ACRS 1690

Issue Date:
February 26, 1980

(FOIA EXEMPTION (b)5)

MINUTES OF THE
235TH ACRS MEETING
NOVEMBER 8-10, 1979
WASHINGTON, DC

CERTIFIED

The 235th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street, N.W., Washington, DC, was convened at 8:30 a.m., Thursday, November 8, 1979.

[Note: For a list of attendees, see Appendix I. Messrs. Bender and Mathis were not present on Saturday, November 10, 1979.]

The Chairman noted the existence of the published agenda for this meeting, and the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that a request to make a public statement had been received, and that this request would be accommodated at an appropriate time. He also noted that copies of the transcript of some of the public portions of the meeting would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC within approximately 24 hours.

I. Chairman's Report (Open to Public)

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

A. Reviewers

The Chairman named Messrs. Lawroski and Shewmon as reviewers for the 235th ACRS Meeting.

B. Resignation of Executive Director for Operations

The Chairman noted that L. V. Gossick, Executive Director for Operations of the NRC, has resigned his position effective no later than February 1, 1980.

C. Honor for ACRS Member

The Chairman noted that ACRS member, Chester P. Siess, was honored by the American Concrete Institute on November 1, 1979.

D. Cancellation of Items from Meeting Agenda

The Chairman noted that the planned reviews of the Diable Canyon and Sequoyah Nuclear Plants have been cancelled because of the pause in licensing directed by the NRC. The anticipated delay

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is for a minimum of five to six months. The NRC has indicated that the reason for the delay is their need for time for response to the Kemeny Commission and the forthcoming Rogovin reports. He also noted that the review of the GE Test Reactor seismic issue has been exempted from this pause because this matter is not Three Mile Island-related.

II. Meeting with Members of the NRC Staff Regarding Current Matters (Open to Public)

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

A. Report on the President's Commission on the Accident at Three Mile Island (Kemeny Report)

W. Kane, NRC Staff, discussed the NRC Staff's preliminary evaluation of the Kemeny Commission Report (see Appendix IV). An outline of W. Kane's discussion is contained in Appendix V.

Members of the Committee requested that a member of the NRC Staff holding a policy-level position discuss this matter with the Committee. Later in the day, H. Denton, Director of the Office of Nuclear Reactor Regulation, discussed the NRC Staff's preliminary position regarding the Kemeny Report. The discussion was keyed to the Committee's interest in the reasoning behind the NRC Staff's pause in the licensing of nuclear power plants now ready for operation. He said that this pause is to allow the NRC Staff to develop new procedures that will conform with those recommendations accepted by the NRC from both the Kemeny and the Rogovin groups. He said that this pause in licensing also provides the NRC Staff with an opportunity to focus its attention on operating plants.

Mr. Lewis questioned the logic of keeping a few new plants off-line, while older plants are still operating. He suggested that the risk from newly operated plants does not significantly increase the overall risk of nuclear power. He observed that this NRC Staff action appears to be more symbolic than useful.

H. Denton said that the NRC Staff wants to be able to evaluate new safety ideas such as containment filtered venting, a core ladle, and other proposed equipment that might increase the available time for evacuation in the event of an accident.

In answer to a question regarding the continued NRC Staff's interest in the continued immediate review of the Offshore Power System, H. Denton indicated that the interest centers around evaluation of the proposed core-ladle that conceivably could be adapted to shore based reactor plants. He said that the NRC Staff also believes that there is an immediate need for an evaluation of the probability and consequence of steam explosions in reactor systems.

R. Baer, NRC Staff, said that an action plan is being developed to allocate Office of Nuclear Reactor Regulation resources. In the near future, the bulk of these resources will be directed toward operating plants. During this period, when enough review has been completed on nonoperating plants, Committee input will be requested.

Mr. Ebersole suggested that rather than a total pause on new plants, licensing these plants to operate at low power could provide an opportunity for utilities to perform start-up testing without substantial risk, and provide a level of testing not done in the past.

B. Clad Swelling During LOCA

R. O. Meyer, discussed the recent issue of fuel element cladding swelling during LOCA, which received national prominence in the press. He presented comparative curves obtained from both NRC Staff and reactor vendor calculations (see Appendix IV). He noted that the issue raised in the public press turned out to be a non-problem. The NRC Staff was caught in a maze of its own rules. Following its practices, the NRC Staff notified the hearing boards that a potential problem might exist with regard to the analyses of water reactor fuel element cladding during postulated LOCAs. This information was transmitted to the boards before an adequate investigation of the analytical practices of the reactor vendors could be made. When the investigation was complete, it was found that no problem had existed. The affecting internal NRC procedure requires that the NRC Staff interpret all safety issues conservatively, and immediately notify licensing boards. This procedure requires that the NRC Staff release information to the public whenever technical information from outside the NRC Staff is sought. In this case, major publicity was given to a non-problem.

C. Combination of Dynamic Loads

J. Knight, NRC Staff, reported to the Committee progress that the NRC Staff is making with respect to the development of stress loading criteria particularly for Mark II containments. He discussed the acceptance criteria for Mark II piping systems (see Appendix VII). Research programs on this matter are currently being completed at Brookhaven National Laboratory. He said that since there are a number of areas where loads must be combined, the NRC Staff directed its efforts toward developing the best method for this combination, rather than determining whether loads should be combined overall. The question currently addressed is whether the probability of exceedence of the design criteria is acceptable.

Mr. Siess suggested that the NRC Staff should use a number of real time histories regarding the seismic input for the combination of loads rather than the artificial time history currently being used, to determine whether the results of the calculations using the artificial time histories are realistic rather than needlessly overconservative.

D. Potential Unresolved Safety Questions on Interactions Between Non-Safety-Grade Systems and Safety-Grade Systems

(For background material relating to this discussion, see Appendix VIII.)

P. Check, NRC Staff, discussed the chronology of NRC Staff actions relating to potential safety questions on interactions between non-safety-grade and safety-grade systems, the basis for continued operation of plants despite the lack of review of this matter, initial findings of the review, current related activities, and future plans (see Appendix IX). He noted that although the Committee had requested a discussion of systems interactions resulting from steam line breaks outside containment, he would present information viewing the subject in a broader manner, consistent with a letter to the nuclear power industry from H. Denton, dated September 17, 1979 (see Appendix IX). He summarized the initial situation as being one in which the NRC Staff had a safety concern, but could identify no event that led to an unacceptable consequence.

Mr. Ebersole requested that the NRC Staff investigate a postulated accident in which a 10-in. steam line, supplying the steam-driven HPCI pump turbine, ruptures and dumps its steam in an emergency operating area. A proposed memorandum to the EDO on this matter was deferred for further discussion. Mr. Okrent agreed to redraft this proposed memorandum for Committee consideration at the 236th ACRS Meeting (December).

P. Check said that the NRC Staff has screened all licensees' submittals, and that there is general acknowledgment in the industries' responses that the issue deserves further study.

The NRC Staff's initial findings continue to be that safety problems have not been identified yet. He differentiated between safety problems and safety concerns, noting that safety concerns involve mostly uncertainty that is derived from a lack of information.

