIS RECOMMENDATION IS SIMILAR TO RECOMMENDATION NO. 9 IN NUREG-0585, I.E., EVALUATING INTERACTIONS OF NON-SAFETY AND SAFETY SYSTEMS AND PROPER QUALIFICATION OF SAFETY SYSTEMS);

(B) OPERATIONAL PROCEDURES AND TRAINING – (THIS RECOMMENDATION PERTAINS TO NEW GUIDELINES AND PROCEDURES TO MORE READILY IDENTIFY SMALL-BREAK LOCA'S AND TRAINING INVOLVING THE USE OF SEVERAL PARAMETERS, THE SATURATION METER AND DIRECT POSITION INDICATOR ON PORV'S IN DIAGNOSING SMALL-BREAK LOCA's);

(C) McGUIRE CONCERN – (THIS MATTER INVOLVES THE FAILURE OF A PORV SUPPLIED BY CCI ON A SPECIFIC PLANT DURING TESTING. BECAUSE OF THE SPECIFICITY OF THIS MATTER, THE EFFECTS OF THE "FIX" WERE NOT ASSESSED);

(D) FAILURE TO CLOSE "EAL" – (THIS RECOMMENDATION INVOLVES THE PROMPT REPORTING OF PORV FAILURES IN CONFORMANCE WITH THE EMERGENCY ACTION LEVELS STATED IN NUREG-0610); AND

(E) SHORT-TERM LESSONS LEARNED (STLL) RESEARCH – (THIS RECOMMENDATION INVOLVES THE TESTING OF RELIEF AND SAFETY VALVES IN CONFORMANCE WITH RECOMMENDATION NUMBER 2.1.1 OF NUREG-0578, INCLUDING THE TESTING OF VALVES UNDER THEIR EXPECTED DYNAMIC OPERATING CONDITIONS SUCH AS TWO-PHASE FLUID SLUG FLOW).

SUMMARY OF THE ESTIMATED EFFECTIVENESS OF THE B&O RECOMMENDATIONS IN REDUCING THE LIKELIHOOD OF SMALL-BREAK LOCA'S IN OPERATING PLANTS

I. PROBABILITY (P) OF SUCH LOCA'S PRIOR TO THE B&O RECOMMENDATIONS:

A. $P \sim 10^{-1}$ FOR B&W PLANTS.

B. $P \sim 6 \times 10^{-3}$ FOR C-E AND W PLANTS.

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II. PROBABILITY (P') OF SUCH LOCA'S SUBSEQUENT TO THE SHORT-TERM RECOMMENDATIONS:

A. $P' \sim 5 \times 10^{-3}$ FOR B&W PLANTS.

B. $P' \sim 2 \times 10^{-3}$ FOR C-E AND W PLANTS.

III. PROBABILITY (P'') OF SUCH LOCA'S SUBSEQUENT TO THE LONG-TERM B&O RECOMMENDATIONS:

A. $P'' \sim 2 \times 10^{-4}$ FOR B&W PLANTS.

B. $P'' \sim 10^{-4}$ FOR C-E AND W PLANTS.

LOGIC DIAGRAMS USED TO ASSESS THE EFFECTIVENESS OF THE BULLETINS AND ORDERS TASK FORCE'S RECOMMENDATIONS IN REDUCING THE LIKELIHOOD OF SMALL-BREAK LOCA'S DUE TO STUCK-OPEN PORV'S

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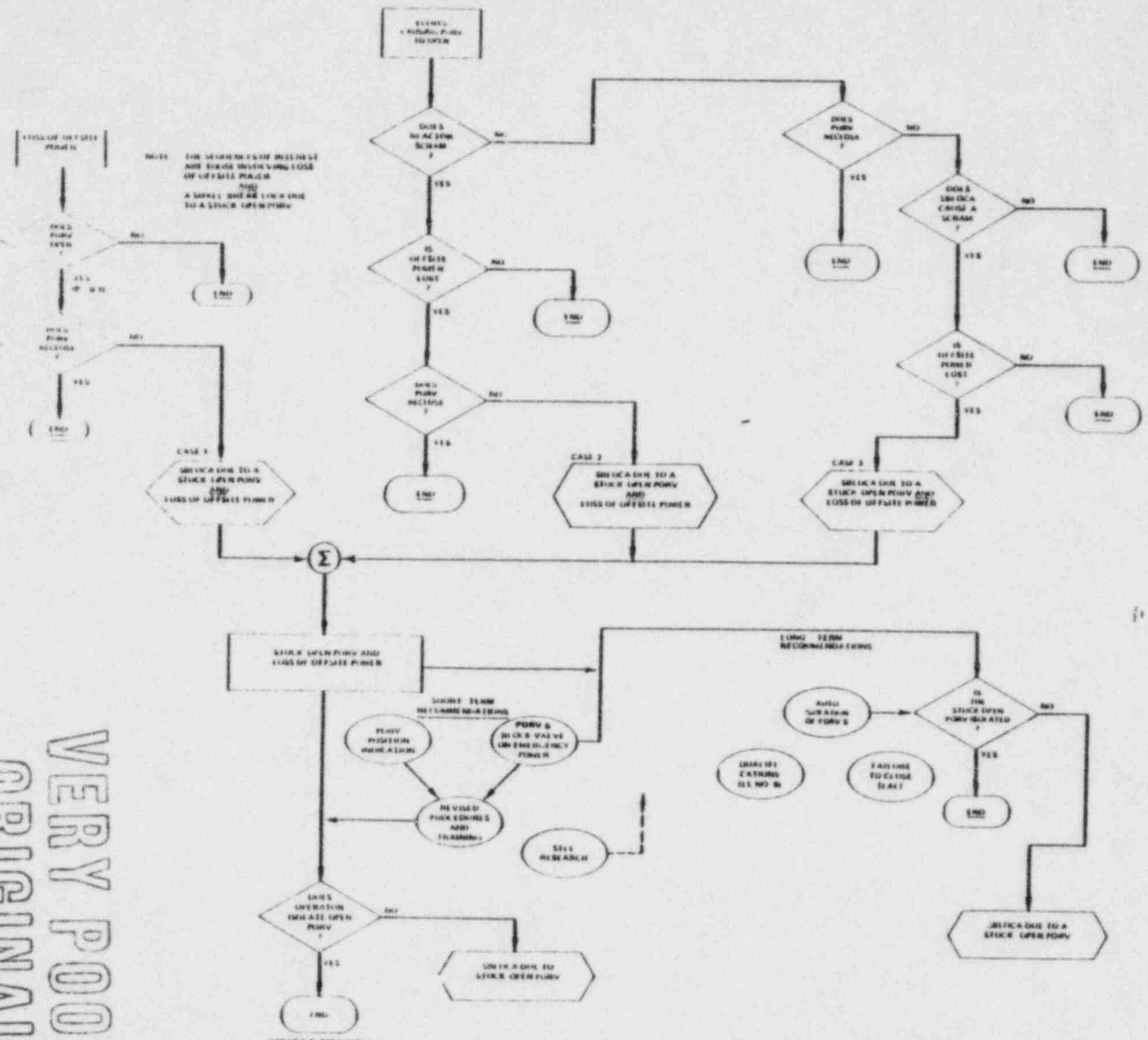
NOTE: The Probability Numbers Used in the Logic Diagrams are Based on Limited Statistical Data and the Author's Engineering Judgment, Both of Which are Subject to Bias. Thus, a Wide Band of Uncertainty Exists on the Actual Results Obtained. The Actual Results, Therefore, Should be Considered to be a First Approximation of the Improvements.

ASSESSMENT OF CITED RECOMMENDATION IN REDUCING THE LIKELIHOOD OF SMALL-BREAK LOCA'S DUE TO STUCK-OPEN PORV'S



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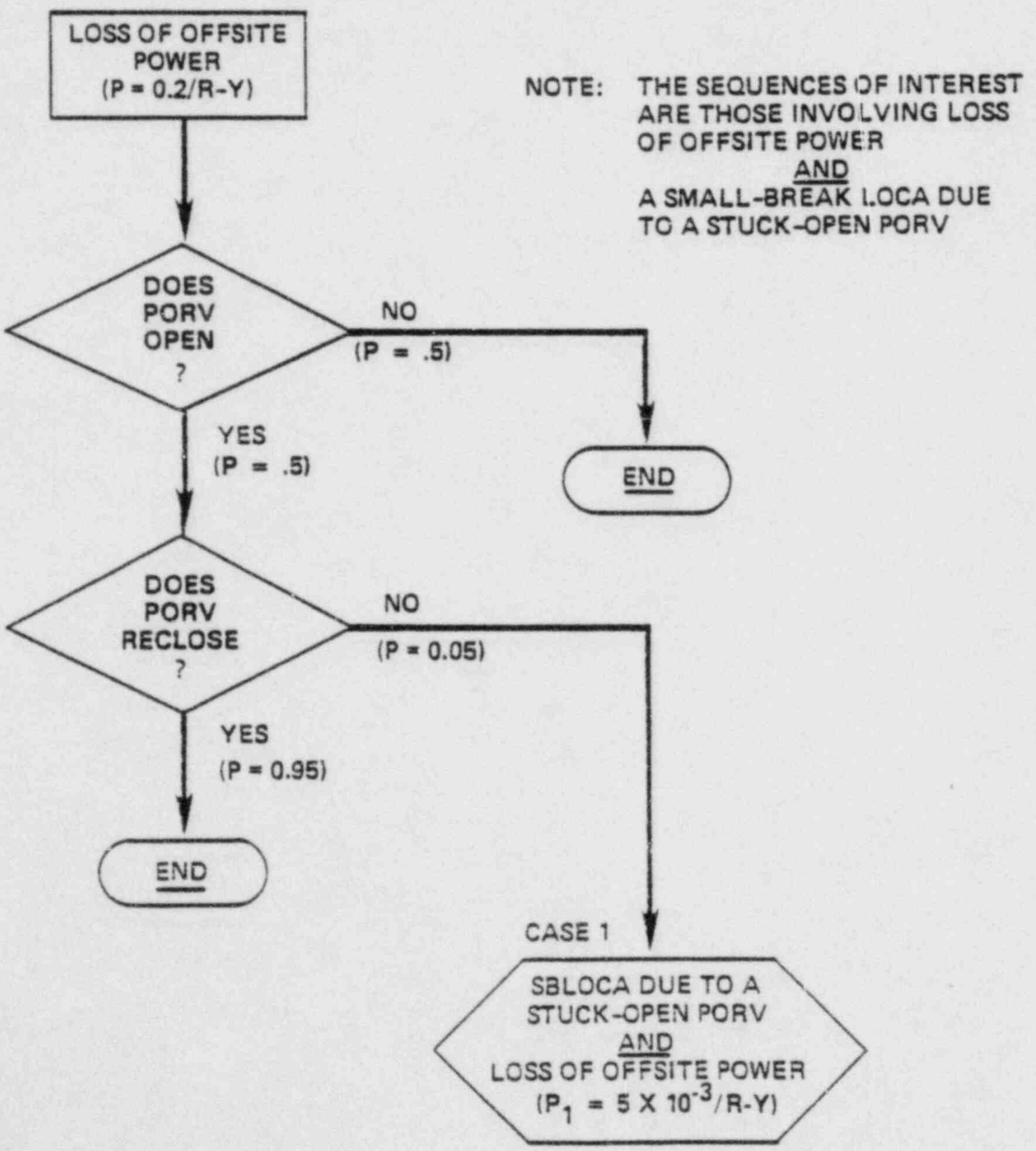
VERY POOR ORIGINAL

SHEET 1

(PRE-TMI)
INITIATING EVENTS LEADING TO STUCK-OPEN PORVS IN B&W PLANTS



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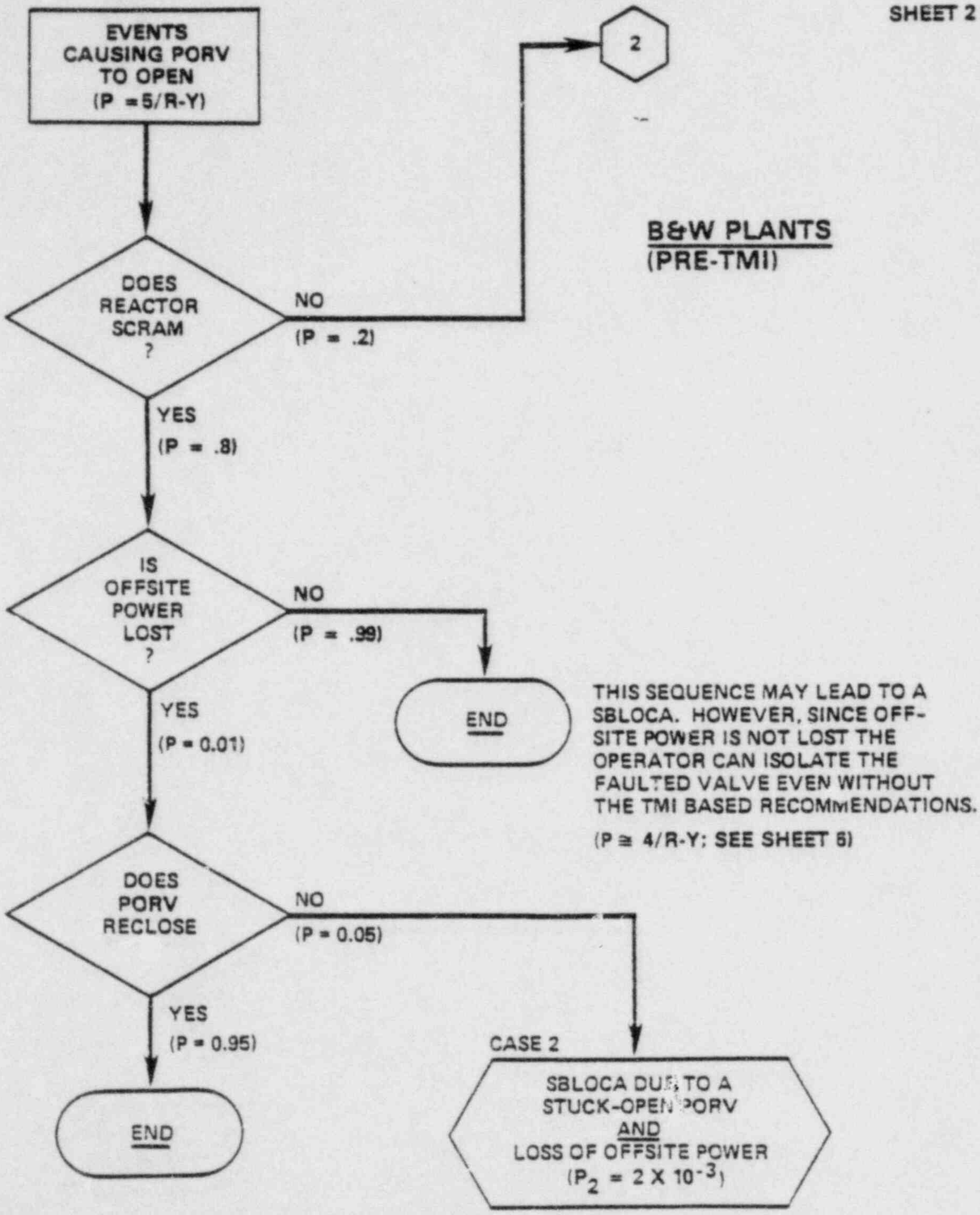
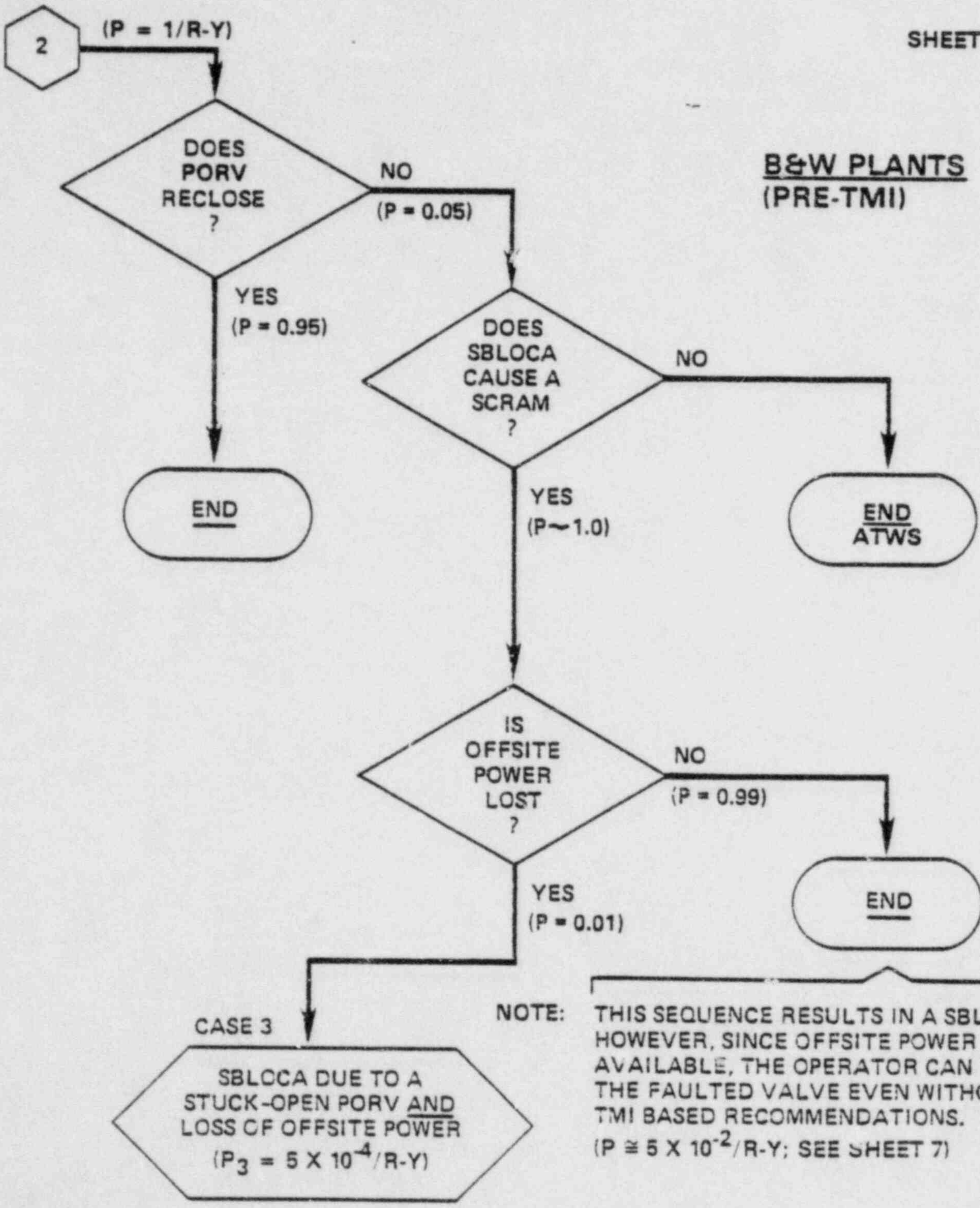

EVENTS CAUSING PORV TO OPEN (P = 5/R-Y)
2
B&W PLANTS
(PRE-TMI)
DOES REACTOR SCRAM ?
NO (P = .2)
YES (P = .8)
IS OFFSITE POWER LOST ?
NO (P = .99)
END
THIS SEQUENCE MAY LEAD TO A SBLOCA. HOWEVER, SINCE OFF-SITE POWER IS NOT LOST THE OPERATOR CAN ISOLATE THE FAULTED VALVE EVEN WITHOUT THE TMI BASED RECOMMENDATIONS.
(P ≅ 4/R-Y; SEE SHEET 6)
YES (P = 0.01)
DOES PORV RECLOSE
NO (P = 0.05)
YES (P = 0.95)
END
CASE 2
SBLOCA DUE TO A STUCK-OPEN PORV AND LOSS OF OFFSITE POWER (P_2 = 2 X 10^{-3})

SHEET 3

<u>B&W PLANTS</u>
(PRE-TMI)

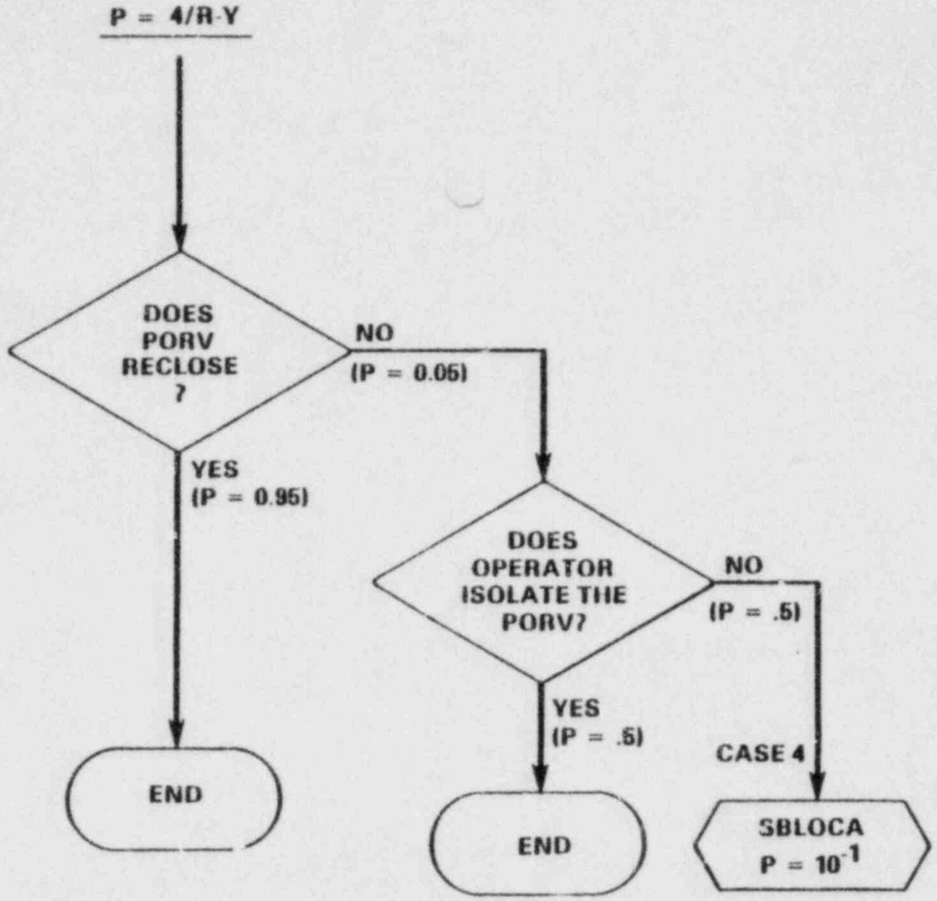


NOTE: THIS SEQUENCE RESULTS IN A SBLOCA; HOWEVER, SINCE OFFSITE POWER IS AVAILABLE, THE OPERATOR CAN ISOLATE THE FAULTED VALVE EVEN WITHOUT THE TMI BASED RECOMMENDATIONS.
(P ≅ 5 X 10^{-2}/R-Y; SEE SHEET 7)

B&W PLANTS (PRE-TMI)

INITIATING SEQUENCE: (FROM SHEET 2) **SHEET 6**

PORV OPENS; REACTOR SCRAMS;
OFFSITE POWER REMAINS INTACT.



PRIOR TO B&O RECOMMENDATIONS

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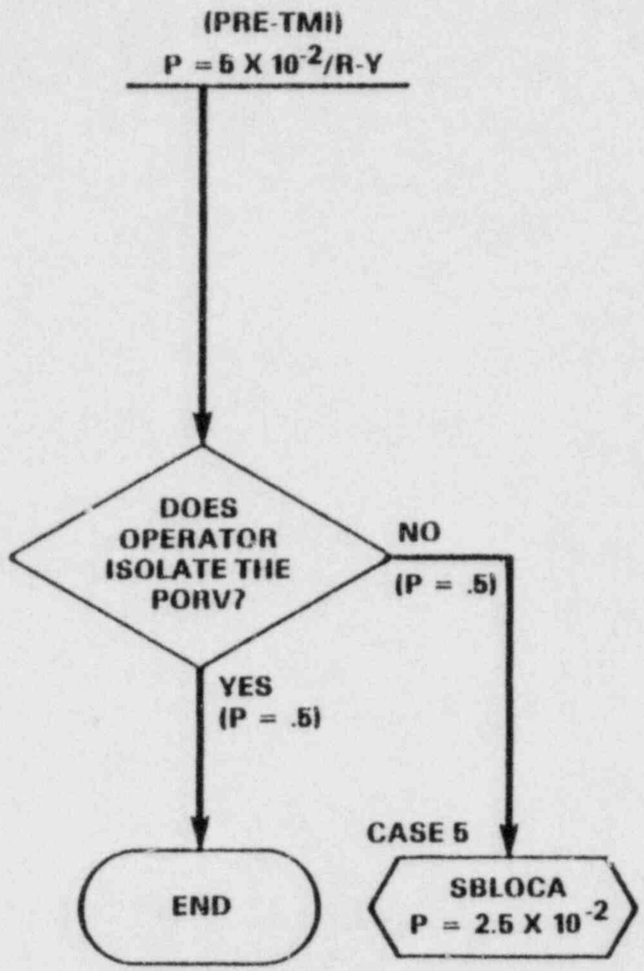
B&W PLANTS (PRE-TMI)

INITIATING SBLOCA SEQUENCE: (FROM SHEET 3)

SHEET 7

PORV OPENS; REACTOR DOES NOT SCRAM; PORV DOES NOT RECLOSE; REACTOR SCRAMS ON SBLOCA; OFFSITE POWER REMAINS INTACT.



PRIOR TO B&O RECOMMENDATIONS

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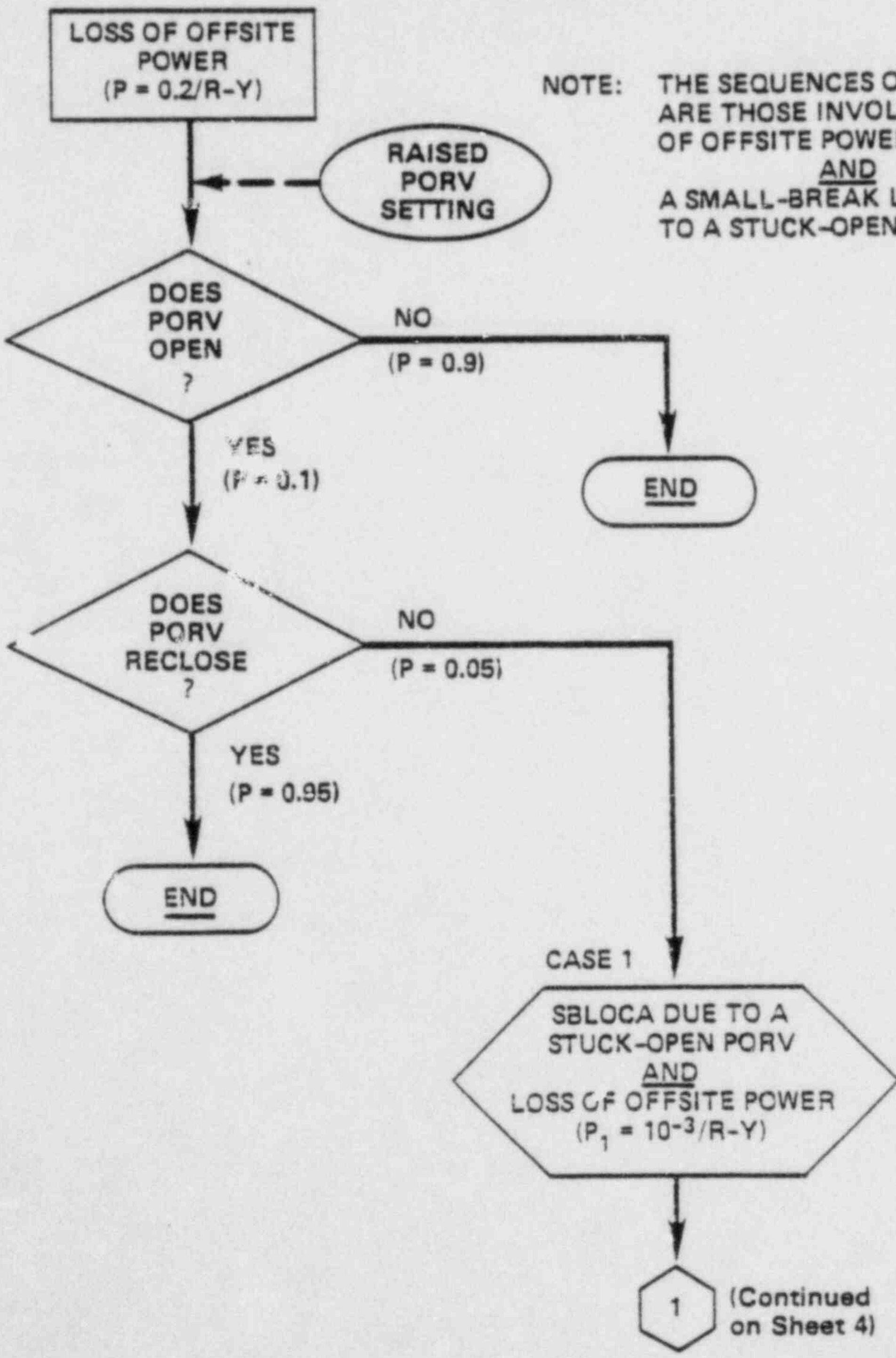
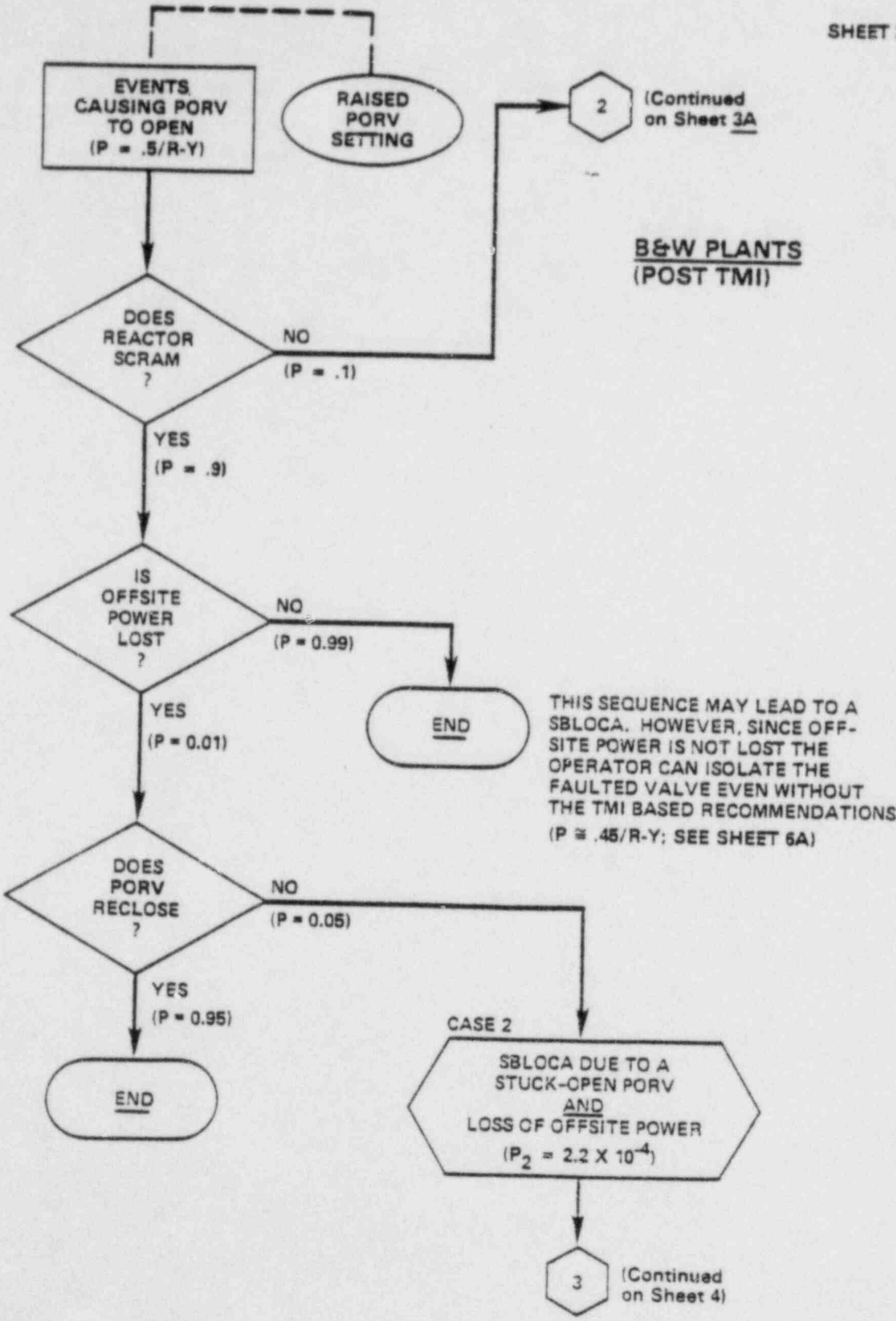
SHEET 1A

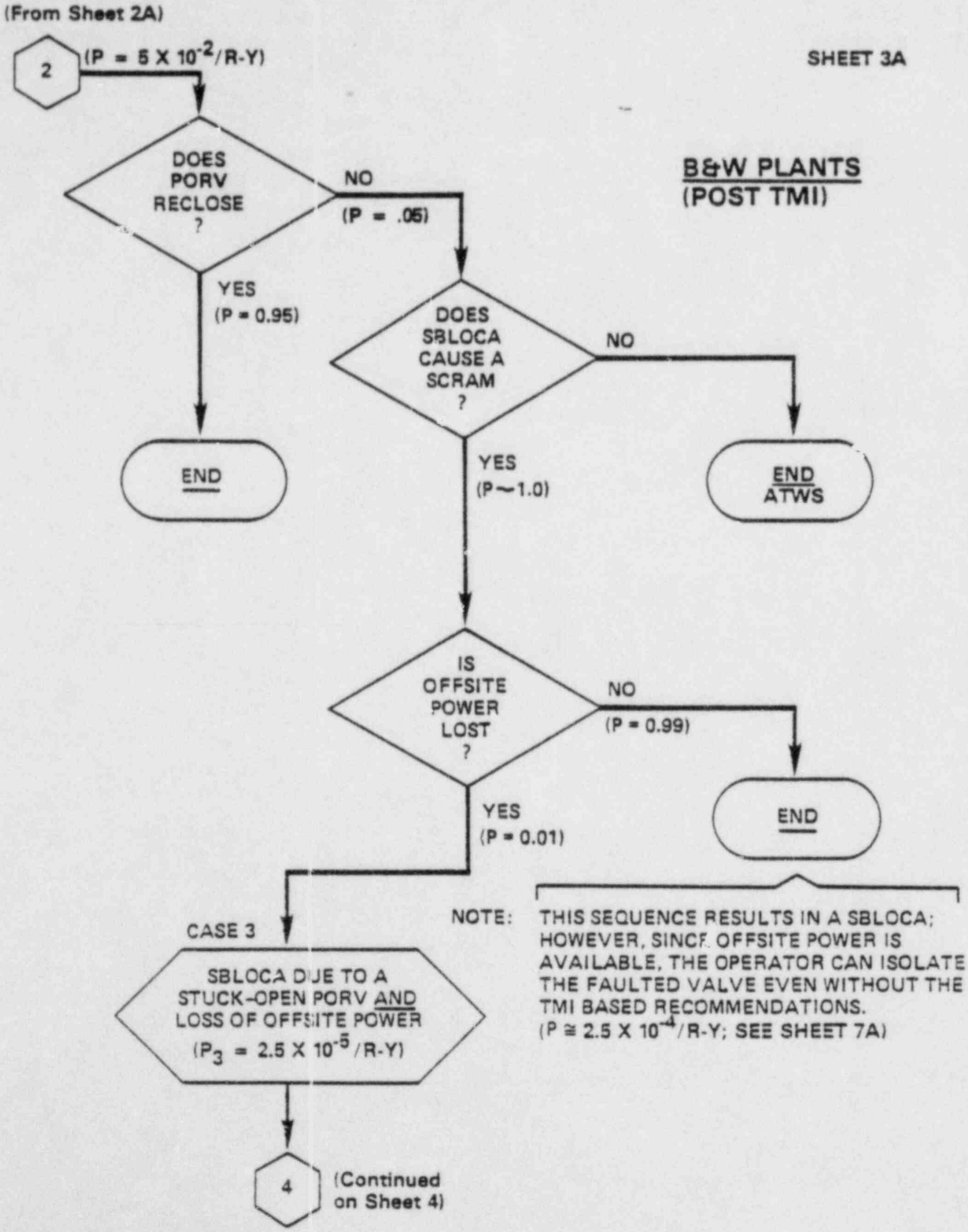
(POST TMI) INITIATING EVENTS LEADING TO STUCK-OPEN PORVS IN B&W PLANTS



NOTE: THE SEQUENCES OF INTEREST ARE THOSE INVOLVING LOSS OF OFFSITE POWER AND A SMALL-BREAK LOCA DUE TO A STUCK-OPEN PORV

(Continued on Sheet 4)


EVENTS CAUSING PORV TO OPEN (P = .5/R-Y)
RAISED PORV SETTING
2
(Continued on Sheet 3A
B&W PLANTS (POST TMI)
DOES REACTOR SCRAM ?
NO
(P = .1)
YES
(P = .9)
IS OFFSITE POWER LOST ?
NO
(P = 0.99)
END
THIS SEQUENCE MAY LEAD TO A SBLOCA. HOWEVER, SINCE OFF-SITE POWER IS NOT LOST THE OPERATOR CAN ISOLATE THE FAULTED VALVE EVEN WITHOUT THE TMI BASED RECOMMENDATIONS.
(P ≅ .45/R-Y; SEE SHEET 6A)
YES
(P = 0.01)
DOES PORV RECLOSE ?
NO
(P = 0.05)
YES
(P = 0.95)
END
CASE 2
SBLOCA DUE TO A STUCK-OPEN PORV AND LOSS OF OFFSITE POWER
(P_2 = 2.2 X 10^{-4})
3
(Continued on Sheet 4)

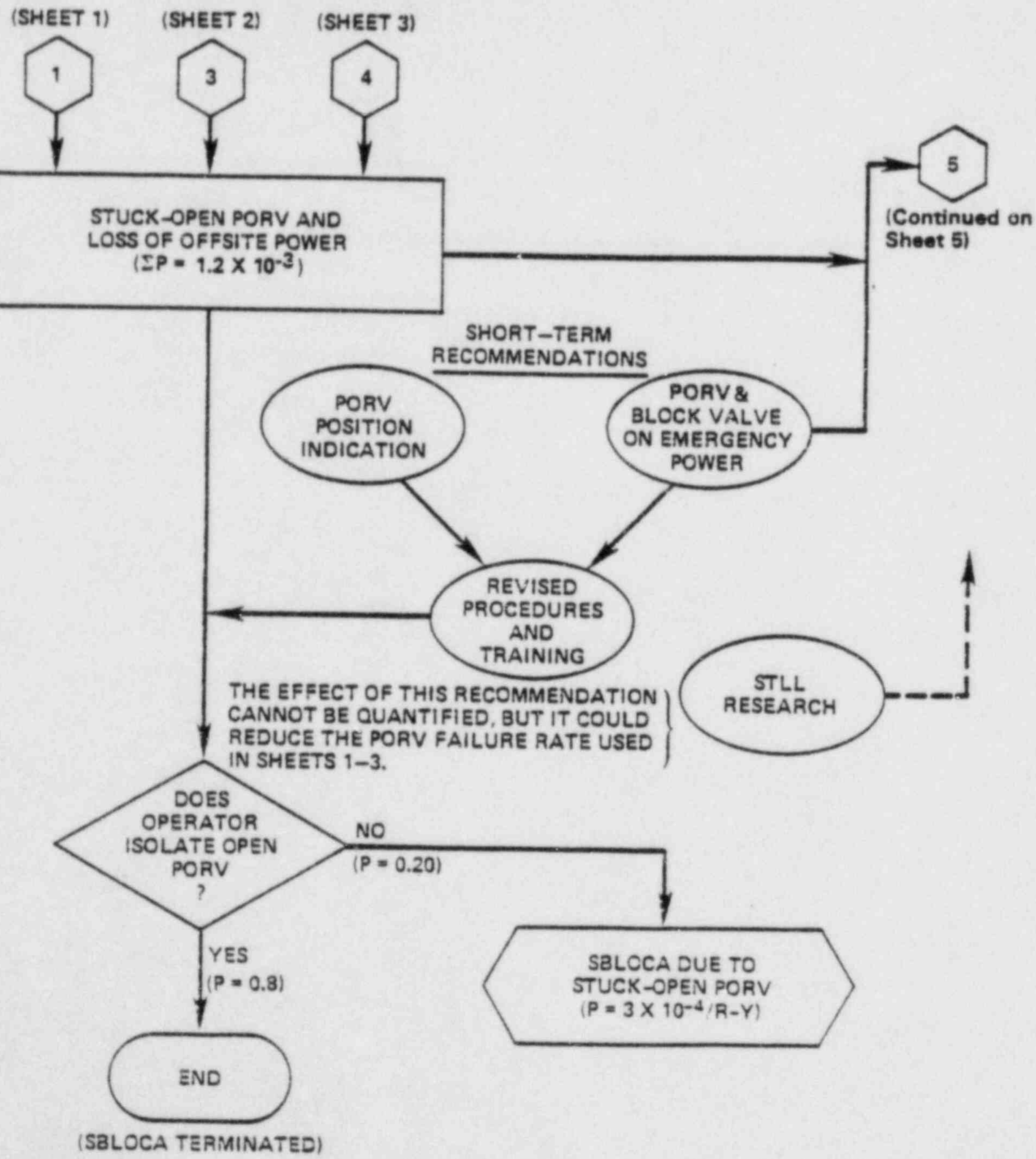


(From Sheet 2A)
SHEET 3A
B&W PLANTS
(POST TMI)
2
(P = 5 X 10^{-2}/R-Y)
DOES PORV RECLOSE ?
NO
(P = .05)
YES
(P = 0.95)
END
DOES SBLOCA CAUSE A SCRAM ?
NO
END ATWS
YES
(P~1.0)
IS OFFSITE POWER LOST ?
NO
(P = 0.99)
END
YES
(P = 0.01)
CASE 3
SBLOCA DUE TO A STUCK-OPEN PORV AND LOSS OF OFFSITE POWER
(P_3 = 2.5 X 10^{-5}/R-Y)
NOTE: THIS SEQUENCE RESULTS IN A SBLOCA; HOWEVER, SINCE OFFSITE POWER IS AVAILABLE, THE OPERATOR CAN ISOLATE THE FAULTED VALVE EVEN WITHOUT THE TMI BASED RECOMMENDATIONS.
(P ≅ 2.5 X 10^{-4}/R-Y; SEE SHEET 7A)
4
(Continued on Sheet 4)

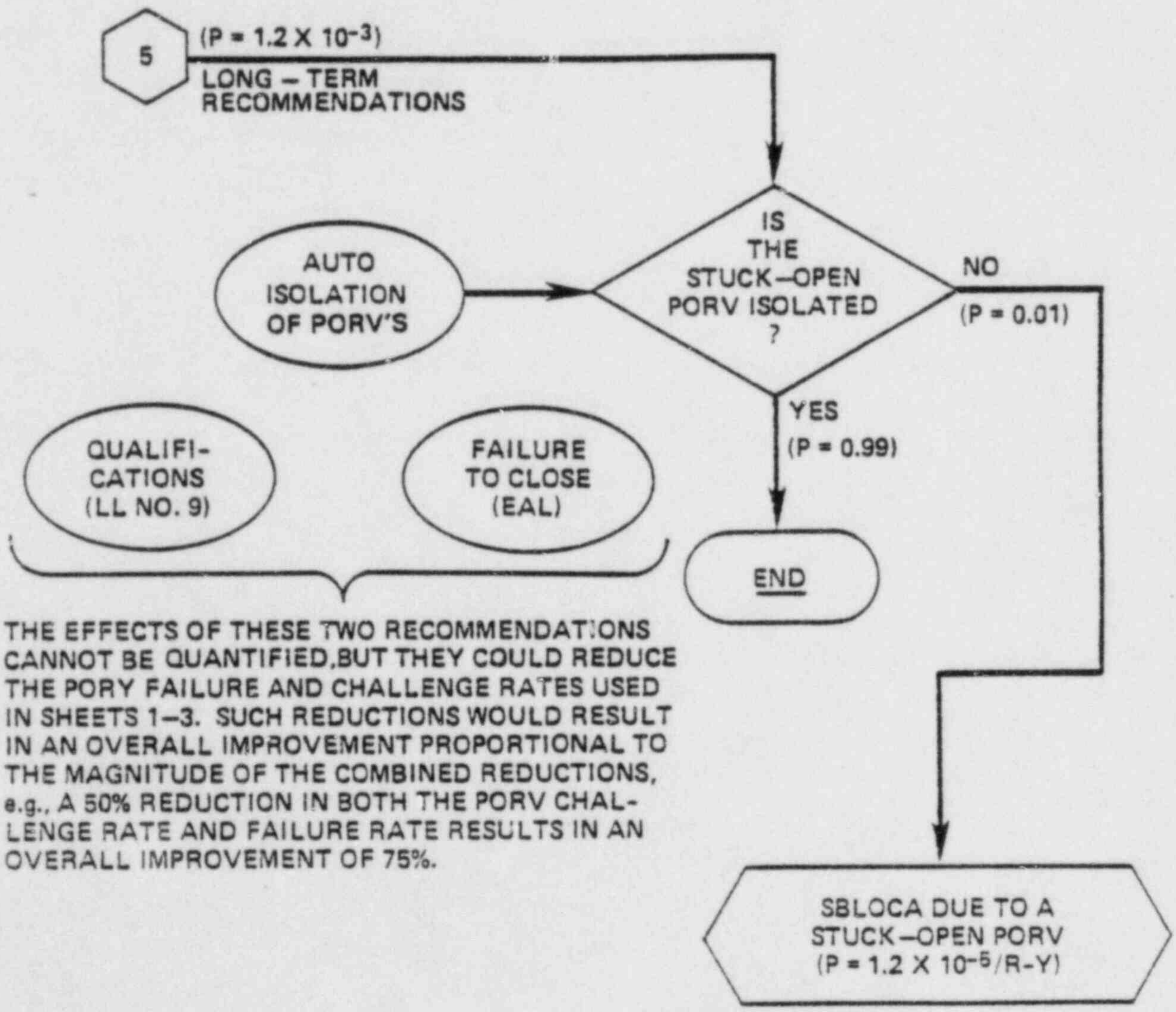
(POST TMI)

SHEET 4

ASSESSMENT OF REDUCTION OF SBLOCA'S DUE TO STUCK-OPEN PORV'S IN B&W PLANTS



(POST-TMI)
ASSESSMENT OF REDUCTION OF SBLOCA'S DUE TO STUCK-OPEN PORV'S IN B&W PLANTS

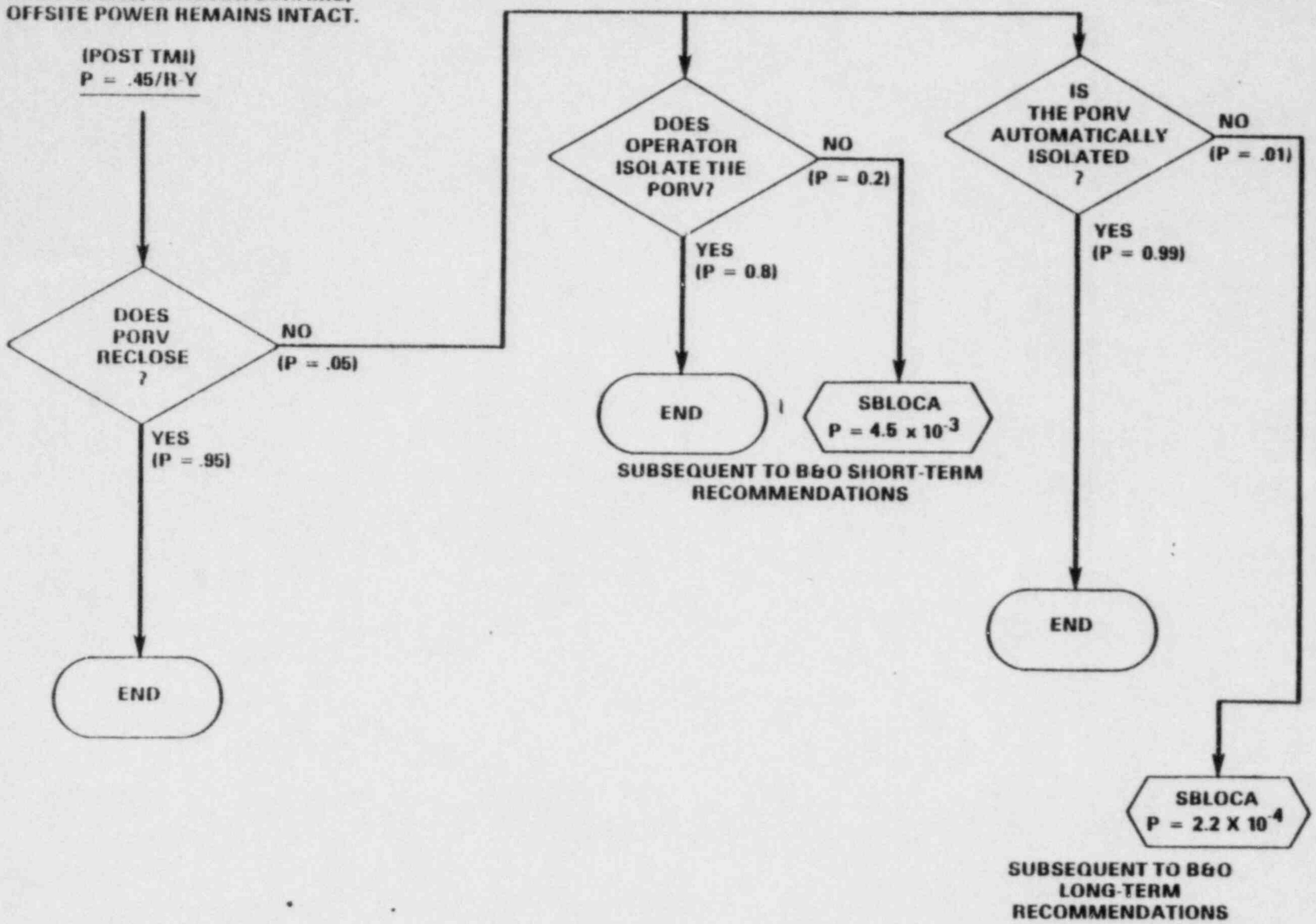


B&W PLANTS (POST TMI)

INITIATING SEQUENCE: (FROM SHEET 2A)

SHEET 6A

PORV OPENS; REACTOR SCRAMS;
OFFSITE POWER REMAINS INTACT.



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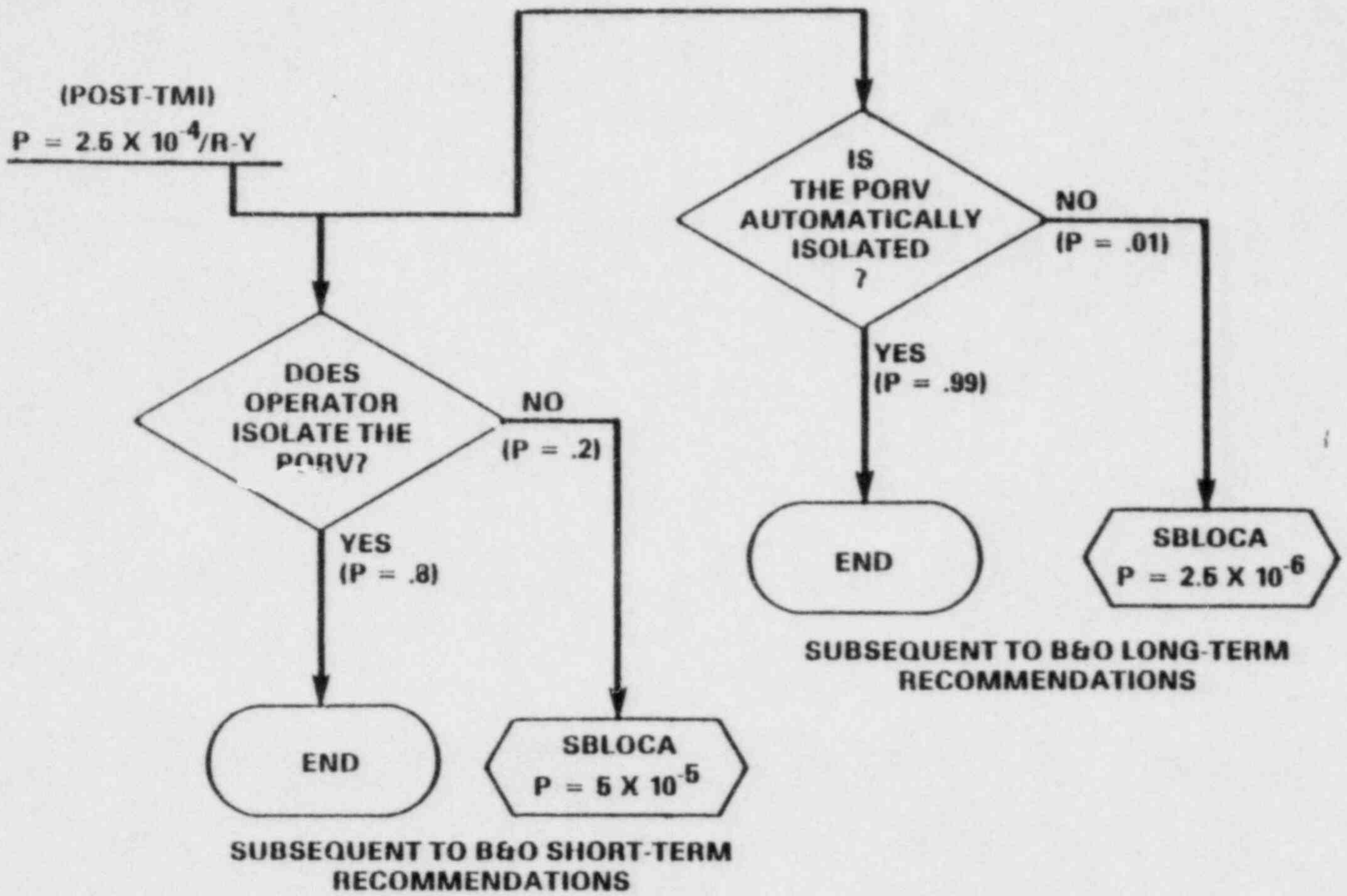
B&W PLANTS (POST TMI)

INITIATING SBLOCA SEQUENCE: (FROM SHEET 3A)

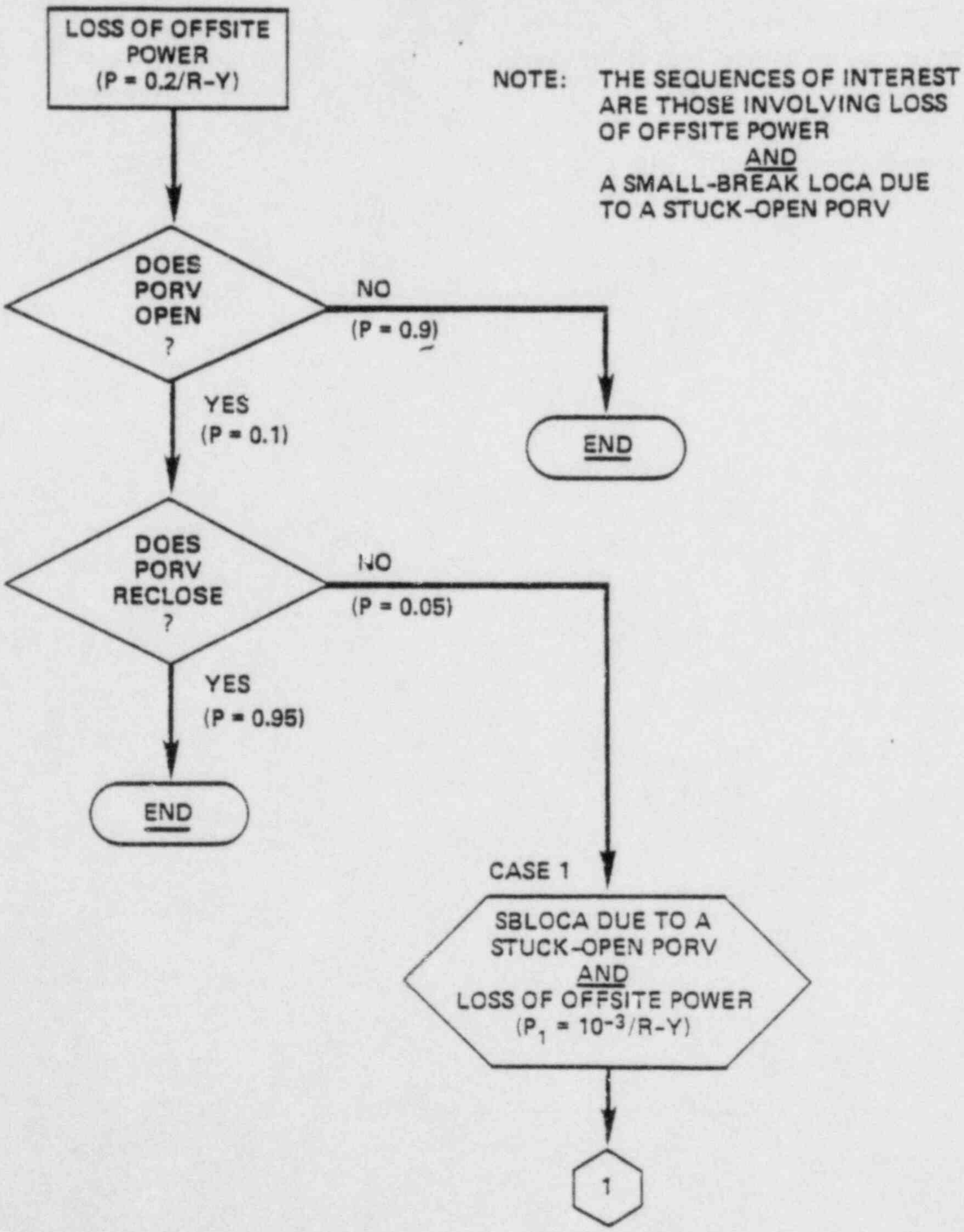
SHEET 7A

PORV OPENS; REACTOR DOES NOT SCRAM; PORV DOES NOT RECLOSE; REACTOR SCRAMS ON SBLOCA; OFFSITE POWER REMAINS INTACT.



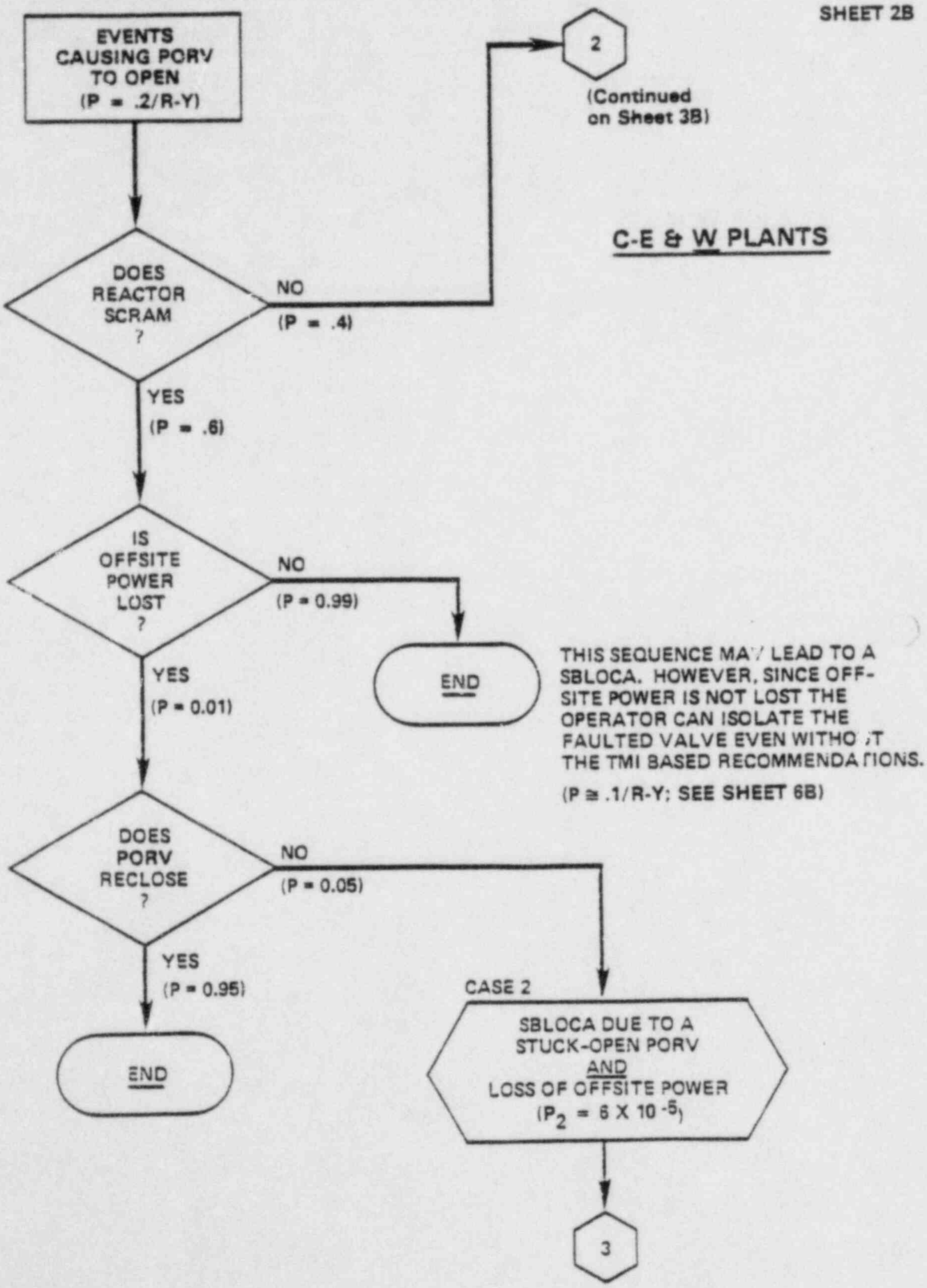
SHEET 1B

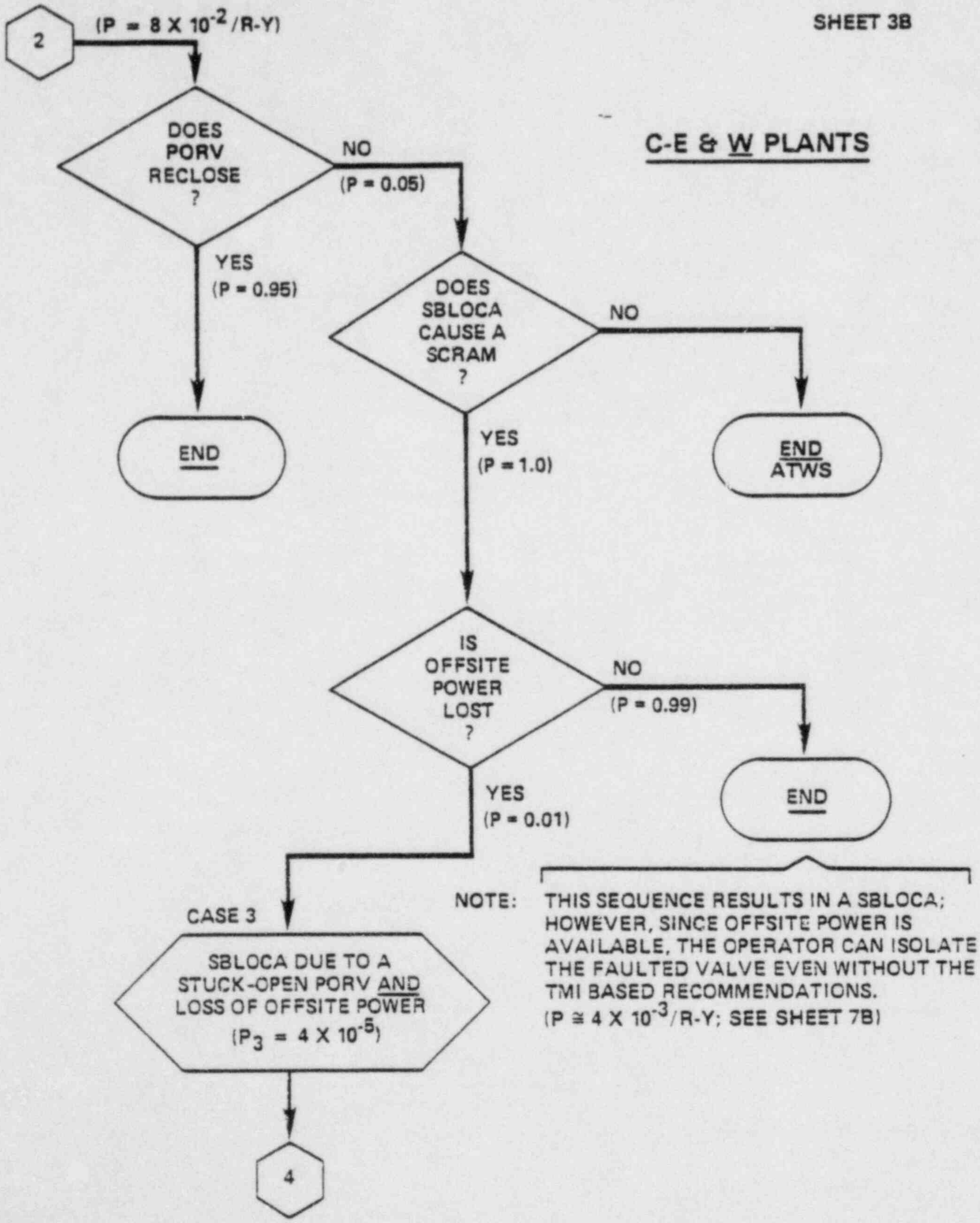
INITIATING EVENTS LEADING TO STUCK-OPEN PORV'S IN C-E AND W PLANTS



SHEET 2B

C-E & W PLANTS

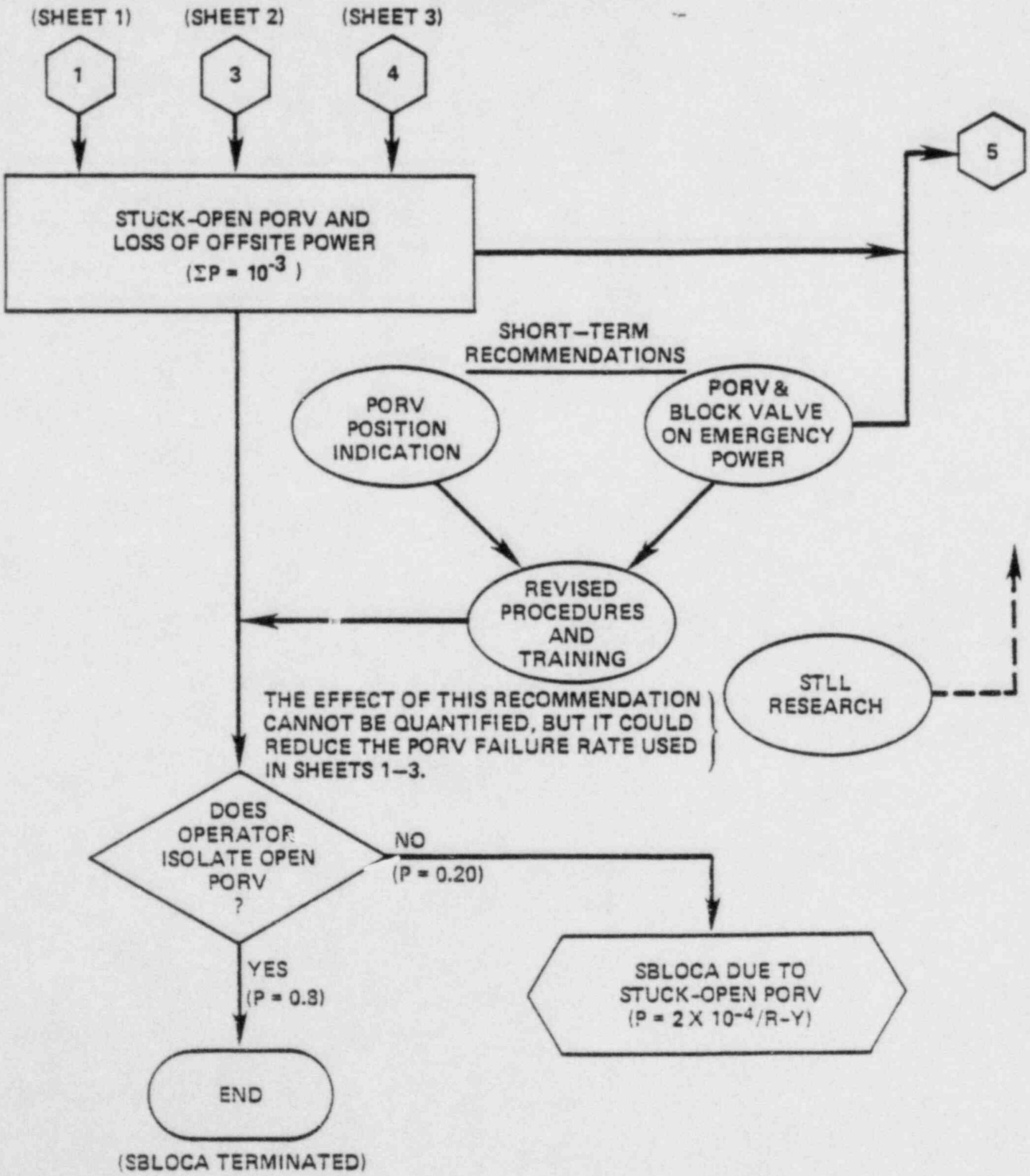



2
(P = 8 X 10-2/R-Y)
DOES PORV RECLOSE ?
NO
(P = 0.05)
YES
(P = 0.95)
END
DOES SBLOCA CAUSE A SCRAM ?
NO
END ATWS
YES
(P = 1.0)
IS OFFSITE POWER LOST ?
NO
(P = 0.99)
END
YES
(P = 0.01)
CASE 3
SBLOCA DUE TO A STUCK-OPEN PORV AND LOSS OF OFFSITE POWER
(P3 = 4 X 10-5)
4
NOTE: THIS SEQUENCE RESULTS IN A SBLOCA; HOWEVER, SINCE OFFSITE POWER IS AVAILABLE, THE OPERATOR CAN ISOLATE THE FAULTED VALVE EVEN WITHOUT THE TMI BASED RECOMMENDATIONS.
(P ≅ 4 X 10-3/R-Y; SEE SHEET 7B)

C-E & W PLANTS

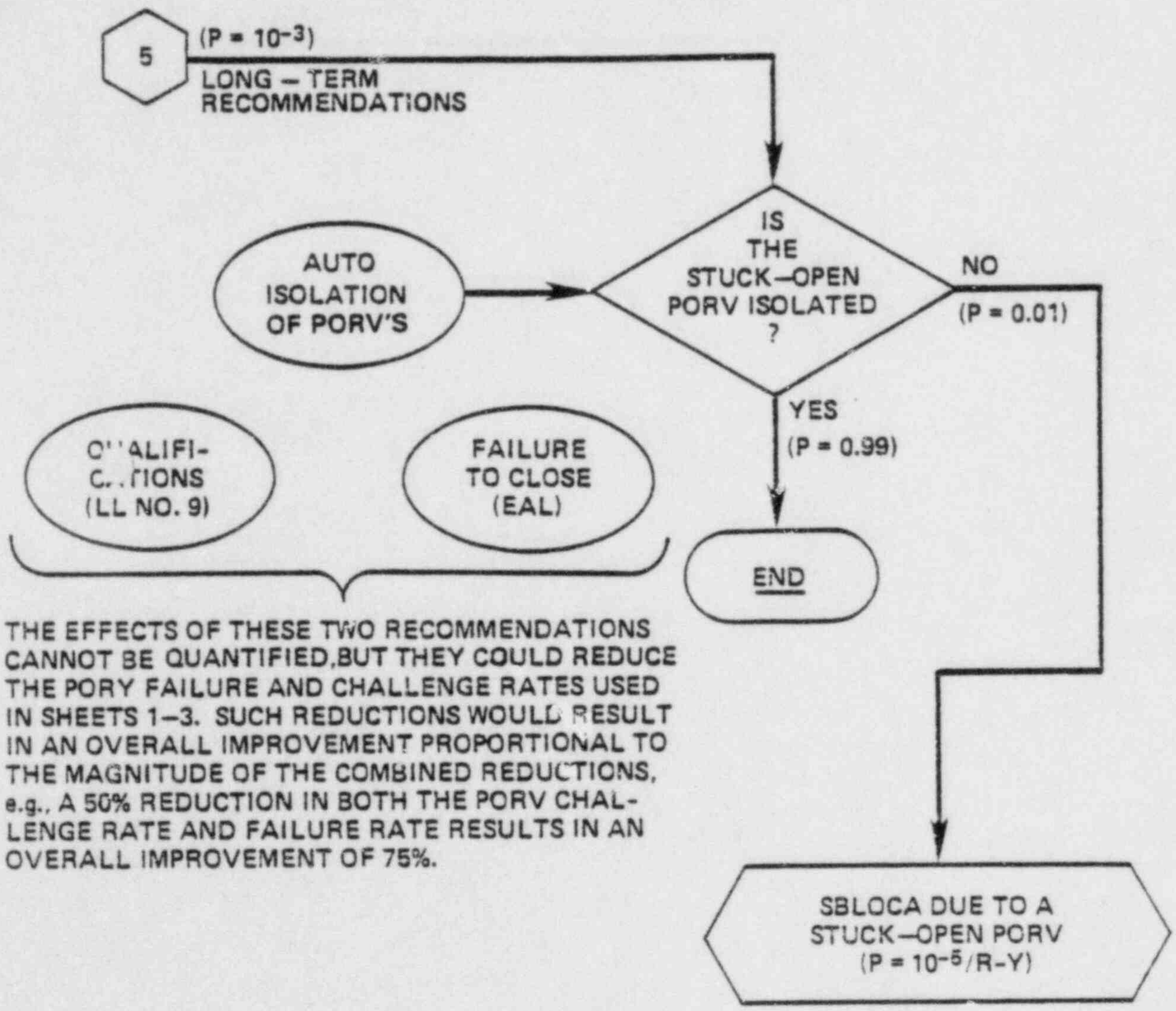
SHEET 4A

ASSESSMENT OF REDUCTION OF SBLOCA'S DUE TO STUCK-OPEN PORV'S IN C-E & W PLANTS



SHEET 5A

ASSESSMENT OF REDUCTION OF SBLOCA'S DUE TO STUCK-OPEN PORV'S IN C-E & W PLANTS

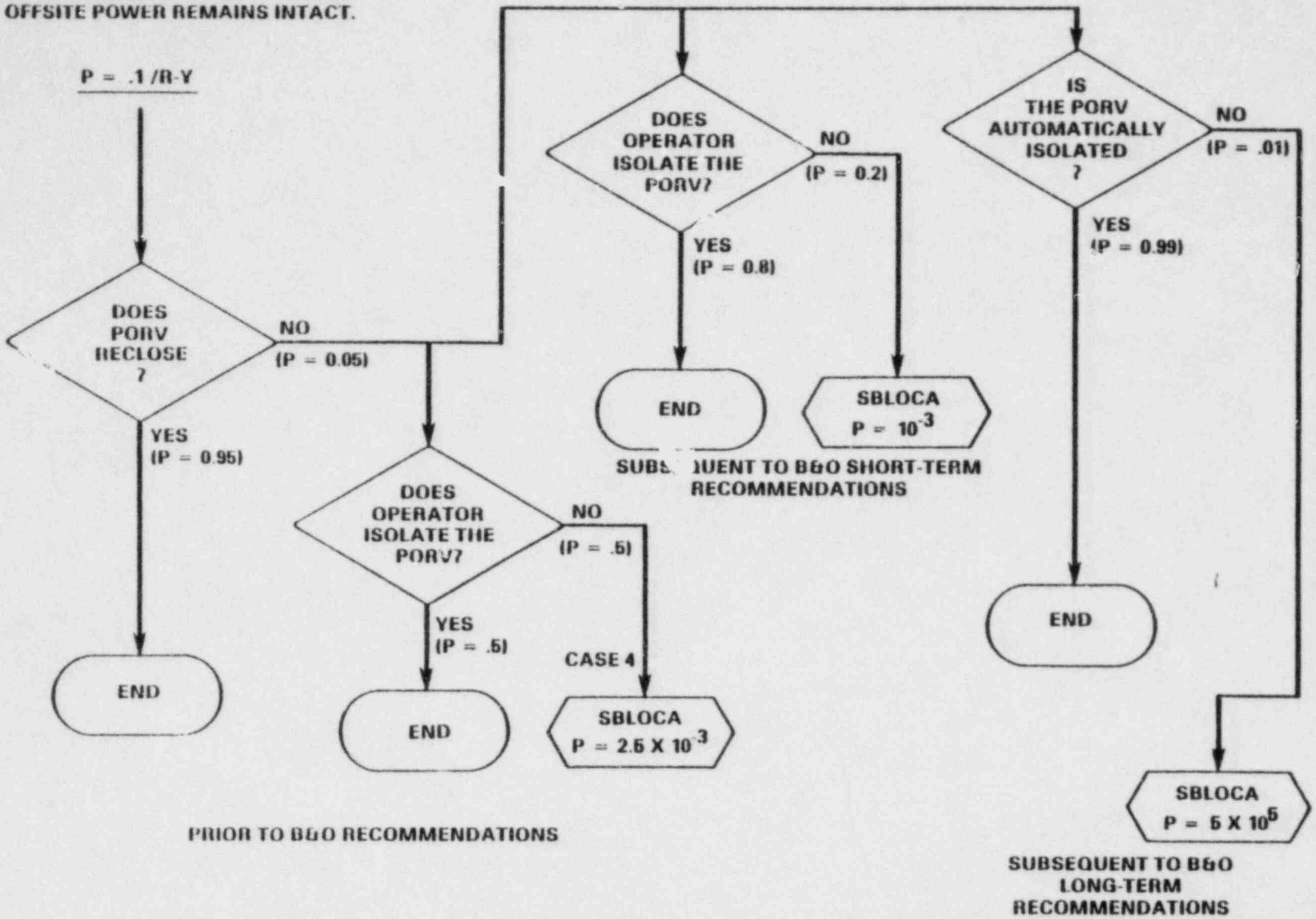


C-E & W PLANTS

INITIATING SEQUENCE: (FROM SHEET 2B)

SHEET 6B

PORV OPENS; REACTOR SCRAMS;
OFFSITE POWER REMAINS INTACT.



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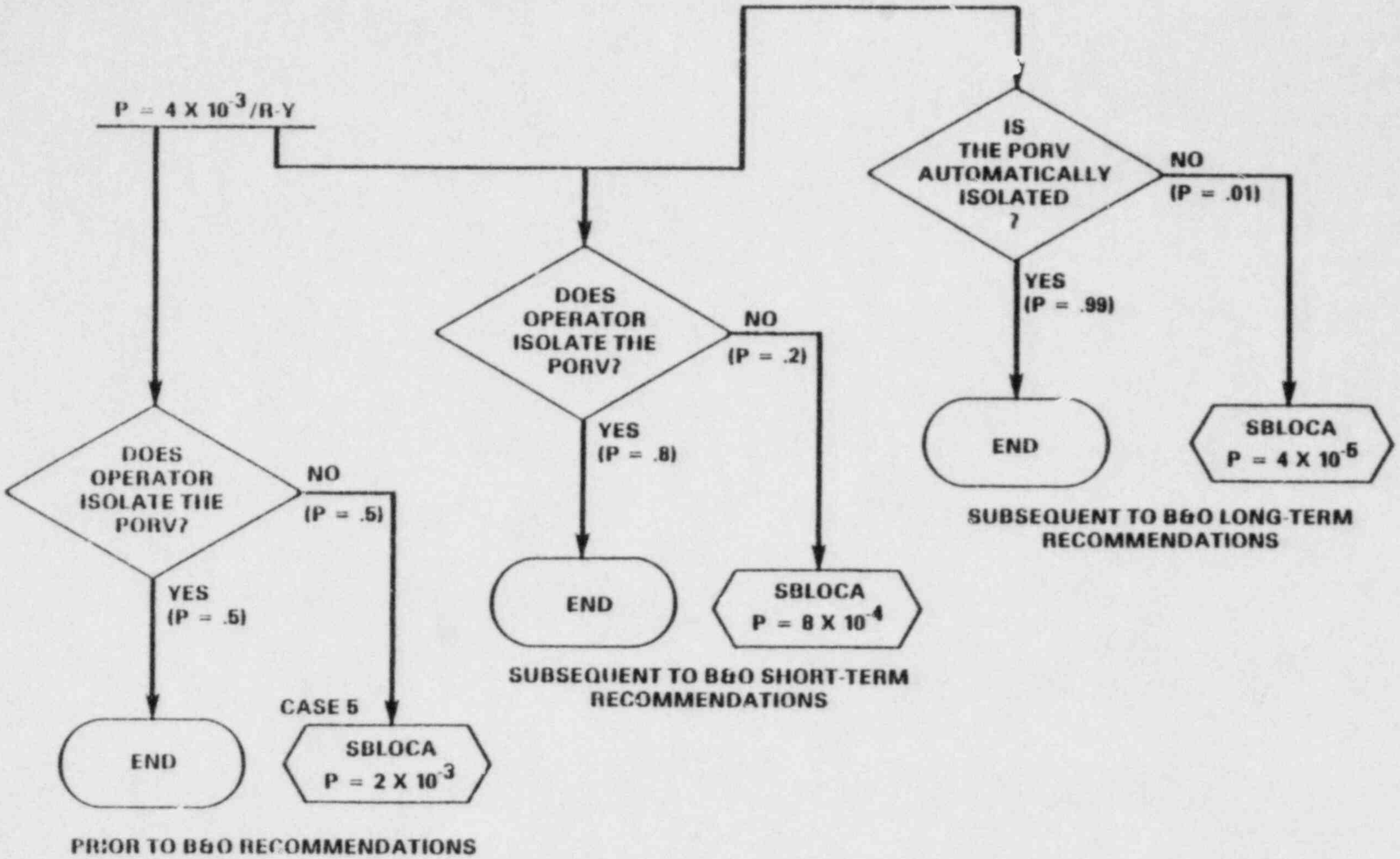
C-E & W PLANTS

INITIATING SBLOCA SEQUENCE: (FROM SHEET 3B) **SHEET 7B**

PORV OPENS; REACTOR DOES NOT SCRAM; PORV DOES NOT RECLOSE; REACTOR SCRAMS ON SBLOCA; OFFSITE POWER REMAINS INTACT.



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ESTIMATED EFFECTIVENESS OF THE B&O RECOMMENDATIONS IN REDUCING THE LIKELIHOOD OF SMALL-BREAK LOCA'S IN OPERATING PLANTS DUE TO STUCK-OPEN PORV'S

A. LIKELIHOOD OF SUCH EVENTS PER REACTOR-YEAR IN B&W PLANTS:

CASE	PRIOR TO TMI	AFTER SHORT-TERM RECOMMENDATIONS	AFTER LONG-TERM RECOMMENDATIONS
1	5×10^{-3}	2×10^{-4}	10^{-5}
2	2×10^{-3}	4.4×10^{-5}	2.2×10^{-6}
3	5×10^{-4}	5×10^{-6}	2.5×10^{-7}
4	10^{-1}	4.5×10^{-3}	2.2×10^{-4}
5	2.5×10^{-2}	5×10^{-5}	2.5×10^{-6}
TOTAL	$\sim 10^{-1}$	$\sim 5 \times 10^{-3}$	$\sim 2 \times 10^{-4}$

B. LIKELIHOOD OF SUCH EVENTS PER REACTOR-YEAR IN C-E AND W PLANTS:

CASE	PRIOR TO TMI	AFTER SHORT-TERM RECOMMENDATIONS	AFTER LONG-TERM RECOMMENDATIONS
1	10^{-3}	2×10^{-4}	10^{-5}
2	6×10^{-5}	1.2×10^{-5}	6×10^{-7}
3	4×10^{-5}	8×10^{-6}	4×10^{-7}
4	2.5×10^{-3}	10^{-3}	2.5×10^{-5}
5	2×10^{-3}	8×10^{-4}	5×10^{-5}
TOTAL	$\sim 6 \times 10^{-3}$	$\sim 2 \times 10^{-3}$	$\sim 10^{-4}$

Appendix XX

PROJECT STATUS REPORT
ACRS SUBCOMMITTEE ON SURRY UNIT #2
STEAM GENERATOR REPLACEMENT PROGRAM
MEETING ON JANUARY 23, 1980

PURPOSE: The Surry Subcommittee reviewed the Unit 2 Steam Generator Replacement Program at a meeting on October 28, 1978. The Subcommittee Chairman made a report at the November 2-4, 1978 ACRS meeting and recommended that the Committee invite VEPCO and the NRC Staff to an ACRS meeting after the steam generators in Unit 2 are replaced, but before it returns to power and before the Unit 1 replacement program begins. The Committee at its December 6-8, 1979 meeting decided that a Subcommittee meeting should be held to discuss any significant unanticipated problems encountered during the Unit 2 program and VEPCO's ability to meet the projected estimates for control of man-rem exposure and generation of radioactive wastes. Also, it was suggested that the Subcommittee review the changes being made in equipment and procedures at Surry to ensure that a repeat of the previous steam generator degradation problems do not occur.

STATUS: Surry Unit No. 2 was shutdown on February 4, 1979 and defueled for steam generator replacement. The steam generator replacement program, including testing, is now essentially complete. The expected startup date is April 15, 1980; however, VEPCO is presently holding discussions with the NRC Staff concerning moving the startup date to some time in February 1980. Items which need to be resolved before startup are the Seismic Show Cause Order, the anchor bolt, I&E Bulletin and as-built verification of piping supports.

COMMENTS: (Conclusions drawn from November 2, 1979 Progress Report):

o The total man-rem exposure expended for Unit No. 2 steam generator replacement remains below the original estimate established prior to commencement of work

o Actual radioactive liquid effluents exceed the estimate

o Airborne releases remain well below the estimate

o Solid radioactive waste generated exceed the volume and activity estimates.

THINGS TO BE ACCOMPLISHED: The Subcommittee chairman will make a report to the Committee at the February 7-9, 1980 meeting indicating whether the Lessons Learned from Surry Unit No. 2 Steam Generator Replacement Program are/are not adequate to allow the commencement of the replacement of steam generators on Unit No. 1. The Subcommittee chairman will also comment on whether Unit No. 2 is/is not ready to restarg following the Steam Generator Replacement Program.

PROPOSED SUMMARY OF THE JANUARY 23, 1980 MEETING OF THE SURRY STATION, UNIT 2 SUBCOMMITTEE

PURPOSE: The purpose of the meeting was to discuss the Steam Generator Replacement Program on Surry Station, Unit 2.

ATTENDEES: The ACRS Members in attendance were: H. Etherington, Subcommittee Chairman, P. Shewmon, D. Moeller, and M. Bender. ACRS consultant R. Dillon was also in attendance.

MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS:

1. The Unit 2 Steam Generator Repair Program (SGRP) started February 3, 1979 and was completed December 31, 1979 (37 weeks actual versus 26 weeks planned). Unit 2 startup is scheduled for May 1980 following completion of the seismic show cause order and some other things not related to the SGRP. Unit 1 Steam Generator Replacement is scheduled to begin in June 1980.

2. VEPCO informed the Subcommittee that they had no major significant problems during the Unit 2 SGRP; however, they did mention numerous "unanticipated events and problems," both technical and administrative, which they plan to remedy on the Unit 1 SGRP.

3. The following estimated versus actual labor, exposure, and rad-waste value comparisons for the Unit 2 SGRP were discussed:

	ESTIMATED	ACTUAL
Labor	233,588 manhours	871,643 manhours
Man-rem exposure	2,066 man-rem	2,140 man rem
Liquid effluent volume	2.3×10^6 gallons	3.0×10^6 gallons
Liquid effluent activity	0.344 curies	0.497 curies
Noble gas effluents	Negligible	101.3 curies*
Gaseous iodine effluents	4.53×10^{-3} curies	6.88×10^{-6} curies
Gaseous particulate effluents	3.12×10^{-3} curies	1.32×10^{-3} curies
Solid waste volume	26,236 ft^3	57,820 ft^3
Solid waste activity	18.9 curies	63.6 curies

*Resulted from defueling operations

A-38 L

4. The NRC Staff has concluded, that although the Unit 2 SGRP took longer than estimated it was well performed, and that Unit 1 steam generator replacement can be started as scheduled.

5. VEPCO told the Subcommittee that due to the equipment and operating procedure changes which have been made and/or planned, they feel confident they will not have a recurrence of the steam generator problems previously encountered.

FUTURE MEETINGS: No future Subcommittee meetings with VEPCO on the Unit 1 or 2 SGRP are planned. VEPCO has been requested to show a 40-45 minute videotape of the actual Unit 2 SGRP operations at the February 7-9, 1980 meeting.

Appendix XXI

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

February 1, 1980

ACRS Members

APPENDIX XXI
ISSUES RAISED RE WOLF CREEK SEISMIC DESIGN

WOLF CREEK NUCLEAR PLANT SEISMIC DESIGN

The Wolf Creek Nuclear Plant will utilize the SNUPPS standing plant design. All Category 1 structures associated with this part of the design will be designed to withstand an SSE of 0.2g and an OBE of 0.1g. There has been some controversy with the strength of the concrete used in the base. The NRC has reviewed the design of the as-built basemat and has concluded that the basemat can withstand the 0.12g SSE. This was established during the CP review. The applicant has performed a finite-difference analysis of the basemat for a 0.2g SSE and has shown that the stress falls within acceptable limits. However, the applicant has not performed the analysis using an alternate approved method as required by NRC procedures and has not obtained NRC's concurrence that the basemat will withstand 0.2g. It is highly probable that the basemat can be qualified for 0.2g.

The Category 1 structures falling outside of the SNUPPS design are designed the 0.12g SSE and the 0.06g OBE. These structures are:

(a) The ultimate heat sink dam (a submerged earth dam, partitioning off a portion of the cooling pond)

(b) The essential service water system (ESWS) pumphouse

(c) The ESWS discharge structure

(d) The ESWS pipes

(e) The ESWS electrical duct banks and pull boxes.

The plant site is characterized by rock and shallow soil deposits. All SNUPPS Category 1 Structures, the ESWS pumphouse, and the ESWS discharge structure are founded on competent rock or lean fill concrete.

The NRC Staff in the CP review has accepted an MMVII earthquake as the SSE. The Trifunac/Brady (1975) intensity acceleration relationship currently used by the NRC Staff associates 0.1g to 0.2g accelerations to the MMVII and MMVIII intensity [illegible]. A graphic representation of this and other intensity/accelera[illegible]relations is attached.

for R. Muller
Senior Staff Engineer

for R. Savio
Staff Engineer

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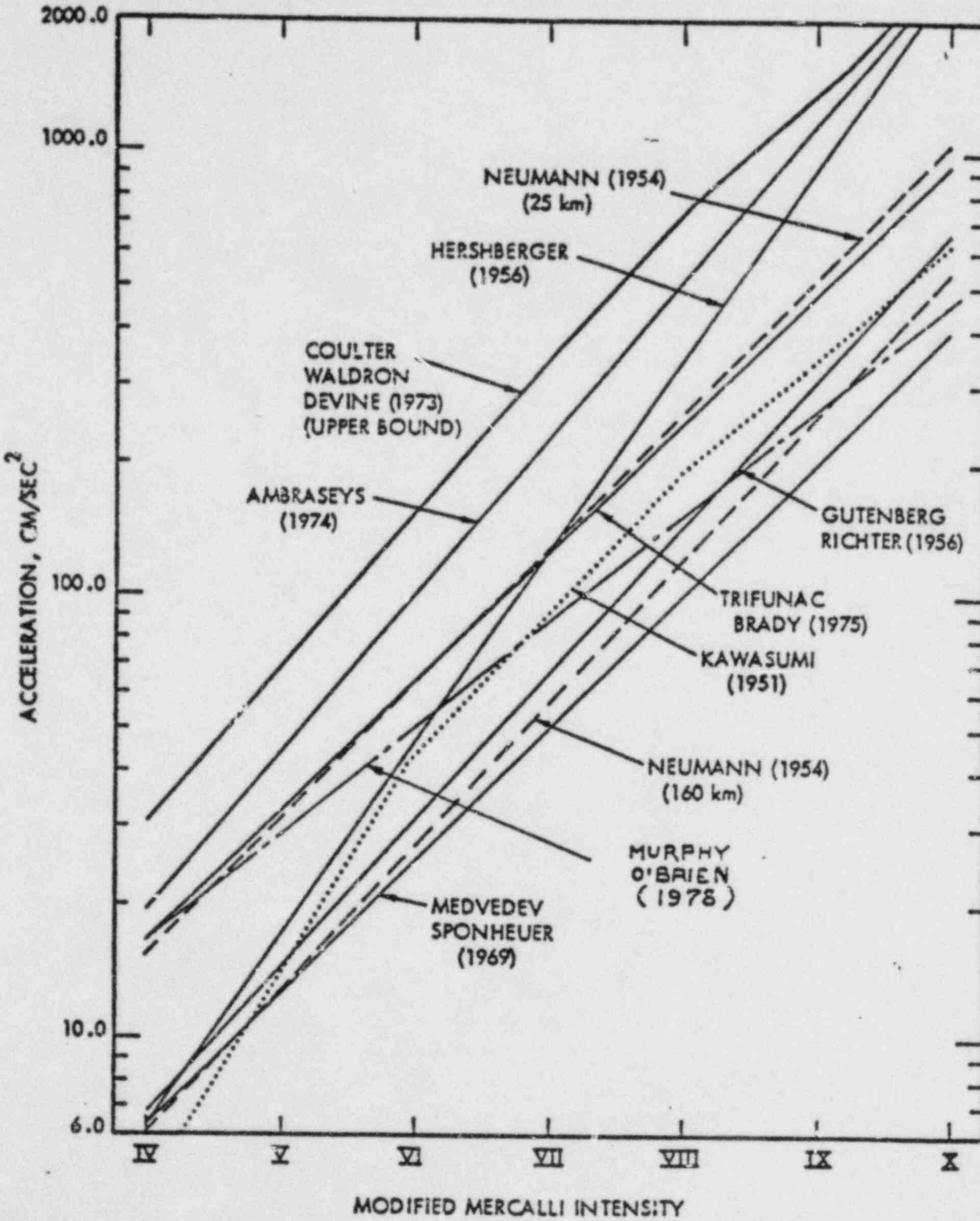

Figure 1 Graphic Representation of Selected Intensity/Acceleration Correlations

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

January 8, 1980

R. F. Fraley, Executive Director
ACRS

WOLF CREEK SEISMIC QUESTIONS RAISED BY MR. WILLIAM WARD'S PETITION TO THE COMMISSIONERS

The Staff's response to the Ward petition has just been received. (Attachment 1).