III. Meeting with Members of the NRC Staff Regarding NUREG-0600, "Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement" (Open to Public)

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

A. Three Mile Island Subcommittee Report

Mr. Etherington, Subcommittee Chairman, discussed the salient points of NUREG-0600 as identified at the October 31 subcommittee meeting (see Appendix X). He noted that the report is limited to a review of licensee actions prior to and during the TMI-2 accident, and to efforts to control releases.

Mr. Lewis stated that there may be a need to allow licensees to violate technical specifications during emergency conditions. (For ACRS consultant's reports on NUREG-0600, see Appendix XI.)

B. General Discussion

V. Stello, NRC Staff, stated that it had never been the intent of Inspection and Enforcement (IE) to blame the operators for the TMI-2 accident. The blame will have to fall on many including the NRC, the reactor vendor, the licensee, and the operators. The accident could have been prevented by any one of them. He said that it is the limited scope of NUREG-0600 that has allowed the misinterpretation of IE's intent. As far as the assessment of both causes and blame, the Rogovin and Committee reports probably will shed some light; the Kemeny report does shed some light.

V. Stello discussed some of the material contained in the letter from him and H. Denton to the Chairman of the NRC, dated October 6, 1979, and included in the background material in Appendix X. He concluded that the main problem is the human interface, and that IE's main emphasis is being directed to that problem.

Mr. Ebersole suggested that General Public Utilities and Metropolitan Edison, the Licensee, prior to operation, should have studied the plant on an engineering basis to learn of its deficiencies, especially with respect to instrumentation.

C. I.E. Investigation Scope and Method

J. Allen, NRC Staff, discussed the scope and method of the IE investigation into the TMI-2 accident (see Appendix XII). He said that the investigation has taken four months, and involved two groups of inspectors with seven men in each group. Over 200 people were interviewed. The goals of the investigation were to obtain the facts, and to evaluate the performance of the licensee. The period investigated was from the closure for maintainance of the feedwater valves on March 26 through the restarting of the main circulation pumps on March 28. IE did not evaluate the actions of the NRC or other agencies, did not review the regulatory process, legislative authorities, rules and regulation, safety research or the licensing program.

D. Operational Aspects

R. Martin, NRC Staff, discussed the operational aspect of TMI-2 during the period investigated (see Appendix XIII). He noted that TMI-2 had a normal maintainance history, and that all surveillance was current and normal, and in excess of technical specification requirements. He said that the NRC Staff cannot assess when core damage occurred, because there was no recognition during the transient that the core was uncovered. All actions were based on the belief that there was core cooling at all times.

Mr. Etherington noted that three individuals had separately predicted this type of accident on the basis of mechanical deficiencies, but that none of them had considered human error.

E. Licensee Performance

A. Gibson, NRC Staff, discussed the IE investigation into the performance of the licensee. He discussed the objectives of the study, the scope, the investigating team organization, sources of information, preaccident conditions, initial emergency response, the TMI emergency plan, the emergency organization, and various plant design and operating parameters (see Appendix XIV). He noted that operators did not interpret pressure--temperature readings in a manner consistent with retrospective conditions. He noted that in this accident, one of the problems was that radioactive materials were transferred from the primary system to the makeup purification system, thus transferring these materials from the containment to the auxiliary building, and providing a pathway for release of these materials to the environment.

In answer to a question, A. Gibson said that there were no technical specifications limiting the number of instruments that were permitted to be in the shop for repair at any one time.

Mr. Ebersole noted that the reactor heat removal system is not capable of handling highly radioactive water; this is a generic problem of pressurized water reactors.

F. ACRS Comments

The Committee provided its comments on NUREG-0600 in a report (see Appendix XXIX).

[Note: Members were provided copies of the IE Investigation Report No. 50-320/79-10, in which report IE proposed a fine of Metropolitan Edison for the TMI-2 accident (see Appendix XV)].

IV. Meeting With Members of the NRC Staff Regarding Draft 1 of Proposed Regulatory Guide 1.97 (Rev. 2), "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Open to Public)

[Note: Sam Duraiswamy was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

Mr. Siess, Regulatory Activities Subcommittee Chairman, discussed the Background for the current Revision 2 of Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, and noted the receipt of comments from several Members (see Appendix XVI), from several ACRS consultants (see Appendix XVII), and from several vendors (see Appendix XVIII). He noted that although this regulatory guide has not been "officially sent out for comment", some comments have been received regarding this draft.

B. NRC Staff View

A. S. Hintze, NRC Staff, said that the Staff desires ACRS input into this draft Regulatory Guide and its concurrence with the Guide prior to sending it to the industry and the public for comments. He reviewed the history of the Guide from 1973 through 1979, noted the objections to the specificity of the original Guide, and said that this revision has been rewritten to provide guidelines only, and leaves the selection of the actual instrumentation to be used to the licensees. He said that there has been some objection to "open-endedness" and ambiguity, and that no limit has been put on the accidents for which instrumentation must be provided. He noted that Revision 1 was issued in 1977, but that after issuance, licensees were reluctant to implement the Guide, especially for high-range instrumentation. The basis of the objections was that regulations did not require protection beyond Class-8 accidents. He said that the current Revision 2 takes into account Task Group actions and TMI-2 lessons learned.

A. S. Hintze noted that Regulatory Guide 1.97 (Rev. 2) parallels American Nuclear Society Standard 4.5 (Draft 4). He noted that the following requirements are in the Regulatory Guide, but not in the Standard:

- all concerns should be addressed by the instrumentation,
- an approach to breach of cladding, pressure boundary, and containment is included,
- all Design Basis Accidents should be included in the instrumentation,
- Type D system monitoring should be included,
- a list of accident variables should be included, and
- the length of time that instrumentation should be functional (200 days for the Regulatory Guide).

In answer to a question, V. Benaroya, NRC Staff, said that the adoption of Regulatory Guide 1.97 (Rev. 2) will increase the instrumentation price by a factor of two over that required by Regulatory Guide 1.97 (Rev. 1).

E. Wenzinger, NRC Staff, said that to date the NRC Staff has received only fragmented answers from the nuclear industry. He said that the required instrumentation is generally available. The problem is the requirement for environmental and seismic qualification of the instruments. He said there is a technological problem regarding the method for pressure vessel level instrumentation.

Mr. Okrent noted that the draft Regulatory Guide 1.97 (Rev.2) does not require gamma monitoring capability in containment. He suggested that the NRC Staff look at the possibility of semicontinuous monitoring of cesium or other radionuclides.

C. Statement From Member of Public

The Chairman noted that a request had been received for a statement to be made by a representative from the nuclear industry. X. Polanski, Commonwealth Edison Company, also representing the American Nuclear Society, would present that statement.

X. Polanski noted the opposition of the American Nuclear Society to the approach followed in Regulatory Guide 1.97 (Rev. 2). He proposed, instead, the approach used in ANS Standard 4.5 (Draft 4), noting it is expected that this Standard will be adopted by late Spring 1980. He discussed the objections to accident monitoring instrumentation, and the ANS' concerns regarding Regulatory Guide 1.97 (Rev. 2) (see Appèndix XX).