The Staff does not completely agree with the Kansas State Geological Survey (KSGS) report but points out that even so, the KSGS results fall within the envelope considered by the Staff in arriving at a 0.12g SSE.

At the 235th ACRS Meeting, the Committee asked that Dr. Maxwell be provided with additional data. (See Attachment #5), and if his subsequent report warranted, the matter was to be reviewed further by the Extreme External Phenomena (EEP) Subcommittee.

Maxwell has indicated that the "assigned SSE value of 0.12g is still reasonable for the Wolf Creek site". However, he adds that if 0.20g is required for Tyrone, it appears it should also be required for Wolf Creek.

Although the ACRS, subsequent to the Wolf Creek review, recommended in a letter on North Anna 1 & 2, that 0.2g be used for all plants east of the Rockies, there seems to be no indication in the consultant reports that safety considerations require further review by the EEP Subcommittee.

1

Attachment:
(1) Memo R.E.Jackson to Olan Parr dated Jan. 4., 1980
Subject "Staff Response to William H. Ward letter on Seismic Issues at Wolf Creek."
(2) J. C. Maxwell Consultant report of Oct. 1, 1979
(3) J. C. Maxwell Consultant report of Nov. 26, 1979
(4) P.Pomeroy Consultant report of Oct. 29, 1979
(5) Extract from Minutes of November ACRS Mtg.

cc: ACRS
T. G. McCreless
R. Savio
J. C. McKinley
M. W. Libarkin

A-390

J. Apneuc

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

RECEIVED 1-1016

1980 JAN 7 AM 11 09

January 4, 1980

U.S. NUCLEAR REG. COMM. ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MEMORANDUM FOR: Olan D. Parr, Chief
Light Water Reactors Branch No. 3, DPM

FROM: Robert E. Jackson, Chief
Geosciences Branch, DSS

SUBJECT: STAFF RESPONSE TO WILLIAM H. WARD'S LETTER ON SEISMIC ISSUES AT WOLF CREEK

Enclosed is the Staff response to William H. Ward's petition to the Commissioners requesting at least a partial suspension of the construction permit for the Wolf Creek Generating Station. This Staff response is an expanded background for the seismic issue mentioned in footnote 6 of the July 12, 1979 Director's Decision under 10 CFR 2.206. This Director's Decision by Victor Stello, Jr., IE, denied Mr. Ward's petition. It stated that the seismic issues contained in Mr. Ward's letter were previously considered by the Staff and do not alter the Safe Shutdown Earthquake at the Wolf Creek site. Based upon the enclosed evaluation of Mr. Ward's concerns and recent Staff licensing decisions, we conclude that the 0.12g Safe Shutdown Earthquake is adequately conservative and therefore recommend that Mr. Ward's request for at least a partial suspension of the construction permit for Wolf Creek be denied. Dr. Phyllis Sobel, Geophysicist, prepared this evaluation. She was assisted by Leon Reiter, Section Leader.

Original Signed by
R. E. Jackson

Robert E. Jackson, Chief
Geosciences Branch
Division of Systems Safety

Enclosure:
As stated

cc: w/enclosures

J. Knight
S. Varga
R. Jackson
L. Reiter
R. McMullen
P. Sobel
M. Licitra
J. Harbour
J. Lieberman
S. Burns
R. Rothman
H. Lefevre
R. Muller
D. Vassallo
H. Thornbury, IE
W. Reinmuth, IE
M. Schumacher, IE

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STAFF RESPONSE TO WILLIAM H. WARD'S LETTER ON
SEISMIC ISSUES AT WOLF CREEK

On June 29, 1979, William H. Ward, Attorney for the Mid-America Coalition for Energy Alternatives, wrote the NRC Commissioners to advise them of several seismic issues affecting the Wolf Creek site and to request at least a partial suspension of the construction permit (Attachment). It is the purpose of this Staff response to address Mr. Ward's concerns.

Concern 1. A report by the Kansas State Geological Survey (KSGS), NUREG/CR-0294, concludes that the 1867 Manhattan earthquake was at least intensity VII-VIII (MM). Mr. Ward states that this earthquake was used as the basis for the Safe Shutdown Earthquake (SSE) and that the SSE was based on the assumption that the 1867 Manhattan earthquake could occur on the Nemaha Ridge at its closest approach to the Wolf Creek site, 50 miles. In light of the new information developed by the KSGS, the .12g horizontal acceleration SSE does not now appear to be conservative to Mr. Ward.

Response. The Staff has reviewed the report by KSGS and still finds the 1867 Manhattan earthquake to be intensity VII (MM). The assignment of intensity VII-VIII is based upon an 1877 report of liquefaction on a farm on the floodplain of the Kansas River. That observation was assigned intensity VIII and placed close to the epicenter by the Kansas Geological Survey. Liquefaction is very dependent upon local site conditions and may occur in isoseismal areas that may otherwise be associated with intensities less than VIII. The staff agrees with the standard references, such as Earthquake History of the United States (1973), which list this earthquake as an intensity VII (MM).

In the Safety Evaluation Report (SER) for the Wolf Creek site, the Staff chose a Safe Shutdown Earthquake (SSE) of intensity VII (MM). This intensity was based on:

1. The maximum earthquake that could occur in the Nemaha Uplift at its closest approach to the Wolf Creek site.
2. The maximum random earthquake in the region (for example, the 1956 Catoosa, Oklahoma earthquake).

The Staff's analysis did not involve the direct use of the 1867 Manhattan earthquake since a larger earthquake (intensity greater than VIII and less than X) was assumed to occur on the Nemaha Uplift. This larger earthquake was already assumed to occur at the closest approach of the Humboldt Fault to the Wolf Creek site. Therefore, the results of the Staff's analysis (an SSE of intensity VII) are not modified by the KSGS results.

Concern 2. The size of the appropriate Wolf Creek SSE can be determined by reference to the SER for another of the SNUPPS units, Tyrone. Both Tyrone and Wolf Creek are located in the Central Stable Region Tectonic Province. The Tyrone SSE is 0.2g horizontal acceleration.

Response. The Staff's assessment of the SSE at both Wolf Creek and Tyrone considered both the maximum random earthquake and the maximum earthquake that could occur on a nearby structure. The staff has evaluated the SSE at Wolf Creek and Tyrone in light of more recent licensing decisions. As a result of this evaluation we see no evidence that the SSE at Wolf Creek is unconservative or that it is inconsistent with recent licensing decisions.

1. Random earthquake at Tyrone.

The Tyrone site is near the town of Durand in western Wisconsin. The site is in the Central Stable Region Tectonic Province. In the Tyrone SER (1975), the Staff considered the intensity VII-VIII Anna, Ohio earthquake of 1937 as the largest earthquake in the Central Stable Region which could not be reasonably associated with known geologic structure. Using the Trifunac-Brady (1975) empirical relation between intensity and ground acceleration, the mean vibratory ground acceleration corresponding to MM intensity VII-VIII is 0.2g. This evaluation of the largest random earthquake near the Tyrone site is conservative and similar to recent licensing decisions made for other sites in the Central Stable Region. The Staff, however, recognizes significant variations in the historic seismicity among subregions of this large structural tectonic province. Based on the low level of seismicity in the vicinity of the Tyrone site and had the licensee given sufficient supportive bases, the Staff may have considered an intensity lower than VII-VIII (MM) more appropriate for the random earthquake.

2. Maximum earthquake on the Midcontinent Geophysical Anomaly and its effects at the Tyrone site.

For the purpose of establishing the SSE at the Tyrone site, the Staff evaluated the effects of the maximum earthquake associated with the Midcontinent Geophysical Anomaly (MGA) on the Tyrone site (SER, 1975). The Staff assumed that an intensity VIII earthquake could occur on structures associated with the MGA. In the SER the Staff assumed that at its closest approach to the site, i.e. 45 miles, the intensity at the site due to attenuation would be reduced to intensity VII-VIII. Using current intensity-attenuation relationships for the Central Stable Region (Gupta and Nuttli, 1976) attenuation of the effects of the intensity VIII event at the closest point on the MGA to the Tyrone site, i.e. 45 miles, results in a site intensity less than VII. Using the Trifunac - Brady (1975) empirical relation between intensity and ground acceleration, the mean vibratory ground acceleration corresponding to MM intensity VII is 0.12g.

3. Random earthquake at Wolf Creek.

The Wolf Creek site lies in southeast Kansas in the Central Stable Region Tectonic Province. In the Wolf Creek SER, the Staff considered the maximum random earthquake to be intensity VII (MM). This position was reiterated in a more recent Staff decision in the same region-the Black Fox site in eastern Oklahoma (SER, 1977). The Staff recognized the low level of seismicity in the vicinity of the Black Fox site and considered the maximum random earthquake to be intensity VII.

4. Maximum earthquake on the Nemaha Uplift and its effects at the Wolf Creek site.

For the purpose of establishing the SSE at the Wolf Creek site, the Staff evaluated the effects of the maximum earthquake associated with the Nemaha Uplift (NU) on the Wolf Creek site (SER, 1975). The Staff assumed that intensities greater than VIII and less than X could occur on the Nemaha Uplift. In a more recent Staff decision for the Black Fox site (SER, 1977), the Staff found that an earthquake of intensity VIII was a more reasonable maximum event on the NU, based on similarity with other structures in the Central Stable Region which have associated seismicity. (This Staff decision was supported by the Black Fox Licensing Board Decision "Partial Initial Decision Authorizing Limited Work Authorization," LBP-78-26, 8 NRC 102, 111 (1978), Aff'd ALAB - 573, Slip Op. at 40 (Dec. 7, 1979)). Using current intensity-attenuation relationships for the Central Stable Region (Gupta and Nuttli, 1976), attenuation of the effects of the intensity VIII event at the closest point on the NU to the Wolf Creek site, i.e. 50 miles, results in a site intensity less than VII.

Conclusion

Therefore, based upon our evaluation of the SER's for the most recent licensing decisions, we conclude that it is not necessary to have the same SSE at the Tyrone and Wolf Creek sites. Applying a current intensity-attenuation relation at both sites, a site intensity of VII is an adequately conservative value for the effects of the maximum earthquake on significant nearby structures. At the Tyrone site the maximum random earthquake was conservatively chosen to be intensity VII-VIII but the Staff could have considered a lower intensity based on the low level of seismicity in the vicinity of the site. At the Wolf Creek site credit was given for the lower level of seismicity in the vicinity of the site and the maximum random earthquake was considered to be intensity VII.

Analysis of NRC Sponsored Research Programs Affecting the Wolf Creek Site

The KSGS report mentioned in Ward's letter is part of a cooperative geologic, seismic, and geophysical research program by several state geological surveys that is seeking to define the structural setting and tectonic history of the Nemaha Uplift and the Midcontinent Geophysical Anomaly in order to provide the bases for a more realistic appraisal of the earthquake risks in the siting of nuclear facilities in the North American Mid-Continent. This information is used as a basis for continuing research and as input to the evaluation of seismic risk in the region within and around the Nemaha Uplift. The research effort ~~thus far has increased our current~~ data base and our understanding of earthquake phenomena in the vicinity of the Nemaha Uplift; however, this information has not indicated a need to modify any previous licensing decisions.

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As part of this cooperative research program, the NRC is funding a five year detailed study of the sources of seismicity in the Nemaha Uplift area. The results of work completed in Phase I is currently being reviewed. Therefore, it is too early to assess the impact on nuclear power plant licensing. The total impact of the five year study cannot be assessed until the overall program is completed and synthesized with seismic monitoring data. The preliminary results are being considered in the development of a tectonic province or seismic zoning map of the eastern U. S.

Mid-America Coalition For Energy Alternatives

[illegible]

Attachment

June 29, 1979

Joseph Hendrie, Chairman
Peter Bradford, Commissioner
Victor Gilinsky, Commissioner
Richard Kennedy, Commissioner
John Aherne, Commissioner
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Commissioners:

I wrote you on January 19, 1979, on behalf of my client asking that you suspend the construction permit for the Wolf Creek project in view of documented quality control problems specifically related to the base mat of the reactor containment building. You responded by publishing a notice of our request in the Federal Register.

This letter is to advise of certain new determinations with respect to the seismic character of the area and to renew our request for at least a partial suspension of the construction permit in view of the significance of those determinations in conjunction with existing unresolved issues regarding base mat integrity.

Your attention is directed to a report of the Kansas State Geological Survey (KSGS) prepared under contract to your Division of Reactor Safety Research, Office of Nuclear Regulatory Research, entitled "A Revised and Augmented List of Earthquake Intensities for Kansas, 1867-1977" NUREG/CR-0294, August, 1978. The report details the conclusion of the KSGS that the largest historical earthquake in Kansas occurred at a different location and was of a different magnitude than had been previously believed. This earthquake was used as the basis for the design of the non-standardized Category I (safety related) portions of the plant. Commonly known as the 1867 Manhattan earthquake and thought to have been of the size Modified Mercalli VII, its epicenter was assumed to have been approximately 22 miles northwest of Manhattan, Kansas. The applicants argued that the earthquake was related to a presumed "zone of weakness" associated with the contact of the Keweenawan mafic volcanic belt and the Nemaha Ridge (Nemaha Uplift). The nearest

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approach of the zone, according to the SNUPPS PSAR Wolf Creek Addendum, is 75 miles from the Wolf Creek site. On that basis the applicants urged the adoption of a safe shutdown earthquake (SSE) with a .10g horizontal acceleration.

Finding insufficient basis for the applicants' assumption that the earthquake was related to such a zone of weakness, your staff apparently insisted that the SSE be based on the assumption that the 1967 Manhattan earthquake could occur on the Nemaha Ridge at its closest approach to the Wolf Creek site, 50 miles. Such an assumption would, concluded the staff, yield a safe shutdown earthquake of .12g, and the site was licensed accordingly.

In light of the new information developed by the KSGS concerning the size of the 1867 earthquake and the actual location of its epicenter, and recent microseismicity recorded along the long inactive Humbolt Fault, the postulated .12g horizontal acceleration safe shutdown earthquake does not now appear to be conservative. The KSGS report concludes, on the basis of extensive review of historical records, that the 1867 "Manhattan" earthquake was at least a Modified Mercalli VII-VIII -- stronger than the MM VII that both applicants and staff had assumed. It concluded also that its epicenter was in the Wamego vicinity, and was, accordingly, associated with the Humbolt Fault. The Humbolt Fault defines the eastern boundary of the Nemaha Ridge and passes within 50 miles of the Wolf Creek site. In addition, since January, 1978, numerous microearthquakes have occurred along the trace of the Humbolt Fault north of the Wolf Creek site and south in Oklahoma. While the KSGS has not yet concluded that this means stress is building in the vicinity of the nearest approach of the fault to the plant site, they site successful earthquake prediction experience elsewhere in the country which indicates that such is often the case.

The size of the appropriate safe shutdown earthquake for the Wolf Creek site can be determined by reference to your staff's Safety Evaluation Report for another of the SNUPPS units, Tyrone. Both Tyrone, in Wisconsin, and Wolf Creek are located in the Central Stable Region Tectonic Province. The following Tyrone SER discussion elucidates the reason for setting the Tyrone SSE at .2g horizontal acceleration:

> "Based on historical accounts, the area of the Central Stable Region in which the Tyrone site is located is seismically very quiet. No historical earthquakes have been reported within 100 miles of the site, and only ten earthquakes of intensity MM IV or greater have been reported within 200 miles of the site. The nearest historical earthquake in the vicinity of the Tyrone site, which occurred sometime between 1865 and 1870, had an estimated

intensity MM VI-VII and occurred slightly more than 100 miles west of the site.

* * *

"The Midcontinent Geophysical Anomaly is located approximately 45 miles northwest of the Tyrone site. This feature corresponds to a region characterized by gravity and magnetic anomalies, which over much of its extent, coincide with mapped basement faulting. The Midcontinent Geophysical Anomaly extends generally from the Lake Superior region south-west through Minnesota, across Iowa, and into Kansas where it trends into the Nemaha Uplift. The largest historical earthquakes which have been located along this feature have had reported epicentral intensities of MM VIII. However, as has been noted above, the characteristics associated with at least one of these intensity MM VIII events, the Keewenaw Peninsula earthquake of 1906, would indicate that the intensity level may have been influenced by local geology. If it is assumed that an intensity MM VIII earthquake could occur on structures associated with the Midcontinent Geophysical Anomaly at its closest approach to the site; i.e. 45 miles, the intensity at the site due to attenuation would be reduced to intensity MM VII-VIII.

* * *

"In 1954 Neumann developed an empirical relationship between earthquake intensity and ground acceleration. More recently Trifunac and Brady (1975) have published a relation between intensity and acceleration which was developed using many additional observations. Trifunac and Brady's data essentially corroborate the relationship published by Neumann. Utilizing either the Neumann or the Trifunac-Brady relation between intensity and acceleration, the mean acceleration corresponding to intensity MM VII-VIII is 0.2g. Based on this analysis we consider 0.2g to be the appropriate acceleration for the seismic design of the proposed plant at the Tyrone site." pp. 2-16, 17, 18

With respect to the base mat of the Wolf Creek reactor building, the significance of setting the safe shutdown earthquake at .2g horizontal acceleration is substantial. Your staff has been unable to conclude that the 90-day concrete cylinder tests, which showed that the base mat concrete failed to meet the design specification of 5000 pounds per square inch, were in error. Accordingly, it ordered the applicants, who carry the burden of proof on all

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such matters, to show that the concrete is of sufficient strength, on the basis that the 90-day tests are assumed to be accurate. The Wolf Creek architect/engineer, the Bechtel Power Corporation, performed the reanalysis by first determining that actual concrete strength as shown by the 90-day tests was 4460 pounds per square inch (by working backward from the acceptance criteria) and then by performing computer simulations to show that the base mat was adequate at that strength to permit the safe shutdown of the plant even if it is subjected to a horizontal acceleration of .2g -- greater than the .12g earthquake for which, as noted above, the Wolf Creek site is licensed.

The standardized portion of all SNUPPS plants must be built to be shut down safely after a .2g earthquake. The Bechtel Report notes that this safe shutdown earthquake is "controlled by a site other than Wolf Creek", but does not specify which one. The Report states that the use in the reanalysis of the greater than required .2g assumption "is consistent with the general methodology used for the project, is in accordance with the commitments made in PSAR Section 3.7 and provides additional conservatism." "Seismic loads were conservatively determined at the SNUPPS envelope "g" level, which is considerably higher than that for which the site is licensed", states the Report in its conclusion. We submit that the reanalysis was, for the reasons discussed above, not conservative -- that the Bechtel Report shows, if it is valid, only that the base mat is not expected to crack during the largest probable earthquake, if the concrete undergoes no deterioration.

However, no allowance is made in the Bechtel Report for normal deterioration of the base mat due to routine plant operation. In addition, evidence exists that the base mat concrete is presently undergoing spontaneous deterioration due to some as yet unknown cause.

As you are aware, some of the 90-day test results were lower than the 28-day test results. Unless the reason for this anomaly is explained, it constitutes evidence that deterioration is taking place -- evidence which, under your agency's rules, it is the responsibility of the applicants to refute. Yet, on June 7, 1979, your staff issued a summary of the public meeting held in Burlington, Kansas on May 15, 1979, to review with the applicants the Bechtel Report and the base mat problem generally, a principal conclusion of which was:

"1. There is no clear cut answer as to why some of the 90-day cylinder test results are lower than 5000 pounds per square inch. Neither is there a clear cut answer as to why some of the 90-day strength results

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are lower than those obtained with the 28-day cylinders."

We understand that your staff has now enlisted the technical services of the U.S. Army Corps of Engineers in an effort to illuminate the deterioration issue, and that several factors and combinations of factors are being investigated.

We are aware of one such possibility, which we communicated to your staff two months ago. It involves the possible presence of opaline in the aggregate portion of the concrete mixture. Opaline has, after numerous investigations, been determined to be responsible for the unusual phenomenon attending concrete made with river sand aggregate taken from northern Kansas rivers, including the Kaw, or Kansas, River: the concrete tends to expand and weaken over time, although this effect is seemingly somewhat unpredictable. It is our understanding that the source of the fine aggregate for the Wolf Creek base mat was originally to have been a limestone quarry near Ottawa, Kansas, operated by the Haworth Company, but that Daniels, the Wolf Creek general contractor, with the assumed knowledge of the applicants, changed the source to Kaw River sand, to be supplied by Holiday Sand and Gravel of Bonner Springs, Kansas. The change precipitated a lawsuit by Holiday, which is pending in Coffey County. We do not know that your staff has addressed this.

Accordingly, we inquire whether the ultimate source of the aggregate was properly approved by your staff and whether the presence of opaline aggregate has been determined and evaluated for its significance to the deterioration issue.

In sum, (1) the largest historical earthquake in Kansas was bigger than your staff and the applicants were aware and took place on a fault which passes 50 miles from the plant site, which is only now known to be active, and which may be developing a "seismic gap" in the vicinity of the nearest approach to the plant, (2) no evidence exists that the base mat could survive such an earthquake after a period of wear and tear due to normal plant operations, or at any time if spontaneous deterioration is taking place, and (3) evidence that such deterioration is taking place exists. It is therefore imperative that those making decisions about the Wolf Creek project know all that can possibly be known about the nature of the concrete in the base mat. We ask that you provide us a complete explanation of all the steps taken by you, other governmental agencies, the applicants or their agents to determine whether deterioration of the base mat can be expected.

Finally, we ask that you take action on our petition of January 19, 1979, concerning the Wolf Creek construction permit. It is your staff's position, expressed repeatedly, that the applicants' decision, without staff authorization, to remove the voluntary "hold" placed on construction of the containment building, would cause the staff to seek an immediate order from you, which they expect would be granted, requiring that such work be stopped. In fact, a vice-president of applicant KG&E advised your staff in writing at the time of the May 15, 1979 Burlington meeting that they indended to resume concrete placement in the reactor containment building within a few days. It is our understanding that "jawboning" by your staff dissuaded them. It remains our position that a partial construction permit suspension is the only effective way for your agency to protect the public interest in this situation, and we hereby renew our request that you act accordingly.

Very sincerely yours,

William H. Ward
Attorney for MACEA

WHW:bw

cc: Domenic Vassallo, NRC
Roger Boyd, NRC
Olan Parr, NRC
Carl Seyfrit, NRC
H. D. Thornburg, NRC
Stephen H. Lewis, Esq., NRC
S. J. Chilk, Secretary, NRC
Jay E. Silberg, Esq.
Kansas Congressional Delegation

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EVALUATION REPORT REGARDING THE

CONCRETE STRENGTH OF THE

REACTOR BUILDING BASE MAT

WOLF CREEK GENERATING STATION

On December 12 and 13, 1977, the Wolf Creek reactor building base mat was placed as a monolithic pour of approximately 6600 cubic yards of concrete. At the end of the 90-day curing period, thirty-four out of a total of sixty-six sets of concrete cylinders tested exhibited strengths below the specified concrete strength of 5000 pounds per square inch. Thirty sets of the concrete cylinders tested at 90 days had strengths which were lower than the strengths previously determined for the same batch of concrete after 28 days. The 5000 pounds per square inch strength for the concrete was specified by Bechtel (architect-engineer for the plant), in conjunction with other design parameters (e.g., base mat thickness and rebar arrangement), in order to satisfy the design criteria specified in the Wolf Creek Preliminary Safety Analysis Report (PSAR). These criteria require that the base mat be able to withstand, without impairment of its structural integrity or its safety function, the specified design loads and loading combinations.

Subsequently, the applicant conducted several investigations to determine the possible causes of the anomaly and submitted the results of the investigations in a report, dated October 26, 1978. The applicant concluded in its report that the 90-day strength of the concrete in the reactor building base mat was above 5000 pounds per square inch and that the apparent low strength of a portion of the 90 day cylinders was attributed to errors in testing.

The matter was investigated by the NRC Office of Inspection and Enforcement (IE) with the help of an outside consultant. As a result of the investigation, IE determined that the conclusions made by the applicant in its report of October 26, 1978, were not sufficiently supported by the facts contained in the report. Detailed findings of the investigation performed by IE are described in a report, dated February 16, 1979. Subsequently, the applicant performed additional studies in order to resolve the issues and concerns expressed by the NRC staff. At our request the applicant also performed a reanalysis of the base mat, based on the concrete strength indicated by the results of the 90-day cylinder tests, to determine if the design stresses are within allowable limits and whether the base mat design satisfies all commitments made in the Wolf Creek PSAR.

Additional tests were performed by the Construction Technology Laboratories of the Portland Cement Association on 48 concrete cylinder remnants previously tested at 90 days and on 26 cylinder remnants previously tested at 28 days. Cement compression strength tests were also performed on four cement samples. The additional concrete tests consisted of compressive strength tests on two-inch cubes sawed from the cylinder remnants, and petrographic examination and chemical analysis of a selected group of cylinder remnants. All of these test results are described in detail in reports submitted by the applicant by letters, dated February 28, 1979 and May 3, 1979. In addition, the Structures Laboratory of the Corps of Engineers, USAE Waterways Experiment Station, Vicksburg, Mississippi, conducted a petrographic examination of concrete thin sections and documented its conclusions in a report, dated July 2, 1979.

We have completed our review of the results of the tests performed by the Portland Cement Association and the evaluation performed by the Corps of Engineers. Based on our review of the test data, we conclude that there is no evidence of degradation of concrete strength nor is there any sign of substandard or faulty cement. However, we cannot conclude that the low 90-day strengths obtained with the cylinder tests are attributed to testing machine factors or testing conditions as claimed by the applicant. We note that the 90-day cylinder strength test results correlate very well with the two-inch cube compressive strength test results. In fact if both sets of results are plotted, the two curves would almost be parallel. Because of this excellent correlation between the strengths of cubes and cylinders, we conclude that the 90-day cylinder test results should be considered as valid and that these results should be used in assessing the load carrying capacity of the base mat.

The applicant completed the requested reanalysis of the base mat and the results were submitted by letter, dated May 10, 1979. In order to perform the reanalysis, the applicant first determined a concrete strength for the mat based on the 90-day cylinder test results by utilizing the established acceptance criteria in Section 4.3 of American Concrete Institute (ACI) Standard 318-71. We concur with the applicant that the resultant strength is 4460 pounds per square inch.

The reanalysis of the base mat was then performed in accordance with the original design commitments of the Wolf Creek PSAR by using the calculated concrete strength of 4460 pounds per square inch. A seismic soil-structure interaction analysis was performed by using the computer code FLUSH based on a finite element approach. Since the Wolf Creek plant is one of the five SNUPPS standard plant units, the SNUPPS envelope design earthquake ground motions of 0.20g for the safe shutdown earthquake (SSE) and 0.12g for the operating basis earthquake (OBE) were used to generate the seismic design forces for the base mat. The SSE and OBE for the Wolf Creek site are 0.12g and 0.06g, respectively.

The seismic forces generated by the finite element approach were compared with those generated by another established method of analysis, the fixed base approach, to demonstrate that the seismic loads used in the reanalysis are conservative for the Wolf Creek plant. For the fixed base approach, the Wolf Creek site specific design earthquake ground motions of 0.12g for the SSE and 0.06g for the OBE were used as input motions.

The results of the reanalysis by both approaches indicate that the base mat meets the design criteria for the Wolf Creek facility and that the tensile stresses of the reinforcing steel are controlling. Thus, the base mat design is controlled by tension and the load carrying capacity of the mat is governed primarily by the amount of reinforcing bars provided in the base mat. Lowering the specific concrete design strength from 5000 to 4460 pounds per square inch has very little effect on the load carrying capacity of the base mat.

Based on our review of the test results and the results of the reanalysis, we conclude that the base mat concrete strength has not retrogressed, that the strength of the base mat meets the original design criteria in the Wolf Creek PSAR, and that the mat will withstand the specified design loads and loading combinations without impairment of its structural integrity or its safety function.

JOHN C. MAXWELL
GEOLOGIST
3322 WESTERN HILLS DRIVE
AUSTIN, TEXAS 78731
512-454-1951

October 1, 1979

Mr. Harold Etherington
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Mail Stop H 1016
Washington, D. C. 20555

Subject: Letter of June 29, 197 , to the Commissioners, U.S.N.R.C., from William H. Ward, Attorney for Mid-America Coalition for Energy Alternatives

Dear Mr. Etherington:

Mr. Ward's letter raises substantative questions on which Mr. Muller asked me to comment. Undoubtedly, the Staff has considered these questions at length. My comments are based on material immediately available to me in Austin.

1. Validity of the choice of 0.12g for the Wolf Creek project.

The selection of an SSE of 0.12g was controlled by the 1867 Manhattan earthquake, MM intensity VII, epicenter approximately 20 miles northwest of Manhattan, Kansas. Additional research carried forward by the Kansas Geological Survey, under the sponsorship of the N.R.C., indicated that the epicenter lay east of Manhattan, closer to the Wolf Creek site, and approximately on the trace of the eastern boundary fault zone of the Nemaha buried uplift. The intensity was also reevaluated and raised to VII - VIII. Apparently, the evidence for these changes is described in NUREG/CR - 0294, which was not immediately available to me. The new position and intensity are, however, listed in NUREG/CR - 0?76, which I do have in my file.

Assuming both the newly located 1867 earthquake epicenter and higher intensity are valid, then the SSE for the Wolf Creek Project should be redetermined on the basis of a similar earthquake occurring 50 miles WNW of the project site along the eastern margin of the Nemaha structure. I must agree with

WOLF CREEK
J. MAXWELL

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Mr. Ward's contention that the regional structural setting of the Wolf Creek project is similar to that of Tyroné, for which the SER recommended an SSE of 0.2g horizontal acceleration. This value would seem to be about right for Wolf Creek also.

2. The principle thrust of Mr. Ward's letter is to call attention to the possible deterioration of concrete in the base mat of the reactor containment building, especially with regard to increased seismic risk related to the reevaluation of the 1867 Manhattan earthquake. The undesirable effects of opaline silica (usually as chert or chalcedony sand grains and pebbles) on the strength of concrete are well known. I'm sure the staff is evaluating this situation. In any case I have no basis for further comment.

Cordially,

John C. Maxwell

John C. Maxwell
Consultant to A.C.R.S.

VERY POOR ORIGINAL

A-410

JOHN C. MAXWELL
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NG-W

November 26, 1979

Mr. Harold Etherington
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Mail Stop H 1016
Washington, D. C. 20555

Dear Mr. Etherington:

Subject: Additional comments on my letter of October 1, 1979 referring to the Wolf Creek Project

At the time of writing my letter of October 1, 1979 I did not have available to me NUREG/CR-0294, containing information on damage reports relating to the 1867 Manhattan, Kansas earthquake. I have now had the chance to examine this material. It appears that the authors have thoroughly researched the available newspaper sources and have produced a reasonable isoseismal map for this earthquake (figure 3, page 17 in the NUREG/CR-0294). On the basis of this study the location of the 1867 epicenter was shifted about 20 miles southeastward to the vicinity of the subcrop of the north-south trending Humboldt fault, bordering the buried Nemeha uplift on the east side. It was further suggested that the intensity of the earthquake was in the range of VII-VIII rather than VII as previously listed.

The isoseismal lines shown in figure 3 are reasonably well established for the eastern half of the figure, but the shape and extent of the western half is unknown. Note also the highly elongated intensity VII zone, which could extend westward or southwestward for an unknown distance. From reading the accompanying damage reports, it seems to me that the choice of a location for the epicenter within this 150-mile long zone of intensity VII is quite arbitrary. For example, at the western end of the zone, location 28, it was noted that: "train on Pacific Railroad violently rocked by shock, locomotive was stopped and train men abandoned cab for fear the boiler was about to blow up," and to the east, at location 13, Leavenworth, Kansas, "man shaken off load of hay; two contiguous buildings lifted up, separated two inches, settled back; nearly everything toppled over in private homes; several chimneys overthrown." At the site chosen for the epicenter near Wamego (number 30 on figure 3) the damage to buildings appeared to

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be comparable to that observed throughout this long zone. The apparent reason for locating the epicenter here, other than its nearness to the trace of the Humboldt fault, was an observation that "on the farm of John Cotton . . . . during the earthquake the earth opened and water was thrown out of the opening in considerable quantities. At another place not far distant from the above, the earth opened and fire and smoke issued out. So one of our papers states". To this phenomenon the writers assign a questioned intensity VIII. If this were a bonafide major occurrence of sand boils in unconsolidated sediments, then an assignment of intensity VIII would probably be justified. However, in view of the very similar degree of damage throughout the isoseismal VII zone, and the lack of evidence of greater damage in the Wamego area it would seem to me more probable that the observed phenomena were related to escape of marsh gas and accompanying water. I see no compelling reason for raising the intensity in this area to VIII.

If the 1867 earthquake resulted from movement on the Humboldt fault, it would indeed be logical to assume similar motion could occur along the general extension of that fault southward, about 75 kilometers west of the Wolf Creek site. Figure 3 of NUREG/CR-0666 shows that this fault is discontinuous and the continuity and displacement diminishes in a southerly direction. Furthermore, the locations of the three earthquakes for which isoseismals were prepared (1867, 1875, and 1906) lie on an east-west trend approximating the VII isoseismal zone of the 1867 earthquake. The center trace of the isoseismal zone and eqicenters of the 1867, 1875 and 1906 earthquakes are plotted on figure 4, NUREG/CR-0666 (photostat attached). The possibility that the 1867 and 1906 earthquakes are related to the basic intrusive rocks of the Keweenawan mafic igneous belt, as initially postulated by the applicant is quite apparent. To me it seems more likely, however, that these and other earthquakes shown on figure 2, NUREG/CR-0294 occurred along an easterly trending fault or zone of faults approximately paralleling the VII isoseismal zone of the 1867 earthquake. Multiple shocks accompanying that earthquake, plus one aftershock some 25 hours later, also seem to have been distributed along this zone.

A map summarizing all known earthquakes for eastern North America, intensity I through XII for the period 1534-1971 has been published by Lynn Sykes (Reviews of Geophysics and Space Physics, November, 1978, page 648). Two observations of particular interest for the Wolf Creek area are: (1) that the 20 or so counties in the southeast corner of Kansas, and the adjacent areas in Missouri and Oklahoma, have been free of observed seismic activity, and (2) that the most obvious seismic zones trend east-west to east-southeast, reflecting the Ouachita-Wichita structures and the Manhattan and other earthquake trends in Kansas and adjacent states. There is no obvious seismic reflection of the north-south trending Nemaha structure. It seems unlikely, therefore, that large earthquakes of the Manhattan, Kansas type will occur southward along southerly extending faults corresponding to the Humboldt fault of northern Kansas and southern Nebraska. For this reason and because I find the evidence for an intensity VIII assignment for the 1867 earthquake to be unconvincing, I believe the assigned SSE value of 0.12g is still reasonable for the Wolf Creek site.

In his letter of June 29, 1979 to the NRC Commissioners Mr. William H. Ward, attorney for MACEA brought up another point, also touched in my October 1, 1979 letter, namely that the regional structural setting of the Wolf Creek Project is similar to that of Tyrone, for which the SER recommended an SSE of 0.2g horizontal acceleration. Both the Wolf Creek and Tyrone sites are in the Central Stable Region tectonic province. Both are also in regions which are seismically quiet. No historical earthquakes have been reported within 100 miles of the Tyrone site and only 10 earthquakes of intensity IV or greater have been reported within 200 miles of that site. The SER (page 2-16) says, "in our review of the vibratory ground motion potential for the Tyrone site, we took the position that an intensity MM VII-VIII event could occur at the site based on the criteria defined in appendix A to 10 CFR part 100. In our evaluation the resulting acceleration value due to this intensity would be 0.2g". Using the same line of reasoning it is apparently true that an intensity of 0.2g would now be similarly applied to the Wolf Creek site by the staff. This appears to be a matter of accepted policy, not specifically required by the tectonic setting of the Wolf Creek site.

Respectfully submitted:

John C Maxwell

John C. Maxwell
Consultant

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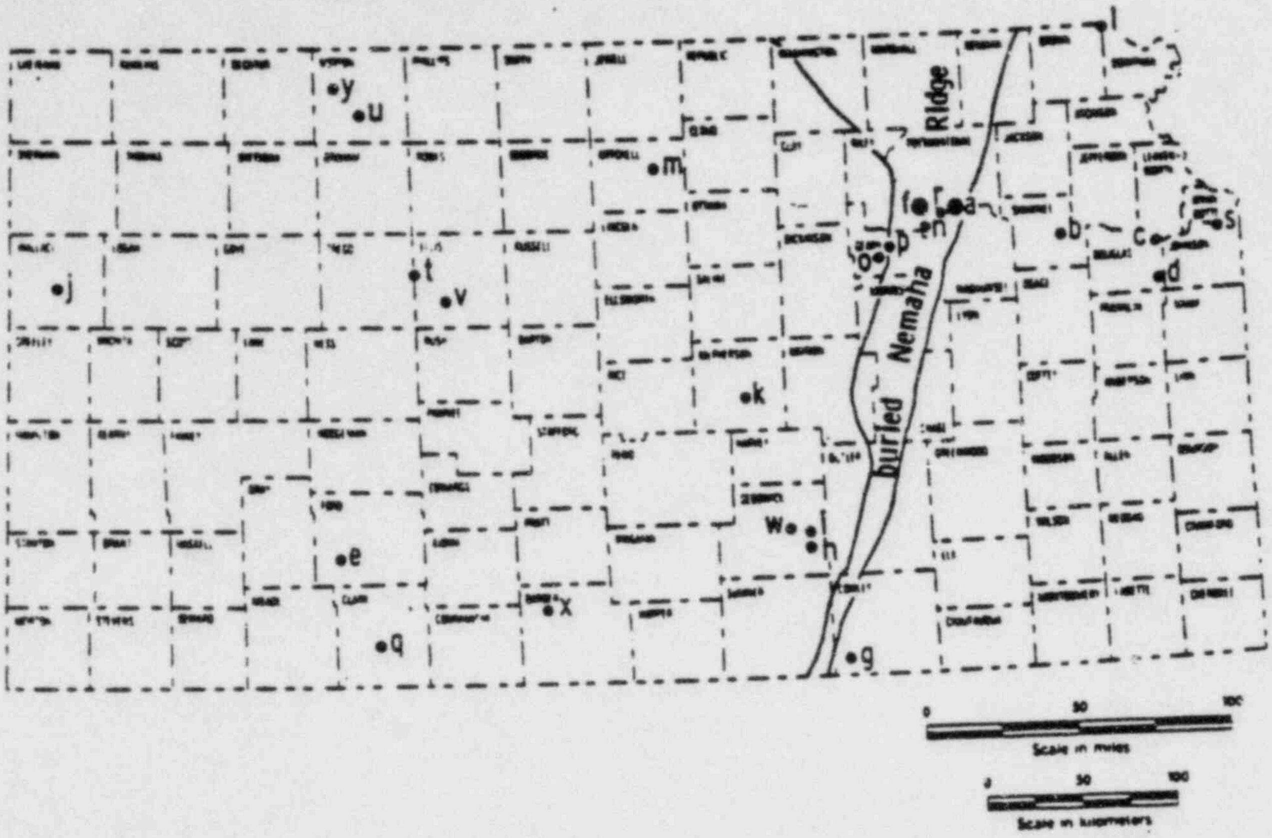
EARTHQUAKES IN KANSAS



Explanation

a	1867	VIII	f	1906	VII	k	1927	V	p	1929	V	u	1933	V
b	1875	V	g	1907	IV	l	1927	VI	q	1929	V	v	1942	IV
c	1881	III	h	1919	IV	m	1928	IV	r	1929	V	w	1948	IV
d	1903	II	i	1919	IV	n	1929	V	s	1931	VI	x	1956	VI
e	1904	IV	j	1926	?	o	1929	V	t	1932	VI	y	1961	V

FIGURE 2. LOCATIONS AND DATES OF EARTHQUAKES IN KANSAS, 1867-1977 (DUBOIS AND WILSON, 1978).

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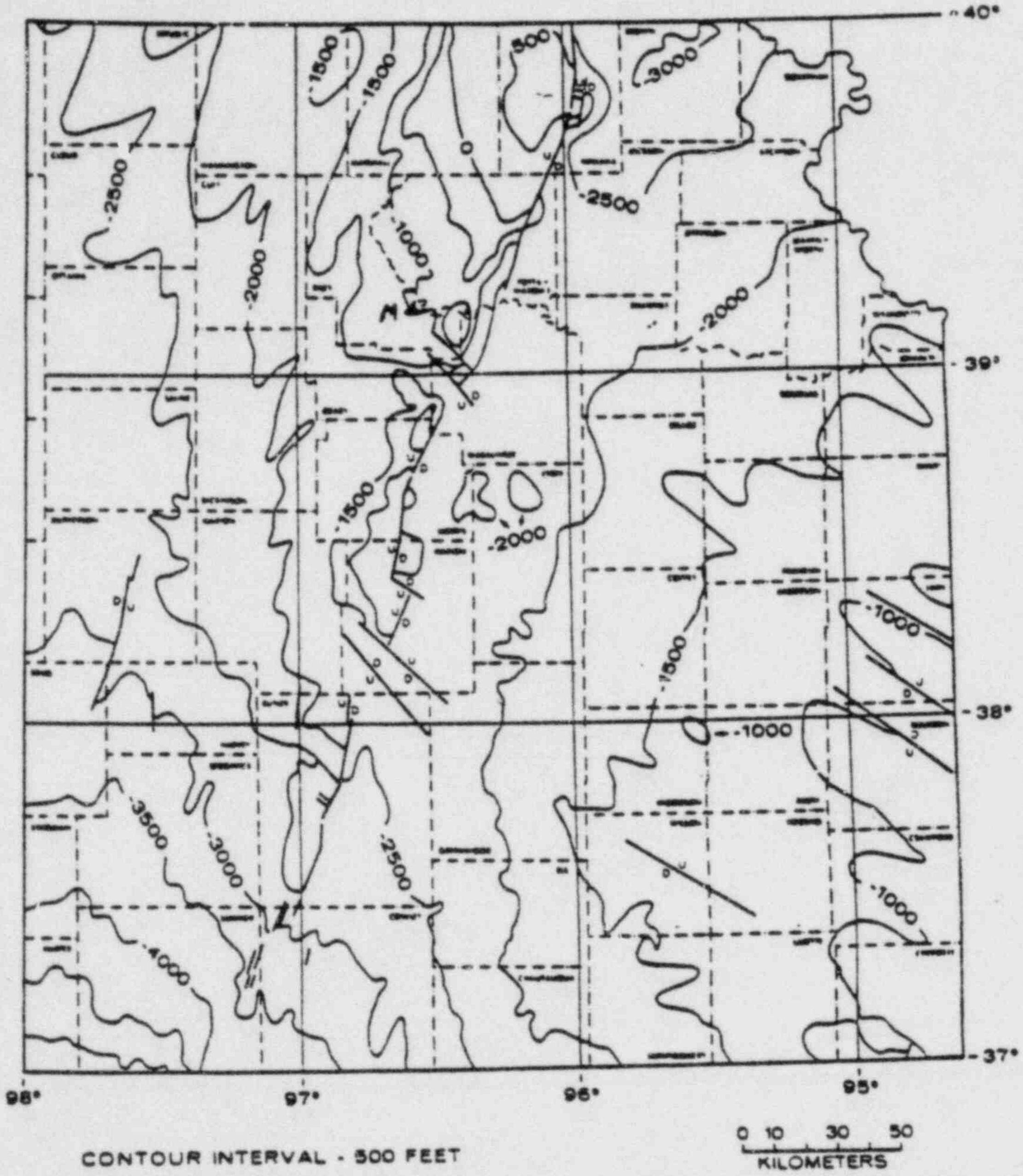

FIGURE 3. GENERALIZED CONTOUR MAP OF THE TOP OF PRECAMBRIAN BASEMENT ROCKS IN KANSAS (AFTER COLE, 1977).

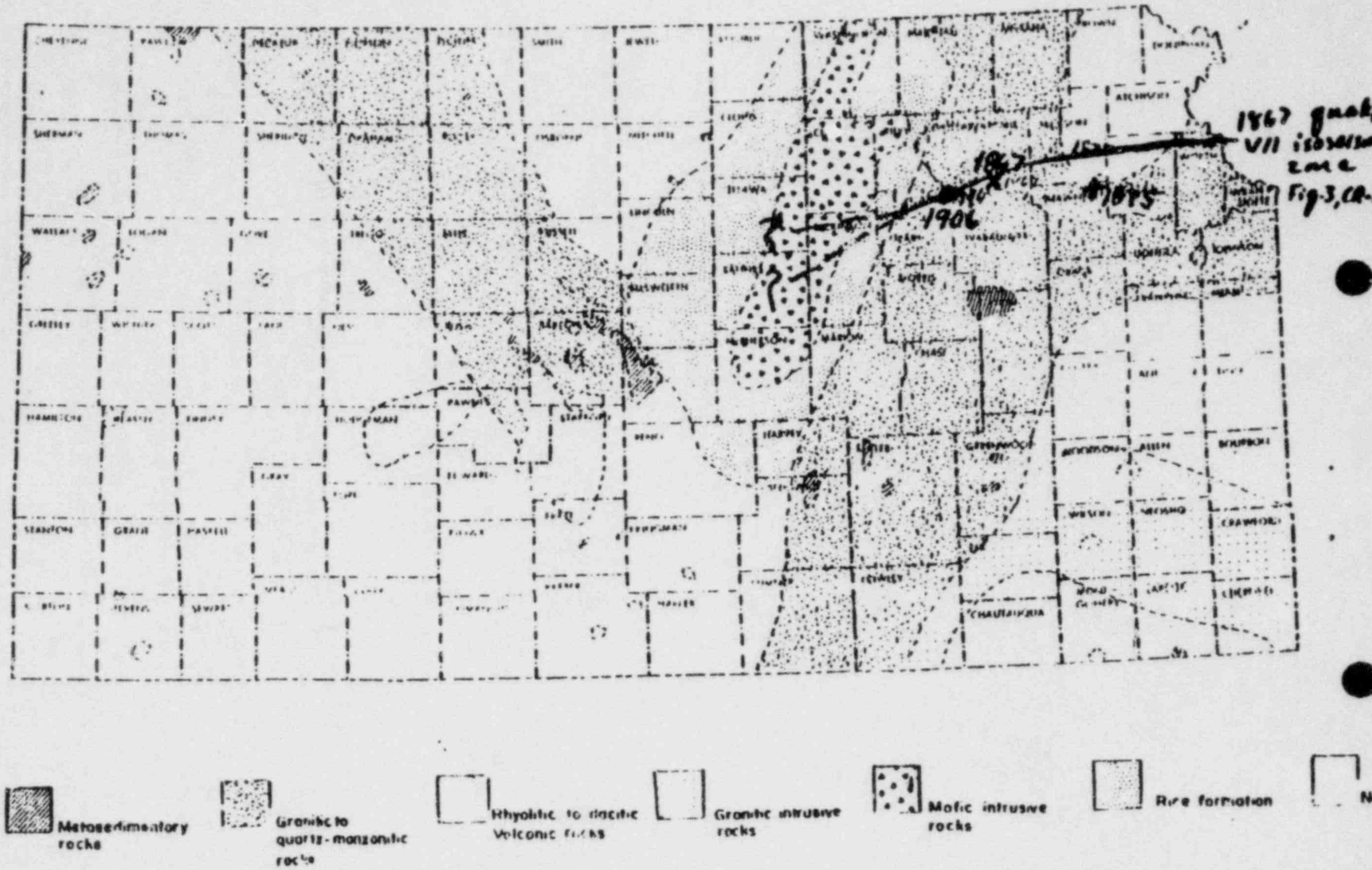

FIGURE 4. GEOLOGIC MAP OF PRECAMBRIAN BASEMENT ROCKS IN KANSAS (AFTER BICKFORD AND OTHERS, IN PRESS).

NUREG/CR-294

A-416

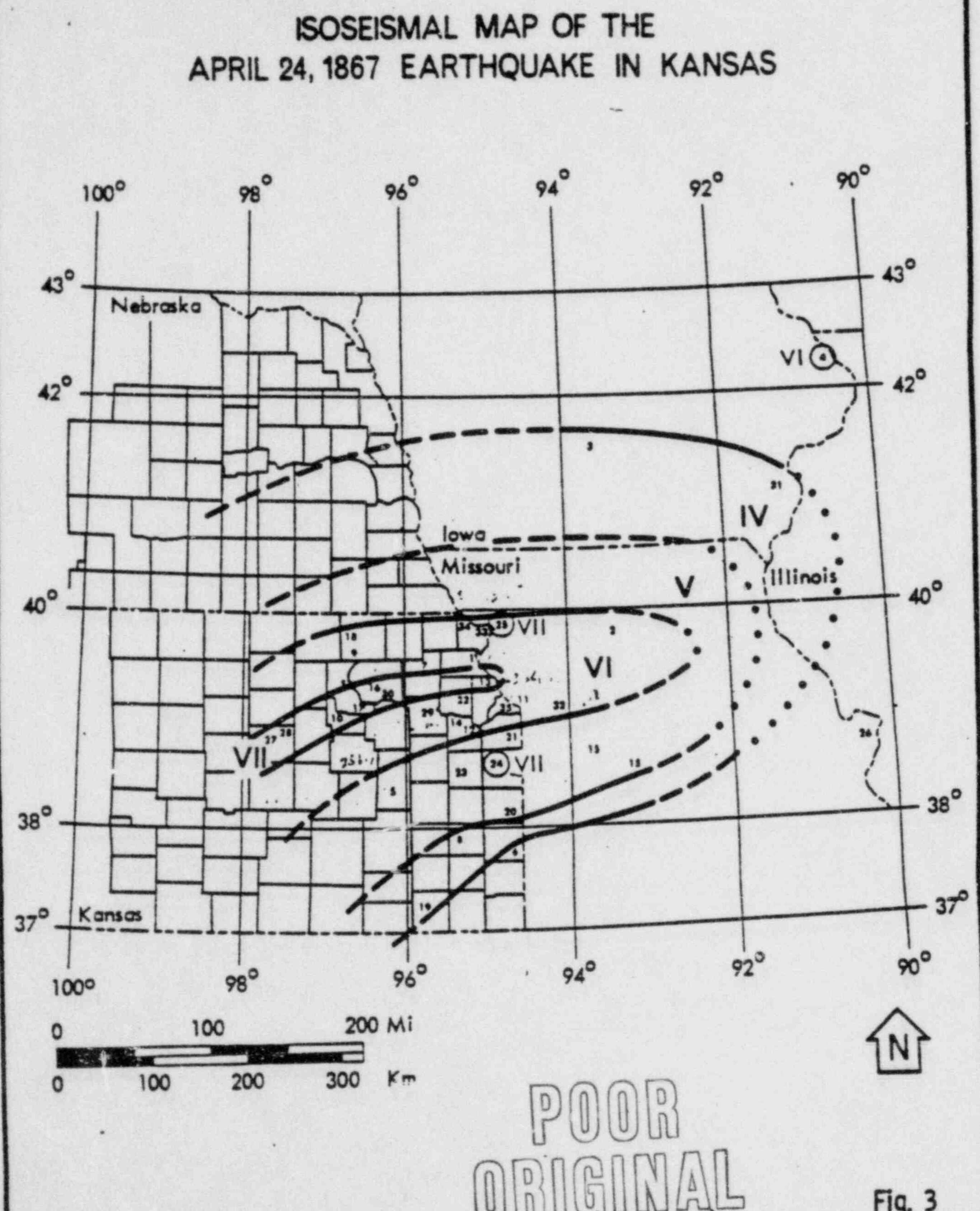
ISOSEISMAL MAP OF THE
APRIL 24, 1867 EARTHQUAKE IN KANSAS
100°
98°
96°
94°
92°
90°
43°
42°
40°
38°
37°
Nebraska
Iowa
Missouri
Illinois
Kansas
IV
V
VI
VII
0
100
200 Mi
0
100
200
300
Km
N
POOR
ORIGINAL

Fig. 3

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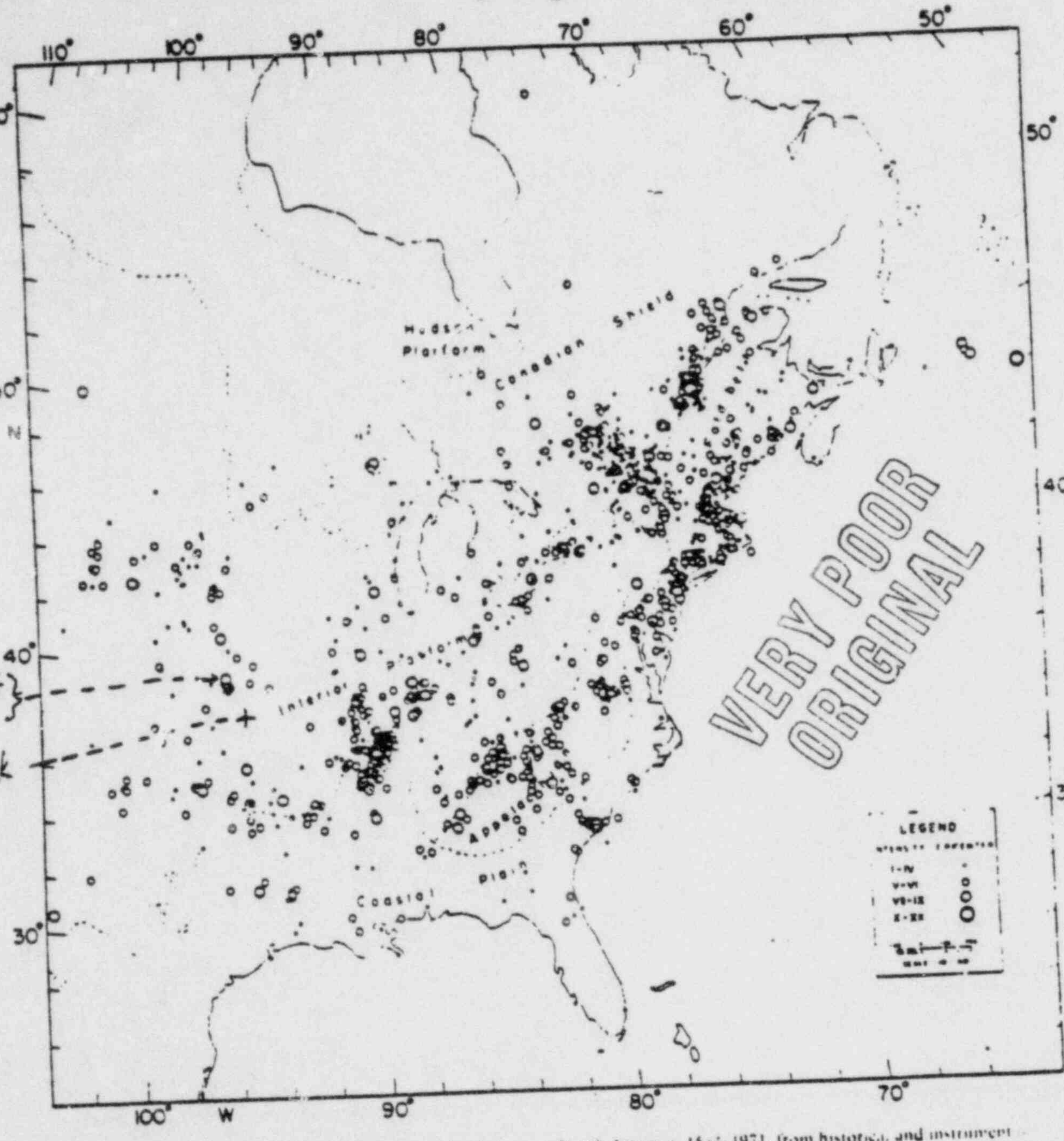

Fig. 14. Distribution of reported earthquakes in eastern North America, 1534–1971, from historical and instrumental data [after *York and Oliver*, 1976]. Note activity along the Appalachian fold belt, the northwest trending zone in New England and southern Quebec, and the northwesterly trend in South Carolina.

the existing evidence from instrumental locations argues for a region of low activity in Vermont. Thus new and historic data indicate that the Boston-Ottawa seismic zone is composed of two distinct zones of high activity: one extending from offshore Massachusetts into central New Hampshire and another extending northwest from northern New York State to Kirkland Lake, Ontario. As will be discussed later, however, alkalic rocks postdating the opening of the western Atlantic extend across the gap in seismic activity in Vermont and western New Hampshire.

Younger Igneous Rocks in New England and Southern Quebec

Although major tectonism and magmatism are commonly thought to have ended in most of the eastern United States in the Triassic, igneous rocks with ages postdating the initial separation of North America from Africa are found in the White Mountain magma series of New England, in the Monteregian Hills of southern Quebec, and along the New England seamount chain (Figure 18). The White Mountain magma series extends NNW across New Hampshire and ranges in age from about 200 m.y. (the initial stage of rifting of North America from Africa) to about 100 m.y. [*Foland and Faul*, 1977]. A small percentage of the radiometric ages are between 220 and 235 m.y. The Monteregian Hills, a group of alkalic mafic and ultramafic rocks in southern Quebec, trend WNW from north of New Hampshire to Montreal. Carbonatites and diatremes are found in the western part of the Monteregian province [*Gold*, 1967]. The $^{87}Sr/^{86}Sr$ ratio of 0.704 obtained by *Fairbairn et al.* [1963] for rocks of the Monteregian province

NOTE: "Pages A-419 -424 were duplicates of A-413-418 and
have therefore been removed"

RONDOUT ASSOCIATES, INCORPORATED

P.O. Box 224, [illegible] Ridge, New York 124[illegible]

RECEIVED

1979 NOV 5 PM 1 03

October 29, 1979

U.S. NUCLEAR REG. COMM. [illegible] REACTOR [illegible]

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Att'n: Ragnwald Muller, Senior Staff Engineer

Gentlemen:

Pursuant to your letter of October 18, 1979, I have conducted an assessment of the contentions contained in the June 29, 1979, letter to the Commission from the Mid-America Coalition for Energy Alternatives (MACEA) regarding the Wolf Creek Site in Kansas. In addition to the documents which you provided, I have consulted the documents listed in Appendix I below in the course of this investigation. As I have pointed out by telephone, my field of expertise is seismology, and I offer no opinion with regard to the base mat strength contentions.

Addressing specifically the seismological contentions, we will consider the following:

1. <u>The size of the April 24, 1867, earthquake.</u> The standard references list this earthquake as a Modified Mercalli Intensity VII (Coffman and von Hake, 1973; Docekal, 1970) and the Wolf Creek applicant used this value in the PSAR (pg. 2.5 - 101a). DuBois and Wilson of the Kansas Geological Survey (KGS) in their publication, "A Revised and Augmented List of Earthquake Intensities for Kansas, 1867-1977" - NUREG/CR-0294 have listed the event as Modified Mercalli Intensity VII-VIII based on their evaluation of published reports. All of the reports and the Dubois and Wilson intensity map are reproduced in Appendix II. Of the 35 numbered citations in their text, one (No. 30) is partially reproduced here.

"*Special Report from 3 mi. S. in Wabaunsee Co. - "on the farm of John Cotton,.... during the earthquake, the earth opened and water was thrown out of the opening in considerable quantities. At another place not far distant from the above, the earth opened and fire and

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smoke issued out. So one of our papers states." '

This report was published in the Topeka Commonwealth for April 24, <u>1877</u>, ten years after the event (in the Ten Years Ago Today' column according to the text of the NUREG document - pg. 4.) DuBois and Wilson on page 4 of the NUREG document state that this is a report of earthquake fountains or liquifaction and that this is a criterion for M.M. Intensity VIII. Thus, in the listing, they attach an VIII? rating to this report. This is the only report of an Intensity VIII effect in their report and thus their assigned rating of VII-VIII depends entirely on this report.

There are several problems with the report and the conclusion regarding intensity drawn from it; namely:

1. The original report itself is dated 10 years after the event took place.
2. The report states that an opening of the ground occurred and water was thrown out in considerable quantities. Intensity VIII criteria as presently formulated are "changes in flow or temperature of springs and wells. Cracks in wet ground and on steep slopes," while Intensity IX criteria involve "conspicuous cracks in ground. In alluviated areas, sand and mud ejected, earthquake fountains, sand craters." If DuBois and Wilson believe, as stated on pg. 4, that this is a report of earthquake fountains and liquifaction, it should have been rated as Intensity IX.
3. In assigning intensities and preparing an intensity map, one normally finds an isolated report or two that fall outside the range of values reported in the area. An isolated report like this one should be treated with caution. One might assign an Intensity VIII? to the report, but <u>to change the earthquake intensity on the basis of that isolated report alone is not justified</u>.

<u>Conclusion</u>: The earthquake should still be considered as Intensity VII.

2. <u>The location of the April 24, 1867, event.</u> The standard references place the earthquake 22 miles northwest of Manhattan. On the basis of the same report cited above (No.30) and on an account of horses falling down during the earthquake at Louisville, KS, (a few miles north of Wamego), DuBois and Wilson relocated the epicenter to "the general Wamego-Louisville-Manhattan area with the possible epicenter in the area of liquifaction mentioned above." The authors had pointed out earlier that "recent airphotos indicate that the probable location of the area of liquifaction was on the floodplain of the Kansas River closely adjacent to the subsurface trace of the Humboldt Fault." The authors after a complete review of all the original sources of Merriam's references for the 1867 event were not able to find any felt reports which would justify placement of the epicenter 22 miles northwest of Manhattan. An examination of any of the intensity maps published for this event indicate a wide area where the epicenter might be located. The location of DuBois and Wilson is not unreasonable given all the uncertainties involved.

<u>Conclusion</u>:

The 1867 may or may not have been located east of Manhattan (rather than 22 miles to the northwest of Manhattan), and, if it was, the epicenter could be spatially associated with the Humboldt fault trace.

3. <u>Microearthquakes associated with the Humboldt fault and "the development of a seismic gap"</u>

From 1 December, 1977, through the 2nd of August, 1979, the seismic network operated by the Kansas Geological Survey (and funded by the Nuclear Regulatory Commission) has recorded 39 microearthquakes occurring in Kansas and the adjacent states of Missouri and Nebraska. The latest annual technical report of the Kansas Geological Survey to the NRC dated August 1979 (NUREG/CR-R6, RA) indicates "Of these, 13 appear to be spatially associated with the Humboldt fault zone which forms the abrupt east side of the Nevada Uplift (Figure 1)." Figure 1 of that report is reproduced in Appendix III of this report. As can be seen from Figure 1, at least some of the 13 microearthquakes can be spatially associated with the Humboldt Fault. If such association implies a slight activity of the Humboldt Fault, such activity is not surprising. It has undoubtedly existed in the past although only with the advent of the network, can it be pinpointed. This small scale energy release north of the Wolf Creek site and south in Oklahoma can not be inpreted as indicating that stress is building up in the vicinity of the nearest approach of the fault to the plant. In fact, on 25 July, 1979, according to the KGS, an earthquake occurred near Potwin, Kansas, at 38.018°N and 96.983°W. Based on experience elsewhere, as microearthquake monitoring continues for many years, the activity should begin to fill in along the length of the fault. One cannot use even 13 microearthquakes along a fault several hundred kilometers in length to predict "seismic gaps".

Conclusion: Small scale microearthquake activity may be spatially associated with the Humboldt Fault but no evidence exists to support an incipient "seismic gap".

4. Implications of the above for the Wolf Creek Site. The Staff, in its treatment of the Wolf Creek site (SER pg. 2 - 20) addressed the two basic questions with regard to the site:

1. What is the maximum or upper bound earthquake that could occur on the Nemaha Uplift?

2. What is the maximum intensity value for the random earthquake in the region?

1. The Staff concluded that the upper bound earthquake on the Nemaha Uplift was less than Intensity X, and they assumed that it could occur as close as the nearest point of approach of the eastern edge of the Uplift (marked by the Humboldt Fault) to the site or 50 miles away. They concluded that the intensity at the site from the upper bound earthquake would be Intensity VII.

2. They also concluded that the largest random earthquake which could occur at the site was Intensity VII (such as the earthquake in Catoosa, Oklahoma (near Tulsa) in 1956. Therefore, if the April 24, 1867, earthquake, were Intensity VIII and if it were located on the Humboldt Fault trace at its point of closest approach to the Wolf Creek site, the intensity at the site would still be lower than that required by the Staff's analysis.

Conclusion: The Staff's analyses did not involve the direct use of the April 24, 1867 event as the controlling earthquake and the results of their analyses are not modified by the contentions in the MACEA letter of June 29, 1979.

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In summary, four points should be made:

1. The evidence that the April 24, 1867, earthquake should be considered as a Modified Mercalli VII-VIII event is, at best, marginal and should still be considered as a VII.

2. The earthquake of April 24, 1867, may or may not have been located east of Manhattan, Kansas, and, if it was, the epicenter could be spatially associated with the Humboldt Fault.

3. Small scale microearthquake activity may be spatially associated with the Humboldt Fault but no evidence has been brought forth to support the presence of an incipient seismic gap.

4. The Staff's analyses did not involve the direct use of the April 24, 1867 event as the controlling earthquake and the results of their analyses are not modified by the contentions in the MACEA letter.

I will be pleased to provide further information if you require it.

Sincerely yours,

Paul W. P[illegible]eroy

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Appendix I

Documents Utilized in this Study

BIBLIOGRAPHY

1. Earthquake History of the United States, Publ. 41-1 Revised Edition (Through 1970) U.S. Dept. of Commerce, NOAA, EDS, Boulder, Colorado, 1973.

2. Docekal, J. Earthquakes of the Stable Interior, with Emphasis on the Midcontinent, Unpublished Thesis, Univ. of Nebraska, 2 vols., 1970.

3. Steeples, D.W., S.M. DuBois and F.W. Wilson, Seismicity, Faulting and Geophysical Anomalies in Nemaha County, Kansas: Relationship to Regional Structures, Geology, v. 7, No. 3, March 1970.

4. DuBois, S.M. and F.W. Wilson, A Revised and Augmented History of Earthquake Intensities for Kansas, 1867-1977, U.S. Nuclear Regulatory Commission NUREG/CR-0294.

5. Wilson, F.W., A Study of the Regional Tectonics and Seismicity of Eastern Kansas-Summary of Project Activities and Results to the End of the Second Year on September 30, 1978. U.S. Nuclear Regulatory Commission NUREG/CR-0666.

6. Wilson, F.W., Nemaha Uplift Seismotectonic Study-Regional Tectonics and Seismicity of Eastern Kansas, Technical Progress Report October 1, 1978 to September 30, 1979. U.S. Nuclear Regulatory Commission, NUREG/CR-R6, RA.

7. Kansas Gas & Electric Co., Wolf Creek Nuclear Generating Station, Unit Docket No. 50-482, ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

8. Pages from Wolf Creek Generating Station Addendum, Vol. 2 PSAR
   2.5-1 through 2.5-6
   2.5-25 through 2.5-30b
   2.5-83 through 2.5-122A
   TABLE 2.5-11 through 2.5-15
   TABLE 2.5-18a through 2.5-18b
   TABLE 2.5-20 through 2.5-21
   FIGURE 2.5-17c through FIGURE 2.5-18
   FIGURE 2.5-20aa through 2.5-20v
   FIGURE 2.5-33 through 2.5-35c
   FIGURE 2.5-36 through FIGURE 2.5-38d

9. Safety Evaluation Report - NUREG-75/080 related to construction of Wolf Creek Generating Station, Unit No. 1 Docket No. STN 50-482, Sept. 1975 Kansas Gas & Elect. Co./Kansas City Power & Light Co. Pages 2-15 through 2-23.

10. Ltr. to Hendrie etc. from Mid-America Coalition for Energy Alternative/ Wm. H. Ward, Attorney for MACEA

11. Ltr. to Karl V. Seyfrit, from F.R. Brown, Dept. of Army with enclosures.

Appendix II

Earthquake Reports and Intensity Map

for the April 24, 1~~9~~867

Kansas Earthquake

APRIL 24, 1867 MM VII-VIII

(See Figure 3, p. 17, for location of the reports listed below)

Lat: 39°10' Long: 96°18' Near: Wamego (see Fig. 2, a)
Time: 2:30 p.m. Felt Area: 300,000 sq.mi. (8,17,45)**
(777,000 sq.km.)

Where reports are available, the following information has been included:
(a) Time
(b) Duration & Number of Shocks
(c) Direction of Wave Movement

Locality	Assigned MM Intensity	Earthquake Effects
1. Atchison, KS (Atchison Co.) (b) 2 shocks (c) S - N	VI	•Every building rocked to-and-fro •Lamps thrown from tables & mantles •Bottles from drug store thrown down •People fled from buildings to streets •Water in White Clay Creek moved rapidly after a standstill for several days •No damage reported to buildings (5) •Vibration passed westward or northward •Wave moved from south to north •First oscillation followed by heavier more perceptibly felt swell (12,13)
2. Chillicothe, MO (Livingston Co.) (a) 3:30 p.m. (b) one shock	VI	•Severe enough to cause plaster to fall from ceilings of several houses (49)
3. Des Moines, IA (Polk Co.)	VI	•Rocked persons sitting in chairs •Shook buildings (49)
4. Dubuque, IA (Dubuque, Co.) (b) 3 shocks	VI	•Three shocks felt •Openings formed in brick walls •Furniture displaced •Persons in chairs undulated backwards & forwards •Windows rattled, pictures shook, chandelier swayed •Not felt severely by ground floor - much felt by occupants of 2nd and 3rd stories (49) Panic - people fled to the streets Plastering came down in courthouse & other buildings (8,49,50)

**Numbers in parentheses refer to the References at the end of this report.

ocality	Assigned MM Intensity	Earthquake Effects
4. Dubuque, IA (cont.)		Gas burners vibrated like pendulums Cases shook in newspaper room (50)
5. Emporia, KS (Lyon Co.) (a) 2:30 p.m. (b) more than one	V - VI	*Low rumbling sound followed by vibrations *Houses shook, windows rattled *Panic - people fled from buildings *Brick & stone houses more severely affected than frame houses *Small boxes fell off shelves (49)
6. Fort Scott, KS (Bourbon Co.)	II - III	*Slight trembling in buildings, not alarming (49)
7. Holton, KS (Jackson Co.) (a) 2:00 p.m.	VI	*Goods & wares fell off shelves *Shook buildings *People fled to the streets (49)
8. Iola, KS (Allen Co.) (a) 2:45 p.m.	VI	*Shook houses *Rattled crockery (49)
9. Irving, KS (Marshall Co.) (a) 2:30 p.m. (b) lasted 30 sec.	VI	*Rumbling sound heard before shock *Houses shaken severely *Inmates rushed out of doors *Lasted 30 seconds (49)
10. Junction City, KS (Geary Co.) (a) 2:30 p.m.	VI	*Very heavy shock *Rocked buildings to-and-fro, moving several inches (31,49) Destroyed well being dug in town (17,31,49) *Shock seems not to have extended over a quarter of a mile in width (31)
11. Kansas City, KS (Wyandotte Co.)	VI	*Books unshelved *Tables moved *Pendant articles swung (bridles & harness) *Two clock doors suddenly opened *Crack in wall open & shut *Water in tumblers spilled *Plastering shaken off in one or two houses *General panic - people fled to streets *Every movable article of furniture & crockery rattled & shook about (49)

April 24, 1867 (continued)

	Assigned MM Intensity	Earthquake Effects
awrence, KS Douglas Co.) (a) 2:57 p.m. or 2:45 p.m. (b) 2 or 3 shocks	VI	Three shocks felt over a period of 30 seconds (17) Earth trembled & vibrated Doors & windows violently shaken (8,43) *Type thrown down in printer's office *Butcher's spring balance drawn down 1 1/2 lbs. (5,33) Bottles shaken off druggist's shelves (8,17,43) Plaster broken off Loud rumbling noise (6,17,43,49) Three - four loose stones knocked off Unitarian Church (17,49) Rattled crockery, glassware, shook bundles from shelves (8,49) *Building with stone walls 30-inches thick shook very perceptibly *People fled to streets *One stove overturned in a house *Books fell off shelves (49)
. Leavenworth, KS avenworth Co.) (a) 2:30 p.m. (b) 3 shocks felt, 30 sec. duration (c) W - E	VII	Plaster cracked entire length of ceiling - large portion fell to floor *Man shaken off load of hay *Two contiguous buildings lifted up, separated two inches, settled back *Dishes, tumblers knocked off shelves *Visible agitation of water in river *Clocks stopped at 2:30 p.m. *Nearly everything toppled over in private homes *Plaster fell in brick law office, several other buildings *Six-foot saws leaning against wall moved out six inches *Rumbling like thunder (49) *Stove pipe forced apart, some joints overlapping four inches *Several chimneys overthrown *Tables danced, dishes thrown to floor *Piles of sheeting toppled down from counters in post office *Plaster badly cracked in Billiard Hall (40,49) *Woman received electrical shock from spring water, smoke seen to come from bank (34) *Shocks moved from west to east (41)

April 24, 1867 (continued)

Locality	Assigned MM Intensity	Earthquake Effects
14. Lecompton, KS (Douglas Co.) (a) 2:30 p.m. (b) one shock (c) came from SE or NW (2 conflicting reports)	V - VI	*Panic - people fled to streets *Lane University building quivered *Windows & doors danced (49)
15. Lexington & Sedalia, MO (Lafayette & Pettis cos.)	VI	*Felt with equal force at Kansas City, Lexington, Sedalia, St. Joseph (49)
16. Louisville, KS (Pottawatomie Co.)	VII	*Horses fell down in streets *Chimneys toppled & fell (49)
17. Manhattan, KS (Riley Co.) (a) 2:32 p.m. (c) S - N or SW - NE	VII	Two-foot wave observed to move south to north on Kansas River (8,17,34,45,49) Clocks stopped *No wave observed on Blue River *Stacked photographs pitched over to SW Cattle alarmed *Oscillation of houses seemed to approach the "overtopping point" (49) *Inhabitants severely frightened *Some people felt electric shocks (34,49) Stone buildings with weak walls fractured but did not fall (8,17,34,49) *Aftershock occurred between 3 - 4 a.m. Thurs. (one day later) (34)
18. Marysville, KS (Marshall Co.) (a) 2:30 p.m. (b) 1 - 3 minutes	VI	*Temporary alarm on part of a few *Felt by people on first and second floors *Fisherman on Spring Creek felt tree shake, saw all the others trembling Stone high school much shaken - along with desks, stove-pipes, & other furniture (72) Rumbling sound - like heavy trunks being dragged across planks Windows, doors, shutters, stove-pipes, all loose or hanging articles rattled, waved, swung back & forth fearfully (8,72) Bottles & packages rattled, some shaken off shelves & broken (17,72)

L ...Y	Assigned MM Intensity	Earthquake Effects
19. Montgomery Co., KS	V	*Shook buildings *Knocked dishes off shelves *People in moving vehicles did not feel it (witness was Topeka Weather Bureau man in 1906) (54)
20. Mound City, KS (Linn Co.) (a) 3:00 p.m. (b) 15 seconds	V	*Houses violently shaken *Doors opened *Water shaken from buckets *Loose articles tumbled around (49)
21. Olathe, KS (Johnson Co.)	V	*Houses seen to totter, wave back & forth *Shingles on roofs broke loose, fell to ground *Glassware rattled *Deep rumbling sound (49)
22. Oskaloosa, KS (Jefferson Co.) (a) 2:34 p.m. (b) 15 - 20 sec.	VI	*Houses vibrated *Movable items shaken & jostled *Public panic - people fled to streets *Rumbling noise *Cupola of new school house reeled like drunken man (49)
23. Ottawa, KS (Franklin Co.)	V - VI	*Houses emptied of occupants *Buildings shaken (49)
24. Paola, KS (Miami Co.) (a) 3:20 p.m. (b) 50 sec. (c) W → SE motion	VII	*Plaster fell from ceiling of large schoolhouse *Buildings rocked *Large brick building which housed the <u>Republican</u> newspaper office much injured - one side knocked down & destroyed *West to southeast motion *Those in eastern part of town nearly thrown down if standing *Sound - rolling of large train over railroad (49)

Locality	Assigned MM Intensity	Earthquake Effects
25. St. Joseph, MO (Buchanan Co.) (a) 2:35 p.m. (b) 20 sec. (c) E → W & W → E	VII	*Rumbling noise *Shaking of entire surface of terra firma *Drove everyone into streets *Four-story brick buildings shaken from cornice to foundation stone *Windows broken, plastering thrown down *Ladies fainted, men turned pale *Solid brick blocks swayed to & fro like reeds (49) Buildings shook, walls cracked, rocked, jarred (17,49) *Brick walls of new school house, standing on elevated piece of ground where street had been cut down, cracked several feet from ground & bank on which it stood was also rent in a distinct seam (13)
26. St. Louis, MO (St. Louis Co.) (a) 3:00 p.m	II - III	*Shock felt here about 3:00 p.m. (49)
27. Salina, KS (Saline Co.) (a) 2:30 p.m. (b) 10 sec.	III (?)	*Shaking lasted 10 seconds, no damage reported (49)
28. Solomon, KS (Saline Co.) (a) 2:25 p.m.	VII	Train on Pacific RR violently rocked by shock, locomotive was stopped and trainmen abandoned cab for fear the boiler was about to blow up (16,31,45,49)
29. Topeka, KS (Shawnee Co.) (a) 2:45 p.m. (b) 2 shocks (c) SW → NE	VI	*Waves in ceiling of Lincoln College observed to run southwest to northeast (49) *People fled to streets *Stone church rocked (49,53,56) *Ceiling of Methodist Church bent up and down like waves on a pond *Floor heaved & sank lower than its normal level *Horses broke loose from hitching racks & ran toward open country (53,56) All but one glass window broken in schoolhouse "below this city" (10,17)

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Locality	Assigned MM Intensity	Earthquake Effects
30. Wamego, KS (Pottawatomie Co.) (a) about 2:45 p.m.	VI - VII	*Shaking & rocking of every house *General alarm - people fled from buildings *Plaster broken in houses *Glasses shaken from lamps (49)
	VIII (?)	*Special Report from 3 mi. S. in Wabaunsee Co. - "on the farm of John Cotton, ... during the earthquake the earth opened and water was thrown out of the opening in considerable quantities. At another place not far distant from the above, the earth opened and fire & smoke issued out. So one of our papers states". (10) Walls cracked (17)
31. Wapello, IA (Louisa Co.)	IV	*Motion of tremor described as "not violent, but easy swinging, giving one a sensation something like the first effects of a dram of wiskey". (50)
32. Warrensburg, MO (Johnson Co.) (a) 2:50 p.m. (b) 10 sec. (c) SW → NE	VI	*Walls of church heaved "as if moved by a shock from SW" *Glassware shook about *Plastering fell from ceiling *Buildings moved *No damage (?) (49)
33. Wathena, KS (Doniphan Co.) (a) 3:05 p.m. (b) 10 sec.	III (?)	*Small earthquake visited this section at 3:05 p.m. - lasted 10 sec. (49)
34. White Cloud, KS (Doniphan Co.) (b) 2 shocks	V (?)	*Two distinct severe shocks felt (49)
35. Wyandotte, KS (Wyandotte, Co.) (a) 2:00 p.m. or 2:45 p.m. (c) N → S motion	VI	*Doors jarred open *Windows rattled & jarred *People fled to streets *Houses swayed *Dishes shook *People awakened from naps (49)

VERY POOR ORIGINAL

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stionable Report:

Carthage, Ohio

Three mi. S. of Carthage on Miami Canal, an acre of ground sank 10', leaving a perpendicular wall of 10' on all sides (8,17,45,49)

Comments: An estimated felt area of 95,000 sq. mi. is also found in Docekal (17). Equally strong felt reports exist from Leavenworth, Paola, Wamego, Louisville, Manhattan, and Solomon, KS. All of these towns, excluding Leavenworth and possibly Paola, were situated in alluvial valleys which may have served to amplify the effects of the shock. Documentation is limited because of the sparse population in 1867. The isoseismal map (Fig. 3) has been constructed with open contours to the west due to lack of reports in that direction.

VERY POOR ORIGINAL

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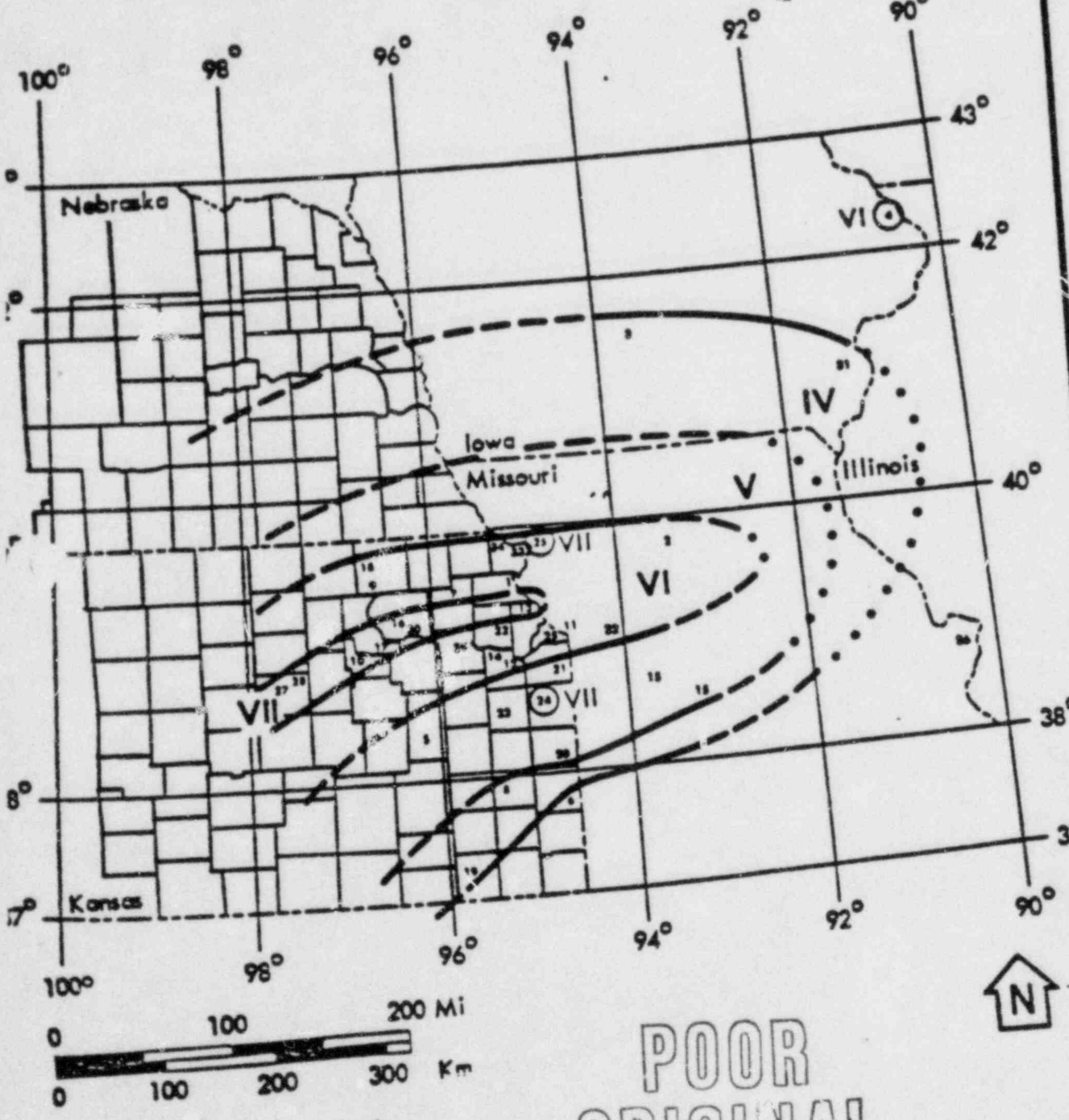
ISOSEISMAL MAP OF THE
APRIL 24, 1867 EARTHQUAKE IN KANSAS



POOR ORIGINAL

Fig. 3

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Appendix III

Figure I from the August 1979 Annual Technical Report of the Kansas Geological Survey (NUREG/CR-R6, RA) Showing the Location of Some of the Microearthquakes Spatially Associated by the KGS with the Humboldt Fault.

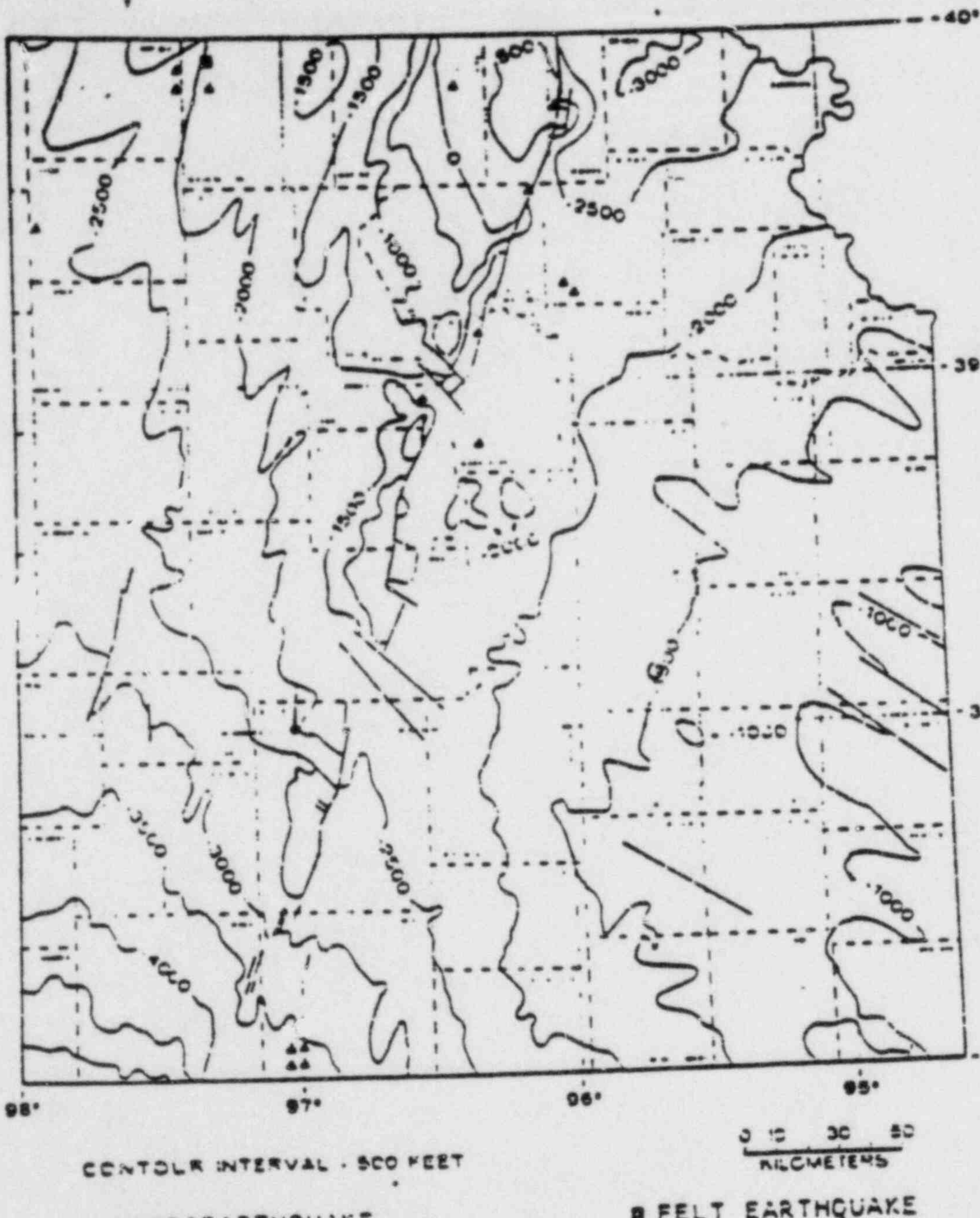

FIGURE 1. GENERALIZED CONTOUR MAP OF THE TOP OF PRECAMBRIAN BASEMENT ROCKS IN EASTERN KANSAS.

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EXTRACT FROM MINUTES OF 235th ACRS MEETING NOV 8-10, 1979

3. Wolf Creek

(For Background information on the concerns regarding site seismicity, see Appendix XXV.)

Mr. Etherington, Subcommittee Chairman, recalled that Wolf Creek is a RESAR-3 SNUPPS plant, for which the Committee wrote a construction permit report on October 16, 1975, and that the permit was issued by the NRC on May 11, 1977. He discussed the reports received from ACRS consultants (see Appendix XXVI), and noted that these reports questioned the conclusions made by the Kansas Geological Survey. He said that these consultants found no evidence of incipient seismic gaps. With respect to the concrete already poured at the site, he said that the Army Corps of Engineers has tested the concrete, and found no significant differences between the "good" and the "poor" specimens; all are of good quality.

Mr. Seiss suggested that the current contentions are derived from deficient tests. He noted that there is no indication of the presence of opaline in the concrete.

The ACRS Staff was requested to provide the ACRS consultant, J. Maxwell, with background information regarding the seismic design basis of the Wolf Creek Nuclear Station for his evaluation of the seismic design criteria of the plant. If Maxwell's report warrants, the Extreme External Phenomena Subcommittee will review the matter further.

ATTACHMENT 5

A-442a

4. Wolf Creek

Mr. Etherington, Subcommittee Chairman, noted that the Subcommittee's attention has been directed to the seismic design basis for Wolf Creek by a letter to the Commissioners from the Mid-America Coalition for Energy Alternatives (see Appendix IX). He said that a report has been received from J. C. Maxwell, ACRS consultant, on this matter (see Appendix X).

The Committee agreed to defer further action regarding the condition of the concrete base mat of the Wolf Creek Generating Station until the NRC Staff Safety Evaluation Report is available. The ACRS Staff was directed to obtain the services of Dr. Pomeroy, ACRS consultant, to further evaluate the adequacy of the seismic design of Wolf Creek.

Appendix XXII

POOR ORIGINAL

Date of Meeting: 2/
Date Issued: 2/

MINUTES OF ACRS PROCEDURES SUBCOMMITTEE MEETING
February 6, 1980
Washington, D.C.

Summary

I) Proposed Procedures for ACRS Participation in NRC Rulemaking Process

The Subcommittee endorsed Attachment 1 as the basis for a change in NRC regulations to cover ACRS participation in the rulemaking process.

II) Proposed Procedures for ACRS Handling of Dissenting Professional Opinions

The Subcommittee endorsed Attachment 2.

III) Proposed Procedures for Management of the ACRS Fellowship Program

The Subcommittee endorsed Attachment 3.

Proposed use of a Brazilian national as an ACRS Fellow as part of an International Exchange Program was discussed. It was agreed that his professional qualifications will be considered to determine if he can contribute to the program.

A policy for annual review of performance and related salary was discussed.

IV) Comments by ACRS Members Mathis and Ebersole (Attachments 4 and 5) Regarding ACRS Procedures

With respect to procedures regarding improved conduct of ACRS meetings (Mathis) and review of Supplementary SER's (Ebersole) the members endorsed the following:

- Meeting notices should state the specific purpose and objectives of meetings.
- ACRS consultants should be more clearly informed regarding what is expected of them at meetings.
- Specific minimum limits should be set regarding receipt of documents prior to meetings (A target of 2 weeks and an absolute minimum of 1 week was suggested.)

A-445

. A method is needed to provide for input from all Committee meetings earlier in the Subcommittee review. Several alternates were suggested:

- Members should identify areas of concern/interest during the discussion of Anticipated Subcommittee Meetings which is scheduled during each full Committee meeting. In order to facilitate this discussion a list of topics and meeting objectives should be provided.

- An initial session would be held with the full Committee and the Subcommittee would pursue those areas identified as needing further attention.

. Discussion during full Committee meetings should give more recognition to the work done during Subcommittee meetings. If a member has not identified topics to be explored by the Subcommittee and has not attended the related Subcommittee meetings, he should refrain from extensive detailed questioning during the full Committee sessions.

. Members should do adequate homework prior to ACRS meetings so they are better able to focus their questions during meetings.

. The ACRS Subcommittee Chairman with the assistance of the cognizant ACRS Staff engineer should examine Supplementary SER's and inform the Committee of areas where the NRC Staff is not implementing ACRS recommendations adequately. The Committee will then take appropriate action.

V) <u>Proposed Reorganization of the ACRS Technical Staff to Provide Improved Support of Committee Activities</u>

Based on the assumption that ten additional permanent, full-time technical staff members, as requested by the Committee is approved, a proposed reorganization of the ACRS technical staff was discussed. Attachment 6 was endorsed.

VI) <u>Proposed Procedures for Improved Interface Between the ACRS and the Commission</u>

To improve the opportunity for better contact/discussion with the Commission and EDO the NRC Chairman and the EDO should both be invited to ACRS meetings each month to discuss regulatory policy, problems, objectives, etc. All of the other Commissioners should be invited to attend sessions with the Chairman where they may have an interest.

A-445

Consideration should be given to a practice where the Chairman's Technical Assistant attends ACRS meetings as an observer.

VII) <u>Recommendations of Special Inquiry Group Regarding ACRS Activities</u>

The suggestion that five ACRS members should be the members of a full-time, independent Nuclear Safety Board was discussed. The Subcommittee agreed that a joint ACRS/NSB should not be supported by the Committee.