D. ACRS Actions

The Committee agreed with the NRC Staff's plan to issue proposed Regulatory Guide 1.97 (Rev. 2), Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, for public comment.

V. NUREG-0606, Resolution of NRC Category A Safety Related Tasks (Open to Public)

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

(For background information, see Appendix XXI.)

S. H. Hanauer, NRC Staff, discussed the plans, effort, and schedule for the attempted resolution of the NRC Staff Category A Safety-Related Tasks as appearing in NUREG-0606 (see Appendix XXII). He said that at the current time, most work to generate and assemble the necessary technical information to resolve the issues is being contracted out. He suggested that the NRC Staff is interested in receiving ACRS input all along the way and that he would attempt to provide the proposed documents to the Committee in a timely fashion so that early, meaningful input into the effort could be received from the Committee. He specifically spoke to the ATWS problem, noting that the NRC Staff's first three reports on ATWS were not adequately coordinated with the Committee, but he promised to do better in the future. He said that ATWS will be resolved by rule making, with a rule based on the information available January 1, 1980 as the basis for the rule. The rule will be based on equipment which can assure that the safety requirements will be met.

Mr. Okrent recommended that it might be useful if the NRC Staff learned what the French are learning regarding their studies on station blackout. He suggested that the ACRS Staff arrange a Committee briefing by the Probability Assessment Staff on the Integrated Reliability Assessment Program.

Mr. Ebersole noted his concern relating to the status of nuclear power plant control rooms. He noted that many of the alarm systems and instruments are not qualified for truly adverse conditions. He noted that no modern control room appears to be available to licensees.

S. H. Hanauer agreed to send copies of reports on proposed resolutions (e.g., NUREG-0613 (Rev. 1), regarding pipe racks in BWRs) to the ACRS at the same time they are released for public comment. This would provide an opportunity for Committee input deemed appropriate during the public comment period. The Committee concurred with this proposed procedure.

The Committee also tentatively accepted the schedule for the resolution of these safety related items as proposed by S. H. Hanauer and NUREG-0606.

VI. Executive Sessions (Open to Public)

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

A. Subcommittee Reports

1. TMI-2 Accident Implications

Mr. Okrent, Subcommittee Chairman, noted that a request had been received from Commissioner Bradford (see Appendix XXIII) for a clarification of the Committee's report of August 13, 1979. Short-Term Recommendations of TMI-2 Lessons Learned Task Force. Mr. Okrent requested, and the Committee concurred, that this memorandum be answered.

[Note: The Committee prepared a report responding to Commissioner Bradford's request at a later time during this meeting (see Appendix XXX).]

2. Anticipated Transients Without Scram

Mr. Kerr, Subcommittee Chairman, said that S. H. Hanauer, NRC Staff, has informed the Subcommittee that information is not being received from licensees at either the rate or detail that is desired by the NRC Staff. He said that BWR owners, however, are committing to pump trips on a specific schedule. The NRC Staff is meeting with reactor vendors. S. H. Hanauer believes that "foot-dragging" is being practiced.

(For information regarding the history and status of the installation of recirculation pump trips in BWRs to mitigate ATWS, see Appendix XXIV.)

[Note: As noted previously in item V above, the Committee concurred with the process and schedule proposed by the NRC Staff for resolution of 29 Category A Task Action items of which ATWS is one.]

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.

4. Procedures

The Chairman noted that at its November 7 meeting, the Procedures Subcommittee discussed the following matters, and proposed that the Committee adopt the following recommendations:

- invite interested Congressmen and their staffs to attend and participate in Reliability and Probabilistic Assessment Subcommittee meetings in the near future, when discussions will be held on comparative risk assessment.

- the ACRS clerical staff and equipment appear to be adequate and provide no problems to the Committee.

- agreed with the NRC open door policy with respect to bringing forth safety concerns or bringing technical disagreements into the open. The ACRS should maintain a similar open door policy; ACRS procedures developed in the past do cover this matter. These matters should be brought to individual Members or the Chairman, and later to the full Committee, if appropriate.

Mr. Lewis suggested that the Committee needs a procedure to close issues brought up by the Staff.

In answer to a question, Mr. Kerr noted that the concerns brought forth by D. S. Basdekas, NRC Staff, will be addressed at a Power and Electrical Systems Subcommittee meeting scheduled for December 13.

5. Regulatory Activities

The Committee concurred with the conclusions drawn by the Regulatory Activities Subcommittee and also in the regulatory position of Regulatory Guide 1.141 (Rev. 1), Containment Isolation Provisions For Fluid Systems, with the condition that the implementation section of this Guide be revised consistent with the TMI-2 Lessons Learned Task Force recommendations (NUREG-0578). The Committee also concurred with the NRC Staff's plan to issue the proposed Regulatory Guide 1.97 (Rev. 2) for public comment (see section III, preceding).

B. Generic Items

The Committee reviewed its generic items list, both resolved and unresolved, and recommended action on most of these items (see Appendix XXVII).

C. Review of Regulatory Function and Process

The committee continued its discussions regarding the NRC regulatory function and process that was begun at the 233rd ACRS Meeting. Mr. Bender agreed to coordinate the rewriting of this paper on the regulatory process as follows:

- Written comments by Members on Draft 4, discussed at this meeting, were to be provided to him by November 16, 1979.

- Mr. Bender will merge the comments and provide copies of Draft 5 to the Members by November 23, 1979.

- Members were assigned to review Draft 5 and provide their comments at the Procedures Subcommittee Meeting on December 5. Individual assignments for this review are

- Chapters 1-3	:	Mr. Etherington
- Chapter 4	:	Mr. Mark
- Chapter 5	:	Mr. Ray
- Chapter 6	:	Mr. Okrent
- Chapter 7	:	Mr. Moeller
- Chapter 8	:	All Members
- Covering Letter	:	All Members

(Note: Members who would be unable to attend the Procedures Subcommittee Meeting were requested to provide their comments to the Executive Secretary so that they could be discussed during the meeting.)

D. NRC Staff Follow-Up on ACRS Requests and Reports

The Executive Secretary noted the receipt of a memorandum, NRC Procedures for the Control of ACRS Requests and ACRS Consultant Reports, from C. J. Heltemes, Jr., NRC Staff (see Appendix XXVIII).

The Committee agreed that the Executive Director should wait for meeting reviewers' and the Chairman's comments on the ACRS' formerly internal publication, Actions, Agreements, Assignments, and Requests, before this publication is provided to the NRC Staff for follow-up in accordance with the procedures outlined in the memo from C. J. Heltemes noted above.

E. Policy and Procedures for Differing Professional Opinions (NUREG-0567)

The Committee authorized the Executive Director to provide the Director of the Office of Management and Programs Analysis, NRC, with a revised paragraph 6.b. for the proposed publication, Policy and Procedures for Differing Professional Opinions (NUREG-0567) as it relates to ACRS participation in this process (see Appendix XXIX.)

F. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 236th ACRS Meeting (December) and several items for future meetings (see Appendix II).

2. Schedule of ACRS Subcommittee Meetings and Tours

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

G. ACRS Reports and Letters

1. Report on NUREG-0600

The Committee prepared a report regarding NUREG-0600, Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement (see Appendix XXX).