A proposal to review the recommendations of the Rogovin Report which have safety significance and compare them with the Kemeny Report and ACRS Report on the Regulatory Process (NUREG-0642) was not endorsed.

VIII) <u>Sustained Performance Award for the ACRS Technical Staff</u>

The Subcommittee endorsed a Sustained Superior Performance Award accompanied by a cash award for members of the ACRS technical staff.

IX) <u>Miscellaneous</u>

- Dr. Carbon suggested that the Committee should designate a Planning Subcommittee to organize future ACRS activities (e.g., time devoted to generic, cosmic, specific issues, research facilities).

- Dr. Plesset noted the inquiry of Mr. Paul Leventhal (Senate Subcommittee on Nuclear Regulation) regarding areas where the ACRS could assist the Subcommittee. The ACRS Executive Director was asked to follow-up regarding this matter.

G: VYDEC:1st disc:A15
DRAFT 2
Fraley/car
2/7/8[illegible]

MEMORANDUM TO: Chairman Ahearne
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Hendrie
Commissioner Bradford

SUBJECT: PROPOSED PROCEDURES FOR ACRS PARTICIPATION IN THE NRC RULEMAKING PROCESS

This paper addresses three aspects of ACRS participation in the rulemaking process:

- The <u>first</u> addresses the implementation of the recommendations of the President's Commission on TMI that:

  The ACRS should have the power to initiate a rulemaking proceeding before the agency to resolve any generic safety issue it identifies.

- The <u>second</u> addresses the procedures by which the ACRS can best participate in and contribute to the development and promulgation of NRC rules which are being formulated by the Commission (see memorandum from S. Chilk to Leonard Bickwit, Jr., dtd. January 16, 1980).

- The <u>third</u> addresses ACRS participation in the hearing process itself which may accompany the development of a particular rule.

POOR ORIGINAL

I. <u>ACRS Participation in NRC Rulemaking - ACRS Recommendations Regarding Promulgation of Needed Rules</u>

In a recent report to the Commission (Dr. Milton Plesset, ACRS Chairman, to Dr. John Ahearne, NRC Chairman dated January 15, 1980, "Recommendations of President's Commission on ACRS Role") the Committee proposed that this be implemented as follows:

> The Committee agrees with the thrust of this recommendation but believes that the Commission would, as a matter of course, initiate a rulemaking proceeding when recommended by the ACRS.

This appears to be an appropriate interpretation of the recommendations of the President's Commission in view of the advisory nature of the Committee and the Committee's belief that the Commission will respond to specific recommendations in an appropriate manner within a reasonable period of time.

<u>Recommendation</u> - Appropriate followup procedures to deal with ACRS reports and recommendations will deal with this matter adequately. A rule change should be promulgated to indicate that the Commission will respond to such recommendations on the public record within a reasonable period of time (e.g., 30 days).

II) <u>ACRS Participation in Rulemaking - Development of ACRS Comments and Recommendations Regarding Proposed Rules and Regulations.</u>

[. Alternate 1] - ACRS comments on proposed final rule after public comments have been incorporated and the hearing process (if held) is complete.

<u>Advantages</u>

1. Makes available to the ACRS the input from public comment and the hearing processes in the preparation of ACRS recommendations.
2. Provides for a single step [illegible] at a time when it is considering a completed [illegible]

<u>Disadvantages</u>

1. Could delay promulgation of the rule if significant changes resulting from ACRS comments must be resolved/incorporated. Could require that the public/comment - hearing process be repeated.

2. Provides for ACRS comment at a time when a major investment in NRC resources has been expended and staff positions have been hardened as a result of extended debate and evaluation.

[. Alternate 2] - ACRS comment on proposed rule after public comments have been received and incorporated but before the hearing process begins.

POOR ORIGINAL

Advantages

1. Makes public comments and staff reaction available to the ACRS in preparing its comments.
2. Provides for a single step ACRS review at a time when NRC staff thinking is well advanced but is still flexible with respect to proposed changes.
3. Provides ACRS input at a t[illegible] recommendations can be evaluated/discussed [illegible] without undue delay in the process and the possible [illegible] reopening the hearing process. Major changes resulting from ACRS comments at this stage could [illegible] for public comment, however.

Disadvantages

1. Provides for ACRS input after a considerable amount of NRC manpower and resources have been expended.

2. Could result in a delay in promulgation of a final rule if the public comment phase must be redone because of major changes resulting from ACRS comments.

. Alternate 3 - Provide ACRS comments during the same period when public comments are being accepted.

Disadvantages

1. Limits the time available for ACRS comments (30/60/90 days) and does not take into account priorities associated with other ACRS assignments.

2. ACRS does not have the benefit of public comments and staff reaction in developing its recommendations.
3. Does not appear to be an appropriate way to make use of an agency advisory Committee.
4. Occurs at a time when considerable staff manpower and resources have been expended but the staff is still flexible with respect to changes.

Advantages

1. Would not result in any delay of the rulemaking process.
2. ACRS input would occur at a time when the staff position is responsive to suggestions.

. Alternate 4 - ACRS comments would be provided before the rule is published for public comment.

Advantages

1. ACRS input would occur at a time when minimum staff resources have been expended and the staff is most responsive to suggestions and guidance.
2. Any delays resulting from evaluation/resolution of ACRS comments could best be accommodated with minimum delays to eventual promulgation of the rule.

Disadvantages

1. ACRS would not actually comment on the proposed rule as it eventually evolves after public comments and/or the hearing process is complete.

Conclusion

All of the above have substantive advantages and disadvantages, however, Alternates 2, 3, and 4 appear to offer the opportunity for ACRS participation without the possibility of Alternate 1 that significant delay could result in the promulgation of a proposal if the ACRS were to make substantive comments so late in the process.

It appears that Alternates 2, [illegible] used to advantage depending on the substance, degree of public interest, degree of prior Committee particip[illegible] etc. and any one or combination of more than one (e.g., Alternates 2 and 4 for example) should be selected [illegible] Staff with the concurrence of the Committee on a case-by-case basis. Input from the ACRS in a two-part proceeding could be at the Subcommittee level during the first phase (Alternate 2) and the full Committee during the second (final/Alt. 4) phase.

Recommendation

That an appropriate revision of NRC rules (10 CFR Part 2 Rules of Practice, 10 CFR Part 7 - Advisory Committees, and 10 CFR Part 50 Licensing of Production and Utilization Facilities) be promulgated reflecting the conclusions noted above.

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III. ACRS Participation in NRC Rule[illegible] - Participation in NRC [illegible] Hearings

The President's Commission on TMI had recommended that the ACRS should be authorized to raise any safety issue in rulemaking proceedings, to give reasons and arguments for its views, and to require formal response to any submission it makes. In addition, any member of ACRS should be authorized to appear and testify in hearings, but should be exempt from subpoena in any proceeding in which he has not previously appeared voluntarily or made an individual written submission.

In its comments regarding this recommendation (M. Plesset ltr. to Chairman J. Ahearne, dated January 15, 1980, "Recommendations of the President's Commission on ACRS Role") the ACRS has noted that with respect to its participation in licensing proceeding, that:

> While the ACRS agrees that additional emphasis should be given to ACRS recommendations during the hearing process, it believes that a more desirable method of achieving this purpose would be to alter the statute to require that all recommendations made by the ACRS on given licensing proceedings be treated as substantive issues during the hearing. In order to protect the advisory role and collegiality of the ACRS, the statute should also specify that neither the Committee nor its members should be involved as a party nor be subject to subpoena in connection with the hearings.

With respect to the proposal that, "Any member of the Committee should be authorized to appear and testify in hearings, ..." the Committee has indicated that:

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> The Committee believes that one [illegible] strengths results from its collegial approach [illegible] this would [illegible] jeopardized if members departed from the collegial forum. Although members can express disagreement with full Committee views by adding separate [illegible] we believe the collective aspect is [illegible] and [illegible] cannot support the recommendation. A member should be free, of course, to participate as an intervenor in his capacity as a private citizen.

It is the position of OGC that a similar rationale would apply to direct participation by the ACRS or its individual members in rulemaking proceedings. Instead of participation as parties to a proceeding, the technical capability of the Committee should be utilized to assist the [illegible] the Commission as the case may be, in specifying [illegible] considered in the hearing and judging of the final [illegible] This kind of participation is reflected in [illegible] procedures recently established for ACRS participation in [illegible] proposed rulemaking on interim storage and ultimate disposal of radioactive waste. The procedures outlined in the attached letter from J. Ahearne to M. Plesset dated January 9, 1980 and Alternate 2 of Part 1 will provide for such ACRS contribution.

With respect to a formal response to ACRS recommendations regardint rulemaking, the response proposed under Item II would fulfill this requirement.

Recommendation

A revision of NRC rules (10 CFR, Part 2, [illegible] 7, and 50) should be promulgated based on the procedu[illegible] above.)

Conclusion

Proposed changes in NRC rulemaking procedures as noted should be implemented.

ADVISORY COMMITTEE ... WASHINGTON, D.C.

DRAFT 2
GASKE car ...

ACRS MEMBERS

DRAFT ACRS PROCEDURES FOR DIFFERING PROFESSIONAL OPINIONS

The following are draft ACRS procedures for handling dissenting professional opinions among NRC Staff Members and among ACRS Consultants.

BACKGROUND

NUREG-0567, Proposed Policy and Procedures for Differing Professional Opinions, dated October, 1979, was published ... with a request for comments. The Committee recommended that ... be revised to read as noted below. The NRC is ... which have been received, and it is assumed for purposes of this memo that paragraph 6.b will be changed to the form ... by the Committee, viz.

"If the differing professional opinion ... potential safety issue within the purview of the Advisory Committee on Reactor Safeguards, an NRC employee may communicate orally or in writing directly with the Chairman or any Member of the ACRS. Such communication may be anonymous. The ACRS will append comments, as appropriate, to all written statements of differing professional opinion and will forward these statements for resolution to the appropriate NRC office director.

An NRC employee may also appear before the ACRS or an ACRS Subcommittee as deemed appropriate by the Committee. The ACRS will assure that all such statements that do not constitute a differing professional opinion are forwarded to the appropriate NRC office director for information."

Proposed Procedures for ACRS Handling Differing Professional Opinions Among NRC Staff Members and ACRS Consultants

A. Handling Dissenting Professional Opinions Among Members of NRC Staff

Differing professional opinions among NRC Staff Members (concerning matters within the purview of the Committee) which are brought to the attention of ACRS Members by NRC Staff Members will be handled as follows:

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ATTACHMENT 2

1. Written statements of differing professional [illegible] provided to the Committee or its members for consideration in accordance with NUREG-0567 should be provided to the ACRS Executive Director who will forward copies to all members with an appropriate cover memorandum.

2. The Executive Director will consult with the [illegible] Chairman and Subcommittee Chairmen as appropriate to determine whether the matter should be assigned to a subcommittee for initial review, brought directly to the full Committee, or whether it is an item of a type that does not warrant Committee consideration.

3. The NRC employee who provided the differing opinion will be advised of the initial disposition at this stage by the [illegible] Executive Director.

4. The NRC employee involved may appear before the Committee or an ACRS Subcommittee as deemed appropriate by the [illegible].

5. If a Member receives an oral communication, [illegible] represents a differing professional opinion by a Member of the NRC Staff, the caller should be asked to provide it in written form ([illegible] if the caller desires to remain anonymous) and provided to the ACRS Executive Director for distribution to all members. It will then receive the same handling as a written communication of this type received from an NRC Staff Member.

6. The ACRS Executive Director will, in accordance with the second paragraph of 6.b above, assure that statements received which do not constitute a differing professional opinion are forwarded to the appropriate NRC Office director by the ACRS Executive Director for information in accordance with NUREG-0567.

The NRC Staff Member involved will be kept informed of the action taken in response to his communication.

B. Differing Professional Opinions Among ACRS Consultants

In regard to ACRS Consultants, it will be the responsibility of the ACRS Subcommittee Chairman, to whom consultants are providing advice, to try to resolve any differing professional opinions by consultants which arise at Subcommittee meetings or are expressed in ACRS consultant reports prepared in support of the Subcommittee activities. In the event this resolution is not practicable at the Subcommittee level, the differing opinions of consultants should be brought to the attention of the full Committee so an appropriate Committee position can be established.

If consultant reports, containing differing professional opinions, are received regarding matters where there is not an obvious Subcommittee Chairman, the Engineer receiving the action copy of the report should bring the matter to the attention of the ACRS Executive Director. The

ACRS Executive Director will then consult with the Committee Chairman and Subcommittee Chairman as appropriate and, when considered appropriate, the Chairman will designate a Subcommittee to resolve the matter.

Since copies of all ACRS consultants' reports are provided to all members by the designated ACRS Project Engineer, no change in distribution procedures/practices are needed.

R. F. Fraley
Executive Director

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

DRAFT [illegible]
M.C.Gaske [illegible]
2/7/80

ACRS MEMBERS
ACRS STAFF
ACRS FELLOWS

Attached is a draft outline of the manner in which the ACRS Fellowship Program will be carried out. The outline has been written to apply to Fellows located in Washington, D. C. and should be complied with, to the degree practical, by ACRS Fellows at other locations.

In order to aid in conducting the Fellowship program, the following designations are made for the Senior Fellows to assist in providing day-to-day technical guidance to the Fellows in carrying out their assignments.

Supervising Sr. Fellow	Abbott	Kastenberg*
Fellows	Bessett Bicket Young	Johnson** Zukor

* maintains liaison with Chen (effective 2/1/80 at Cal Tech.)

** maintains liaison with Griesmeyer (at UCLA)

Supervising Staff Engr.	Libarkin/McCreless
Fellows	Stampelos

R. F. Fraley
ExecutiveDirector

ATTACHMENT 3

POOR ORIGINAL

PROCEDURES FOR ACRS FELLOWS

Background

The ACRS Fellowship Program was established by Congress: "To assist the Advisory Committee on Reactor Safeguards in carrying out its functions, the Committee shall establish a fellowship program under which persons having appropriate engineering or scientific expertise are assigned particular tasks relating to the functions of the Committee." In order to assist the Committee most effectively, the Fellowship Program has been structured so that, when practicable, individual Fellows are assigned to work with Sponsoring ACRS Members whose background and interests are similar to those of the individual Fellow. In those cases where this is not practicable, the Fellows will be assigned to work with designated members of the ACRS Staff.

Member - Fellow Interaction

Fellows have been assigned to work with one Sponsoring ACRS Member or, in a few cases, with more than one [illegible] desirable to have the Members provide technical direction and guidance so that assigned projects are successfully completed in [illegible] consistent with the needs/desires of the Member(s) and the Committee [illegible] to provide the time needed to discuss the scope and [illegible] being performed by the Fellows and to provide needed guidance, Members should consider setting aside time in Washington to provide for such discussion. In addition, Fellows should take the opportunity to visit the Members at their normal duty station, when appropriate, so that necessary discussion can take place. Supervising Members should be informed of difficulties on a timely basis. Much of a successful Fellowship Program will depend on a successful Member-Fellow interaction.

To supplement the guidance by Members, the Senior Fellows or more experienced ACRS Fellows will be assigned, as appropriate, to provide technical guidance on a day-to-day basis during periods when Members are not available.

In regard to the procedural aspects of assignments which are out of the ordinary, for example direct contact with a utility, vendor, etc., to request information, documents, etc., setting up a visit to a reactor site to obtain information, etc., the Assistant Executive Director should be contacted concerning the best method for such action.

In selected cases, Fellows may also be assigned to work directly with members of the full-time permanent ACRS staff. In these cases, Fellows will receive their technical as well as other guidance from the designated ACRS Staff member. In the event a Working Group consisting of ACRS Fellows and permanent ACRS staff members is needed to accomplish a task, supervising ACRS member(s), permanent staff and/or Fellow(s) will be designated on a case-by-case basis.

Work Assignments

Fellows will be assigned to work with Sponsoring ACRS Members or desig-
nated ACRS staff members by the ACRS Assistant Executive Director and will
normally receive their work assignments from the Sponsoring ACRS Member
they are designated to assist.

Fellows will normally receive assignments [illegible] Sponsoring ACRS Member.
These assignments will consist of [illegible] the Committee in
carrying out its statutory functions. [illegible] process, the ACRS
Assistant Executive Director will maintain [illegible] proposed projects,
which takes into account suggestions by all ACRS Members and the ACRS Fellows
themselves. In order to add a work assignment [illegible] list, the proposing
Member/Fellow should provide a brief summary of the work scope and objectives
and an indication of its priority. A major effort (anticipated to involve
more than 12 man-months effort) will be [illegible] the Procedures Sub-
committee by the Assistant Executive [illegible] practicable to
determine if it is a practical assignment, consistent with the intent and
resources of the Fellowship Program.

If Fellows are to be asked to perform short term assignments for other
than their Sponsoring Member (e.g., in a direct support of an ACRS meet-
ing, or in drafting an ACRS report) they will in effect be reassigned on
a short term basis by the Assistant Executive Director. The ACRS Assist-
ant Executive Director will coordinate the assignment with the Sponsor-
ing Member to the degree considered appropriate. The ACRS Office can
give Fellows short term assignments, such as attendance at NRC or other
meetings, a training session, etc., provided it does not appear that such
assignments will interfere unduly with urgent projects for ACRS Members.

For each assignment which is expected to last more than one month, the
designated Fellow should develop a general plan of attack (e.g., litera-
ture survey, visit to field installations, digest and evaluation of infor-
mation, etc.) with proposed target dates. This outline will be based on
discussions as needed between the Fellow and the person requesting the
work.

A list will be published by the ACRS Office (Assistant Executive Director)
monthly of the major projects on which Fellows are working. To assist in
the preparation of this report, ACRS Fellows should keep the ACRS Assist-
ant Executive Director informed of progress regarding their assignments.
(Attachment A should be completed by the Fellows and turned in to Thurston
Faulder weekly.)

Document Management

.11 ACRS Fellows should ensure that [illegible] by them is marked "OUO, Prepared for Internal ACRS Use [illegible] if it con- tains their advice, opinions, or recommendations. If the material con- tains proprietary or classified information, see H. Alderman for the proper identification of the document. All Fellows are encouraged to communicate with the ACRS Members for whom they are working including providing the Members with preliminary draft material which the Fellow has written. Further distribution of the document for purposes of re- view (e.g., obtaining the comments of ACRS consultants) or publication will be at the discretion of the Supervising Member. If a document is to be made available to the general public and/or is to be published as a NUREG report, it should be given a [illegible] as noted below.

Contacts

As indicated above, Fellows should maintain contact with Member(s) they are assigned to assist to ensure that they are carrying out their work assignments in the desired manner. Fellows are free to con- tact ACRS staff members and working level [illegible] staff members to discuss their assigned projects. If in doubt [illegible] you may/should con- tact, see the Assistant Executive Director for guidance and assistance.

Publication of Material

Those reports deemed appropriate for publication (e.g., as a NUREG docu- ment) or public distribution* will be determined by the Member for whom the work was performed. This will ordinarily be material which conforms to one of the following:

1. Contains significant new analysis and/or data.

2. Represents a major consolidation of information.

3. Is a major report containing significant recommendations.

Arrangements will be made by the Assistant Executive Director to have a review of the above type material performed by two ACRS Members and/or consultants.

The Member requesting work by an ACRS Fellow will be expected to evaluate recommendations by the Fellow in connection with the assignment and intro- duce them as appropriate into Committee deliberations.

----

* Material written by ACRS Fellows and requested under the FOIA will be retained or released in accordance with the usual procedures for handling documents of this type.

ATTACH

ACRS FI...

Name

Activity*/ | % of effort

1.
2.
3.
4.
5.
6.
7.
8.
9.
10.
11.
12.

----

*/ Indicate items on which work was performed and % of time expended on each. Also, indicate attendance at meetings, training, leave, holiday, etc., so that the total equals 100%.

POOR ORIGINAL

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

January 12, 1980

8005150206

MEMORANDUM TO: M. Plesset, Chairman, ACRS

FROM: W. M. Mathis

SUBJECT: ACRS PROCEDURES

After observing the conduct of our business during the past week, I am convinced that improvement can be made in our methods and procedures.

For example, our meeting topics have consumed far more time than scheduled, have accomplished very little if anything, and have shown our subcommittee meetings on these same topics to have been essentially a waste of time.

First, we do not follow through or subcommittee activities because each member of the Committee feels he must dig through all the details to satisfy his personal interest for which he cannot accept the opinion of others. This method of project or activity review makes the subcommittee concept of review and recommendation impractical so other approaches should be considered.

Second, each meeting should be preceded with a very definite stated "Purpose."

The purpose may be any of the following, for example:

- information
- action; i. e., letter
- guidance for further work
- report for whomever or whatever

Third, with a purpose clearly defined, the presentations can be engineered to fulfil the purpose by instructing the participants in what is desired for such subjects as:

- scope
- depth
- recommendations
- status
- schedule
- priority
- research needs
- etc.

POOR ORIGINAL

ATTACHMENT 4

Fourth, Committee members must do their homework and decide what they want to hear - what participation is necessary on their part - how much detail on what subjects they wish to investigate - and when making requests or recommendations the Committee must assume responsibility for the actions required and be accountable for the merit of the results to be accomplished.

Fifth, we have been critical of the planning of others but have no plans for our own activities. Possibly we would do a better job if we attempted to better program our objectives and goals for some future time scale, i. e. define the more important safety issues which we believe should receive attention by NRC and on a suggested schedule for accomplishment.

The above thoughts are rough and can stand much refinement and amplification, but I hope they are provocative enough to initiate more thought on the part of the entire Committee. In short, we can do a better job and need to get on with it. Possibly, we could chew on this in February.

cc: ACRS Members
R. F. Fraley

NUCLE[illegible]
ADVISORY CO[illegible]
[illegible] D.C. 2[illegible]

February 4, 19[illegible]

Ray:

Your remark concerning the issue of [illegible] supplementary SER following the ACR[illegible] other things[illegible] convert the generally [illegible] ACRS advice into partial or c[illegible] suggests that, as a ritual proc[illegible] review that SER in this cont[illegible] or disapprove the SER.

It is not my impression [illegible] but why not? I realize this puts a [illegible] on ACRS (which it deliberately [illegible]) but somehow when we are really convinced that an issue should be satisfactorily resolved, [illegible] avoid a decision-making role except in a final administrative sense.

J. Ebersole
J. Ebersole

POOR ORIGINAL

ATTACHMENT 5

DRAFT 1:[illegible]

Proposal for Reorganization, Reassignment and Establishment of New Functions (ACRS Project Staff)

Background:

Current Staff

Management	1	
Supervisory	2	
Professional engineering	9	
Secretarial	5	
Engineering Aide	1	18

Proposed Addition:

	10	
		28

Proposed New Organization:

(assuming the addition of 10 full-time, permanent positions)

Management	1
Supervisory	3
Professional Engineering	16
Secretarial	7
	27
Asst. Exec. Dir. - Analysis	1*

* Note: One senior member of the present Project Staff will be reassigned to provide technical supervision of the ACRS Fellowships Program.

POOR ORIGINAL

ATTACHMENT 6

A. Proposed Functional Organization

Operating Reactors and Licensing actions (including the periodic preparation of Category B summary reports.

GE Reactors	2
Westinghouse Reactors	2
B&W Reactors	1
CE Reactors	1
	6

This group would also be responsible for Committee activities in connection with evaluation of LERs, Requests and Recommendations and Reactor O...

The remaining 10 professional Staff would be assigned as follows:

(assignment would include preparation of periodic status reports covering areas assigned, handling all Subcommittee Meetings in areas assigned, etc. In the event of a conflict, Branch Chief, etc. could cover meeting. ... existing standing and ad hoc Subcommittee functional assignments have been divided in some cases)

1. Concrete and Concrete Structures
   Metal Components
   Spent Fuel Storage and Design (structural)
   Safeguards and Security (structural)

2. ECCS
   ATWS
   Fluid Dynamics
   Combination of Dynamic Loads

3. Reactor Fuel
   Core Performance
   Spent Fuel Storage and Design (physics)

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*NOTE: Based on the first, imperfect, report from the MFS system, about 1/3 of the current 9 engineers' activity is in the areas to be assigned in expanded form to the new personnel, and would not have to be duplicated. Six seems like the proper number, therefore.

POOR ORIGINAL

4. Power and Electrical System
Plant Arrangements
Fire Protection
Safeguards and Security ([illegible] separation, other [illegible]
features)

5. Radiobiological Effects and Site Evaluation
Extreme External Phenomena
Transportation of Radioactive Materials

6. Generic Items (Include Task Action Plans for dealing with Generic Items)
Regulatory Activities

7. Three Mile Island 2 Accident Implications (include action plan)
Three Mile Island 2 Accident [illegible] Orders

8. Reactor Safety Research (include LOFT reactor)
Improved Safety Systems
Advanced Reactors

9. Waste Management
Enrichment Plants

10. Reliability and Probabilistic Assessment
Consideration of Class 9 Accidents
Single - Failure Criterion

Appendix XXIII

NUREG-0657

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1981

A Report to the Congress
of the United States of America

Manuscript Completed: February 1980
Date Published: February 1980

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

February 15, 1980

The Honorable Walter F. Mondale
The President of the Senate

The Honorable Thomas P. O'Neill, Jr.
The Speaker of the House

Gentlemen:

I am pleased to transmit herewith the Advisory Committee on Reactor Safeguards' report to the Congress on the Nuclear Regulatory Commission's safety research program for fiscal year 1981. This report is required by Section 29 of the Atomic Energy Act of 1954 as amended by Section 5 of Public Law 95-209.

Chapter 1 is intended to serve as the Executive Summary.

A copy of this report is being sent to the Chairman of the Nuclear Regulatory Commission.

Respectfully submitted,

Milton S. Plesset

Milton S. Plesset
Chairman

PREFACE

This report has been prepared in response to the requirement by the Congress that the Advisory Committee on Reactor Safeguards (ACRS) "undertake a study of reactor safety research and prepar and submit annually a report containing the results of such study."

Previous reports have been submitted in December 1977 and December 1978.* As requested by the Congress, submittal of this report has been delayed until the Administration's budget for FY 1981 has been submitted to the Congress and reviewed by the ACRS.

As in previous reports, the ACRS has interpreted the words "reactor safety research" to include safety-related research in all phases of the nuclear fuel cycle and power plant operation, and excluding only that having to do with non-safety-related environmental concerns.

Chapter 1 includes an introduction and a summary of the principal recommendations of the ACRS regarding the Nuclear Regulatory Commission (NRC) safety research program and the proposed levels of funding. It is intended to serve as an Executive Summary.

Chapter 2 discusses briefly the implications of the accident at Three Mile Island, Unit 2 (TMI-2), as they relate to the research program, and lists several areas comprising new directions in research. A more detailed discussion of these implications and new directions is given in Chapter 1 of the ACRS report to the NRC on its budget request for FY 1981 and its supplemental request for FY 1980.**

The remaining chapters of this report present specific comments on the individual decision units of the research program, and include some assessments of priorities, where this was possible, and recommendations regarding new directions and levels of funding.

All references to funding in this report relate to funds budgeted for program support. Funds allocated for NRC personnel, administrative support, and equipment have not been included.

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*Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, NUREG-0392, December 1977.
1978 Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, NUREG-049[illegible], December 1978.
** Comments on the Nuclear Regulatory Commission Safety Research Program Budget, NUREG-0603, July 1979.

TABLE OF CONTENTS

1. INTRODUCTION AND RECOMMENDATIONS

1.1 Introduction

1.1.1 Implications of the Accident at TMI-2

In its report to the NRC Commissioners in July 1979 (NUREG-0603), the ACRS stated its belief that the accident at TMI-2 had major implications for the safety research program and that both more and different research would be required in FY 1980 and in the future to answer the questions that had been raised by the accident and to improve reactor safety.

In that report, the ACRS recommended and discussed in some detail several new directions in research resulting from the accident at TMI-2 and requiring early implementation in FY 1980 and FY 1981.* These recommendations are listed in Chapter 2 of this report. Many are being implemented in FY 1980 and more are planned for FY 1981. Nevertheless, the ACRS believes that research in three areas warrants greater emphasis than is now planned by the NRC Staff; these are:

Studies of the courses of serious accidents

Studies of molten core retention and steam explosions

Studies of plant operations and of systems behavior, particularly in various shutdown-heat-removal modes

The ACRS believes that additional funding, as recommended below and elsewhere in this report, is required in order for these areas to be studied with the depth and timeliness they deserve.

1.1.2 Priorities

In its December 1978 report to the Congress (NUREG-0496), the ACRS stated that it might be possible to develop a hierarchy of priorities within the various program areas and perhaps even across the entire safety research program. The ACRS attempted to do this for this report with but limited success. Nevertheless, priorities have been suggested in several areas, as indicated below.

(a) Generally high priorities have been assigned to most, but not all, of the research inspired by the accident at TMI-2, and especially to research in the new directions indicated above and in Chapter 2.

*The pertinent material from NUREG-0603 is included herein as Appendix B.

(b) Within each program area (Chapters 3-15), high or low priorities have been indicated where possible.

(c) Priorities among program areas have been indicated by the changes in levels of funding recommended by the ACRS in Section 1.2 and Table 1. A reduced level of effort has been accepted for the LOFT program, a moderate increase has been recommended for the program in Probabilistic Risk Assessment, and a relatively large increase has been proposed for research on Improved Reactor Safety. These programs all relate closely to the accident at TMI-2. In addition, the ACRS has recommended a significant increase above the Administration's proposal for research on Advanced Reactor Safety.

1.1.3 Relation to FY 1980 Budget

The recommendations made herein, especially those relating to funding levels, are based on the assumption that the NRC budget for safety research programs will be supplemented for FY 1980 by all or a substantial portion of the $27 million requested for additional research related to the accident at TMI-2.

1.2 Recommendations

The principal recommendations of the ACRS are summarized briefly in this section, with references to the appropriate chapters in which more detailed recommendations, discussions, and justifications are provided.

1.2.1 General Recommendations

The proposed budget for research program support is shown in Table 1. The numbers at the left of the table refer to the chapters in this report in which each budget decision unit is discussed.

The ACRS recommendations regarding funding levels for FY 1981 are shown in the table. The ACRS believes that a safety research program addressing the questions raised by the accident at TMI-2, together with the continuation of essential ongoing programs, cannot be carried out within the limits of the proposed budget and that an increase of at least $20.9 million is needed.

The increase recommended by the ACRS is proposed for the following three areas:

Advanced Reactors, including Fast Reactors and Advanced Converters	$13.0 million
Risk Assessment	2.4
Improved Reactor Safety	5.5
	$20.9 million

Reductions in other program areas to offset these increases were considered by the ACRS but were found to be undesirable in view of the increased efforts in most of those areas required to study the implications of the accident at TMI-2. Nevertheless, the ACRS believes that the recommended increases in the programs on Risk Assessment and Improved Reactor Safety should be provided even if this requires a reduction in other portions of the overall safety research program.

Specific comments supporting the recommended increases, and other major recommendations of this report, are given in the following sections:

1.2.2 Advanced Reactor Safety Research (Chapter 9)

The proposed budget provides only $5 million for this program, an amount not sufficient to bring the current program to an orderly conclusion.

In its previous reports to the Congress (NUREG-0392 and NUREG-0496), and in its report to the Commission in July 1979 (NUREG-0603), the ACRS has consistently supported an NRC research program related to the safety of advanced reactors. This recommendation has been based on the perception that many of the current safety problems associated with light-water reactors (LWRs) have resulted from the fact that safety research lagged behind reactor development. The Administration has presented a proposal to defer indefinitely the development of liquid metal fast breeder reactors (LMFBR), and to provide for no work at all related to other concepts. If the Congress agrees with this indefinite deferral, the ACRS would agree that there is no need to maintain an advanced reactor safety research program. On the other hand, if the deferral is to be short term, or if the development program in FY 1981 is to continue at a pace similar to that of the last few years, then the arguments presented earlier for an NRC program are still valid. Finally, if the expectation is that a move to exercise the LMFBR option in the next 10-20 years will be accommodated by importing foreign technology, it is important that the NRC program of safety research on advanced reactors be maintained to ensure an adequate technical basis for U.S. regulatory standards, guides, and criteria.

If it is decided that the development of advanced reactors is to be continued, the ACRS considers it essential that a concomitant safety research program be carried out by the NRC at a funding level of at least $18 million for FY 1981. Specific comments regarding the content of such a program are presented in Chapter 9.

1.2.3 Risk Assessment (Chapter 14)

This program already includes many projects relating to the questions raised by the accident at TMI-2; it is being expanded in FY 1980, and the ACRS considers further expansion in FY 1981 to be highly desirable. The importance of probabilistic risk assessment as a guide to, or as a basis for, licensing requirements and decisions is recognized by the ACRS, and its use by the NRC Staff is dependent to a significant degree on the efforts of the Probabilistic Analysis Staff.

The ACRS considers this program to be of high priority and recommends that it be funded at a level of at least $15 million in FY 1981.

1.2.4 Improved Reactor Safety (Chapter 15)

This program was initiated at the request of the Congress but has been funded at a grossly inadequate level for the past three years. The current programs and those proposed for FY 1981 relate directly to improvements in safety that have become even more prominent as a result of the accident at TMI-2.

The ACRS recommends that funding of this program for FY 1981 be at a level sufficient to permit aggressive programs on alternate decay heat removal concepts, vented and filtered containments, and improved in-plant accident response. In addition, scoping studies of other likely projects should be undertaken as a basis for planning future programs of similar nature and import.

Funding at a level of at least $10 million for FY 1981 is recommended.

1.2.5 LOFT Facility (Chapter 4)

The proposed level of funding of $43 million for LOFT is $6.3 million less than the ACRS commented on favorably in its report to the NRC Commissioners in July 1979 (NUREG-0603).

Since the LOFT program for FY 1980 and FY 1981 has been reoriented almost completely toward research relating to small loss-of-coolant accidents and transients in response to the accident at TMI-2, the ACRS believes that the higher level of $49.3 million could be used to good technical effect in FY 1981; and also, in view of the large unavoidable expenses required for the upkeep and operation of this facility, that the higher level would be more cost-effective.

Nevertheless, the ACRS can accept the reduced level of funding in this area in view of the needs for the increases recommended above in Sections 1.2.3 and 1.2.4, and for increases already incorporated in other areas to address the matters mentioned in Chapter 2.

1.2.6 New Directions in Research (Chapter 2)

The accident at TMI-2 has indicated clearly the need for new directions in the NRC safety research program, and the programs for FY 1980 and FY 1981 have been reoriented to various degrees in those new directions.

The ACRS has indicated in most of the following chapters of this report how and where these new directions have been or should be incorporated into the proposed research program. In most areas, the ACRS believes that these new directions can be accommodated by reallocation of resources without additional funding beyond that proposed. The exceptions are Risk Assessment and Improved Reactor Safety, for which additional funds have been recommended above.

1.2.7 Increased Personnel for Waste Management (Chapter 12)

All of the foregoing recommendations have related to funding for program support and have not addressed other portions of the research budget or manpower requirements. However, the research program in Waste Management is growing at such a rate (supported by the ACRS) that there is now reason to be concerned about the ability of the current staff to manage the program effectively. For this reason, the ACRS recommends that the NRC be authorized and funded to expand the technical staff for this program by at least five people above the fifteen now proposed.

1.2.8 Confirmatory Research vs. Research to Improve Reactor Safety (Chapter 15)

The Energy Reorganization Act of 1974, which effected the separation of the Atomic Energy Commission (AEC) into the separate entities of the NRC and Energy Research and Development Administration (ERDA), established within the NRC an Office of Nuclear Regulatory Research (RES).

In view of the legislative history (Conference Report), the role of RES was accepted by the NRC as being limited to "confirmatory research," presumably as opposed to the "developmental research" that had dominated the AEC program.

Several attempts were made by RES to define "confirmatory research." The need for definition, however, was in effect obviated by the issuance of a directive that the research must be responsive to and, except in special circumstances, initiated by "user" needs. Users were considered to be principally the other NRC program offices but to include also the Commission, Atomic Safety and Licensing Boards, Atomic Safety and Licensing Appeal Board, ACRS, Congress, the technical community, and the public.

The limited nature of "confirmatory research" was recognized implicitly by the Congress when, in its Budget Authorization Act for NRC for FY 1978 (PL 95-209), it amended the Energy Reorganization Act of 1974 to require the NRC to develop a long-range plan for the "development of new or improved safety systems" for nuclear power plants. Such a plan was developed and submitted to the Congress in April 1978 (NUREG-0438).

In its reviews of the NRC safety research program during the past three years, the ACRS has found it increasingly difficult to distinguish between "confirmatory research" and "research to improve reactor safety." As a result of the accident at TMI-2, the distinction has become even less obvious and, clearly, less important. Several of the proposals by NRC and others resulting from the accident at TMI-2, which will now become matters for licensing decisions, were previously relegated to the category of "research to improve reactor safety" and were inadequately funded as a separate program area (See Chapter 15).

The ACRS believes that the distinction that has been made between "research to improve reactor safety" and "confirmatory research" is no longer useful. The ACRS suggests, therefore, that the Congress review the legislative charter of the NRC research program and eliminate this distinction. Inasmuch as the purpose of the NRC is to ensure safe design, construction, and operation of nuclear power plants, it should follow that any research is legitimate that contributes to that purpose without compromising the credibility or independence of the NRC. The ACRS urges consideration of this matter by the Congress.

TABLE 1

PROPOSED FY 1981 BUDGET
(In millions)

CHAPTER	DECISION UNIT	PROPOSED	ACRS RECOMMENDATION
3.	Systems Engineering	38.0	38.0
4.	LOFT	43.0	43.0
5.	Code Development	14.2	14.2
6.	Fuel Behavior	27.9	27.9
7.	Primary System Integrity	14.3	14.3
8.	Seismic, Engineering, and Site Safety	16.9	16.9
9.	Fast Breeder Reactors	5.0	18.0
9	Advanced Converters	0.0	
10.	Reactor Environmental Effects	12.2	12.2
11.	Fuel Cycle		
12.	Waste Management	13.6	13.6
13.	Safeguards	4.9	4.9
14.	Risk Assessment	12.6	15.0
15.	Improved Reactor Safety	4.5	10.0
	Total Program Support	$207.1	$228.0

2. IMPLICATIONS OF THE THREE MILE ISLAND ACCIDENT AND NEW DIRECTIONS IN RESEARCH

In Chapter 1 of NUREG-0603, the ACRS discussed the need for new directions in research and identified about a dozen areas which required either a much greater emphasis, a major change in orientation, or early initiation. (See Appendix B). The ACRS also identified several areas warranting a new research emphasis in its "Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2," May 16, 1979 and its report entitled "Studies to Improve Reactor Safety," August 14, 1979. The areas recommended for new directions in research include the following:

Anomalous transients and small loss-of-coolant accidents
Studies of the courses of serious accidents
Molten core retention
Steam explosions
Siting
Plant operations
Transient simulation
Systems behavior
Inadequacies in the single-failure criterion
Water chemistry and crack growth
Disturbance analysis

The ACRS believes that these and other new directions in research are of major importance. The ACRS supported the general levels of funding proposed by RES in July 1979 with the expectation that a large-scale reorientation of the previously planned program would ensue.

Although some reorientation has resulted, the ACRS does not believe that the pace or the extent of redirection of the research program has been adequate in all cases. In particular, of the matters identified above, the following warrant considerably greater emphasis than is planned by the NRC Staff and/or is called for in the proposed budget:

Studies of the courses of serious accidents

Studies of molten core retention and steam explosions

Studies of plant operations and of systems behavior, particularly in various shutdown-heat-removal modes

The ACRS believes that additional effort and funding should be devoted to these areas in FY 1981 and that, together with the programs on probabilistic risk assessment and on research to improve reactor safety, these efforts in new directions should receive first priority in the NRC research program.

3. SYSTEMS ENGINEERING

3.1 Scope

Research in this area includes experimental studies of transients initiated by small breaks in the primary coolant system and studies of transients originating in the main secondary steam system. Research includes the study of thermal-hydraulic effects, steam-water interactions, reactor core flow blockage, and multi-dimensional flow phenomena. These studies aid in the development of computer code models and in the review of operating reactor requirements and experience.

3.2 General

The proposed budget includes items of large financial commitment of a long-range nature. Nevertheless, the research program in the costly facilities involved has been very effectively adapted to changing views regarding the important problems in LOCA-ECCS. In the past, the greatest emphasis in loss-of-coolant accidents was placed on the large break in the primary reactor loop. It is now perceived that small breaks are not only more probable but their consequences require additional experimental study and analysis. The small break is prototypical of transients, most of which arise in the secondary loop, which could lead to core uncovery. Core uncovery can occur also without the occurrence of an actual "break." Current proposals to use "feed and bleed" as a means of emergency cooling involve a controlled loss of coolant through the power-operated relief valve and/or the safety valves. The planned program addresses these problems and may be expected to make important contributions to their resolution.

3.3 Comments

3.3.1 Semiscale

The NRC budget supplement request for FY 1980 included funds which would provide a significant and useful upgrade of this facility. The upgrade included improved heat insulation, improved pump performance, and an improved secondary loop configuration. It has been possible to simulate the Westinghouse loop configuration and the upgrade will make possible the simulation of the secondary loop in the Babcock and Wilcox type of pressurized water reactor (PWR). Improvements in the representation of the secondary side of a PWR and in the height relationships in primary and secondary loops which will be provided are decisive for physically acceptable studies of small breaks and natural circulation.

The FY 1981 budget covers the experimental studies on transients in Westinghouse type loops begun in FY 1980 and will initiate the same kind of experimental studies on the Babcock and Wilcox type loops.

The Semiscale program will include studies of transients induced by small breaks, and a survey of transients initiated on the secondary side. The transients initiated on the secondary side are the most common sources of challenges to the ECCS. The ACRS recognizes the value of these programs in Semiscale.

Semiscale is an integrated test facility, but not in the sense that Semiscale data can be translated directly to full-size PWRs. If so translated, the Semiscale data can be misleading, and for this reason the ACRS urges that Semiscale be separated from the licensing path. The important contributions of Semiscale are of two kinds; first, Semiscale tests contribute to the general understanding of the pertinent physical phenomena; and, second, these tests make an important contribution to reactor code development.

3.3.2 Blowdown and Reflood Heat Transfer (BDHT)

A significant facility in this program is the Two Loop Test Apparatus (TLTA) which is an integrated test facility that is presumed to do for boiling water reactors (BWRs) what Semiscale does for PWRs. Test results from TLTA have been translated directly to prediction of the behavior of full-scale BWRs. The ACRS believes that this translation to full-scale BWRs is unfortunate and a misuse of the results. The scaling behavior in TLTA has not been adequately analyzed and using it to predict the performance of full-scale systems can be quite misleading. The ACRS urges strongly that results from this facility not be injected into the licensing path. The ACRS objects, in particular, to the series of small break tests proposed to answer questions raised by the accident at TMI-2. The limits of applicability of TLTA test results to full scale plants should be considered carefully in advance of any such tests, and the test results them selves should be used as recommended above for Semiscale test results -- for their contributions to code development and to the understanding of the essential physical phenomena.

While TLTA has received some upgrade, the ACRS believes that an extensive, further upgrade is necessary and urges that this be pursued.

Another program in this category is the spray test facility at Lynn, Massachusetts, which is a 30-degree sector of the spray installation in a BWR. Steam-water interaction effects will be studied in this facility and the results will be of importance.

3.3.3 3-D Flow Distribution

This large and continuing item is being modified to relate more effectively to present perceptions of some of the most significant problems in reactor safety. The ACRS concurs in its continuation, since the results will be useful and a strong commitment has been in place for several years to participate in this international (FRG-Japanese-U.S.) study of reactor safety features.

3.3.4 Model Development Program

This program consists of small projects in various university laboratories. The ACRS encourages this kind of program as being useful and productive; at the same time the program provides an interaction with an important part of the engineering and scientific community.

3.3.5 Operational Safety

The past and current research program in Operational Safety was initiated on an ad hoc basis as a result of operating experience or particular regulatory requirements. The ACRS believes that the program to date has been useful. However, in NUREG-0603, the ACRS recommended that the NRC develop a systematic research program on the safety implications of procedures for operation, maintenance, testing and surveillance. In addition, the ACRS recommended that an NRC safety research program on systems behavior should be developed. The ACRS believes that priority should be given to the initiation of a broad research program on operational aspects of reactor safety.

3.3.6 Natural Circulation Capability of PWR Systems

Heat removal by natural circulation is a critically important safety consideration during some shutdown transients. During loss of all AC power transients in some PWRs it is the only means of transferring fission product decay heat from the reactor core to the heat removal system, short of coolant boiling in the core. Transition from natural circulation to boiling may be necessary during such transients. An experimental program is needed to establish a better understanding of this process. It might utilize a combination of facilities such as nuclear power stations operated at low power levels, LOFT, separate-effects facilities (U.S. and foreign), and some visualization-type, bench scale experiments. A list of variables to be investigated should be established and an experimental program should be planned for this purpose. This work should have high priority.

3.4 Recommendations

The research in this area should be funded at the level requested. At least some of the studies included in the program on Operational Safety will contribute (or can be adapted to contribute) to the objectives called out in Chapter 2; and, as detailed plans are developed for the work to be undertaken, these should be directed as far as possible to contribute further to those objectives.

4. LOFT

4.1 Scope

LOFT is an integral test facility designed to study the effects of large and small breaks on a scaled PWR. The research budget includes funds for the LOFT experimental program as well as for the continued upkeep and operation of the facility.

4.2 General

The redirection of the LOFT program to small break tests, begun in FY 1980, will extend through FY 1981. In addition, one large break test may be conducted.

4.3 Comments

Previous remarks have indicated that the Semiscale facility does not provide results suitable for direct translation to full scale; although LOFT is appreciably larger than Semiscale, it has other limitations which likewise inhibit direct translation to full scale. LOFT has a short core, and height relationships are not preserved in the rest of the system. Such features limit the use of this facility to the study of basic physical effects and to contributions to code development and verification. It should be pointed out also that some extreme transients cannot be studied in LOFT because of its nuclear core with its decay heat. Semiscale, with its electric power source, need not be so limited in studies of extreme transients.

LOFT should be a useful facility if used with good physical and engineering judgment. However, the ACRS believes that care must be exercised in translating the results from LOFT to commercial reactors.

4.4 Recommendations

The support level of $43.0 million for LOFT is $6.3 million less than the ACRS commented on favorably in NUREG-0603. The ACRS believes that the LOFT program could use the higher figure of $49.3 million effectively in FY 1981; and also, in view of the large unavoidable expenses required for the upkeep and operation of this facility, that the higher level would be more cost-effective. However, the ACRS can accept the proposed support level of $43.0 million for LOFT on the basis that the $6.3 million reduction is restored to the total reactor safety research program and is used to support greatly accelerated programs in research to improve reactor safety and to initiate, or substantially augment, the new directions in research recommended in NUREG-0603 and discussed in Chapter 2 of this report.

5. CODE DEVELOPMENT

5.1 Scope

The objective of this program is the development, for predictive purposes, of computer codes for the quantitative analyses of reactor transients and accidents.

5.2 General

While the objective of the program has not yet been achieved, the program has made some progress.

5.3 Comments

The principal computer code of choice, TRAC, suffers so far from incomplete knowledge of some of the necessary physical parameters. It should be pointed out that a fairly complete description of the possible physical situations in a reactor transient is required for the microscopic description used in TRAC. This microscopic description leads to long running times and thereby limits a rapid survey of the many possible transients. While an effort is under way to develop a fast running version of TRAC, the ACRS believes the RELAP-5 computer code, which is already somewhat faster than TRAC, also should be developed to provide a second fast running code.

The ACRS believes that the program is progressing reasonably well in view of the difficulty of the task.

The ACRS supports the code development program.

5.4 Recommendations

The ACRS recommends the continued development of RELAP-5 as another general code of potential value. The ACRS recommends also that a strong program be initiated for the development of methodology and techniques that would facilitate the implementation of more sophisticated reactor simulators, not necessarily limited to real-time analysis. This would enable a more detailed understanding of the course of events in complex transients that include multiple failures and operator intervention. The ACRS believes that the proposed budget is adequate to include these developments without additional funds.

6. FUEL BEHAVIOR

6.1 Scope

This research program provides experimental data for independent assessment of reactor fuel behavior during accidents. The approach used is to develop analytical models through basic experiments on fuel rods conducted out-of-reactor, to assess these models with in-reactor tests, and to better quantify fission product release and transport from fuel under accident conditions.

6.2 General

The TMI-2 accident provided a unique test of fuel behavior under novel and extreme conditions. It is essential that the important results of this "test" be understood by examining the fuel and core from TMI-2. Plans for this must be coordinated with NRC, Department of Energy (DOE), and industry. The priority is high.

6.3 Comments

6.3.1 Clad and Fuel; Fuel Codes

This work is of substantial aid in providing an NRC capability in fuel behavior analysis, and should continue at current levels. However, a greater breadth of input into the physical modeling would be desirable. Work on modeling of severe overheating, which occurred at TMI-2, is encouraged.

6.3.2 In-Pile Testing at Power Burst Facility (PBF)

PBF represents about 60 percent of the total fuel behavior research budget. The information on fuel behavior during reactivity insertion accidents (RIA) is still believed by the Office of Nuclear Reactor Regulation (NRR) to be inadequate. It is not clear that this experimental program has provided information of quantity and significance in proportion to its level of support. If these accidents are of sufficiently low risk (low probability and/or low energy insertions), such research is not necessary. The NRC Staff has not provided the ACRS with a convincing argument in favor of the need for the experiments on fuel behavior during RIA, or for most of the other experiments planned for PBF in FY 1980 and FY 1981.

The ACRS believes that PBF probably can be used for experiments related to flow starvation and fuel melting accidents and urges an early and complete evaluation of the currently proposed PBF program. In the meantime, the ACRS believes that flexibility in reprogramming some PBF funds to other high priority work on steam explosions and core melt should be provided.

6.3.3 Other In-Pile Testing

These are confirmatory programs related to core behavior following a large LOCA. The priority is probably low. The joint U.S.-Canadian research program at the NRU reactor in Canada should be terminated in FY 1983 as planned. Before committing to the multi-national research program at the ESSOR reactor complex in Ispra, Italy, the NRC Staff should be convinced that there are not higher priority NRC research needs.

6.3.4 Fuel Melt

This work currently includes steam explosions and interactions of molten core material with concrete. In NUREG-0496, the ACRS recommended that work on phenomena important to the course of postulated core melt accidents should continue to receive high priority. In Sections 1.2.4 and 1.2.6 of NUREG-0603 (See Appendix B), the ACRS recommended an augmented research program on steam explosions and a conceptual study to examine the practicality of molten core retention within containment. The ACRS recommends that the existing program be reoriented and strengthened accordingly, and furthermore, that it be closely coordinated with work being done on the cause of severe accidents.

6.4 Recommendations

The ACRS recommends that research in the area of fuel behavior be funded at the level requested, but that some funds be redirected from work on transients to work on more severe accidents, including fuel melt.

7. PRIMARY SYSTEM INTEGRITY

7.1 Scope

This program is concerned with maintaining the integrity of the primary system. It is concerned with detecting incipient cracks, predicting their growth, and inhibiting cracking through the control of coolant chemistry.

7.2 General

Design procedures and inspection techniques are amply funded, but the important problem of degradation of system integrity by the plant coolant is not addressed. Cracks have developed in several piping systems, but the worst degradation is clearly in the PWR steam generator.

A costly postmortem examination is planned for a decommissioned steam generator to improve future inspection techniques. The inspection and replacement of steam generators is a major contributor to occupational radiation exposure. The rupture of steam generator tubing is a disconcerting challenge to the safety system.

A program should be started to provide a firm basis for establishing and judging adequate systems for coolant chemistry control. Several of the recent incidents of pipe and nozzle cracking occurred at locations not subject to routine in-service examination. This suggests a need for a systematic review of current thinking about system behavior and degradation and an evaluation of the possible need for redirecting research in order to anticipate or prevent similar cracking incidents in the future.

7.3 Comments

7.3.1 Fracture Mechanics

This ongoing program addresses important questions. It should continue as planned.

7.3.2 Operating Effects

This program consists of two areas: Irradiation Effects and Dosimetry, a valuable well organized program; and Steam Generators a program about which the ACRS has reservations. The main effort of this latter program involves a detailed, destructive examination of one of the steam generators removed from the Surry Power Station. A careful study must be made to determine if a positive contribution can be made by the steam generator study before performing work in addition to that needed to determine the correlation between non-destructive examination indications and tube integrity.

7.3.3 Non-Destructive Examination

This is an expanding program on an important topic. The coherence, as well as coordination with regulatory needs, leaves something to be desired. The program should be funded, but the ACRS urges that the NRR and RES managements improve the coordination of the programs on Primary System Integrity with respect to regulatory needs. Also, emphasis should be placed on how program developments will influence design and practice in plants.

7.3.4 Corrosion and Cracking

A series of problems in operating reactors are related to the control of primary and secondary coolant chemistry. Examples are steam generator degradation culminating in replacement, cracking in primary piping, and cracking in stagnant water lines. The NRC has very limited capability to develop procedures to prevent such problems. The new program on cracking in BWR piping should be broadened to consider the corrosion-accelerated problems found in PWR pressure boundaries. The criteria for water chemistry limits, plant design, and operating procedures required to approach more trouble-free operating conditions should be addressed.

7.4 Recommendations

The ACRS believes that the proposed funding level for this program is adequate. A strong program is needed on improved coolant chemistry control.

8. SEISMIC, ENGINEERING, AND SITE SAFETY

8.1 Scope

This program includes research on extreme external phenomena, such as earthquakes, hurricanes, floods, and tornadoes; on site conditions such as seismology, geology, hydrology and meteorology; on seismic design of nuclear facilities; and on other mechanical and structural engineering aspects of nuclear safety.

8.2 General

The work on extreme external phenomena is well established and is for the most part directed toward seismology and geology. The Seismic Safety Margins Research Program (SSMRP) and the organizational area whose efforts are directed toward mechanical and structural problems have been in existence for about two years.

8.3 Comments

8.3.1 Seismology, Geology, and SSMRP

The research program on seismology and geology and the SSMRP are among the high priority NRC programs. It is important that the SSMRP program be structured to provide input as early as is feasible into the broad safety policy considerations concerning the seismic design bases of nuclear power plants. This should include a timely preliminary evaluation of the seismic contribution to the probability of serious accidents and the principal contributors to uncertainty in such probability estimates.

8.3.2 Hydrology

The program on hydrology should be kept under continuing evaluation to see if the current low level effort is adequate to support possible informational needs arising from the consideration in future siting policy of liquid pathways effects from serious accidents.

8.3.3 Structural and Mechanical Engineering

The programs in structural and mechanical engineering are relatively new. Effort should be devoted to the formulation by FY 1981 of a broad research program responsive to the NRC needs arising from operating reactors and from reactors to be constructed. The research program should include efforts devoted to provide the NRC with an improved capability for design audit and to evaluate the significance of off-design conditions during potential accidents or other severe loading conditions such as a large earthquake, as well as an improved basis of experimental verification of seismic design.

8.4 Recommendations

The ACRS favors long-term growth in the areas included in this program and supports funding at the proposed level.

9. ADVANCED REACTORS

9.1 Scope

This chapter deals with research needed for licensing advanced reactor types, specifically Fast Breeder Reactors and Advanced Converters.

9.2 General

The proposed budget provides $5 million for closing out Fast Breeder Reactor Safety Research in FY 1981 and no funds for Advanced Converter Safety Research.

In earlier reports, including NUREG-0603 and NUREG-0496, the ACRS has consistently supported an NRC research program related to the safety of advanced reactor types. This recommendation has been based on the perception that many of the current safety problems associated with LWRs have resulted from the fact that safety research lagged behind reactor development. The Administration has proposed to defer indefinitely LMFBR development and to provide for no work related to other concepts. If Congress agrees with this indefinite deferral, the ACRS would agree that there is no need to maintain an Advanced Reactor Safety Research Program. On the other hand, if the deferral is to be short-term, or if the development program in FY 1981 is to continue at a pace similar to that of the last few years, then the arguments presented earlier for an NRC program are still valid. Finally, if the expectation is that a move to exercise the LMFBR option in the next 10-20 years will be accommodated by importing foreign technology, it is important that the NRC program of safety research on advanced reactors be maintained to ensure an adequate technical basis for U.S. regulatory standards, guides, and criteria.

9.3 Comments

As in its 1978 report (NUREG-0496) to the Congress, the ACRS recommends that, if the research program is continued, a broader spectrum of possible fast breeder reactor accidents be examined. The Advanced Reactors Safety Research Program proposed for FY 1981 moves very slightly in that direction. The movement should be accelerated as soon as feasible.

It is noted that a significant portion of the work that is being carried out and planned in the Fast Breeder Reactor program will provide benefits to the LWR programs, especially in the area of core melt phenomena. If the Fast Breeder Reactor work is terminated, the LWR programs should be reviewed and the funding augmented as necessary.

If it is decided to support research on advanced reactors, the ACRS has the following comments based on the programs proposed for Fast Breeder Reactors and for Advanced Converters:

9.3.1 Analysis

This is primarily code development and qualification, but includes some work on accident delineation which purports to be responsive to the ACRS recommendations of 1978 that "NRC undertake a comprehensive study of the safety questions that are likely to arise for commercial LMFBRs. The ACRS believes that there is a high-priority need to review all possible sources of serious accidents (e.g., loss of shutdown-heat-removal capability), their probabilities, and their level of seriousness in plants of commercial size. Considerable use of probabilistic analysis techniques should be made. Preliminary conceptual designs should be utilized in the studies as a means for focusing on an integrated approach to the solution of problems such as post-accident heat removal." However, the ACRS also commented concerning the SIMMER computer code. "...it is doubtful that the code can ever be validated in the sense of precise calculations of such parameters as pressure, temperature, energy release, etc. Rather, the ACRS believes that the primary value of the code will lead to increased understanding of the event.... The ACRS expects that reduction of the code development goals will lead to more modest experimental needs and lower costs than previously anticipated." The intent was that additional emphasis be given to investigation of a broad spectrum of accidents. The ACRS does not believe that the proposed FY 1981 allocation provides enough emphasis on other than core disruptive accidents. Attention is directed again to the recommendation quoted in part above. It is believed that both accident delineation and accident prevention should receive greater attention than now seems indicated. In addition, the accident delineation work that is proposed seems to put too much emphasis on the Clinch River Breeder Reactor. However, the SIMMER computer code and the other analytical activities are viewed as important and valuable, and need to be continued at a level adequate to sustain them.

9.3.2 Aerosol Release and Transport

This is a combination of analyses and experiments aimed at an important problem area. The work seems well planned and is producing results.

9.3.3 Materials Interaction

This item includes funds for loop design and fabrication and for a series of fuel tests. It is clear that fuel research needs to be done. While the NRC needs to do work on problems crucial to licensing concerns, more determined effort should be made to have the fuel developers assume a larger part of the investigative burden. In addition, more effort is needed to obtain a more precise formulation of the questions to be asked and how the answers are to be obtained with these facilities.

9.3.4 System Integrity

The proposed program involves testing of the CONTAIN computer code and carrying out a set of experiments associated with molten core retention, core debris coolability, and container cell liner response to accident loads. Some of the work on molten core retention is also useful in connection with licensing concerns of the Floating Nuclear Plant and in consideration of problems associated with severely damaged cores in water reactors generally. The work associated with this item seems appropriate to future needs in the development and licensing of fast breeder reactors. However, the ACRS believes, as recommended in NUREG-0496, that specific attention should be given to the study of alternate containment systems and to conceptual studies of systems for retaining a molten core in containment.

9.4 Recommendations

Funding at a level of $18.0 million is recommended for support of research on Fast Breeder Reactors and Advanced Converters.

10. REACTOR ENVIRONMENTAL EFFECTS

10.1 Scope

Research in this program area includes studies of the buildup of radionuclides within nuclear power plants, the assessment and control of associated occupational radiation exposures, the behavior, transport and control of radionuclides discharged into the environment, and the evaluation and control of associated population exposures. The last two areas apply to both routine and emergency situations.

10.2 General

On the basis of this review, the ACRS has concluded that three areas of research within this subject area are still essentially not addressed by the current program. These are:

(a) Research to determine the basic factors that govern radionuclide buildup in reactor coolant systems, including the influence of operating practices on such buildups.

(b) Research to develop methods for evaluating the effectiveness of measures for removing radionuclides from the primary coolant circuits of operating reactors.

(c) Research on emergency planning.

These needs were commented on in the ACRS report of July 1979 (NUREG-0603). In the case of (a), a research program needs to be developed and funded. In cases (b) and (c), some work is underway but more attention needs to be directed to problems associated with the decontamination of operating reactors and the recovery and reentry phase following an accident.

10.3 Comments

With respect to the level of effort on specific items, the ACRS offers the following comments:

10.3.1 Priority Items Within the Program as Planned

Of the individual projects outlined in the program as planned, the following are considered by the ACRS to have priority:

Development of Mathematical Models for the Transport of Radionuclides in Water and Sediment. Portions of this work that should be emphasized include evaluations of the liquid pathway, particularly as it pertains to radionuclide releases from a nuclear power plant under conditions of a severe accident.

Development of Mathematical Models for the Atmospheric Transport of Airborne Radionuclides. Although models for the transport of airborne radionuclides over short distances are reasonably adequate, there continues to be a need for an improved capability to assess the behavior of airborne releases at moderate (5 to 15 kilometers) and greater distances (16 to 80 kilometers) from nuclear power plants. This is especially important relative to emergency planning, where models are needed to provide projections on a real-time basis.

Occupational Radiation Assessment and Protection. In addition to the work described in Section 10.1 above, there is a need for research on developing better methods for assessing neutron exposures in nuclear power plants. This work should include the application on a routine basis of the newer techniques now available. The ACRS also endorses the program for the incorporation of newer data on the biological behavior of radionuclides into the NRC internal dosimetric models, and encourages the application of probabilistic assessment techniques in the establishment of internal dose limits. In addition, the ACRS strongly supports the efforts to develop better means for providing respiratory protection to radiation workers, such as those involved in decontamination and post-accident recovery operations.

10.3.2 Priority Programs Not Within the Program as Planned

Key items of research relative to emergency planning include:

Accident Source Terms. Better definition of accident source terms is needed. Emphasis should be placed on requirements for instrument systems to provide the definitive types of data necessary for real-time projections of the nature and consequences of a release.

Interdictive Measures. Studies of the full range of interdictive measures with emphasis on their suitability for given sites and means for their improvement are needed. Accompanying this research should be a reevaluation of Protective Action Guides and the initiation of research to develop a better scientific basis for their establishment.

Recovery and Reentry Phases Following Accidents. Further investigations of improved measures that might be implemented in the recovery and reentry phase following an accident must be made. This program should include evaluations of designs and procedures to facilitate the decontamination and recovery of major nuclear power plant systems. It should also include research on procedures to aid decisions by medical and other authorities concerning the affected offsite population; methods for decontaminating and reclaiming offsite land, buildings, and equipment; and the establishment of dose limits or guides for population groups desiring to return to areas that have been evacuated.

10.3.3 Items Within the Research Program that are Considered of Low Priority

The ACRS does not believe that there is an urgent need for emphasis on research to improve the models for describing low level airborne or liquid radionuclide releases from nuclear power plants under routine conditions. This is especially true relative to refinements in the calculations that support 10 CFR 50, Appendix I.

10.4 Recommendations

Overall, the ACRS considers programs within this area to be important and believes that they are adequately funded. However, the ACRS urges that consideration be given to the development of new programs and the reorientation of existing programs as outlined in Sections 10.2 and 10.3 above.

Because related research on many of these topics is underway in other Federal agencies, such as the DOE and the Environmental Protection Agency (EPA), the ACRS urges the NRC Staff to keep abreast of such work and to take full advantage of the findings in helping to meet its own research needs.

11. FUEL CYCLE

11.1 Scope

The fuel cycle safety research program relates to: effluent control, safety system performance, occupational exposure and health, environmental impacts, transportation of radioactive materials, and decommissioning. Some of the projects contained in these six categories concern requirements related to NEPA and are, thus, outside the scope of this review. Some of the other projects pertain principally to regulatory and licensing problems associated with the use of radionuclides in medicine, industry, and research and are thus also outside the scope of this review.

11.2 General

This research program comprises a relatively small percentage, about two percent, of the total NRC research budget. Funds budgeted were $2.8 million in FY 1979, and $3.1 million in FY 1980; $4.4 million has been proposed for FY 1981. The program includes a broad mix of research topics of moderate to high priority.

11.3 Comments

Decommissioning. The ACRS believes that the research on decommissioning of fuel cycle facilities is important and should be funded above the level planned by the NRC. More emphasis should be given to research on the problems of decommissioning or long-term care of shallow land burial sites.

Effluent Control. The ACRS believes that the research effort on effluent control should be augmented. Increased emphasis is needed on the problem of radioactive gaseous wastes with respect to their removal, confinement, and long-term storage or disposal.

Safety System Performance. The ACRS believes that more work is needed to assure adequate performance of safety systems when they are called upon. For example, more information is needed on conditions that adversely affect air filter system capability and on testing methods to confirm that satisfactory performance capability exists.

11.4 Recommendations

The ACRS recommends that more emphasis be given to the research areas cited in Section 11.3; this can be accommodated by decreased effort on transportation research.

12. WASTE MANAGEMENT

12.1 Scope

The NRC waste management research program is directed to the public health and safety problems that result from the handling and ultimate disposal of high and low level radioactive wastes and uranium mill tailings. The potential risk from these activities is an important fraction of the total risk from all operations in the nuclear fuel cycle.

12.2 General

In its 1978 report (NUREG-0496) to the Congress, the ACRS criticized the NRC for the poor formulation and management of research work on waste management problems and for the inadequate rate of progress. Similar criticism was expressed by the ACRS to the NRC in the July 1979 report (NUREG-0603). In NUREG-0603 the ACRS added that upgrading was needed in the NRC research staff capability.

The ACRS believes that the NRC has taken positive steps to improve this situation. Staff capability has improved; however, more is needed. Commendable progress is also being made toward improved assessment and selection of research; further attention, however, is needed on this matter, especially the ordering of priorities. It is also apparent that there is now more interaction and review of research programs by the various groups within the NRC. Further, it appears that the managers of this work in NRC have initiated more effective communication and interaction with DOE, EPA, USGS and other organizations.

12.3 Comments

12.3.1 High Level Waste (HLW)

The ACRS believes that work in this area, including that related to the ultimate disposal of spent fuel, has high priority and that adequate funding is necessary for its timely completion. It is therefore urgent that the NRC develop detailed criteria and procedures needed to evaluate (1) the suitability of a HLW package for ultimate disposal and (2) the licensability of a geologic site as a repository for HLW packages. Some of the unique technical problems with respect to the latter task may require further augmentation of Staff capability, for example, added expertise in geological sciences and engineering.

12.3.2 Low Level Waste (LLW).

The ACRS considers this a high priority item. Although the planned FY 1981 funding by NRC for LLW research once might have been considered to be adequate, certain recent events suggest this may no longer be true. Experience has shown the need for better bases to enhance the guidance of LLW management, such as in recovery operations after a reactor accident. The ACRS believes also that more research is needed to provide better rationale and wider technical bases for the development of detailed site selection criteria and better procedures to assess the suitability of sites and site practices in the licensing and operation of commercial shallow land burial facilities for LLW. This information is urgently needed in order that additional acceptable sites can be expeditiously selected, evaluated, and licensed. Sites tha' are acceptable for is purpose can provide the needed flexibiliy to accommodate large w. ste volumes and shipping route alternatives. Specific areas of research needed relative to LLW management include:

(a) Techniques for reducing the volume of low level waste as well as the exploration of alternatives to shallow land burial as a method for disposal.

(b) Field monitoring equipment to evaluate the acceptability of LLW packages as received at a disposal site, including their content of free standing liquids.

(c) Methods for negating the influence of chelating agents, commonly used in radionuclide decont ination operations, in terms of later migration of radionuclides within and from a disposal site.

(d) Criteria for permitting public access to facilities formerly used in nuclear work, and development of associated field monitoring equipment to assure compliance with the criteria.

12.3.3 Uranium Mill Tailings

Good progress has been reported by DOE and NRC in resolving problems associated with the control of mill tailings. Techniques have been developed that decrease significantly the amount of radioactive releases from the uranium tailings piles. The results of studies to stabilize the tailings from erosion also appear promising. The NRC is developing methods to evaluate the long term effectiveness of these techniques. The ACRS believes that this work warrants the amount of funding requested and that it should have high priority.

12.4 Recommendations

In view of the growth rate of NRC's research in this area (from \$4.5 million in FY 1979 to \$8.6 million proposed for FY 1980 to \$13.6 million requested in FY 1981), the ACRS is still concerned about the adequacy of manpower and expertise available in RES and in the Office of Nuclear Material Safety and Safeguards (NMSS) to manage the program effectively and yet stay abreast of important developments that occur outside the NRC, for example, in DOE and in foreign countries. The ACRS believes that additional staff capabilities and technical expertise are urgently needed in this area and, therefore, recommends that the NRC be authorized and funded to expand the technical staff for this program by at least five people above the fifteen now proposed.

All segments of the waste management research program are of high priority, particularly those on HLW and LLW. The ACRS believes that the requested FY 1981 funding, which represents a substantial increase above that for FY 1980, is warranted; even some further increase for LLW may be justified.

13. SAFEGUARDS

13.1 Scope

This program is concerned with both safeguards and security. Safeguards refers to means of preventing the theft of special nuclear materials (SNM) from fixed sites or during transportation, and also to means of detecting loss or diversion of SNM. Security refers to the protection of nuclear facilities from sabotage or seizure.

13.2 General

It is difficult to compare the urgency of work in this area with that of work in other reactor research areas. The other programs mainly involve the operational safety of reactors, and it is at least possible in principle to assess their relative importance by comparing their possible contributions to reductions in risk. The importance of the safeguards program, on the other hand, is a direct function of the threat level which may be assumed to characterize attempts at theft or sabotage. One way of comparing these projects is to ask how long it would take, with some specified relative funding, to complete the projects now seen to be necessary in the various fields. If these times are not too different, then the relative funding could be said to be in acceptable balance.

At the proposed level of funding ($4.9 million), work on some projects (such as safeguards needs for some possible alternative fuel cycles) will have to be deferred; but work on some of the more immediate needs (such as evaluation of the physical security provisions for reactors) can be completed in two or three years. The proposed level of funding would seem to be close to the minimum acceptable level; but, on the basis of the comparison criterion suggested above, the ACRS considers the proposed funding to be reasonable.

Recently the NRC has moved to consolidate the planning of safeguards work. The Safeguards Technical Assistance and Research (STAR) group, with representatives of the various NRC offices, has been established to monitor all proposals for research or technical assistance projects on safeguards. In addition, responsibility for all safeguards operational activities has been transferred to NMSS. These changes have improved the coherence of the research program.

13.3 Comments

13.3.1 Evaluation Methods

This includes evaluation of the effectiveness of physical protection provisions for fixed sites and for material in transit; materials control and accounting methods (MC&A) for SNM; security force selection and training; and support of the development of regulatory guides and standards. Upgraded rules concerning physical security provisions and guard force requirements have already been issued, and a rule for MC&A is about to appear. Following a trial period to ascertain if modifications are necessary these rules will be turned over to the user offices. All major projects in this program element are expected to be completed by about FY 1983.

In FY 1981, the program will include research on: automated MC&A systems; vital areas in power reactor plants; transport of high level waste; spent-fuel storage and shipment; and assessing the new upgraded rules on the basis of experience with their implementation.

13.3.2 Inspection Methods

This covers work intended to help inspectors evaluate the safeguards provisions in effect; to assess the implementation of the upgraded rules for guards; to develop the methods and tools required for monitoring MC&A performance; and to review and evaluate safeguards contingency plans. Major projects in this program element are expected to be completed by about FY 1983.

In FY 1981, one or more of the new rules will be tested and transferred to the regional inspection staff for operational use.

13.3.3 Alternative Strategies

This program consists of longer range projects. These include: plant design alternatives and damage control measures; vulnerability of spent fuel storage pools; methods for analyzing and dealing with communicated threats; and safeguards requirements for alternative fuel cycles and new enrichment or separation technologies.

In FY 1981, the work proposed will include: techniques for MC&A in process systems; response to communicated threats; source terms resulting from attack on shipping casks in transit; and safeguards requirements for proliferation-resistant fuel cycles.

13.4 Recommendations

All of the projects proposed are needed to meet safeguards requirements. In many instances more rapid progress would be desirable; whereas slower progress would scarcely seem to be acceptable. The ACRS considers the proposed level of funding to be marginal, but adequate.

It is recommended that possible conflicts between desirable safeguards requirements and essential operational safety requirements be identified and resolved, and that human response in the context of proposed mechanical and procedural safeguards provisions should be studied. Work in these areas probably could be accommodated within the proposed budget by some curtailment of that presently planned for other projects under Alternative Strategies (Section 13.3.3).

14. RISK ASSESSMENT

14.1 Scope

This program includes research on probabilistic methodology and software, equipment and human failure rates, nuclear fuel cycle risk, and risk acceptance criteria. In addition, it includes a program of probabilistic analysis in support of licensing and the integrated reliability evaluation program. Furthermore, this program provides risk-based guidance for various other activities in the NRC as well as training in probabilistic methods to the NRC Staff.

14.2 General

The ACRS strongly supports the planned growth in this research program. Furthermore, in its recommendations for new directions in research (Chapter 2), the ACRS identified several additional areas which can logically be located in this program. These matters include studies of the courses of serious accidents, inadequacies in the single failure criterion, and the effects of the considerations of serious accidents on siting criteria. The ACRS therefore believes that this research program should receive some of the funding which can be reallocated from the proposed reduction in the support of LOFT.

14.3 Comments

(a) The Integrated Reliability Evaluation Program has considerable potential for an important contribution to the improvement of the safety of existing reactors and should receive high priority. The nuclear industry should initiate and place emphasis on its own concurrent program in order to more quickly evaluate reactor design aspects which can and should be improved in a timely fashion.

(b) Priority should be given to an evaluation of flood models and to a more realistic examination of potential on-site and off-site effects of a large release of radioactive material, including possible decontamination measures.

(c) The topics relating to reactor systems analysis and licensing support should include the early development of an improved alternative to the single-failure criterion.

(d) The work on nuclear fuel cycle risk should include a focus that will provide the NRC with improved bases for the promulgation of criteria for ultimate disposal of high level wastes.

(e) Other matters which should be addressed include projects to provide information needs outlined in the "Report of the Siting Policy Task Force" (NUREG-0625), and to provide data for assessing the advantages and disadvantages of multi-unit versus single-unit sites.

(f) The ACRS has previously recommended that the NRC attempt to develop quantitative risk acceptance criteria for public comment and for review by the Congress itself. The ACRS believes that this effort should be given high priority, well beyond that afforded it thus far by the NRC.

14.4 Recommendations

The research and applications program in Risk Assessment is of high priority and should be funded at a level of $15.0 million.

15. IMPROVED REACTOR SAFETY

15.1 Scope

The Energy Reorganization Act of 1974, which effected the separation of the AEC into the NRC and ERDA, established within the NRC an Office of Nuclear Regulatory Research (RES). In view of the legislative history (Conference Report), the role of RES was accepted by the NRC as being limited to "confirmatory research", presumably as opposed to the "developmental research" that had dominated the AEC program.

Several attempts were made by RES to define "confirmatory research". The need for definition, however, was in effect obviated by the issuance of a directive that the research must be responsive to and, except in special circumstances, initiated by "user" needs. Users were considered to be principally the other NRC program offices but to include also the Commission, Atomic Safety and Licensing Boards, Atomic Safety and Licensing Appeal Board, ACRS, Congress, the technical community, and the public.

The limited nature of "confirmatory research" was recognized implicitly by the Congress when, in its Budget Authorization Act for NRC for FY 1978 (PL 95-209), it amended the Energy Reorganization Act of 1974 to require the NRC to develop a long range plan for the "development of new or improved safety systems" for nuclear power plants. Such a plan was developed and submitted to the Congress in April 1978 in NUREG-0438, in which 16 research topics were evaluated and 7 of them were proposed for a three-year program estimated in 1978 to cost about $15 million. The recommendations of that report were endorsed by the ACRS in a March 13, 1979 report to the NRC Chairman, and in its reports to the Congress in 1977 and 1978 (NUREG-0392 and NUREG-0496). The recommendations presumably were endorsed also by the Commission and by the other NRC Program Offices.

15.2 General

No funds were available for this program in FY 1978. For FY 1979, the Congress authorized $1.5 million but did not appropriate funds specifically for this purpose. In spite of a lack of appropriated funds, work was begun at various times in FY 1979 using funds from three sources:

FY 1978 unobligated carryover funds	$0.40 million
FY 1979 reprogrammed RES funds	$0.40 million
FY 1979 confirmatory research funds	$0.15 million

These programs address four of the seven proposed areas of research to improve reactor safety.

For FY 1980, the NRC requested $4.3 million for research in this area but this was reduced by the Office of Management and Budget (OMB) to $1.0 million. In addition, OMB stipulated that no funds in this category could be used by NRC to support experimental work, presumably on the assumption that a very limited program of strictly analytical and conceptual research by the NRC would be supplemented by research carried out or funded by the DOE. (The Congress has followed the OMB recommendations in its appropriation for FY 1980.) Following the accident at TMI-2 , the NRC requested a supplemental appropriation for FY 1980 to provide for research and other activities related to it. This request included $27.2 million for research program support. Although RES had requested $3.4 million for research to improve reactor safety, and this level of support had been endorsed strongly by the ACRS in its "Comments on the NRC Safety Research Program Budget" submitted to the Commission in July 1979 (NUREG-0603), the Commission did not include any additional funding for research on improved reactor safety in its request for a supplemental appropriation for FY 1980. At this time, the funds available for this program in FY 1980 amount only to the $1.0 million to which it was restricted by the OMB.

For FY 1981, RES requested $6.6 million, a level also supported by the ACRS in NUREG-0603. In view of the fact that the FY 1980 Supplemental Request for Improved Reactor Safety had not been approved by the Commission, the FY 1981 request was reduced to $4.5 million, presumably to be compatible with the $1.0 million level available for FY 1980. The budget request now before the Congress includes $4.5 million for this program.

As a result, the funds available specifically for the three-year period, FY 1979 through FY 1981, will be only $6.45 million, far below the 1978 estimate of $15 million for a three-year initial program of research to improve reactor safety.

15.3 Comments

The ACRS has indicated repeatedly its strong support for a vigorous and well-funded program of research to improve reactor safety. It offered this view on several occasions prior to the TMI-2 accident and has repeated it since. The importance of two of the projects in the proposed program, those relating to vented and filtered containments and to improved in-plant accident response (human-interactions) have received widespread recognition as a result of the TMI-2 accident; others may prove to be equally important.

15.4 Recommendations

The ACRS repeats its recommendation that a vigorous and well-funded program of research to improve reactor safety be started, on a crash basis if necessary, and assigns it the highest possible priority. The ACRS recommends that funding for the program for FY 1981 should be at a level sufficient to permit aggressive programs on alternate decay heat removal concepts, vented and filtered containments, and improved in-plant accident response, all of which are closely related to the TMI-2 accident. Scoping studies of other likely projects also should be undertaken as a basis for planning future programs. Funding at a level of $10 million is recommended.

The ACRS considers it necessary to call attention to the rather unfortunate distinction that has been made between "confirmatory research" and "research to improve reactor safety". Although the latter has consistently been assigned a high priority by RES in its requests for funds, it has been difficult to provide funds for "research to improve reactor safety" under circumstances where this might have resulted in some reduction of funds for an item of "confirmatory research" believed by one of the "user offices" to be directly related to their own real or perceived needs. It must be noted, however, that this attitude may be changing, and may be expected to change further in the aftermath of the TMI-2 accident. Research on alternate decay heat removal concepts (one of the projects identified in NUREG-0438 as "research to improve reactor safety") has already been started, using "confirmatory research" funds, in response to a request from a "user office". In addition, the concept of a vented and filtered containment has received prominent mention in the "TMI-2 Lessons Learned Task Force Final Report" (NUREG-0585), so that research in this area might also be the subject of a "user request" and, qualify for support from "confirmatory research" funds.

The foregoing comments indicate that the distinction between "confirmatory research" and "research to improve reactor safety", is neither very clear nor useful, if indeed there ever was a legitimate distinction. It is similiarly artificial to stipulate that in its research to improve reactor safety the NRC should not make use of experimental studies where those would seem to be the most advantageous means to follow. The ACRS suggests that the Congress review the legislative charter of the NRC research program and eliminate this distinction. Inasmuch as the purpose of the NRC is to ensure safe design, construction, and operation of nuclear power plants, it should follow that any research is legitimate that contributes to that purpose without compromising the credibility or independence of the NRC. The ACRS urges consideration of this matter by the Congress.

Appendix A

GLOSSARY

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
BDHT	Blowdown and Reflood Heat Transfer
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Administration
FRG	Federal Republic of Germany
FY	Fiscal Year
HLW	High Level Waste
LLW	Low Level Waste
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss-of-Coolant Accident
LOFT	Loss of Fluid Test
LWR	Light Water Reactor
MC&A	Materials Control and Accounting Methods
NEPA	National Environmental Policy Act
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation

OMB	Office of Management and Budget
PBF	Power Burst Facility
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
RIA	Reactivity Insertion Accident
SNM	Special Nuclear Material
SSMRP	Seismic Safety Margins Research Program
TLTA	Two Loop Test Apparatus
TMI-2	Three Mile Island, Unit 2
USGS	United States Geological Survey

APPENDIX B

EXCERPT FROM NUREG-0603

SECTION 1.2 OF CHAPTER 1, ENTITLED

"IMPLICATIONS OF THE ACCIDENT AT

THREE MILE ISLAND, UNIT 2"

1.2 Recommendations for New Directions in Research

In its review of the budget proposals in Parts 2 and 3 of this report, the ACRS has identified a number of areas in which the programs are not yet completely defined in content, but for which the need for research and funding is clear. The following recommendations for new directions in the NRC Safety Research Program are intended to provide guidance to the Commission, to the RES Staff, and to the user offices, that can be utilized in the detailed formulation of research programs for FY 1980, to the extent practicable and for FY 1981, and for the development of requests and plans for FY 1982 and beyond.

The ACRS recognizes that research has already begun in many of these areas, and expects that others will be considered and implemented in a timely fashion. The ACRS believes that this can and should be done without delaying the ongoing budgetary process.

1.2.1 Priorities and Focus

The ACRS believes that the research and regulatory staff of the NRC should, in the reasonably near future, reevaluate the overall priorities, levels of expenditure, and focus of the safety research program. The ongoing program to a large extent reflects priorities that were established several years ago and has been strongly influenced by the single failure concept and research needs arising from detailed studies of design basis accidents. While useful results are being obtained from most ongoing research tasks, it is important that the Staff take a new broad look at the existing and recently proposed levels of support and research directions to evaluate the potential need for major change in emphasis.

The ACRS suggests that the existing structure of the safety research program, which was developed to manage a research program plan established a few years ago, be reviewed to determine whether modifications are appropriate to meet the requirements of the coming years.

Also, the ACRS notes that the focus of the research program has reflected the needs of the NRC regulatory staff as perceived in past years. Here, too, early attention should be given to an evaluation of the priorities of the detailed existing requests as well as requests arising from changed perceptions in safety research priorities.

1.2.2 Anomalous Transients and Small LOCAs

The need for greater emphasis on transients and small LOCAs has been recognized. The ACRS recommended increased effort on transients in its 1977 and 1978 reports to Congress, and emphasized the study of anomalous transients in its Interim Report No. 3 on TMI dated May 16, 1979.

A research program on anomalous transients should have as its focus the need for greater understanding of the probable course of a wide range of possible events leading to severely degraded conditions, in order to provide a better basis for operator training, for improved instrumentation, and for possible on-line computer-diagnostic procedures to aid the operator. Equally, such studies should provide insight into the significance of possible design modifications and into areas of research warranting further study in order to have an appropriate degree of preparedness and background knowledge.

Such a program should receive coordinated guidance by a group including representatives from both licensing and research.

1.2.3 Accident Studies

The NRC should initiate a series of analytical studies to explore the probable course of events and possible potential consequences of a broad spectrum of accidents which go well beyond the current design bases in terms of the damage to the core and the release of radioactivity to the environment via both atmospheric and liquid pathways. In particular, specific studies should be carried out to scope scenarios of serious accidents beginning from the initiating event through to the eventual resting place of a melted core for some of the sequences.

Preliminary guidance for the choice of scenarios to study can be provided by WASH-1400, although the TMI experience showed that many sequences must be considered altered by human intervention at some point. For each scenario, sufficient technical detail should be provided to obtain insight into such matters as the following: to what extent can the probability and consequences of the sequence be quantified; what are the intermediate stages in the sequence, and to what extent may they be affected by human intervention; how serious is the sequence in terms of its effects on human health; where are the trigger points for emergency action, and what are the criteria therefor; etc.?

It is especially important that these studies concern themselves with the identification of significant sequences that have not received sufficient research attention, so that one can develop in advance significant safety procedures, and equipment, and mitigating actions to avoid surprises of the sort that occurred at TMI.

It is expected that such studies would be useful in the specification of instruments to help diagnose and follow the course of an accident, in the identification of new research and development needs, in siting considerations, in modification of containment, etc. An effort on the order of ten man-years is envisioned.

1.2.4 Molten Core Retention

The NRC should undertake a conceptual study to examine the practicality of retaining a molten core within containment or significantly reducing the release of radioactivity via liquid pathways following penetration of the containment foundation, in order to help provide insight into the practicality, benefits and costs of such a safety feature.

1.2.5 Power Burst Facility

The PBF program should be reoriented to emphasize primarily the study of the processes leading to medium and severe core damage in postulated accidents, the possible consequences of considerable molten fuel in the core, and possible measures to mitigate large scale core melt.

1.2.6 Steam Explosions

The ongoing research program on steam explosions should be substantially augmented to gain a better assessment of their potential role in various postulated accident scenarios, as well as possible insight into measures which could reduce the probability of a large scale thermal reaction, if such a reaction is possible.

1.2.7 Siting

A more extensive evaluation should be made of possible offsite consequences via liquid pathways for postulated accidents involving core melt for a broad range of land-based sites whose characteristics are reasonably representative of reactor sites in use, projected for use, or of potential interest in long-term planning. Such an effort has already been initiated as part of the NRC research program. The depth of the program should be sufficient to provide the background information needed for the possible development of hydrologic siting criteria which allow for the possibility and probability of accidents beyond those currently designed for.

A study should be made of the relative and absolute accident risks, with uncertainties, for a wide range of potentially suitable sites. The study should examine the costs and benefits associated with different types of sites and should include the possible interaction of a serious accident in one reactor on other reactors at the site. The intent of the study should be to provide insight into the relative advantages and disadvantages of more remote siting and power parks.

1.2.8 Plant Operations

A systematic effort should be made to identify research needs relating to the safety implications of procedures for operation, maintenance, testing and surveillance. Operating experience should be reviewed to identify existing problems in these areas and to determine problems important to safety.

1.2.9 Transient Simulation in Research and Licensing

Early consideration should be given to augmentation of the range of NRC capability to simulate various postulated transient and accident sequences to varying degrees of sophistication, including but not limited to real time analysis and permitting a simulation of operator action and intervention. Development of such simulation capability should enable a more detailed understanding of the course of events for various transients, and would be useful in the development of improved operator procedures and training, diagnostic instrumentation, and computer-aided guidance to the operator.

1.2.10 Systems Behavior and Interaction

A new research program should be established in systems behavior and interaction which includes an interdisciplinary approach to safety research including electrical, thermal-hydraulic, mechanical, control, and heating, ventilating and air conditioning systems, under operational, transient and accident conditions. Such a program should provide increased insight into the suitability of existing operational limits, the effect of system arrangement on its ability to withstand abnormal transient conditions, and the degree to which system design changes can be made to improve safety in one way without adversely influencing safety or reliability under other sets of conditions.

1.2.11 Application of Probabilistic Methodology

The ACRS recommends emphasis on the application of probabilistic and other methodology to an evaluation of the adequacy of the single failure criterion and to studies of alternate design approaches to systems and groups of systems important to safety in order to provide a better basis for decision making concerning the optimization of plant design for safety.

1.2.12 Water Specification and Crack Growth

The Committee recommends that programs be initiated to develop appropriate water chemistry specifications, particularly in the BWR primary coolant and PWR secondary coolant, and to establish the effect of environmental, fabrication, and operating variables on crack growth rates in the coolant system boundary. Cracking is a recurring problem and the NRC lacks a basis for establishing conservative practices to prevent it.

1.2.13 Disturbance Analysis

The ACRS recommends that both the licensing and research arms of the NRC Staff place considerable priority on the development of methods for real-time analysis of system disturbances, in an effort to provide improved diagnostic information to the operator concerning abnormal sequences and, as possible, to suggest favored courses of action. The ACRS anticipates that the efforts devoted to the development of such disturbance analysis systems will, of themselves, provide considerable insight into reactor behavior which will be useful in design and in operator training.

APPENDIX C

THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards was established as a statutory committee in 1957 by revision of the Atomic Energy Act. The ACRS was charged with the responsibility for review of safety studies and facility license applications submitted to it, and to make reports thereon, advising the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards, and to perform such other duties as the Commission might request. Section 182b of the Atomic Energy Act requires ACRS review of the construction permit and operating license applications for power and testing reactors and spent fuel reprocessing facilities licensed under Sections 103, and 104b or c of the Atomic Energy Act; any application for a research, developmental or medical facility licensed under Section 104a or c of the Act and which is specifically referred to it by the Commission; and any request for an amendment to a construction permit or operating license under Sections 103 or 104a, b, or c which is specifically referred to it by the Commission. The Energy Reorganization Act of 1974 transferred operation of the ACRS from the Atomic Energy Commission to the Nuclear Regulatory Commission.

In 1977, Public Law 95-209 added to its other duties a requirement for the ACRS to undertake a study of reactor safety research and to prepare and submit annually to the United States Congress a report containing the results of this study. The first of these reports was submitted to the Congress in December of 1977.

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U.S. NUCLEAR REGULATORY COMMISSION

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7. AUTHOR(S)	
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16. ABSTRACT (200 words or less)

Public Law 95-209 includes a requirement that the Advisory Committee on Reactor Safeguards submit an annual report to Congress on the safety research program of the Nuclear Regulatory Commission. This report presents the results of the ACRS review and evaluation of the NRC safety research program for Fiscal Year 1981. The report contains a number of comments and recommendations.

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Appendix XXIV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
February 20, 1980

APPENDIX XXIV
LTR TO REP M.K. UDALL ON CONSISTENCY OF COMPONENT FAILURE EXPERIENCE WITH THAT PROJECTED IN WASH-1400

The Honorable Morris K. Udall, Chairman
Committee on Interior and Insular Affairs
House of Representatives
Washington, D. C. 20515

Dear Congressman Udall:

In a letter dated July 27, 1979, you expressed the hope that the study of Licensee Event Reports by the Advisory Committee on Reactor Safeguards would address the consistency of actual component failure experience (e.g. valve failure rates) with that projected in WASH-1400. You also asked the ACRS to determine the probabilities of occurrence that, prior to the events, would have been predicted for the sequences of events that occurred at Davis-Besse on September 24, 1977 and at Rancho Seco on March 20, 1978 on the basis of WASH-1400 failure rates and methodology. In a letter dated August 15, 1979, the ACRS advised you that it would undertake to provide a detailed response to your requests and that it hoped to be able to complete this effort in approximately six months.

Of course, the calculation of the probability of an event sequence, in retrospect, is ill-defined, since it depends entirely upon the ensemble of event sequences in which the one under discussion is embedded. This letter includes what are thought to be reasonable judgments on this point, and the results depend upon these judgments.

With the aid of the NRC Staff, the ACRS invited a large number of institutions in the U.S. and abroad, including the Electric Power Research Institute and the U.S. reactor vendors, to provide data and analyses responsive to your request. Several groups, including the NRC Staff itself, have submitted component failure rate data developed since the compilation was made for the Reactor Safety Study, WASH-1400. The NRC Staff have summarized the new data in Table 1, which also provides the failure rates used in WASH-1400 for the same components and systems. Some of the information in Table 1 is plotted in Figure 1 and illustrates graphically the considerable spread in data obtained and the relative position of WASH-1400. Also of some interest is the considerable variation observed from plant to plant which is illustrated in Figure 2. Only plants which reported any failures are shown in Figure 2; hence, some plants had much higher failure rates than WASH-1400 on certain components while other plants had no failures during the reporting period studied. Although to some degree the observed variation may reflect actual differences from plant to plant, a certain portion of the variation may be due to differences in the reporting requirements specified in the individual plant Technical Specifications and to differences in the responses of reporting personnel.

Turbine-driven pumps generally exhibit a higher failure rate (a factor of 10 to 100) than used in WASH-1400. The NRC Staff is now giving extra attention to this specific item. Furthermore, a large variation in diesel reliability was observed among the various plants.

The NRC Staff believe that the uncertainties in failure rate data are larger than were projected in WASH-1400, and that the general trend is toward somewhat higher failure rates. Their preliminary assessment is that this might produce an increase in their best estimate of core melt probability by about a factor of three.

None of the groups who were invited have provided probabilistic analyses, using WASH-1400 failure rates and methodology, of the Rancho Seco and Davis-Besse transients of March 20, 1978 and September 24, 1977 respectively. The ACRS, therefore, asked three ACRS Fellows to devote effort commensurate with the time available to provide such analyses; the results of their study are included as Attachment A to this letter.

The ACRS believes that the results they obtained are reasonable. It is clear that the manner of treatment of human error can have a very large effect on the results obtained. Also, for the Rancho Seco transient, the numerical results are very sensitive to the context in which failure of control system power is calculated.

The ACRS Fellows also estimated a probability per reactor year of occurrence of the major sequences which were present in the Three Mile Island 2 accident of March 28, 1979. Of some interest in this regard is an observation by representatives of Electricite de France that by applying WASH-1400 methodology they would calculate an overall probability of the order of $3x10^{-7}$ for TMI-2, but when the events were connected by strategic operator errors, they found a probability as high as $6x10^{-3}$.

The ACRS anticipates that, had several institutions provided independent estimates of the probability of the two transients, a considerable variation in their answers would have been likely.

Although the NRC Staff did not analyze the probability of the Rancho Seco transient using WASH-1400 failure rate data and methodology, they did provide the ACRS with two related memoranda, which are enclosed as Attachments B and C for your possible interest.