2. Clarification of ACRS Report of August 13, 1979 on NUREG-0578

The Committee approved a letter to Commissioner Bradford, responding to his request for clarification of certain items contained in the Committee's August 13, 1979 report concerning Short-Term Recommendations of TMI-2 Lessons Learned Task Force see Appendix XXXI).

3. Regulatory Guides

The Committee authorized a memorandum to the Executive Director for Operations informing him of the Committee's concurrence in the regulatory position of Regulatory Guide 1.141 (Rev. 1), with the condition that the implementation section of the Guide be revised to be consistent with NUREG-0578, and with the NRC Staff's plan to issue proposed Regulatory Guide 1.97 as well, for public comment (see Appendix XXXII).

4. Communications with the NRC and NRC Staff

After a brief discussion, the Committee authorized the transmittal of a letter to Commissioner Ahearne regarding ACRS procedures and practices for transmitting recommendations and questions to the NRC Staff and the Commissioners. However, following the meeting, the Chairman deferred transmittal of this letter pending further discussion by the Committee at the 236th ACRS Meeting (December).

5. Issuance of Low Power Licenses

The Committee considered a proposed letter to the Commissioners discussing the possibility of using the current pause in the issuance of operating licenses for intensive low power tests of those plants completed and awaiting full power licenses. The letter was not completed, and is scheduled for further consideration at the 236th ACRS Meeting (December).

6. Scenario for Systems Interactions Study

The Committee considered a proposed letter to the Executive Director for Operations regarding a proposed scenario to be used for a study of systems interactions between safety-related and non-safety-related systems. The letter was not completed, and will be considered further at the 236th ACRS Meeting (December).

7. Identification of NRC Regulations Needing Changes

The Committee considered a letter responding to a request from Commissioner Bradford requesting identification of NRC regulations needing changes (see Appendix XXXIII). The letter was not completed, and will be considered further at the 236th ACRS Meeting (December).

VII. Executive Sessions (Closed to Public)

A. Report of Nominating Committee

The Committee accepted the report of its Nominating Committee regarding nominations of Committee officers for Calendar Year 1980. The Subcommittee proposed Mr. Plesset for Chairman, and Mr. Mark for Vice-Chairman.

The 235th ACRS Meeting was adjourned at 3:40 p.m., Saturday, November 10, 1979.

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Signed at Washington, D.C., this 15th day of October, 1979.

Ian D. Lanoff,

Administrator, Pension and Welfare Benefit Programs, Labor-Management Services Administration, Department of Labor.

[FR Doc. 79-32680 Filed 10-22-79; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Procedures and Administration Subcommittee; Meeting

The ACRS Procedures and Administration Subcommittee will hold an open meeting on November 7, 1979 in Room 1010, 1717 H St., NW, Washington, DC 20555.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), 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 Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangemens can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, November 7, 1979; 5:15 p.m. until the conclusion of business.

The Subcommittee will discuss the scope and procedures for conduct of ACRS business, including ACRS involvement in consideration of differing professional opinions among NRC Staff members.

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 Designated Federal Employee for this meeting, Mr. Raymond F. Fraley, (telephone 202/634-3265) between 8:15 a.m. and 5:00 p.m., EDT before, and EST after, October 28, 1979.

Dated: October 18, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-32571 Filed 10-22-79; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Regulatory Activities; Meeting

The ACRS Subcommittee on Regulatory Activities will hold an open meeting on November 7, 1979, in Room 1167, 1717 H St. NW., Washington, DC 20555. Notice of this meeting was published in the Federal Register October 18, 1979.

In accordance with the procedures outlined in the Federal Register on October 1, 1979 (44 FR 56408) 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 Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, November 7, 1979. The meeting will commence at 8:45 a.m.

The Subcommittee will hear presentations from the NRC Staff and will hold discussions with this group pertinent to the following:

(1) Proposed Regulatory Guide, "Qualification and Production Tests for Safety-Related Snubbers."

(2) Proposed Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

(3) Regulatory Guide 1.141, Revision 1, "Containment Isolation Provisions for Fluid Systems."

Other matters which may be of a predecisional nature relevant to reactor operation or licensing activities may be discussed following this session.

Persons wishing to submit written statements regarding Regulatory Guide 1.141, Revision 1, may do so by providing a readily reproducible copy to the Subcommittee at the beginning of the meeting. However, to insure that adequate time is available for full consideration of these comments at the meeting, it is desirable to send a readily reproducible copy of the comments as far in advance of the meeting as practicable to Mr. Sam Duraiswamy (ACRS), the Designated Federal Employee for the meeting, in care of ACRS, Nuclear Regulatory Commission, Washington, D.C. 20555 or telecopy them to the Designated Federal Employee (202)-634-3319) as far in advance of the meeting as practicable. Such comments shall be based upon documents on file and available for public inspection at the NRC Public Document Room, 1717 H St., NW., Washington, DC 20555.

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 Designated Federal Employee for this meeting, Mr. Sam Duraiswamy, (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EDT before, and EST after, October 28, 1979.

Dated: October 18, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-32572 Filed 10-22-79; 8:45 am]

BILLING CODE 7590-01-M

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 November 8-10, 1979, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published on September 20, 1979 (44 FR 54558).

The agenda for the subject meeting will be as follows:

Thursday, November 8, 1979

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

The Committee will discuss proposed ACRS comments and recommendations regarding the NRC regulatory process.

1:30 p.m.-4:30 p.m.: Diablo Canyon Nuclear Power Station Units 1 and 2 (Open)—The Committee will hear and discuss reports from representatives of NRC Staff, and the Pacific Gas and Electric Company and their consultants/contractors, as necessary, regarding proposed application of experience gained at the Three Mile Island Nuclear Station Unit 2 to the Diablo Canyon Nuclear Power Station.

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

4:30 p.m.-6:30 p.m.: Westinghouse Nuclear Steam Supply System with Ice-Condenser Containment/Sequoyah-McGuire Nuclear Plants (Open)—The Committee will hear and discuss reports from representatives of the NRC Staff, and the Applicants and their consultants

235th

and contractors, as necessary, regarding proposed application of experience gained at the Three Mile Island Nuclear Station Unit 2 to nuclear plants which make use of Westinghouse NSSS with ice-condenser containment of the Sequoyah and McGuire type.

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

6:30 p.m.–7:30 p.m.: Executive Session (Open)—The Committee will hear and discuss the reports of ACRS Subcommittee Chairmen regarding proposed evaluation and plans for resolution of generic safety matters applicable to light-water reactors. Members of the NRC Staff will participate as necessary.

Friday, November 9, 1979

8:30 a.m.–12:30 p.m.: Executive Session (Open)—The Committee will discuss proposed ACRS comments and recommendations regarding the NRC regulatory process.

1:30 p.m.–6:45 p.m.: Meeting with NRC Staff (Open)—The Committee will hear and discuss a presentation regarding the anticipated schedule and proposed procedures for ACRS review of action to resolve NRC unresolved safety issues (NUREG–0606).

The Committee will hear and discuss a presentation regarding system interactions which could result from a steamline break outside containment in nuclear plants using Westinghouse Nuclear Steam Supply Systems.