The ACRS trusts that this letter is responsive to your request.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Attachments:

A. ACRS Fellows Report, "Analysis of Feedwater Transient Sequences in B&W Nuclear Steam Supply Systems," February 7, 1980
B. Nuclear Regulatory Commission Staff Report, "Evaluation of Davis-Besse and Rancho Seco Feedwater Transients on 9/24/77 and 3/20/78 Using WASH-1400 Data"
C. Memorandum from F. Rowsome to R. Fraley, "ACRS Query on Material Relevant to Udall Letter: Davis-Besse and Rancho Seco Transients," February 12, 1980

ACRS Fellows Report, "Analysis of Feedwater Transient Sequences in B&W Nuclear Steam Supply Systems," February 7, 1980

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Attachment A

TABL.

SUMMARY OF CURRENT FAILURE RATE DATA SURVEY

COMPONENT	FAIL MODE	(1) BIBLIS	(2) GENERAL ATOMIC	(3) LER EVALUATION PROGRAM	(4) NCSR	(5) NPRDS	(6) WASH-1400	(7) WESTINGHOUSE	(8) VOLTA	(9) PICKARD, LOWE AND GARRICK	(10) BECHTEL
AUX FEED PUMPS	FAIL TO START FAIL TO RUN		+1E-2(3) 1E-4(3/10)					2.0E-5 1.2E-5			1.7E-5
ECCS PUMPS	FAIL TO START FAIL TO RUN FAIL TO START & RUN	+8.1E-3	+1E-3(3/10)	+2.7E-3 (2.3-12)E-4	+2E-5 3.5E-4	3.9E-5	+1E-3(3) 3E-5(10)	1.5E-5A	+1E-3 (6-200)E-6	+1E-3(3) 3E-5(10)	
MANUAL VALVES	FAIL TO OPERATE FAIL TO REMAIN OPEN (PLUG)		+1E-3(3)		6.[illegible]-6		+1E-4(3)			1E-4(3)	
MOV'S	FAIL TO OPEN FAIL TO CLOSE SPURIOUS OPERATION ALL MODES		[+1E-3(3)]	+2E-3 +8E-4	4.5E-5		+1E-3(3)	{2.5E-6A} 5E-6	+(1-50)E-4 +(1-5)E-3 (3-100)E-7	+1E-3(3) +1E-3(3) 3.5E-6(10)	
SOLENOID VALVES	FAIL TO OPEN FAIL TO CLOSE SPURIOUS OPERATION FAIL TO OPERATE				+5E-4 +5E-4 9E-6		+1E-3(3)	3.5E-7	(2-6)E-6	+1E-3(3) +1E-3(3) 2.5E-6(3)	
AIR-FLUID VALVES	FAIL TO OPEN FAIL TO CLOSE SPURIOUS OPERATION ALL MODES	+1.5E-3 +1.3E-2			2E-5(3)		+3E-4(3)		+(2-500)E-5 +(2-50)E-5	+1.2E-5(2) +1.2E-5(2) 5.5E-6(2)	
VACUUM VALVES	FAIL TO OPEN				2.2E-6		+3E-5(3)		+7E-3	3E-5(3)	
RELIEF VALVES	FAIL TO OPEN FAIL TO CLOSE 10% LIGHT 10% HEAVY PREMATURE OPERATION		+1E-4(10)H +3E-2(3)H 1E-5(3)H	+(4-7)E-3 +(3-6)E-3 3E-6	1.4E-6 1.2E-6 8E-6 3E-6		+1E-5(3) 1E-5(3)	5E-6	+5E-4	+1.3E-3(10) +1.0E-2(10) 3.5E-6(3)	8.4E-6
PILOT RELIEF VLVS	FAIL TO OPERATE FAIL TO RESEAT				4.5E-6			+1E-2			
BELLOWS REL VLV	FAIL TO OPERATE				7E-6						
SAFETY VALVE	FAIL TO OPEN FAIL TO CLOSE SPURIOUS OPERATION	2.1E-7								+1E-5(3) +1E-4(3) 3.5E-6(3)	
CHECK VALVES	REVERSE LEAKAGE FAIL TO OPEN FAIL TO CLOSE SPURIOUS OPERATION FAIL TO OPERATE		3E-6(3/10) +1E-4(3)	5E-7 +5E-5 +2E-4	1.4E-5 8.4E-6 3.2E-5		+1E-4(3)	+1.2E-5	+5E-3 1E-5	2.9E-6(1.2) +1E-4(3)	1.6E-7 1.6E-6
PIPES >3"	RUPTURE ALL MODES		3E-10(100/30)/S		1.6E-10/S 2.4E-9/S	1.5E-5	1E-10(30)/S			1E-10(30)/S	1E-10
PIPES <3"	RUPTURE ALL MODES		3E-10(100/30)/S		1.6E-10/S 2.4E-9/S	7.7E-6	1E-9(30)/S		8E-7	1E-9(30)	1E-9
SCRAM RODS	FAIL TO SCRAM					1.9E-7	+1E-4(3)		+(1-25)E-6	+1E-4(3)	
ELECT. CLUTCH	FAIL TO OPERATE PREMATURE DISENGAGEMENT				1.8E-5		+3E-4(3) 1E-6(10)				
MECH. CLUTCH	FAIL TO OPERATE PREMATURE DISENGAGEMENT				6E-6		+3E-4(3)				
GASKETS	LEAKAGE				4.2E-6						
CONTROL ROD DRIVE FUNCTION						3.8E-6					

COMPONENT	FAIL MODE	(1) BIBLIS	(2) GENERAL ATOMIC	(3) LER EVALUATION PROGRAM	(4) NCSR	(5) NPRDS	(6) WASH-1400	(7) WESTINGHOUSE	(8) VOLTA	(9) PICKARD, LOWE AND GARRICK	(10) BECHTEL
BATTERY	ALL MODES				7E-6	3.5E-7				1.4E-6(9.8)	
	NO OUTPUT									1.4E-8(9.8)	
BATTERY SYSTEM	FAILURE ON DEMAND								+2E-6		
	FAILURE						3E-6(3)				
	ALL MODES		6E-6(3/3)/P		7E-6						
BATTERY CHARGER	ALL MODES					2.2E-6				1.5E-6(29.9)B	
DIESEL GENERATOR	FAIL TO START	+4.2E-3	+3E-2(2/10)	+2.2E-2	+3E-2		+3E-2(3)	3E-5	+(5-50)E-3	+3E-2(3)	+9.2E-3D
	FAIL TO RUN		3E-4(3)				3E-3(10)	3.3E-4	7E-4	3E-3(10)	1.3E-5D
	OVERALL FAILURE				1.3E-3						
CIRCUIT BREAKERS	FAIL TO OPEN				5.8E-7				+(1-100)E-4	+2.3E-4(8.9)	
	FAIL TO CLOSE				1.2E-6				+5E-6	+1.0E-6(10)	
	SPURIOUS OPERATION		1E-6(3)		1.2E-7		1E-6(3)		5E-6	4.3E-8(10)	
	FAIL TO OPERATE		+1E-3(3)		2.3E-6	7.5E-7	+1E-3(3)	+3E-6		+4E-4(8.9)	
RELAYS	FAIL TO OPERATE				9.1E-7	3.3E-7		2.7E-6	(2-5)E-6	+3.5E-6(4)	
	FAIL TO ENERGIZE				3.4E-7		+1E-4(3)				
	SPURIOUS OPERATION								(3-10)E-7	5.7E-8(35)	
MAN. SWITCHES	FAIL TO OPEN						{+1E-5(3)}			+1.5E-8(71.1)	
	FAIL TO CLOSE									+5.0E-9(67.1)	
	SPURIOUS OPERATION									6.6E-8(1.4)	
	FAIL TO OPERATE				3E-7						
TORQUE SWITCHES	FAIL TO OPERATE						+1E-4(3)				
PRESSURE SWITCH	FAIL TO OPERATE				3.5E-5		+1E-4(3)			2E-7(3.3)	
	PREMATURE OPERATION									9.4E-8(36.5)	
LIMIT SWITCHES	FAIL TO OPEN									+2.1E-6(1.9)	
	FAIL TO CLOSE									+6.2E-7(1.9)	
	SPURIOUS OPERATION									4.2E-6(1.9)	
	FAILURE TO OPERATE				2.5E-6		+3E-4(3)		+2E-5		
LIQ. LEV. SENSOR	FAIL TO OPERATE				3.5E-5			<4E-6	(5-10)E-6	4.4E-6(2.7)	
	PREMATURE OPERATION										
PRESS. SENSOR	FAIL TO OPERATE				3.4E-5			<6E-7	(5-20)E-6	1.7E-7(5.7)	
	OUT OF LIMITS		1E-5(10)								
TEMP. SENSOR	FAILURE				7.5E-5				(5-10)E-6	1.5E-6(5.8)	
	OUT OF LIMITS		3E-5(3)								

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NOTES.

1. LETTER SUFFIXES ON FAILURE RATES DENOTE THE FOLLOWING:
   A - UPPER 95% CONFIDENCE BOUND
   B - RATE FOR STATIC BATTERY CHARGER
   P - PER PLANT HOUR
   S - PER SECTION OF PIPE
   D - FOR SIZE CLASS 1750-2000 KW DIESEL-GENERATORS
   H - FAILURE DATA FOR HELIUM

2. THE NUMBER OR NUMBERS IN PARENTHESES FOLLOWING FAILURE RATES DENOTE THE RANGE FACTORS. FOR EXAMPLE (XX/YY) MEANS ONE SHOULD MULTIPLY THE MEDIAN VALUE BY XX TO OBTAIN THE UPPER 95% CONFIDENCE BOUND AND DIVIDE THE MEDIAN BY YY TO OBTAIN THE LOWER 5% CONFIDENCE BOUND. A SINGLE NUMBER IN PARENTHESES INDICATES THE RANGE FACTOR IS FOR BOTH THE UPPER AND LOWER BOUND.

3. A "+" preceeding a failure rate denotes failure-per-dema
   All other failure rates are failure-per-hour.

DEFINITION OF TERMS

Biblis - Biblis Nuclear Plant in Federal Republic of Germany
NCSR - Provided by the National Center of Systems Reliability - United Kingdom
Volta - Provided by Dr. Guiseppe Volta - Ispra
LER - Provided by Licensee Event Report Data Evaluation
GA - Provided by General Atomic Company
Pickard- Provided by Pickard, Lowe, and Garrick
MOVs - Motor Operated Valves
RVs - Relief Valves
DGs - Diesel Generators
CBs - Circuit Breakers

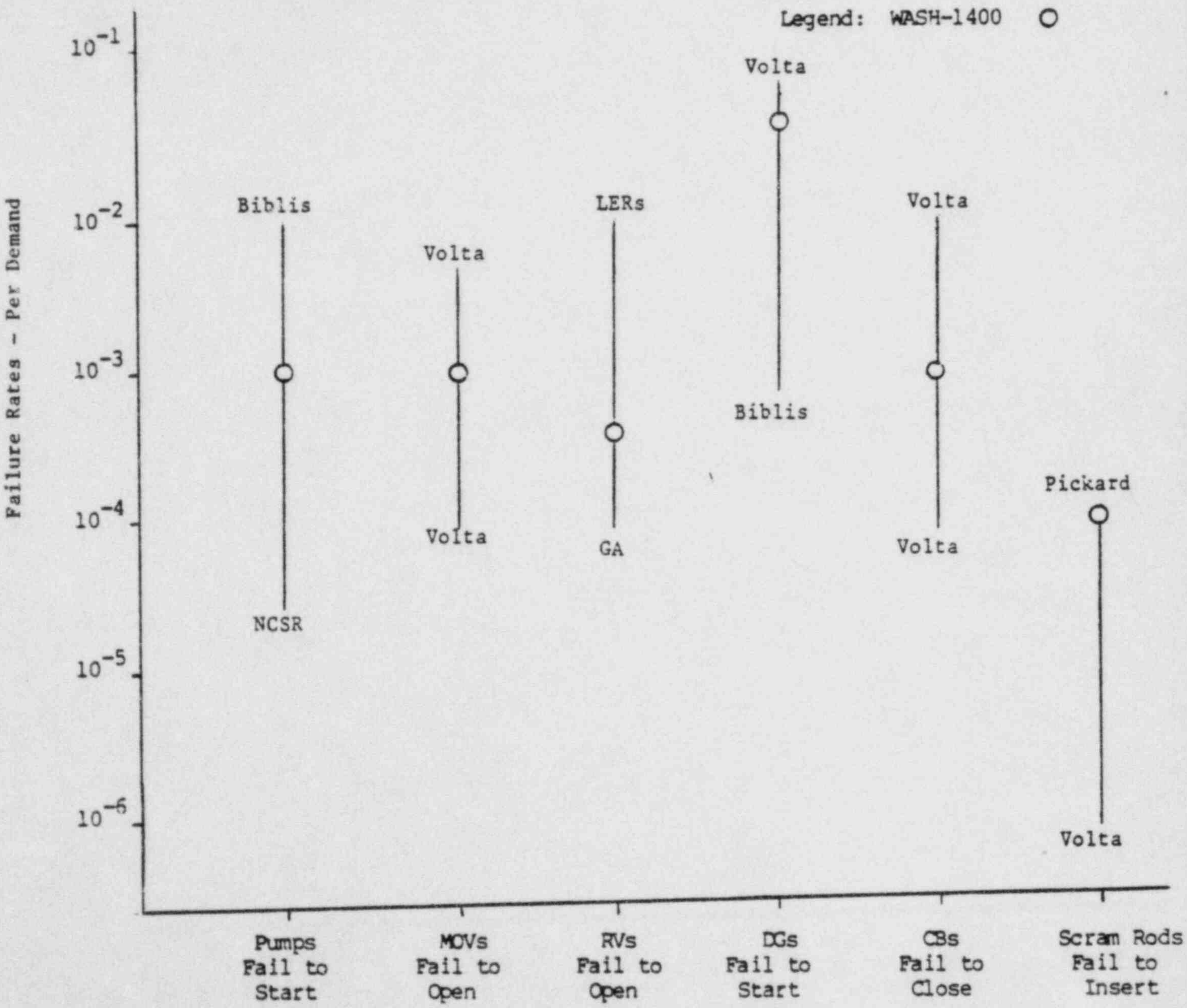

Figure 1. Data Point Estimate Extremes

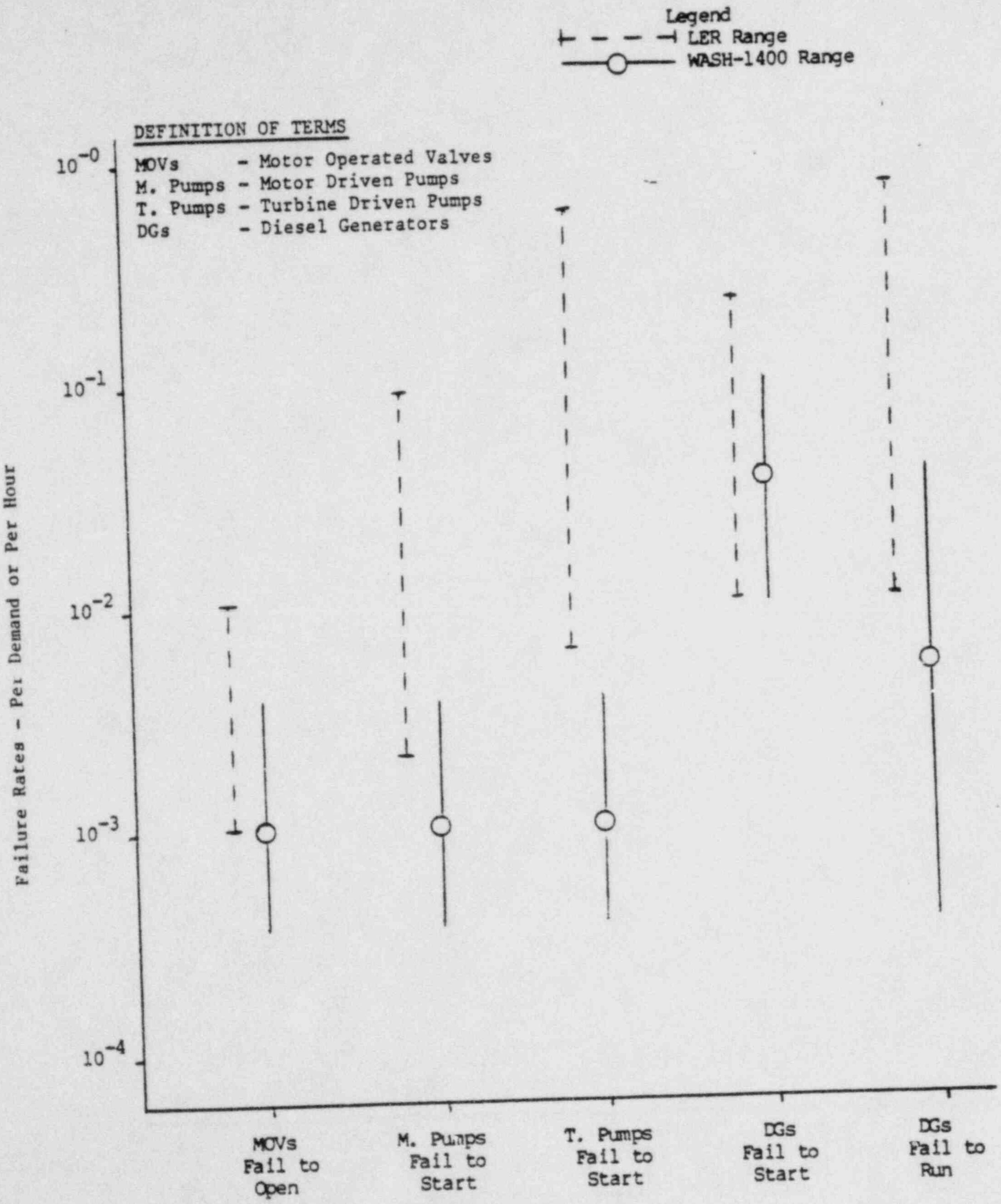

Figure 2. Plant to Plant Variation

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

February 7, 1980

David Okrent, Chairman, Subcommittee on Reliability and Probabilistic Assessment

ANALYSIS OF FEEDWATER TRANSIENT SEQUENCES IN B&W NUCLEAR STEAM SUPPLY SYSTEMS

To aid in the Subcommittee's work in formulating a response to Congressman Udall's letter of July 29, 1979, please find attached a draft of our analysis of the Three Mile Island, Rancho Seco, and Davis Besse events. Using the WASH-1400 event trees and data directly gives meaningless results because several important features of the sequences are omitted. Using an event tree which we constructed for B&W feedwater transients, and using WASH-1400 methodology and data, we obtain the following:

Rancho Seco	1.2×10^{-4}/B&W reactor year
Davis Besse	1.2×10^{-3}/B&W reactor year
Three Mile Island	1.5×10^{-4}/B&W reactor year

A major uncertainty is the characterization of operator behavior. It appears that with appropriate use of WASH-1400 methodology and data, events of this type would be anticipated.

The study will be distributed to all Subcommittee Members and appropriate consultants.

If you have any questions, please do not hesitate to call us.

Edward Abbott
Edward Abbott, ACRS Senior Fellow

John H. Bickel
John Bickel, ACRS Fellow

William Kastenberg
William Kastenberg, ACRS Senior Fellow

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ANALYSIS OF FEEDWATER TRANSIENT SEQUENCES IN B&W NUCLEAR STEAM SUPPLY SYSTEMS

E. Abbott, J. Bickel & W. E. Kastenberg
ACRS Fellows

I. INTRODUCTION

This study uses event tree analysis, and existing WASH-1400 methodology and data to determine various sequence probabilities for three different events which have occurred in plants with a B&W Nuclear Steam Supply System. The events evaluated are the March 29, 1979 Accident at Three Mile Island (TMI), the March 30, 1978 Loss of Instrument Power Transient at Rancho Seco (RS) and the September 24, 1977 Depressurization Transient at Davis Besse (DB). The sequence of events at RS and DB are given in Appendix A. The events are generically classified as loss of main feedwater. The TMI and DB events are similar in that the sequence of events (i.e., the separate plant and operator actions) are comparable up to the point of the operator manually blocking the power operated relief valve (PORV). The RS event is similar only in that the initiating event resulted in a loss of main feedwater. The plant and operator actions, however, are different from TMI and DB.

In the first part of this memo, a heuristic analysis of feedwater transients in B&W plants prior to TMI is given. This is followed by an analysis using the data, event trees and sequences contained in WASH-1400 for the S2 small break LOCA (break diameter ≤ 2") and for the T-transient.* It must be recognized, however, that WASH-1400 utilzes event sequences characteristic of the Westinghouse Nuclear Steam Supply System and its associated protective and engineered safeguard systems. In the last part of the study, we develop a feedwater transient

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* A glossary of abbreviations is given in Table I (page 3).

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event tree sequence unique to B&W plants valid prior to April 1979. This tree is applicable to B&W plants where the PORV is designed to lift prior to RPS trip during a feedwater transient.

TABLE I

GLOSSARY OF TERMS

AFWS - Auxiliary Feedwater System

CHRS - Containment Heat Removal System

CSIS - Containment Spray Injection System

CSRS - Containment Spray Recirculation System

CVCS - Chemical Volume Control System

ECI - Emergency Coolant Injection

ECR - Emergency Coolant Recirculation

EP - Electric Power

DB - Davis Besse

ICS - Integrated Control System

HPIS - High Pressue Injection System

LOCA - Loss of Coolant Accident

NNI-Y - non-nuclear instrumentation power bus Y. (power supply for instruments not associated with the determining of the fission rate in the core)

PCS - Power Conversion System

PORV - Power (or pilot) operated relief valve

Psi - Pounds per square inch

P_X - probability of failure for system X. (e.g., P_k = probability the RPS system fails to insert the reactor's control rods)

PWR - Pressurized Water Reactors

RCS - Reactor Coolant System

RHRS - Residual Heat Removal System

RPS - Reactor Protective System

RS - Rancho Seco

S2 - small break LOCA event tree of WASH-1400 for a PWR

SFRCS - Steam Feedwater Rupture Control System

SHA - Sodium Hydroxide Addition

SR - Safety Relief

SSR - Secondary Steam Relief

T - Transient Event Tree of WASH-1400 for a PWR

TE - Transient Event

TMI - Three Mile Island

VO - Valve Opens

VR - Valve Recloses

WASH-1400 - The Reactor Safety Study NUREG-75/014.

II. HEURISTIC ANALYSIS OF B&W FEEDWATER TRANSIENTS

As stated above, the sequence of events at Davis Besse (DB) and Rancho Seco (RS) are given in Appendix A. The Three Mile Island (TMI) accident is similar to the DB transient up to the last event where the stuck open PORV is isolated at DB but not at TMI. As discussed later in this development, the time frames are however, somewhat different.

Examination of the sequences given in Appendix A yields the following heuristic analysis:

1. The events for TMI and DB are determined by: a) the frequency of feedwater transients in PWRs $\sim$ 3 per reactor year, b) the fact that in B&W plants prior to April 1979, a feedwater transient causes the PORV to open independent of AFWS operation, and c) failure of the PORV to close (3×10^{-2} per demand). Hence this family of transients would be initiated on the order of 9×10^{-2} per reactor year.

2. The eventual outcome of this sequence depends upon a) whether or not the PORV is gagged at the time of transient initiation (50% of the time it is), b) operator action in not interrupting the HPIS, and c) isolating the PORV if it fails to close.

3. For DB the PORV was not gagged, the operator interrupted the HPIS and did isolate the PORV. In order to estimate the frequency of the outcome, the probability of these three events must be obtained. A telephone survey of B&W plants by the authors revealed that the PORV is gagged 50% of the

time. The operator action is more difficult to obtain. WASH-1400 (Appendix III) states that the probability of operator failure under stress is:

0.9 - 5 minutes after a large LOCA

0.1 - 30 minutes after a large LOCA

0.01 - several hours later

The average error rate, in a high stress situation is given as 0.2 to 0.3.

In addition, if P is the probability of operator error, and the number of people present is n, then P^n is given as the probability of a collective error. In practice, the final decision rests with the shift supervisor so that n can vary between 1 and 3 depending on his influence. (See Appendix B)

One problem (among others) in using this data is that it is not clear that the operator made an error in defeating the HPIS. That is, the procedure followed called for interruption of HPIS with high level indicated in the pressurizer. In that case, it may have been the procedure that was in error, and the operators failed to recognize it.

Using a probability of 0.5 for the chance of a gagged PORV, $(0.3)^3 = 0.027$ for defeating the HPIS after several minutes, and using $1-(0.1)^3 = .999$ for successfully blocking the PORV at 20 minutes yields a frequency for DB

$$DB = (9\text{x}10^{-2})(0.5)(0.027)(0.999) = 1.2\text{x}10^{-3}$$

4. At TMI, the PORV was not gagged, the operator interrupted the HPIS and the PORV was not isolated. Since the decay heat load was greater at TMI than DB, the failure to block the PORV occurred sooner. The operator

should have recognized that the PORV had stuck open by the time the quench tank rupture disk blew (about 15 minutes into the transient). This yields an estimate of the error probability of $(.5)^3$. Hence at TMI

$$TMI = (9x10^{-2})\ (0.5)(0.027)(.125) = 1.5x10^{-4}$$

5. For Rancho Seco (RS), the init. ating event (loss of non-nuclear instrumentation) was estimated to be $8.6x.0^{-3}$ per reactor year*. Since this loss initiated the feedwater transient, this value is used, rather than the 3 per reactor year used for DB and TMI.

Since the PORV was gagged (0.5), the operators throttled the HPIS (0.027) and the code safety valves opened and closed as required (1.0), the frequency of this event is estimated as

$$RS = (8.6x10^{-3})(0.5)(0.027) = 1.2x10^{-4}$$

In the next section, an attempt is made to map these events on the WASH-1400 event trees.

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*/ Because of the difficulty in estimating the specific failure of the non-nuclear instrumentation (NNI-Y) power supply in the absence of a detailed fault tree analysis, the failure rate for low power, solid state devices was used. It should be noted that the final result is very sensitive to this failure rate and should be viewed as representing the family of NNI failures.

III. WASH-1400 EVENT TREES

In this section, we have attempted to trace the Davis-Besse (DB), Rancho-Seco (RS) and Three Mile Island (TMI) events on the WASH-1400 Transient (T) and Small Break LOCA (S2) event trees shown in Figures 1 and 2. Mapping the sequences occurring at DB and RS on the WASH-1400 T tree without any modification yields sequence TM, which does not result in core melt, and was subsequently omitted from the dominant risk sequences in WASH-1400. Mapping TMI on the T tree yields: (a) sequence TMLQU if no credit is given for the return of the Auxiliary Feedwater System (AFWS) or TMU if credit is given for AFWS. Both paths do not give credit for actuation of the High Pressure Injection System (HPIS). With HPIS actuation, the corresponding paths are TM and TMLQ (See Figure 1). Several problems arise when trying to evaluate these events in terms of this event tree. For the DB and RS events, sequence TM does not differentiate between the failure of the PORV to close at DB and the initially gagged PORV at RS. Second, the sequence is for all transient initiated events and hence does not identify the initial loss of non-nuclear instrumentation (power bus NNI-Y) induced by human action which resulted in the feedwater transient and in the loss of indicators during the transient at RS. Lastly, for DB and TMI, the tree fails to include the fact that the PORV will lift regardless of the availability of the auxillary feedwater supply in B&W plants, and, therefore, neglects the possibility that the PORV fails to close.

For the DB and RS events, the frequency of sequence TM for all feedwater transients would be given by:

$$P_{TM} = P_T\ (1-P_K)\ P_M\ (1-P_Q)\ (1-P_U)\ (1-P_W).$$

Based on WASH 1400 data, P_T = 3 feedwater transients per reactor year, $P_M = 1$ (failure to recover the main feedwater system within minutes) and assuming $(1-P_i) = 1$ we obtain

$P_{TM} = 3$ per reactor year.

For TMI, the appropriate sequence (taking into account the return of the AFWS) is TMU with

$$P_{TMU} = P_T\ (1-P_K)\ P_M\ (1-P_Q)\ P_U$$

Hence $P_{TMU} = 3 \times P_U$ per reactor year where P_U is the unavailability of the HPIS. Since HPIS was available, but the operators interrupted its operation, P_U is chosen as $(0.3)^3$ which is in the range of WASH-1400 numbers for operator error. Hence for this sequence

$$P_{TMU} = 8.1 \times 10^{-2} \text{ per reactor year.}$$

Again, this tree neglects failure of the PORV to close.

In WASH-1400, it is suggested that transients, for which the PORV fails to close, should be treated as a small break LOCA, and the event tree S2 be used (Figure 2). Since the LOCA is terminated at both DB and RS, (the PORV is finally blocked at DB and the code safety valve reseats at RS), these events become sequence S_2 with a frequency of 3 per year.

Mapping the TMI event on the small break LOCA tree yields sequence S_2D. The initiating frequency S2 is given by

$$S2 = 3 \text{ feedwater transients/year} \times 10^{-2} \text{ failure to close/demand *}$$

$$= 3 \times 10^{-2} \text{ S2 events/yr.}$$

Using a HPIS unavailability of $(0.3)^3$ due to operator error, TMI becomes

$$P_{TMI} = 8.1 \times 10^{-4}\text{/year}$$

Failure to block the PORV is not included in the tree and the PORV failure to close on demand number comes from Appendix V, page V-38 of WASH-1400.

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* WASH-1400 states this number has an error factor of 10.

For the particular feedwater transient at Rancho Seco, the probability of loss of non-nuclear instrumentation (which led to loss of feedwater) and the probability that the loss was attributable to human error should be obtained.

Data from WASH-1400 on loss of non-nuclear instrumentation is about 8.6 x 10^{-3}/reactor year. Hence the Rancho Seco initiating event may be on the order of 8.6 x 10^{-3}/reactor year.

IV. APPLICATION OF A B&W EVENT TREE TO TMI, DB AND RS

A unique event tree was developed for feedwater transients in B&W plants which is different from those used in WASH-1400. The differences between the WASH-1400 - PWR and the B&W PWR were described in Section III.

The sequence of events at TMI is well known and not presented here. The events follow along sequence #5 on the attached event tree and are self-explanatory (Figure 3). The sequence of events for Davis Besse follows sequence #6 on the event tree. The sequence of events for Rancho Seco follows sequence #14 on the event tree.

The probabilities and failure rate data shown below were obtained from WASH-1400 except for those marked with * and **. The uncertainty in P_Q, and P_Q were also obtained from B&W data. The uncertainty in the other probabilities are difficult to obtain because they depend on human errors, operating procedures, etc., and have not been ascertained. Hence, the final results could have large error bounds.

The probabilities for the significant events in the event tree are:

P_T - 3 per reactor year (WASH-1400, Appendix V, pg. V-34)

* $P_P = .5$

** $P_{Q'} = 3 \times 10_{-2}$ ($\pm 1 \times 10^{-2}$)

$P_Q = 3 \times 10^{-2}$ ($\pm 1 \times 10^{-2}$)

$P_{U'} = (.3)^3$ (WASH-1400, Appendix III, page III-60)

$P_{Q''} = (.5)^3$ " " " " " " (for TMI)

$P_{Q''} = (.1)^3$ " " " " " " (for DB)

For TMI the probability is as follows:

$P_{TMI} = P_T \times P_U \times (P_Q) \times (P_{U'}) \times (P_{Q''})$

$= 3 \times .5 \times 3 \times 10^{-2} \times (.3)^3 (.5)^3$

$= 1.5 \times 10^{-4}$/year

For DB the probability is as follows:

$P_{DB} = P_T \times P_P \times P_Q \times (P_{U'}) \times (1-P_{Q''})$

$= 3 \times .5 \times (3 \times 10^{-2}) \times (0.3) \times (1-(.1)^3)$

$= 1.2 \times 10^{-3}$/year

For the Rancho Seco event, the probability of the loss of an instrument bus leading to a feedwater transient must be used for P_T. Using WASH-1400 data, the failure rate of low power solid state devices is:

1×10^{-6}/hr or 8.6×10^{-3} per year.

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* The P_P value was obtained from a telephone survey of B&W plants and their estimate of the frequency of defeating the PORV by blocking or gagging.

** Obtained from B&W

The probability of the RS family of events is then estimated as

$$P_{RS} = P_{NNI} \times P_P \times P_U,$$
$$= 8.6 \times 10^{-3} \times .5 \times (.3)^3$$
$$= 1.2 \times 10^{-4} \text{ per reactor year.}$$

These results are summarized as follows:.

TABLE II.

	WASH-1400		B&W
	T	S_2	Feedwater Transient
TMI	8.1×10^{-2}	8.1×10^{-4}	1.5×10^{-4}
DB	3	*	1.2×10^{-3}
RS	8.6×10^{-3}	*	1.2×10^{-4}

It is important to recognize that the largest uncertainty is in characterization of operator action. WASH-1400 states that if P is the probability of operator error, then P^n is the probability of error if the number of personnel in the control room is n. Because of the supervisory nature of the shift supervisor, the probability may be between P and P^n. This report uses .3 for HPIS unavailability as an average for the initial one-half hour for all three sequences. Failure to block the PORV is given a probability at .5 at fifteen minutes and .1 at thirty minutes. This report does not evaluate in detail the resultant error in the calculations because of a lack of data on operator action. The values chosen are considered to be within the ranges of WASH-1400, and consistent with the methodology.

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*Does not apply.

V. CONCLUSIONS

After mapping the TMI, DB and RS events on the WASH-1400 Transient and Small Break LOCA trees, constructing an event tree for B&W Feedwater Transients, and employing the WASH-1400 data, the following is concluded:

1. As shown in Table II, the values obtained from a B&W transient tree differ from those obtained from the T and S_2 event trees in WASH-1400 because the latter trees do not include the necessary features as discussed above.

   As noted in Section II, the WASH-1400 event trees cannot be used since the PORV lifts during a feedwater transient. This clearly shows that the strict use of these event trees to other PWRs yield erroneous results. This should be obvious because the trees in WASH-1400 are unique to the Surry Plant which is a Westinghouse PWR.

   The values obtained above could have been obtained prior to the event sequences discussed because the data, knowledge of the transients and methodology were known. The only requirement to complete a similar study would have been development of a unique event tree for B&W plants.

2. The consequences of these sequences of events depend upon the exposure history of the core. At DB, the plant was operating at low power with fresh fuel. At TMI, the plant was operating at full power well into the fuel cycle. The time allowed to block the PORV and for re-initiating HPSI before the core is uncovered was different in each case. These time differences are reflected in the characterization of operator action.

3. The NRC will construct event and fault trees for individual plants under the Integrated Reliability Evaluation Program (IREP). The individual licensees, however, could easily perform similar studies using available failure rate data and developing a unique event tree for their respective plants. This would immediately focus upon needed areas of improvement in operations and provide an independent check to IREP.

APPENDIX A

Sequence of Events

The sequence of events for Davis Besse is:

T - A spurious initiation of Steam Feedwater Rupture Control System (SFRCS) isolates the steam generators and starts the auxiliary feedwater pumps.

P - The pressure rise in the primary system causes the Power Operated Relief Valve (PORV) to open.

K - The control room operator manually trips the reactor because the pressurizer level is outside (high) of the operating range.

L - Both auxiliary feedwater pumps start but only one feeds a generator due to binding in the throttle linkage in the other pump's turbine control system.

P;Q- Code safety valves do not lift as the PORV is relieving reactor coolant pressure.

Q - The PORV "simmers" due to a missing relay in the closing circuit and after nine cycles it sticks open.

U - Safety Features Actuation System (SFAS) initiation on low RCS pressure starts the HPI pumps.

U'- The operator cycles the HPI pumps to maintain pressurizer level.

Q"- The operators recognize that the PORV is stuck open and shut the block valve.

The sequence of events for Rancho Seco is:

T - The loss of one of the two non-nuclear instrumentation fuses (NNI-Y) causes the Integrated Control System (ICS) to sense a loss of BTU output and isolates the feedwater system.

P - The primary system pressure rise would have caused the PORV to open but it was gagged shut.

K - The reactor trips on high RCS pressure.

L - The operator manually initiates main feedwater after realizing the NNI-Y failure has blocked the initiation of the auxiliary feedwater system (the auxiliary feedwater pumps initiates automatically on SFAS actuation later on in the transient.)

P - The increased RCS pressure causes one of the two code safety valves to open at a pressure less than maximum setpoint of 2500 psi. The subsequent decrease in RCS pressure causes a SFAS initiation (HPI and AFWS start).

Q'- The power safety valves reseat.

U'- NNI-Y is restored. The operators recognize an excessive cooldown ($> 100^{\circ}$ F/hr) has resulted. They throttle HPI and auxiliary feed flow to reduce rate of cooldown.

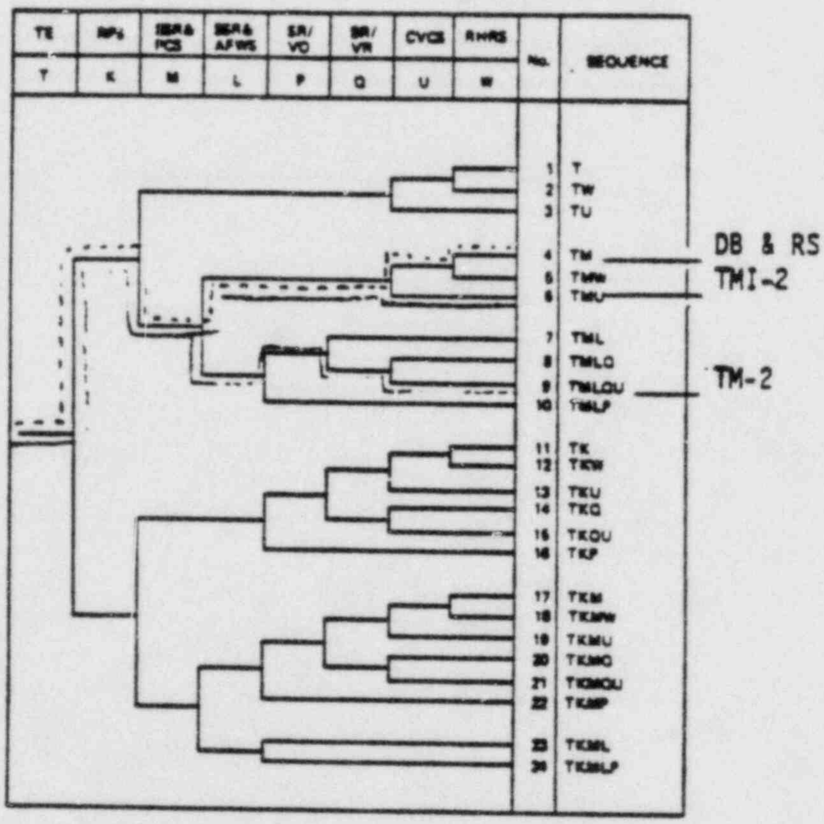

FIGURE I 4-14 PWR Transient Event Tree

FIGURE 1

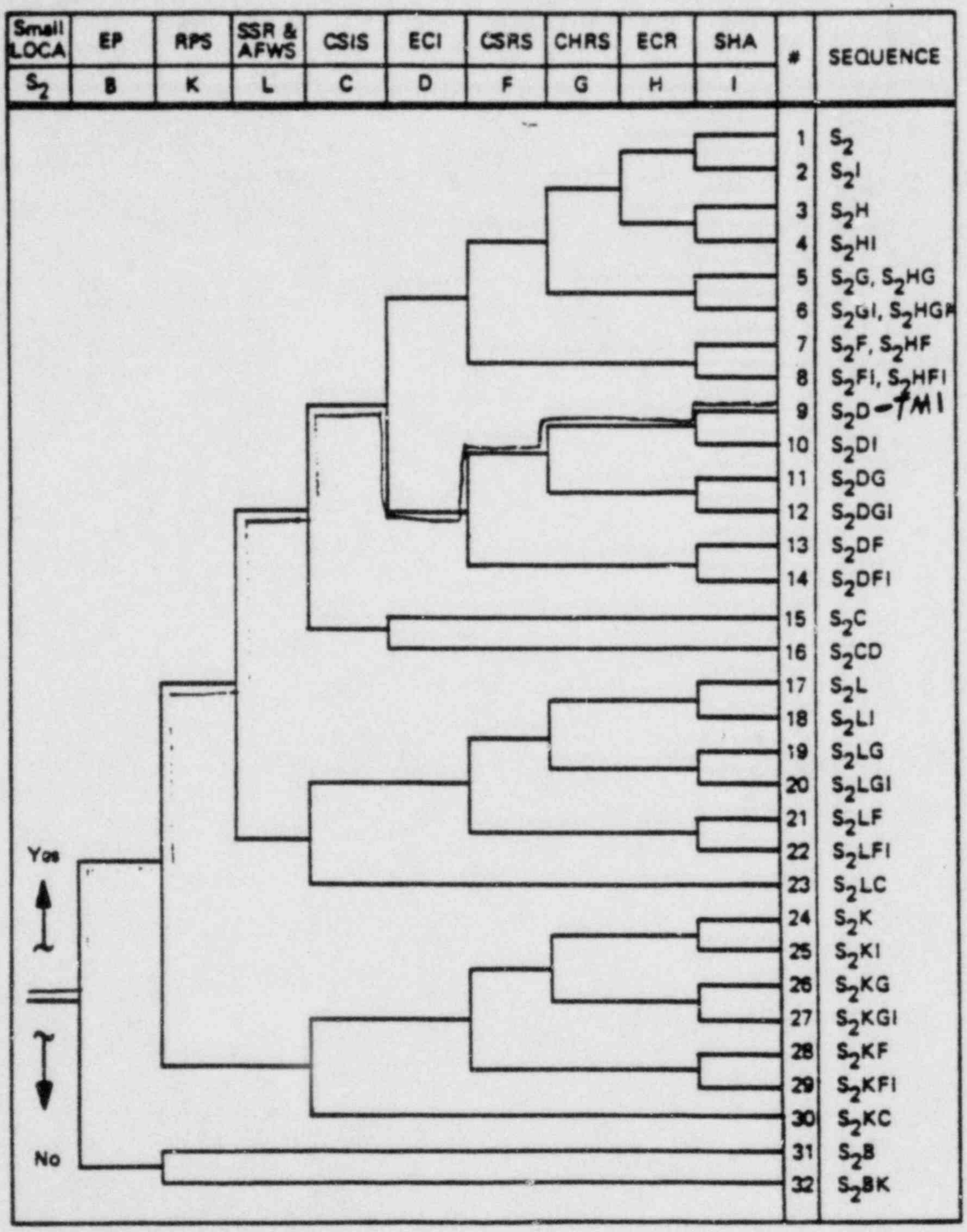

FIGURE I 4-4 PWR Small LOCA (S2, 1/2-2 inch diameter) in RCS

FIGURE 2

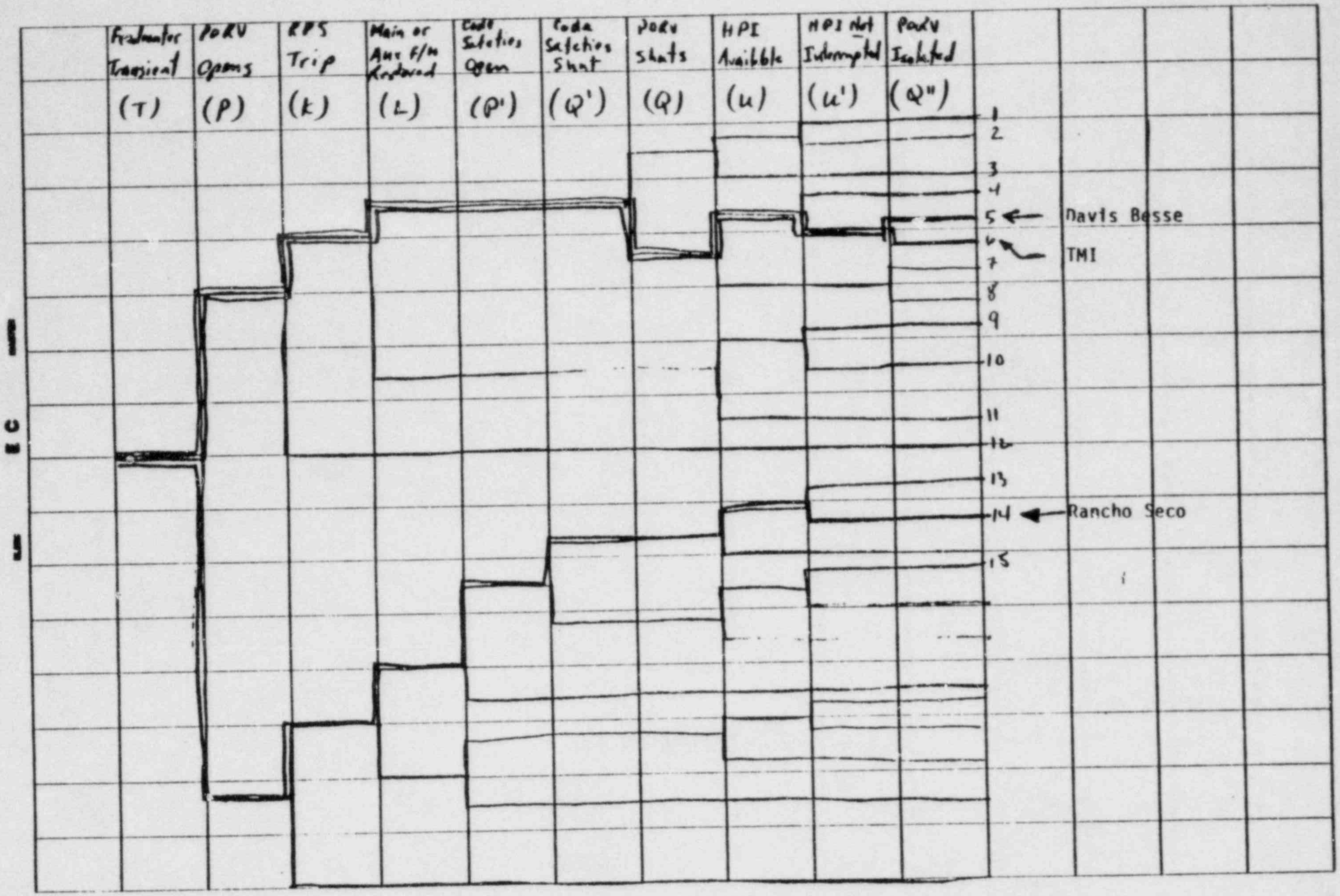

FIGURE 3. B&W FEEDWATER TRANSIENT EVENT TREE.

APPENDIX B
OPERATOR ERROR

The rationale for characterization of operator error in WASH-1400 can be demonstrated as follows. Let p_f be the probability of operator failure and let p_s be the probability of operator success. Then

$$p_s + p_f = 1 \tag{1}$$

as it should. Suppose there are n operators in the control room. Let P_f be the probability the n operators make a "collective" error. In WASH-1400, P_f is given by

$$P_f = (p_f)^n \tag{2}$$

Since probability must be conserved, the probability that the n operators make a "collective" success, denoted P_s is

$$P_s = P_f = 1-(p_f)^n \tag{3}$$

To understand the implications of such an approach consider the following:

let $p_f = 0.1$ (individual failure), n=3.