The Committee will hear and discuss a presentation regarding proposed revision of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

The Committee will hear and discuss a report regarding NUREG–0600, "Investigation Into the March 28, 1979 Three Mile Island Accident" by the Office of Inspection and Enforcement.

6:45 p.m.–7:15 p.m.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee on the NRC report NUREG–0625, "Report of the Siting Policy Task Force."

Saturday, November 10, 1979

8:30 a.m.–4:00 p.m.: Executive Session (Open)—The Committee will continue its discussion of proposed ACRS comments and recommendations regarding the NRC Regulatory process.

The Committee will discuss proposed reports to the NRC on the Diablo Canyon Nuclear Power Station Units 1 and 2; Westinghouse Nuclear Steam Supply Systems with Ice-Condenser Containment of the Sequoyah-McGuire type; NUREG–0600, "Investigation Into the March 28, 1979 TMI Accident"; and proposed changes to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following An Accident".

The Committee will discuss proposed replies to NRC Commissioners regarding follow-up and implementation of ACRS recommendations.

The Committee will hear and discuss reports of its Subcommittees on Three Mile Island Accident Implications and the implementation of recommendations resulting from this accident.

The Committee will also hear and discuss reports of its Subcommittees on the Wolf Creek Nuclear Power Station (seismic design), and anticipated transients without scram.

The Committee will discuss the proposed schedule for future ACRS activities, nominees for ACRS Officers for CY 1980, and will complete discussion of miscellaneous items considered during this meeting.

Portions of this session may be closed as necessary to protect information, the release of which would represent an unwarranted invasion of personal privacy. Portions will also be closed as necessary to discuss Proprietary Information applicable to matters noted above.

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 release of which would represent an unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)).

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 F. Fraley (telephone 202/634–3265), between 8:15 a.m. and 5:00 p.m. EDT. (EST after October 28, 1979).

Dated: October 18, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-32665 Filed 10-22-79; 8:45 am]
BILLING CODE 7590-01-M

[Docket Nos. 50-321-SP and 50-366-SP]

Georgia Power Company, et al. (Edwin I. Hatch Nuclear Plant, Units 1 and 2) (Spend Fuel Expansion); Order Setting Special Prehearing Conference

On August 15, 1979, the Nuclear Regulatory Commission published in the **Federal Register** a notice of a proposed issuance of an amendment to Facility Operating License Nos. DPR–57 and NPF–5 that had been issued to Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Association of Georgia, and City of Dalton, Georgia (the licensee) for the operation of Edwin I. Hatch Nuclear Plant, Units Nos. 1 and 2 (the facilities) located in Appling County, Georgia. 44 FR 47820. The proposed amendment would allow an increase in storage capacities of the spent fuel pools of from 840 to 3,181 fuel assemblies in Hatch No. 1 pool and from 1,120 to 2,845 fuel assemblies in Hatch No. 2 pool.

The notice provided that the licensee might file a request for a hearing and that any person whose interests might be affected by the proceeding might file a request for a hearing in the form of petition for leave to intervene pursuant to 10 CFR 2.714 by September 14, 1979. On that date, a petition to intervene was received from Georgians Against Nuclear Energy (GANE), which was amended on October 4, 1979 to include an affidavit of a member claiming to reside approximately 44 miles from the facilities. The petition has been opposed

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

Revised: November 6, 1979

DETAILED SCHEDULE AND OUTLINE
FOR DISCUSSION
235TH ACRS MEETING
NOVEMBER 8-10, 1979
WASHINGTON, DC

Thursday, November 8, 1979, Room 1046, 1717 H Street, NW, Washington, DC

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

- 1.1) 8:30 A.M.-10:15 A.M. - Chairman's Report (MWC/RFF)
  - 1.1-1) 8:30 A.M.-8:35 A.M.: Request from Comm. Bradford re. ACRS Report on Short Term Recommendations of TMI-2 Lessons Learned Task Force
  - TAB -------------------- 1.1-2) 8:35 A.M.-8:45 A.M.: Civil penalty resulting from TMI-2 accident
  - 1.1-3) 8:45 A.M. - 9:30 A.M.: Report of President's Commission on TMI
  - 1.1-4) 9:30 A.M. - 9:45 A.M.: Rpt. re. clad swelling during LOCA transient
  - 1.1-5) 9:45 A.M. - 10:15 A.M.: Rpt. re. combination of dynamic loads
- TAB ---------------- 1.2) 10:15 A.M. - 1:00 P.M.: Discuss proposed ACRS comments & recommendations regarding the nuclear regulatory process (MB/RFF)

1:00 P.M. - 2:00 P.M. — LUNCH

2) 2:00 P.M. - 6:30 P.M. — Executive Session (Open)

ACRS Subcommittee Reports on:

- TAB --------------- 8.2) 2:00 P.M.-2:30 P.M.: Proposed revision of NRC Regulatory Guides (No. 1.97 and 1.141) (CPS/SD)

TAB --------------- 9.2) 2:00 P.M.-3:00 P.M.: TMI-2 Accident Implications-clarification of ACRS Report on Short Term Recommendations (DO/RKM)

TAB --------------- 9.3) 3:30 P.M.-3:45 P.M.: Anticipated Transients Without Scram (WK/PB)

TAB --------------- 9.4) 3:45 P.M.-4:15 P.M.: Wolf Creek Nuclear Power Station - seismic design (HE/RM)

4:15 P.M.-4:30 P.M.: ACRS Procedures (MWC/RFF)

TAB --------------- 1.2) 4:30 P.M.-6:30 P.M.: Discuss proposed ACRS comments regarding the nuclear regulatory process (MB/RFF)

TAB --------------- 4.1) Reports by ACRS Subcommittee Chairmen regarding proposed plans for reevaluation of ACRS generic safety matters applicable to light-water reactors (MB/et al.)

Friday, November 9, 1979, Room 1046, 1717 H Street, NW, Washington, DC

3) 8:30 A.M. - 9:15 A.M. Meeting with NRC Staff (Open)

TAB --------------- 6.2) Report re. system interactions resulting from steamline breaks outside containment

5) 9:15 A.M. - 1:15 P.M. Executive Session (Open)

The Committee will discuss proposed comments and recommendations regarding the nuclear regulatory process (MB/RFF)

1:15 P.M. - 2:15 P.M. LUNCH

6) 2:15 P.M. - 6:45 P.M. Meeting with NRC Staff (Open)

TAB ---------------- 6.3) 2:15 P.M.-3:15 P.M.: Report regarding proposed revision of NRC Regulatory Guide 1.97, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following An Accident

TAB ---------------- 6.1) 3:15 P.M. - 3:45 P.M.: Resolution of NRC Category A Safety Related Tasks (NUREG-0606)

TAB ---------------- 6.4) 3:45 P.M. - 6:45 P.M.: Report regarding NUREG-0600, Investigation into the March 28, 1979 Three Mile Island Accident by the NRC Office of Inspection and Enforcement

7) 6:45 P.M. - 7:15 P.M. Executive Session (Open)