It follows that:

$p_s = 1-0.1 = 0.900$ (individual success)

$P_f = (0.1)^3 = 0.001$ (collective failure)

$P_s = 1-(0.1)^3 = 0.999$ (collective success)

The possible operator actions are:

$p_f\ p_f\ p_f = (0.1)^3 = .001$

$p_f\ p_f\ p_s = (0.1)^2\ (0.9) = .009$

$p_f\ p_s\ p_f = (0.1)(0.9)(0.1) = .009$

$p_f\ p_s\ p_s = (0.1)(0.9)^2 = .081$

$p_s\ p_f\ p_f = (0.8)(0.1)^2 = .009$

$p_s\ p_s\ p_f = (0.9)^2\ (0.1) = .081$

$p_s\ p_f\ p_s = (0.9)(0.1)(0.9) = .081$

$p_s\ p_s\ p_s = (0.9)^3 = .729$

1.000

Hence, WASH-1400 can be interpreted as follows:

a) For a "collective" failure, all n operators must be in error.

b) For a "collective" success, at least <u>one</u> operator must take correct action.

With this interpretation, $P_s \neq p_s^n$ i.e. all operators are correct.

As stated in the report, the shift supervisor should have the final word ... however, to be consistent with the WASH-1400 approach

$$P_f = p_f^n \text{ and } P_s = 1-(p_f)^n$$

is used, with the interpretation given above.

Nuclear Regulatory Commission Staff Report, "Evaluation of Davis-Besse and Rancho Seco Feedwater Transients on 9/24/77 and 3/20/78 Using WASH-1400 Data"

Attachment B

A-500

A. INTRODUCTION

In this report we have evaluated the Loss of Main Feedwater transients which occurred at Davis-Besse-1 on 9/24/77 and at Rancho Seco on 3/20/78 and compared them with the accident at Three Mile Island-2 on 3/29/79. A summary is provided of the Davis-Besse and Rancho Seco events. The behavior of important safety systems is compared. An event tree for Loss of Main Feedwater transients is provided, and each transient sequence is identified in the context of the event tree, WASH-1400 data.

Certain caveats should be made. First, WASH-1400 was performed for the Westinghouse-designed Surry plant, not a B&W reactor. We have not done the kind of major in-depth analysis here that was done for WASH-1400. Such an analysis would require considerable effort and funds. Second, it should be recognized that there are significant uncertainties in the WASH-1400 data. Third, the evaluation refers to pre-TMI system behavior and transients.

B. DISCUSSION OF DAVIS-BESSE TRANSIENT

1. Event Summary - Davis-Besse

On September 24, 1977 a series of events occurred at the Davis-Besse Unit 1 which resulted in depressurization of the primary system from a normal operating pressure of 2150 psi to 900 psi in approximately eight minutes, and the release of approximately 11,000 gallons of water in the form of steam within the containment through the pressurizer quench tank rupture disc.

On the afternoon of Saturday, September 24, 1977 the main turbine was shut down to repair a leak in a pressure sensing connection on a steam line from the turbine governing valves to the turbine inlet. The reactor was being held critical at approximately 9% thermal power.

At 2134 hours, a spurious half trip occurred in the Steam Feedwater Rupture Control System (SFRCS). This caused the startup feedwater valve on the No. 2 steam generator (which is the normal feed path at this power level) to close. Closure of this valve resulted in a low No. 2 steam generator level, which then resulted in a normal full trip of the SFRCS for this condition and initiation of the SFRCS. SFRCS initiation closes both main steam isolation valves and initiates feed-water flow to both steam generators from their individual steam-driven auxiliary feedpumps.

The half trip and resulting full trip of the SFRCS caused a reduction in heat removal from the primary system and a corresponding temperature/pressure rise in the primary system. The pressure rise in the primary system caused the pressurizer power relief valve to lift. This valve then rapidly oscillated closed-to-open approximately nine times and remained in the full open position. The chattering of the relief valve was caused by the physical absence of a relay in the valve control logic circuitry. The relay normally provides for a deadband between "open" and "close" setpoints. An empty relay socket was found in the logic cabinet after the event.

The temperature rise in the primary system caused an increase in the pressurizer level, and the operator manually tripped the reactor on high pressurizer level approximately two minutes after the half trip on the SFRCS occurred.

The pressurizer power relief valve, in the full open position, rapidly reduced the primary system pressure, and a Safety Features Actuation

System (SFAS) trip occurred at the 1600 psi setpoint of the primary system. The power relief valve discharge goes to the pressurizer quench tank, which became overloaded and overpressurized, and approximately 4 1/2 minutes after reactor trip the rupture disc in this tank relieved due to overpressure, venting the steam into the containment. Approximately 20 minutes after reactor trip, the operators diagnosed the reason for the primary system depressurization as being the power relief valve, and from the control room closed the motorized block valve ahead of the power relief valve, terminating the loss of primary coolant into the containment.

Subsequent operator action using makeup pumps and high pressure injection pumps stabilized the primary system pressure and pressurizer level and a controlled shutdown to cold shutdown conditions followed.

The major physical damage from the incident was to the reflective metal insulation on the lower part of the No. 2 steam generator, which received the jet of steam coming from the pressurizer quench tank. A ventilating duct in the area of the quench tank was dimpled and required straightening. Twenty-three panels of reflective metal insulation required replacement. Entry into the containment was made at 0550 Sunday, September 25, 1977 for cleanup operations.

Another event occurred in the course of this incident that did not contribute materially to the above events, but did result in the No. 2 steam generator going dry. This was the failure of the No. 2 auxiliary feedpump to come up to full speed (3600 rpm) following the SFRCS trip. This feedpump came up to approximately 2600 rpm and stayed at this level with no flow to the steam generator until approximately 12 minutes after reactor trip, when the operators placed its control in manual and brought it up to full speed (commencing feedwater flow to the steam generator).

2. Key Systems Behavior - Davis-Besse

An important fact to bear in mind while discussing the Davis-Besse transient of 9/24/77 is that only one full-power day of operation had been accumulated at the time of the event (see Table 1). This means that considerably less decay heat was being generated in the core than was the case at TMI-2. In addition, the Davis-Besse reactor was only at 9% power when the main feedwater was lost. A high pressure reactor trip did not occur (it did at TMI in 9 seconds), confirming the slower, milder nature of the Davis-Besse transient.

Operator reaction to the transient was effective. Although the pressurizer level increased off-scale in the first ten minutes, the operators apparently realized the pressurizer level increase was misleading and caused by steam formation in the primary system. However, the operators did turn off the HPI pumps (just as at TMI) after only three minutes of operation.

The pressurizer relief valve stuck open early in the transient. The operators diagnosed this problem and closed the block valve after 21 minutes into the transient. At TMI a similar problem took 138 minutes to diagnose. The ability to diagnose and take remedial action in 21 minutes helped to terminate the Davis-Besse transient with a minimum of damage.

3. Event Tree Evaluation - Davis-Besse

The events at Davis-Besse on 9/24/77 can be depicted in an event tree (Figure 1). The Davis-Besse transient is #2 on the event tree. This may be compared with sequence #3 which is the TMI-2 sequence. The event tree is for a category of transients which begin with a loss of all main feedwater (TM). In the case of Davis-Besse, this was apparently initiated by a faulty input buffer in the logic control of the Steam Feedwater Rupture Control System.

WASH-1400 estimated three of these feedwater transients to occur per year at each reactor. In the 12 months prior to the TMI-2 accident, the average number of feedwater transients at B&W reactors was three per year (see Table 2), confirming the WASH-1400 value. It should be noted that a larger number of feedwater transients occur in the first few years of operation, and a smaller number after that. Perhaps 2 to 3 times this number might be appropriate for early operation. Plants which have operated longer than a few years may average 1 to 2 feedwater transients per year.

Within about ten seconds after the main feedwater system had tripped, increasing reactor pressure caused the pressurizer relief valve to open. This valve then failed to close, causing a small LOCA. The WASH-1400 failure rate estimated for this failure mode was 1×10^{-2} per demand with a factor 10 uncertainty up and down. More recent data in light of the TMI-2 accident indicate three relief vale failures in this mode in about 150 demands, or a failure rate (to reclose) of $\sim 2 \times 10^{-2}$ per demand, again confirming the WASH-1400 failure rate.

At the same time that the relief valve was opening in the primary system, the auxiliary feedwater system was being aligned to the steam generators and auxiliary feedwater flow had commenced successfully shortly thereafter. About 30 seconds later, the operator tripped the reactor manually because of rising pressurizer level.

Reactor pressure did not reach the setpoint of the pressurizer safety valves and they were not called on to open. The ECCS system automatically actuated on low pressure (1600 psi) in the High Pressure Injection (HPI) mode about 1 1/2 minutes after the pressurizer relief valve stuck open. After the HPI system operated successfully for about three minutes, the operator manually terminated HPI. Because of the nature of the transient, this was regarded as successful operation of ECCS. The probability of this category of transient occurring in a B&W reactor, as predicted using WASH-1400 failure data, is estimated as follows:

3	x	1×10^{-2}	=	3×10^{-2} per reactor year
Loss of Main Feedwater/yr.		Relief Valve Fails to Close		

C. DISCUSSION OF RANCHO SECO TRANSIENT

1. Event Summary - Rancho Seco

On March 20, 1978 an excessive cooldown transient was experienced while operating at 70% power (IE Report 50-132). Non-nuclear instruments were lost including steam generator and pressurizer levels and all RCS temperatures. Loss of RCS hot leg temperature input to the ICS caused termination of feedwater flow. Reduced heat removal in the steam generators caused RCS temperature and pressure to increase. The reactor tripped on high RCS pressure followed by a turbine trip. The secondary sides of both steam generators emptied due to operation of condenser bypass valves, atmospheric dump valves and auxiliary steam loads. Although normal control room indications were lost, the computer typewriter will print alarms when setpoints are reached. In addition, selected plant parameters can be monitored on the ICS computer printout. With the aid of computer indication, pressurizer level was maintained by manual operation of a high-pressure injection pump. "A" steam generator level control initiated emergency feedwater injection (level control was actually lost at time zero, but the channel drifted slowly downward while "B" channel drifted slowly upward). The turbine-driven auxiliary feedwater pump had started on loss of feedwater flow.

RCS cooldown started as a result of emergency feedwater flow to "A" steam generator and possibly main feedwater pump flow (manually operated). Decreasing RCS pressure (1600 psig) actuated HPI pumps and the motor-driven auxiliary feedwater pump. Full auxiliary feedwater was initiated to both steam generators. The RCS reached a minimum of 1475 psig and was then increased and maintained at 2000 psig by manual control of an HPI pump.

Restoration of the non-nuclear instrumentation restored all lost indications and controls. Operating personnel secured the auxiliary feedwater pumps and started RCS pressure reduction using the pressurizer spray.

2. Key Systems Behavior - Rancho Seco

The incident at Rancho Seco on March 20, 1978 involved a loss of main feedwater due to operator-induced failure in the ICS non-nuclear instrumentation. The incident was aggravated by the fact that (1) the plant ICS reacted to erroneous instrument readings causing delays in initiating AFW injection and subsequently allowing excessive AFW injection, and (2) the operators had a very limited number of instrument readings which they could trust to manually bring the plant to an orderly shutdown. Since the reactor was at 70% power and had logged considerable operating time (3 1/2 years of commercial operation), the decay heat to be removed was significant, similar to TMI-2.

Auxiliary feedwater was not available for seven minutes after MFW trip. However, this delay was not as serious as at TMI-2 because there was no small LOCA in progress; i.e., a pressurizer safety valve had opened and closed properly.

The transient was eventually brought under control by the operators' diagnosis of which electrical circuit breakers had opened, and then closing them.

3. <u>Event Tree Evaluation - Rancho Seco</u>

The Reactor Safety Study (RSS) stated that on the average a plant can expect about three main feedwater losses of a few minutes duration per year. This value was obtained from the operating experience available at the time the RSS was in progress. The <u>nature</u> of the three main feedwater losses per year was not discussed in great detail. Therefore, the breakdown of the various causes of feedwater transients (such as the Rancho Seco incident) in quantitative terms is not provided in the RSS.

The NRC has investigated feedwater transients at B&W plants and has reported this information in NUREG-0560. At least five of the main feedwater losses attributable to ICS-related failures or malfunctions were identified in that document. Among these is the Rancho Seco incident. There were many other main feedwater losses which licensees felt were not significant enough to be reportable. It is not known how many of these were ICS or non-nuclear instrumentation failure related. The average failure rate of main feedwater for B&W plants subsequent to RSS was reconfirmed at three per year.

The RSS identified several potential transient-initiating events which are associated with the loss of feedwater. Among those identified were the loss of main feedwater pumps and malfunction of control, loss of condensate pumps, loss of A.C. power to the feedwater system, and others. The probability of occurrence of any one specific initiating event may be small. However, when assembled into appropriate categories, the net probability of a given type of transient may be considerable. In this regard, the probability of the event at Rancho Seco is a small part of the larger probability that the main feedwater system will be lost.

This transient may be classified as belonging to sequence #1 on the event tree shown in Figure 1. However, this ICS/NNI initiated transient could have been more severe than it was. That is, the loss of NNI which resulted in erroneous instrument readings delayed the automatic injection of AFW; perhaps even more significant, operator information on the status of the plant was severely limited throughout the transient. The erroneous instrument readings eventually "drifted" to the point of AFW injection some seven minutes into the transient even though the steam generator was apparently dried out by the end of the first minute. It appears that the capability existed at all times for manual action to initiate AFW injection. If erroneous instrument readings or manual actions had never initiated AFW injection, this event would have followed the path of sequence 10 in Figure 1.

Another sequence of significance for this initiating event is sequence #3. If a pressurizer relief valve had become stuck open, this event could have been worse than the TMI-2 sequence, depending on operator actions, because of the additional problem of a lack of instrument readings. However, the specific initiating event, ICS/NNI failure or malfunction, may be somewhat less likely than main feedwater losses due to other causes. Using WASH-1400 data, the overall sequence #1 would have a probability of occurrence of three times per year per plant; the specific (and potentially more severe) case where the loss of NNI is the cause is expected to be a much smaller subset of this category.

TABLE 1

COMPARISON OF THREE B&W REACTOR INCIDENT EVENT SEQUENCES

	TMI-2 (3/29/79)	DAVIS BESSE (9/24/77)	RANCHO SECO (3/20/78)
REACTOR POWER	97%	9%	70%
REACTOR HISTORY	IN COMMERCIAL OPERATION THREE MONTHS.	~1 FULL POWER DAY OF OPERATION.	IN COMMERCIAL OPERATION 3 1/2 YEARS.
TURBINE	TRIPPED IMMEDIATELY.	DOWN ALREADY.	TRIPPED AFTER 5".
REACTOR TRIP	AUTOMATIC AFTER 8" ON HI REACTOR PRESSURE (2355 PSI).	MANUAL (1 MIN. 47") BECAUSE OF RISING PRESSURIZER LEVEL.	AUTOMATIC AFTER 5" ON HI REACTOR PRESSURE.
MFW	BOTH PUMPS TRIP IMMEDIATELY.	1 PUMP TRIP IMMEDIATELY 1 PUMP TRIP 58" LATER.	REDUCED TO ZERO FLOW BY FAULTY ICS SIGNAL (SOME MFW INITIATION BY OPERATOR PROBABLE AFTER 7 MIN.).

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TABLE 1 (CONT.)

	TMI-2 (3/29/79)	DAVIS BESSE (9/24/77)	RANCHO SECO (3/20/78)
AFW	NO AFW FOR 8 MIN.	1 PUMP/SG WORKING WITHIN 46". 1 PUMP "UNAVAILABLE" (TURBINE DEGRADED). AVAILABLE MANUALLY AFTER 12 MIN.	NO AFW FOR 7 MIN.
PRESSURIZER RELIEF VALVE	OPENED AFTER 3" AND STUCK OPEN. BLOCK VALVE CLOSED AFTER 138 MIN.	OPENED AFTER 1 MIN. 6", CYCLED RAPIDLY 9 TIMES IN 23" AND STUCK OPEN (STEM GALLING). BLOCK VALVE CLOSED IN 20 MIN.	GAGGED CLOSED. SRV OPENED AND CLOSED PROPERLY
PRESSURIZER	SEVERELY MISLEADING LEVEL INDICATION.	LEVEL INCREASED OFF SCALE.	NO LEVEL PROBLEM.

TABLE 1 (CONT.)

	TMI-2 (3/29/79)	DAVIS BESSE (9/24/77)	RANCHO SECO (3/20/73)
ECCS	HPI AUTOSTARTED (1600 PSI) AT 2'02". 1 PUMP TRIPPED AFTER RUNNING 2 MIN. 36". OTHER PUMP THROTTLED TO MINIMUM FLOW.	HPI AUTOSTARTED (1600 PSI) AT 2 MIN. 57" AND PERMITTED TO RUN FOR 3 MIN. 5". MANUAL SHUTDOWN BECAUSE PRESSURIZER LEVEL NORMAL.	HPI MANUAL AND INTERMITTENT DURING FIRST 13 MIN. THEN AUTOSTART (1600 PSI)
INSTRUMENTS	MOST O.K.	O.K.	ONLY PRESSURIZER LEVEL AND RCS PRESSURE TRUSTED BY OPERATORS DURING FIRST 75 MIN.

TABLE 2

WASH-1400 FAILURE RATES

	FAILURE RATE
1. MAIN FEEDWATER (TM)	3/YR
2. REACTOR TRIP (K)	3.6×10^{-5}/D*
3. AUXILIARY FEEDWATER (L)	3.7×10^{-5}/D*
4. PRESSURIZER RELIEF VALVE OPENS (P_1)	1×10^{-2}/D
5. SAFETY VALVES OPEN (P_2)	3×10^{-5}/D
6. PRESSURIZER RELIEF VALVE CLOSES (Q_1)	1×10^{-2}/D
7. SAFETY VALVES CLOSE (Q_2)	1×10^{-2}/D
8. ECCS - HI PRESSURE INJECTION (C)	3.7×10^{-3}*
9. ECCS DEGRADED OPERATION (C^1)	$>3.7\times10^{-3}$*

*ANALYSIS UNIQUE TO SURRY

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Memorandum From F. Rowsome to R. Fraley,
"ACRS Query on Material Relevant to Udall
Letter: Davis-Besse and Rancho Seco Transients,"
February 12, 1980

Attachment C

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

February 12, 1980

MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Frank H. Rowsome, Deputy Director
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research

SUBJECT: ACRS QUERY ON MATERIAL RELEVANT TO UDALL LETTER:
DAVIS BESSE AND RANCHO SECO INCIDENTS

The following question was posed by Congressman Udall's letter of July 27, 1979:

"Please determine the probabilities of occurrence that, prior to the events, would have been predicted on the basis of WASH-1400 failure rates and methodology as to the probabilities of the sequences of events that occurred at Davis Besse on September 24, 1977 and at Rancho Seco on March 20, 1978."

Needless to say, the predictive probability for a particular historical event can have any value between one and zero depending upon the breadth of the class of events that is taken to represent it. In most cases, a few classifications appear to be "natural" in the sense that "vertabrates" are a natural and distinct grouping of animals. However, there are commonly several levels of event resolution at which one might consider the problem, analogous to the heirarchy of biological classifications: kingdom, phylum, ... , species.

I shall attempt to address Congressman Udall's question using the level of event sequence resolution most natural to WASH-1400, while attempting to sketch answers to several more useful questions, such as:

- Did WASH-1400 consider or predict accidents of this type?
- Could WASH-1400 methods have alerted analysts to the possibility of such accidents if the methods had been applied to the affected plants?
- What improvements in WASH-1400 methods or data are needed to properly consider such sequences in risk assessment?
- Can WASH-1400 methods serve a useful function in analyzing actual experiences?

The Davis-Besse incident, the Rancho Seco incident, and the accident at TMI all entailed feedwater transients, i.e., cessation in the normal delivery of feedwater to the steam generators. The Reactor Safety Study estimated that feedwater transients can be expected to occur between once a year and ten

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times a year at each nuclear plant. The best estimate in WASH-1400 is three feedwater transients per reactor year. There were roughly 30 reactor years of experience accumulated at B&W reactor plants as of March 28, 1979, the date of the accident at Three Mile Island. WASH-1400 would have lead us to expect between 30 and 300 feedwater transients, most likely about 100 feedwater transients at B&W plants up to that time. In fact, there were about 150 feedwater transients at B&W plants, in good agreement with WASH-1400 failure rate data.

In two of the incidents, the September 24, 1977 incident at Davis Besse and the accident at Three Mile Island, the pressurizer relief valve opened and failed to close, giving rise to a small loss-of-coolant accident (LOCA). WASH-1400 identified this possibility and estimated that the probability that a pressurizer relief valve, having once opened, would fail to close at somewhere between .001 and .10, with .01 (a one percent chance) as the most likely value. On the other hand, the pressurizer relief valve opens only very rarely during feedwater transients at Westinghouse plants, the kind studied in WASH-1400. Therefore, the Reactor Safety Study did not predict a high expected frequency for failed-open pressurizer relief valves initiated by feedwater transients. Had a WASH-1400 type analysis been performed for a B&W plant and had the authors recognized that almost all feedwater transients cause the opening of this valve in B&W plants (before the TMI-inspired changes), then the analysis would have predicted between zero and five (most likely one) occurrences of a stuck open pressurizer relief valve following a feedwater transient in the 30 B&W reactor years. In fact, there were two: Davis Besse on September 24, 1977 and Three Mile Island on March 28, 1979.

The Reactor Safety Study (RSS) did not attempt to distinguish by probability the many types of faults that can give rise to feedwater transients. These were lumped together in one broad category. However, the RSS did acknowledge that some of the failure mechanisms that can trigger a feedwater transient might also compromise the reliability of the systems called upon to respond to the feedwater transient. One example of such common-cause failures was found to be important to the risk in WASH-1400; it is the loss of all AC power at the station. The failure mechanisms responsible for the March 20, 1978 incident at Rancho Seco was a failure of the "Non-Nuclear Instrumentation" DC power supplies. It is also a common-cause failure that both triggered the feedwater transient and also compromised the reliability of the backup auxiliary feedwater system.

Although this class of common mode failures was described and one example was found to be important in WASH-1400, nothing quite like this scenario was found for Surry in WASH-1400. The Surry plant does not depend upon non-safety grade equipment for the autostart of its auxiliary feedwater system. Therefore, Surry is immune to the class of accidents in which non-safety grade instrument power supply failure trips main feedwater and defeats the normal autostart of emergency feedwater.

At Rancho Seco the failure of the autostart of auxiliary feedwater (AFW) was not regarded as a principal cause for concern emerging from the incident, although the risk assessment perspective suggests that it should have been high among the warning flags raised by the event.

It should be noted that the auxiliary feedwater pumps were started at the outset and that their discharge control values did receive two "open" commands. The first of these occurred when one of the faulted steam generator level signals happened to drift into the range triggering AFW delivery. The second occurred after the overcooling commenced in response to the ECCS actuation signal. Thus, neither of these signals could be counted upon to mitigate the initiating event.

In the event that WASH-1400 methods had been applied to Rancho Seco, it is unlikely that the specifics of the short circuit and fuse failure would have been considered that led to the NNI-Y power supply failure. However, it is reasonable to expect that such a study would have identified the dependency of the auxiliary feedwater autostart system upon the Integrated Control System, and the dependence of both the ICS and the instruments upon the NNI buses.

In summary, the RSS did identify events of the broad class represented by the DB and TMI incidents: feedwater transients with stuck open pressurizer relief valves. The RSS did identify the class and some examples of common mode failures that cause a feedwater trip and degrade the reliability of the auxiliary feedwater system, as at Rancho Seco, but it did not and could not have been expected to predict the right frequency of occurrence for these classes of accidents at B&W plants. A risk assessment of B&W plants might reasonably have been expected to have identified the high susceptibility to transient-induced LOCA intrinsic in the B&W design - the frequent challenge of the pressurizer relief valve that lead to the Davis Besse and TMI accidents. Had the risk assessment been coupled with a careful review and adequacy assessment for operator emergency procedures, the susceptibility of plants to accidents such as TMI or the Rancho Seco incident could have been foretold.

Risk assessment methods also provide a useful framework for organizing the "what if" questions surrounding an actual, historical incident. Application of these techniques can be used to help identify the safety significance of operating occurrences.

Frank H. Rowsome

Frank H. Rowsome, Deputy Director
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research

COMMITTEE ON INTERIOR AND INSULAR AFFAIRS
U.S. HOUSE OF REPRESENTATIVES
WASHINGTON, D.C. RECEIVED

July 27 1979

Dr. Max Carbon, Chairman
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Carbon:

I understand that the ACRS is nearing completion of its examination of Licensee Event Reports. I would hope that the report of this inquiry would address the questions of the consistency of actual component failure experience with that projected in WASH-1400; e.g. whether valve failure experience approximates the failure rates used in the WASH-1400 calculations, etc. In addition, please determine the probabilities of occurrence that, prior to the events, would have been predicted on the basis of WASH-1400 failure rates and methodology as to the probabilities of the sequences of events that occurred at Davis-Besse on September 24, 1977 and at Rancho Seco on March 20, 1978.

Thank you for your assistance.

Sincerely,

Mo Udall

MORRIS K. UDALL
Chairman

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Appendix XXV

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

February 11, 1980

APPENDIX XXV
COMMISSION ADOPTION OF PARTS OF NUREG-0660, DRAFT 2

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: NUREG-0660 DRAFT 2, "ACTION PLANS FOR IMPLEMENTING RECOMMENDATIONS OF THE PRESIDENT'S COMMISSION AND OTHER STUDIES OF THE TMI-2 ACCIDENT"

Dear Dr. Ahearne:

On February 7, 1980, during its 238th meeting, the ACRS received additional information from Messrs. Denton and Mattson on the status of the Action Plans and the requirements for near term operating licenses (NTOL). The Committee was advised that a large number of NTOL items, including the TMI-2 related NRC Bulletins and Orders, had been approved as a minimal set earlier that day.

The ACRS believes that its input into this process has been largely ignored by the Commission and is concerned that the "rush to judgment" on those important matters may result in, at worst, error, and at best inefficient use of resources important to safety.

During its January 1980 meeting, the ACRS had received a briefing on the Draft Action Plans (following a subcommittee meeting on the same subject) and sent you a letter, noting the lack of priorities within the Plans and the lack of an adequate method to establish such priorities. We further stated that we expected to see and to review the Plans when this had been accomplished.

In view of our letter, the ACRS was surprised to learn that the Staff had requested, and the Commission had approved, a large set of NTOL items without ACRS comment, while an ACRS meeting was in progress. While the Committee recognizes the needs and pressures for action, we believe it is important to be sure that a reasonable rationale exists for the setting of priorities, that there is reasonable assurance that there are no adverse safety effects from new requirements, and that the limitations on total resources have been carefully factored into the decision making.

A principal concern is that a very large number of operational and hardware changes are being mandated with, in most cases, little analysis to establish their safety relevance or impact. Design and operational stability is itself a safety asset and, confident though we are in the engineering judgment of the Staff, we think that there would be merit in ACRS review before, not after adoption.

The ACRS will not be ready to provide its advice on the recommendations of the Bulletins and Orders Task Force until it can hold an additional Subcommittee meeting which will include a discussion of questions that have been raised by reactor vendors and operators.

Messrs. Denton and Mattson also stated on February 7 that they were not sure whether the ACRS would be asked to comment on the final Action Plans before the Commission was asked for its approval. The NRC Staff schedule for the availability of Draft 3 of the Action Plans is not firm. The ACRS is planning to meet with the NRC Staff on the Action Plans at its March meeting if the Committee receives Draft 3 in time. However, there appears to be the element of a timing problem which the Commission must consider in deciding whether, how, and when ACRS input in the decision-making process will be obtained.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

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Appendix XXVI

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

February 14, 1980

APPENDIX XXVI
REPORT ON NUREG-0625 REPORT OF SITING POLICY TASK FORCE

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: NUREG-0625, "REPORT OF THE SITING POLICY TASK FORCE"

Dear Dr. Ahearne:

The purpose of this letter is to provide you with ACRS comments on the "Report of the Siting Policy Task Force" (NUREG-0625). In preparing these comments, the Committee had the benefit of discussions with the NRC Staff at a Subcommittee meeting on October 17, 1979 and at the full Committee meeting on January 10-12, 1980.

Siting Goals

In the abstract of the Report it is stated that a number of changes in siting policy have been recommended in order to accomplish the following goals:

1. To strengthen siting as a factor in defense in depth by establishing requirements for site approval that are independent of plant design considerations.

2. To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and distribution criteria.

3. To require that sites selected will minimize the risk from energy generation.

In connection with the third goal, the Siting Policy Task Force states that, "The selected sites should be among the best available in the region where new generating capacity is needed. Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large regions of the country. This is because energy generation from any source has its associated risk, with risks from some energy sources being greater than that of the nuclear option."

The ACRS agrees with these goals but has some qualifications which are stated below. It is interesting to note that these goals are, in part, similar and are in part complementary to some siting policy recommendations made by the ACRS in a report* to the Atomic Energy Commission prior to the adoption of 10 CFR Part 100. In that report the ACRS stated the following:

1) Everyone off-site must have a reasonably good chance of not being seriously hurt if an unlikely but credible reactor accident should occur.

2) The exposure of a large segment of society in terms of integrated man-rems should not be such as to cause a significant shortening of the average individual lifetime or a significant genetic damage or a significant increase in leukemia - should a credible reactor accident occur.

3) There should be an advantage to society resulting from locating a plant at the proposed site rather than in a more isolated area.

4) Even if the most serious accident possible (not normally considered credible) should occur, the numbers of people killed should not be catastrophic.

However, the AEC Part 100 Siting Criteria were written so as to provide greater flexibility in the choice of sites than was implicit in these ACRS recommendations and permitted the substitution of engineered safety features for distance. In the decade following adoption of Part 100 in 1962, sites were accepted having surrounding population densities less than or roughly equivalent to that typified by Indian Point Unit 1 which had been approved in 1956. Although the engineered safety features provided in nuclear plants were judged to be sufficient to restrict estimated offsite doses to the specified limits, these estimates were based on the stylized calculations of Part 100 which assumes a large fission product release to an intact containment. Historically, with regard to the engineering design requirements for nuclear power plants located on sites near the borderline of acceptability, the ACRS has recommended additional measures to prevent accidents and to mitigate their effects. In recent years, sites approved for nuclear power plants have had surrounding population densities substantially less than those of Indian Point Unit 1.

With regard to the goals discussed above, the ACRS agrees that siting, as a factor in the defense in depth philosophy, should be strengthened. However, the ACRS believes that any minimum requirements for parameters such as the exclusion zone radius, surrounding population density, or distance from population centers should be established, if possible, within the framework of an overall Nuclear Regulatory Commission safety philosophy for future reactors.

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* ACRS letter to the Honorable John A. McCone, Chairman, USAEC dated October 22, 1960, Subject: REACTOR SITE CRITERIA

Such a philosophy should be based on preestablished Commission objectives for acceptable risk both to individuals and society. This will, of necessity, include consideration of matters such as the potential effects of a broad spectrum of reactor accidents, the identification of an ALARA criterion for the reduction of risk from accidents, and a general statement of policy concerning the objectives to be sought in reactor design with regard to the prevention and the mitigation of accidents.

The establishment of demographic-related site criteria will inevitably require a considerable amount of judgment. However, the choice will be less arbitrary if made within the framework of an overall NRC safety policy. The ACRS believes that an overall NRC safety philosophy is also needed in connection with the third objective of the Task Force, namely that of selecting sites to minimize the risk from the utilization of electricity generating sources.

The ACRS believes that well-founded nuclear power plant siting policy and practice are a national as well as a regional need. The Committee suggests that as part of a broad approach to LWR siting, the NRC should explore the possible development of a nationwide program to identify a bank of near-optimal sites regionally distributed for various types of energy-generating plants. By combining considerations of acceptable risk, the risks from various energy sources, and the national needs for energy, together with other relevant factors, a better long-term basis for determining appropriate criteria for LWR siting should be possible. In the absence of such a broad approach, the ACRS recommends that changes to past siting policy be interim in nature and be designed primarily to provide an acceptable basis for near-term decision making.

Task Force Recommendations

The Siting Policy Task Force has made nine recommendations, each of which is followed by a discussion which elaborates on the recommendation, frequently suggesting specific parameters and occasionally a significant additional recommendation. In this report the ACRS will deal primarily with the recommendations themselves, unless otherwise stated.

Recommendation 1

This is the principal recommendation of the Report. It proposes that Part 100 be revised to change the way in which protection is provided for accidents. The recommendation is very general in form and requires the addition of specifics to be meaningful.

Part 1 of the recommendation proposes the specification of a fixed minimum exclusion distance based on limiting the individual risk from design basis accidents. The ACRS believes that the specification of a minimum exclusion distance should include consideration of the risk from all accidents, not just design basis accidents. It should include consideration of the number of reactors at the site. Any long-term criterion concerning a minimum exclusion distance would best be established within the framework of a general NRC policy on LWR safety. Interim guidance could be determined with the benefit of information developed from NRC Staff studies and information submitted during a proposed rulemaking on interim changes in the site criteria.

Part 2 of the recommendation proposes a fixed minimum emergency planning distance of ten miles. The ACRS generally supports this recommendation with the understanding that appropriate attention would be given to potential problems at greater distances.

Part 3 recommends the incorporation of specific population density and distribution limits that are dependent on the average population of the region. The ACRS believes the wording of this recommendation is vague and it could be interpreted to be excessively restrictive or very permissive with regard to demographic requirements. Additional information is needed to establish interim criteria of this sort within the context of an NRC rule. Among the factors which require consideration are the following:

(a) If some regions of the country are permitted to employ higher maximum population densities, should there be any additional requirements for such plants in design, operation, or emergency planning? If not, what basis will be provided for designating regionally dependent acceptable risks?

(b) Should the NRC place a similar or a substantially greater emphasis on improbable, large accidents in its siting (and design) requirements than is utilized for other new societal activities posing hazards similar in magnitude and probability?

(c) How should the effectiveness of emergency measures, such as evacuation, sheltering and decontamination, be ascertained and factored into a judgment concerning minimum exclusion and emergency planning distances?

(d) Should meteorology not be given consideration in regard to the development of siting criteria?

Part 4 recommends removal of the requirement to calculate radiation doses as a means of establishing minimum exclusion distances and low population zones. The ACRS agrees with the Task Force that the approach used for the past two decades has not provided enough emphasis on site isolation. The Committee believes that the emphasis on engineered safety features to meet Part 100 for the postulated accident without direct consideration of other, more serious possibilities has led to a less-than-optimum approach to safety. However, if the recommendation of Part 4 is adopted, some alternative means of determining the need and adequacy of engineered safety features will be required.

In summary, although the ACRS agrees that the specification of minimum exclusion and emergency planning distances and population density and distribution limits is a commendable objective, and that interim criteria should be developed, the Committee believes that the adequacy of such parameters will depend on the safety related design and operational requirements and on the effectiveness of emergency measures. Also, the ACRS believes the establishment of such parameters involves the assumption of some accepted band of risk which should be specified. While the ACRS is not opposed to removal of the Part 100 requirement for calculation of radiation doses or to the specification of regionally dependent acceptable population densities, the Committee believes these matters need in-depth evaluation.

Recommendation 2

This recommendation proposes minimum standoff distances for potential hazards posed by man-made activities and natural characteristics. The Committee believes that such a recommendation is appropriate but the list is incomplete. For example, LNG terminals are included but not LPG. Similarly, hazardous cargo on rivers is not mentioned.

In addition, the proposed approach lacks an adequate rationale for specific numbers suggested. A distance of at least 12.5 miles from all capable faults, with no distinction as to fault size, is proposed, as is a specification that no reactor sites located on a flood plain should be closer than five miles downstream of a major dam. The reason why either of these two proposed numbers is suitable is not clear to the ACRS. For example, dams many miles away could be equally or more dangerous to a nuclear plant; on the other hand, small capable faults nearer than 12.5 miles might not pose significant design problems.

It is noted that the recommendation does not provide standoff distances between nuclear plants. The potential adverse influence of one plant on its neighbors in the event of a serious accident requires consideration in design.

Recommendation 3

This recommendation would change Part 100 to require reasonable assurance that interdictive measures are possible to limit groundwater contamination resulting from Class 9 accidents. The ACRS supports the recommendation. However, the Committee notes that the current wording is subject to a range of interpretations which could include, for example, the necessity for developing interdictive measures for particulate fallout or rainout that could result in groundwater contamination. The Committee recommends that the wording of the recommendation be made more explicit.

Recommendation 4

This recommendation is very general, merely stating that Appendix A to 10 CFR 100 should be revised to better reflect the evolving technology in assessing seismic hazards.

However, in the discussion section, the Task Force recommends that specific guidance be removed from Appendix A and placed in Regulatory Guides.

The ACRS agrees that the NRC criteria for seismic siting should be revised and perhaps expanded. This clearly will require changes in Appendix A. The ACRS believes that Regulatory Guides can be used to provide increased guidance on the interpretation and application of the criteria.

The ACRS has in the past worked closely with the NRC Staff on the development of seismic siting criteria, and expects to continue to do so in the future and to provide comments on the specific changes as they are developed and proposed. At this time, however, the ACRS cannot agree that all specific guidance can be removed from the criteria, in the absence of a quantitative safety goal.

Recommendation 5

This recommendation relates to post-licensing changes in offsite activities but does not specify what population/time period would be used. For example, would it be the present population, that at the projected end of life of the plant, or an average over the time period during which the plant will be operated? This should be clarified. The recommendation also does not specify what is considered to be a "significant increase in risk." Another consideration that might be taken into account is the nature and use of the land surrounding a site. Whether neighboring land is used for residential or industrial purposes, and whether it is fertile land or a desert, could also be important.

Recommendation 6

This recommendation pertains to methods for compensating for unfavorable site characteristics. The Committee suggests that the phrase, "unfavorable characteristics requiring unique or unusual design," be clarified. Many characteristics that are "unfavorable" can be readily compensated for by design, including some of an "unusual" nature. Design features to provide permanent site improvements should be permissible when suitably reliable. Perhaps these problems could be solved by deleting the word, "unfavorable," and substituting the word, "unproven," for "unique or unusual".

Recommendation 7

This recommendation relates to the timing of site reviews. The ACRS suggests that this recommendation could be improved by substituting the word "decision" for "approach" (in the third line).

Recommendation 8

This recommendation relates to the role of a state agency in approving a site for a nuclear power plant. The ACRS has no comments on this item.

Recommendation 9

This recommendation is to develop common bases for comparing the risks from all external events. The ACRS supports the general concept and would, if practical, extend it to internal events as well. The Committee believes that this concept represents a good long range goal; however, recognizing the complexity of the task, the Committee recommends that priority be given to those areas thought either to introduce the greatest risk or to provide the best opportunities for improvements in safety.

The Committee will be pleased to discuss the above items with you if you desire. In the meantime, we trust these comments will be helpful to you and the NRC Staff.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Appendix XXVII

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

February 13, 1980

APPENDIX XXVII
REPORT ON ACCEPTANCE CRITERIA FOR MARK I
CONTAINMENT LONG-TERM PROGRAM

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: NRC ACCEPTANCE CRITERIA FOR THE MARK I CONTAINMENT LONG TERM PROGRAM

Dear Dr. Ahearne:

During its 238th meeting, February 7-9, 1980, the Advisory Committee on Reactor Safeguards reviewed the NRC Acceptance Criteria for the Mark I Containment Long Term Program. This matter was considered at ACRS Fluid Dynamics Subcommittee meetings held on May 23, 1978, November 28-30, 1978, September 13-14, 1979, and November 16, 1979. During its review, the Committee had the benefit of discussions with representatives of the NRC Staff and the Mark I Owners Group.

The NRC Acceptance Criteria for the Mark I Containment Long Term Program are intended to establish design basis loads that are appropriate for the anticipated life of each Mark I BWR facility and to restore the originally intended design safety margins to each Mark I containment system.

The Mark I program was initiated in 1975 in response to loss of coolant accident and safety relief valve (SRV) dynamic loads identified by the General Electric Company during the course of performing large scale testing for the Mark III pressure-suppression containment in 1972-1974. A period of reevaluation resulted in issuance of the Short Term Program Acceptance Criteria in December 1975 which established interim design bases for continued operation of the Mark I BWRs. The Acceptance Criteria for the Long Term Program have been developed from a program of small and full scale tests in two and three dimensional geometries.

The Mark I Owners submitted proposed loads in the "Mark I Containment Program Load Definition Report" in December 1978 and detailed the methods to be used in plant unique analyses in the "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide." Following review of the available information, the NRC Staff determined that certain changes and clarifications to the criteria proposed by the Mark I Owners were necessary. The NRC Staff technical requirements were delineated in the "NRC Acceptance Criteria for the Mark I Containment Long Term Program" issued in October 1979 and also in several additions to the acceptance criteria as discussed during the 238th ACRS meeting. The additions to the Acceptance Criteria were intended, in part, to alleviate some of the difficulties the Mark I Owners had in calculating credible structural responses to SRV actuations.

The Committee recognizes the thoroughness of the efforts taken by the NRC Staff and the Mark I Owners to resolve the generic Mark I issues and believes that the NRC Acceptance Criteria and additions, as proposed, provide a suitably conservative basis for performing the Long Term Mark I Containment structural response analyses. The Mark I Owners indicated that they continue to have significant difficulty in calculating credible structural responses to some SRV loads and they would like to continue to work with the Staff on a generic basis to resolve these difficulties. The NRC Staff would like to complete the generic Mark I program and resolve any remaining problems as they arise from the plant unique analyses. The Committee believes that the individual Mark I Owners can work with the Staff to resolve any additional difficulties that may arise from the plant unique analyses as modifications are being made to the containment structures.

The Committee believes that the Staff should assure the adequacy of the requirements for verification of the design, fabrication, and inservice inspection of the Mark I containment modifications and, in particular, the SRV discharge piping in the wetwell airspace. Further, in the interim period while the Mark I modifications are being performed, the Staff should investigate the potential for and consequences of a failure in the SRV discharge piping in the wetwell airspace for the existing designs. The Committee wishes to be kept informed on this matter.

The Committee believes that, with due consideration to the above items, the generic Mark I Long Term Program can be concluded and the modifications to the individual Mark I BWRs can be implemented on a reasonable schedule over the next 18 months.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:
1. General Electric Company, "Mark I Containment Program Load Definition Report," Revision 0, NEDO 21888, December 1978.
2. General Electric Company, "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide," NEDO 24583, December 1978.
3. U.S. Nuclear Regulatory Commission, "NRC Acceptance Criteria for the Mark I Containment Long Term Program," October 1979, and additions included in the February 8, 1980, transcript of the 238th ACRS Meeting.

Appendix XXVIII

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

February 13, 1980

APPENDIX XXVIII
QUALIFICATIONS OF RADIOACTIVE WASTE
SYSTEM OPERATING PERSONNEL

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: QUALIFICATIONS OF RADIOACTIVE WASTE SYSTEM OPERATING PERSONNEL

Dear Dr. Ahearne:

Traditionally, the radioactive waste (radwaste) facility at a commercial nuclear power plant receives less operational and maintenance attention than safety-related or power-generating equipment. This is due largely to the emphasis placed on the latter by the NRC and the utilities' management. During the preoperational and startup test phases of a plant, very little radioactive material is produced, and therefore, mistakes made in operation of radwaste systems are of minor significance. As the plant commences its commercial phase, however, the proper management of radwaste problems becomes more acute as the quantity of radioactive material to be processed increases.

Since plant organizations are developed prior to commercial operation, utilities rarely recognize or anticipate that future problems may occur. Furthermore, once the plant's organization is established, corporate management approvals of staffing changes must be obtained, which can be difficult, especially when additional personnel are requested.

Operation of the radwaste system at nuclear power plants is frequently assigned to personnel at the entry level. The requirements for such a position are normally a high school diploma and a passing grade on the radiation protection examination for plant workers. In general, such personnel are under the supervision of more experienced radwaste system operators. Depending on the operating staff turnover rate, however, the people who supervise the radwaste system may frequently serve in such a position for only a short period of time. Personnel turnover also frequently results in the responsibility for the radwaste system being assumed by a higher level supervisor who has many competing duties. A review of Licensee Event Reports has shown that many mistakes occur in radwaste systems, ranging from equipment damage to inadvertent radionuclide releases. There may well be a cause and effect relation here.

These problems have been addressed by a few utilities, but additional changes are needed to reduce the number of errors and to improve equipment reliability. The root cause of the radwaste operating problems appears to be the failure of utilities' management to recognize the unique operational problems of such systems.

The Committee believes that NRC evaluation of utilities' organizational arrangements should include consideration of the unique problems associated with the onsite management of radioactive waste and that this should be addressed both at the corporate and plant staff level.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Appendix XXIX

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

February 11, 1980

APPENDIX XXIX
LOW PRESSURE TURBINE DISK CRACKING

William J. Dircks, Acting Executive Director for Operations

SUBJECT: LOW PRESSURE TURBINE DISC CRACKING

Recently, numerous reports have been received of cracks in Westinghouse low pressure turbine disc assemblies. These cracks have been found in the keyway and bore regions of the disc. In some cases these cracks are approaching the critical flaw size as calculated using fracture mechanics methods. This concern is aggravated by the fact that many of these plants have their tur[illegible]nted in a tangential manner where a postulated turbine fa[illegible]ould result in missile strikes on the containment.

The ACRS recommends that the NRC Staff reevaluate the problems associated with turbine failures with regard to the probability and consequences of turbine missiles and their potential to damage the containment building and other safety related structures, equipment, and systems. Please let me know when the Staff can be ready to discuss this matter so that we may schedule a meeting with the Committee.

Raymond F. Fraley

Raymond F. Fraley
Executive Director

Appendix XXX

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS MEMBERS' USE

1. H. R 6390, A bill to amend the Atomic Energy Act of 1954.

2. Ltr, R. H. Bucholz, Mgr, BWR Licensing Systems, General Electric Co, to D . F. Ross, NRC, Bulletins and Orders Safety Evaluation Report Draft Recommendations for BWR's dtd Feb 7, 1980.

3. Memo, L. V. Gossick, EDO, NRC to NRC Commissioners, TMI Action Plan -- Prerequisites for Resumption of Licensing, dtd Jan 5, 1980.

4. Three Mile Island, A Report to the Commissioners and to the Public, (Vol I), Mitchell Rogovin and George T. Frampton, Jr., NRC: Special Inquiry Group.

5. Memo, J. F. Ahearne, Chmn NRC to M. S. Plesset, Chmn ACRS, on ACRS participation in NRC rulemaking on storage and disposal of radioactive waste, dtd Jan 29, 1980.

6. Letter, L. E. Minnick, Yankee Atomic Electric Co. to Harold Denton, Director, NRR, NRC, Comments on Final Report of the TMI-II Lessons Learned Task Force (NUREG-0585), dtd Nov 26, 1979.

7. Memo, G. G. Zech to D. G. Eisenhut, Acting Director DOR, NRC, Summary of Meeting Held on December 5, 1979, with Commonwealth Edison Company (CECo), Consolidated Edison Company of New York (CONED) and Power Authority of the State of New York (PASNY) Regarding the Zion Station Units 1 and 2 and the Indian Point Units 2 and 3 Facilities; dtd Jan 10, 1980.