TAB ---------------- 7.1) Report of ACRS Subcommittee on the NRC Siting Policy Task Force Report (DWM/RM)

Saturday, November 10, 1979, Room 1046, 1717 H Street, NW, Washington, DC

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

TAB ---------------- 8.2) 8:30 A.M. - 10:00 A.M.: Discuss proposed ACRS reports/comments on:

- NUREG-0600, TMI-2 Accident Investigation
- Proposed revision of NRC Regulatory Guides

TAB -------------- 9.1) 10:00 A.M.-11:00 A.M.: Proposed replies to Comm. Ahearne and Comm. Bradford re. implementation of ACRS recommendations (DWM/RFF)

8.3) 11:00 A.M. - 11:15 A.M.: Report of Nominating Committee

8.4) 11:15 A.M.-11:30 A.M.: Future Schedule

9.6-1) Anticipated subcommittee activity

8.4-1) Anticipated full Committee activity

8.5) 11:30 A.M.-12:00 Noon

8.5-1) Complete items discussed during this meeting

(Note: Portions of this session will be closed as necessary to discuss Proprietary Information applicable to these matters.)

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APPENDIXES

TO

MINUTES OF THE 235TH ACRS MEETING

NOVEMBER 8-10, 1[illegible]

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Max W. Carbon, Chairman
Milton S. Plesset, Vice-Chairman
Myer Bender
Jesse C. Ebersole
Harold Etherington
William Kerr
Stephen Lawroski
Harold W. Lewis
J. Carson Mark
William M. Mathis
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
John H. Austin
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
Jean A. Robinette
Richard P. Savio
John Stampelos
Peter Tam
Hugh E. Voress
Harold Walker
Gary Young
Dorothy Zukor

NRC ATTENDEES

235TH ACRS MTG.

Thursday, November 8, 1979

Robert Baer, Nuclear Reactor Regulation
Tolbert Young, Jr., Inspection and Enforcement
Joseph Martore, NRR
William Kane, NRR
J. Knight, NRR
Paul Check, NRR
R. Martin, NRR
J. Allen, IE

Friday, November 9, 1979

John Angelo, Div. of Project Management, LWR #1
R. Baer, NRR
E. C. Wenzinger, Office of Stds. Development
G. Guppy, SD
G. Yohas, IERI
V. Stello, IE
A. S. Hintze, RSSB
V. Benaroya, NRR
S. H. Hanauer, NRR
A. Gibson, IE

PUBLIC ATTENDEES

235TH ACRS MTG.

Thursday, November 8, 1979

Jeff Mapes, Scripss League Newspapers, 1395 Nat'l Press Bldg., Wash., DC
Betsy Taylor, NIRS, 1536 16th Street, NW, Wash., DC 20036
Mark B. Whitaker, So. Carolina Electric, Columbia, SC
James A. Wactor, S. C. Electric & Gas Co., Columbia, S.C.
T. Martin, NURECH, Vienna, VA
Stave Wyncoop, McGraw-Hill, Wash., D. C.
M. Rood, Neuhouse News, Wash., DC
R. L. Stright, Nuclear Projects, Inc., Rockville, MD
R .eyse, EPRI
R. P. Smith, McGraw-Hill
A. Weisbard, SPP&T

Friday, November 9, 1979

B. Montgomery, Bechtel Power Corp., Balt., MD
R. H. Leyse, EPRI, Rockville, MD
R. P. Smith, McGraw-Hill
Xavier Polanski, ANS 4.5
David A. Sommers, ANS 4.5
J. Gillin, Justice
D. Zachery, Private
J. McEwen, Jr., Stafco
N. Knowles, Shaw, Pittman

Saturday, November 10, 1979

J. J. Ray, (Mrs.), Self
C. Davidson, Self, Alexandria, VA
G. A. Blanc, PG&E, 34 Sullivan Drive, Moraga, CA
J. Silberg, Shaw Pittman Potts & Trowbridge, Bethesda, Md
Leyse, EPRI, Rockville, MD
W. H. House, II, Bechtel, Frederick, MD

APPENDIX II

ACRS FUTURE MEETING AGENDA
AS OF NOVEMBER 13, 1979

236th ACRS Meeting, December 1979

Report on Regulatory Process (M. Bender et. al.)	All day Thursday
Response to Commissioner Bradford's request regarding changes in NRC Rules and Regulations	2 hrs
Recommendations in Kemeny Commission Report relating to ACRS	2 hrs
Recommendations in Kemeny Commission Report relating to Safety Issues	2 hrs
President's Response to Kemeny Commission Report (tent.)	2 hrs
TMI-2 Lessons Learned -- Final Report	4 hrs
Proposed revision of NRC List of Category A Task Action Plans	1 1/2 hrs
Meeting with NRC Commissioners	2 hrs
Proposed "pause" in licensing	
ACRS LER Report	
ACRS RSR Report	
ACRS Position/Comments regarding the proposed "pause" in Licensing	1 hr
HPSI Steam Turbine Steamline Break (potential interaction) (tent.)	1 hr

The following items were discussed briefly but were not scheduled for the December meeting. They will be proposed/scheduled for future meetings as appropriate.

Bulletins and Orders Small Break Analysis

Siting Policy Task Force Report

Subcommittee Report on BWR Pipe Cracking regarding Comments by BWR Owners Group

Subcommittee Report on LaCrosse Spent Fuel Pool Modifications

Health Physics Assessment of Conditions at TMI-2

Appendix A (Cont.)

NRC Basis for upgrading LERs to AORs

Briefing by C. J. Heltemes regarding proposed activities of the NRC Operations Evaluation and Analysis Group

Surry 2 Steam Generator Replacement

GETR — Site geology/seismicity

ACRS procedures for review of proposed power level increases

Report by Dr. Plesset regarding Mk I and Mk II containment

ACRS comments/report regarding combination of dynamic loads

Briefing regarding activities of the Nulcear Safety Analysis Center

ACRS comments regarding proposed rule on Fire Protection provisions at nuclear facilities

Proposed procedures for ACRS handling of Dissenting Professional Opinions, allegations regarding Reactor Safety, etc.

Application of the FNP core ladle to land based plants

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
November 10, 1979

APPENDIX III

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS AND TOURS

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

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

MW Libarkin

M. W. Libarkin
Assistant Executive Director
for Project Review

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

A-6

November

14	GETR (San Francisco, CA) EI; WK, MB, DO
15-16	Extreme External Phenomena (Los Angeles, CA) RS; DO, MC, CM, WM
16	Fluid Dynamics (San Francisco, CA) AB; MP, HE
17	FNP (Los Angeles, CA) GQ; DWM, HE, WM, DO, MP, PS
29-30	Advanced Reactors (Albuquerque, NM) RS; WK, MC, CM, MP, PS

December

3	Reactor Operations RKM; HE, WM
4	TMI-2 Impli, Re Nuc. Power Plant RKM; DO, MC, WM
5	Relia. & Probabil. Assess. GQ; DO, MB, JE, HL, CM
5 (am)	Plant Arrangements RKM; MJ, JE, SL, CM, MP, JR
5 (1:30 pm)	*Procedures and Administration RFF; MC, SL, MP
5 (pm)	Fire Protection PT; MB, JR, HE, JE
6-8	236th ACRS Meeting
13	Power & Elec. Sys. GQ; WK, JE, CM, JR
18-19	ECCS (Bull & Ord)(Los Angeles, CA) AB/PB; MP, WM, HE, PGS, DO (tent)
19	Waste Management PT; SL, DWM, WK, JCM, JR, WM

January 1980

2 ~~8~~ 9	RSR TGM; DO, HE, WK, SL, CM, MP, PS, CS
9/ ~~8~~ 16-17 (p.m.)	Metal Components (ORNL tentative) EI; PS, MB, HE
10-12	237th ACRS Meeting
	LER & Reliab/Prob. Ass'ment Subcom. to be scheduled during January.

*NOTE: In order to complete their contributions to the Review of Regulatory Functions and Processes, the following members should be prepared to leave other meetings for a short time on a schedule to be arranged: HE, DO, JR, CM. Any members who are unable to attend at all should provide their comments to Ray Fraley prior to the meeting.

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

November 1, 1979

MEMORANDUM FOR: Lee V. Gossick, EDO

FROM: Harold R. Denton, Director, NRR

SUBJECT: PRESIDENT'S COMMISSION REPORT ON TMI ACCIDENT

We are in substantial agreement with the findings and recommendations of the President's Commission Report that fall within NRR's purview, with only a few exceptions. Since the accident, we have developed considerable momentum in requiring many changes that turn out to be consistent with the recommendations of the President's Commission. Specific preliminary evaluations by the NRR staff of the Report's findings and recommendations are given in Enclosures A through F.

The President's Commission made 44 recommendations in seven categories. Of these 44 recommendations, there are 33 that fall sufficiently within the purview of this Office that some of the implementing recommendations or actions could reasonably be expected to originate here. Our preliminary evaluations of the 33 recommendations show that the NRR staff has already initiated improvements relating to all or part of 28 of the 30 recommendations applicable to operating plants, and has taken or recommended action on many other matters within our licensing authority that are not specifically recommended by the President's Commission.

Our actions to date on operating reactors are summarized in Enclosure G. Some of the more significant actions are:

- Directives to implement the improvements recommended by the NRR Lessons Learned Task Force plus additional requirements from the ACRS and the Director of NRR. A final report of the Task Force, with additional recommendations has been completed.
- Requirements for substantial improvements relating to auxiliary feedwater systems, small break LOCA evaluations and operating procedures and training requirements.
- Review of the capabilities of all licensed operators for coping with a TMI-type accident, and recommendations pending before the Commission for a major upgrading of training requirements.

This recommendation is consistent with recommendation A.8 of the President's Commission. The length of this pause, and the degree to which existing resources can be assigned to CPs and OLs after this pause, cannot be well defined until an agency-wide action plan on all TMI-related recommendations is developed and approved by the Commission.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
A - Analysis of Recommendations for NRC
B - Analyses of Recommendations for Utility and Its Suppliers
C - Analysis of Recommendations for Training of Operating Personnel
D - Analysis of Recommendations for Technical Assessment
E - Analysis of Recommendations for Worker and Public Health and Safety
F - Analysis of Recommendations for Emergency Planning and Response
G - NRR Activities to Date on Operating Plants
H - Summary of Related NRR Actions

cc: Howard K. Shapar, ELD ✓

RECOMMENDATION A-2

An oversight committee on nuclear reactor safety should be established. Its purpose would be to examine, on a continuing basis, the performance of the agency and of the nuclear industry in addressing and resolving important public safety issues associated with the construction and operation of nuclear power plants, and in exploring the overall risks of nuclear power.

a. The members of the committee, not to exceed 15 in number, should be appointed by the President and should include: persons conversant with public health, environmental protection, emergency planning, energy technology and policy, nuclear power generation, and nuclear safety; one or more state governors; and members of the general public.

b. The committee, assisted by its own staff, should report to the President and to Congress at least annually.

FINDINGS

This recommendation apparently results from the various findings given in G1; G2; G3; G4; G5; G6; and G8a through i. We generally agree with the findings given in the Report.

RESPONSE

The recommendation is not within the purview of NRR.

RECOMMENDATION A-4

Included in the agency's general substantive charge should be the requirement to establish and explain safety-cost trade-offs; where additional safety improvements are not clearly outweighed by cost considerations, there should be a presumption in favor of the safety change. Transfers of statutory jurisdiction from the NRC should be preceded by a review to identify and remove any unnecessary responsibilities that are not germane to safety. There should also be emphasis on the relationship of the new agency's safety activities to related activities of other agencies. (See recommendations E.2 and F.1.b.)

a. The agency should be directed to upgrade its operator and supervisor licensing functions. These should include the accreditation of training institutions from which candidates for a license must graduate. Such institutions should be required to employ qualified instructors, to perform emergency and simulator training, and to include instruction in basic principles of reactor science, reactor safety, and the hazards of radiation. The agency should also set criteria for operator qualifications and background investigations, and strictly test license candidates for the particular power plant they will operate. The agency should periodically review and reaccredit all training programs and relicense individuals on the basis of current information on experience in reactor operations. (See recommendations C.1 and C.2.)

b. The agency should be directed to employ a broader definition of matters relating to safety that considers thoroughly the full range of safety matters, including, but not limited to, those now identified as "safety-related" items, which currently receive special attention.

c. Other safety emphases should include:

(i) a systems engineering examination of overall plant design and performance, including interaction among major systems and increased attention to the possibility of multiple failures;

(ii) review and approval of control room design; the agency should consider the need for additional instrumentation and for changes in overall design to aid understanding of plant status, particularly for response to emergencies; (see recommendation D.1) and

(iii) an increased safety research capacity with a broadly defined scope that includes issues relevant to public health. It is particularly necessary to coordinate research with the regulatory process in an effort to assure the maximum application of scientific knowledge in the nuclear power industry.

and supervisory licensing functions. Specific examples of some of the work already underway in this area are provided below in our response to recommendation on operator training. The principal actions and recommendations by the staff on operator and supervisor training, qualifications and licensing are found in NUREG-0578, NUREG-0585, SECY-79-330E, and in a revised draft national standard on qualifications of personnel.

b. We agree with the need to significantly broaden the definition of matters relating to safety. Equipment and human factors should both be included. Specific recommendations and actions by the staff in this area are numerous in the past few months; these are principally identified in NUREG-0578, NUREG-0585, where specific equipment was identified for backfit and significant re-evaluation of non-safety classifications of equipment were recommended, and issuance of the Bulletins and Orders Task Force relating to upgrading of auxiliary feedwater systems.

c. We agree with the need for an overall systems engineering evaluation of overall plant design and performance. A restructuring of NRR, including much increased prominence of the human factors in system safety evaluation, is needed. The overall, integrated system evaluation of pending OL applications needs to be defined. However, first priority should go to operating plants. Recommendations by the staff on how to proceed on operating plants were contained in NUREG-0585 and include a year long re-evaluation of all control rooms, an Integrated Reliability Evaluation Program, and a complete failure modes and effects analysis of all non-safety equipment, with all of these reviews giving much increased attention to the people

Recommendation A5 Responsibility and accountability for safe power plant operations, including the management of a plant during an accident, should be placed on the licensee in all circumstances. It is, therefore, necessary to assure that licensees are competent to discharge this responsibility. To assure this competency, and in light of our findings regarding Metropolitan Edison, we recommend that the agency establish and enforce higher organizational and management standards for licensees. Particular attention should be given to such matters as the following: integration of decision-making in any organization licensed to construct or operate a plant; kinds of expertise that must be within the organization; financial capability; quality assurance programs; operator and supervisor practices and their periodic reevaluation; plant surveillance and maintenance practices; and requirements for the analysis and reporting of unusual events.

Findings: This recommendation apparently flows from findings in Section E on the utility and its suppliers. Our comments on these findings can be found in Enclosure B, below.

Response: The staff agrees with this recommendation and has stated this view in NUREG-0578, NUREG-0585 and elsewhere. We have also said that significant changes are required in the attention to operations reliability, and have proposed several specific but important changes in this area - see especially NUREG-0585.

Additional attention to the area of a licensee's competency to discharge his responsibilities is needed.

The staff is reviewing existing NRC requirements for the utility's management and technical competency including organizational and management standards. Sources of pertinent information include IE inspection reports and discussions with inspectors (including resident inspectors), LERS, staff analysis of operational activities and related aspects to identify and define needed organizational and management standards, and discussions with cognizant utility personnel.

The staff has established a program aimed to develop specific criteria for the management and technical capability of utilities to ensure a low likelihood of occurrence of accidents and to respond to accidents. The program includes a comparison of present utility resources against these criteria to identify the specific areas of weaknesses. Finally, the

Recommendation A-6 In order to provide an added contribution to safety, the agency should be required, to the maximum extent feasible, to locate new power plants in areas remote from concentrations of population. Siting determinations should be based on technical assessments of various classes of accidents that can take place, including those involving releases of low doses of radiation. (See recommendation F.2.)

Findings: This recommendation does not appear to be based on any of the Commission s findings, although there are references to siting in the overview section of the report. In the overview section, there is a recommendation that the low population zone (LPZ) concept be abandoned in siting and in emergency planning. Further, a variety of possible accidents should be considered during siting, particularly smaller accidents which have a higher probability of occurring. For each such accident, one should calculate probable levels of radiation release at various distances to decide the kind of protective action that is necessary and feasible. Only such analysis can predict the true consequences of a radiological incident and determine whether a particular site is suitable for a nuclear power plant.

Response: In NUREG-0625, Report of the Siting Policy Task Force, the staff recommended the abandonment of the LPZ approach whereby the low population distance is established on the basis of calculated radiation releases from a design basis accident. NUREG-0625 recommended that the LPZ be replaced by an Emergency Planning Zone. The depth of the emergency planning zone would be fixed by regulation, extending 10 miles from the plant rather than by being based on a calculation of radiation doses. NUREG-0625 also recommended that maximum population densities be established by regulation within the emergency planning zone and for some fixed distance beyond the emergency planning zone. The population density requirements would be based on a generic consideration of the consequences of Class 9 accidents. NUREG-0625 contained no recommendation

Recommendation A-7 The agency should be directed to include, as part of its licensing requirements, plans for the mitigation of the consequences of accidents, including the cleanup and recovery of the contaminated plant. The agency should be directed to review existing licenses and to set deadlines for accomplishing any necessary modifications. (See recommendations D.2 and D.4.)

Finding: This recommendation deals with accident mitigation and post-accident recovery and appears to result from various findings including A14 and A15. The report also indicates that such consideration should be given to the operating plants.

Response: We agree that planning for accident recovery should be given greater attention and note that some actions are already being taken as a result of the requirements flowing from NUREG-0578. In addition, an industry task group dealing with TMI-2 lessons learned is reviewing the capability of plants and industry advisory and support groups to deal with post-accident recovery matters. This recommendation should lead to further study by the staff and formulation of specific licensing requirements for accident recovery planning. We note that NUREG-0585, the final report of the Lessons Learned Task Force, goes well beyond this recommendation for increased planning by calling for training in core melt accident mitigation and rulemaking to establish requirements for design features to mitigate the consequences of degraded core and core melt accidents.

Recommendation A-9 The agency's authorization to make general rules affecting safety should:

a. require the development of a public agenda according to which rules will be formulated;

b. require the agency to set deadlines for resolving generic safety issues;

c. require a periodic and systematic reevaluation of the agency's existing rules; and

d. define rulemaking procedures designed to create a process that provides a meaningful opportunity for participation by interested persons, that ensures careful consideration and explanation of rules adopted by the agency, and that includes appropriate provision for the application of new rules to existing plants. In particular, the agency should: accompany newly proposed rules with an analysis of the issues they raise and provide an indication of the technical materials that are relevant; provide a sufficient opportunity for interested persons to evaluate and rebut materials relied on by the agency or submitted by others; explain its final rules fully, including responses to principal comments by the public, the ACRS, and other agencies on proposed rules; impose when necessary special interim safeguards for operating plants affected by generic safety rulemaking; and conduct systematic reviews of operating plants to assess the need for retroactive application of new safety requirements.

Findings: This recommendation deals with procedures for establishing general rules affecting safety and appears to follow from finding G2. It involves the formulation of rules, resolution of generic safety issues and timely review of existing rules. In addition, the rulemaking process, implementation, and backfitting action for operating plants are discussed. We generally agree with the findings.

Response: In many of the various aspects of other recommendations, staff actions are already ongoing that show staff recognition of the significance of these matters. In the final report on TMI-2 lessons learned, NUREG-0585, the question of safety goals and backfitting criteria are discussed and recommendations set forth. Unresolved safety issues are also addressed in NUREG-0585. The Presidential Commission does not acknowledge the action taken by NRR and the NRC last June

Recommendation A-10 Licensing procedures should foster early and meaningful resolution of safety issues before major financial commitments in construction can occur. In order to ensure that safety receives primary emphasis in licensing, and to eliminate repetitive consideration of some issues in that process, the Commission recommends the following:

a. Duplicative consideration of issues in several stages of one plant's licensing should, wherever possible, be reduced by allocating particular issues (such as the need for power to a single stage of the proceedings.

b. Issues that recur in many licensings should be resolved by rulemaking.

c. The agency should be authorized to conduct a combined construction permit and operating license hearing whenever plans can be made sufficiently complete at the construction permit stage.

d. There should be provision for the initial adjudication of license applications and for appeal to a board whose decisions would not be subject to further appeal to the administrator. Both initial adjudicators and appeal boards should have a clear mandate to pursue any safety issue, whether or not it is raised by a party.

e. An Office of Hearing Counsel should be established in the agency. This office would not engage in the informal negotiations between other staff and applicants that typically precede formal hearings on construction permits. Instead, it would participate in the formal hearings as an objective party, seeking to assure that vital safety issues are addressed and resolved. The office should report directly to the administrator and should be empowered to appeal any adverse licensing board determination to the appeal board.

f. Any specific safety issue left open in licensing proceedings should be resolved by a deadline.

Findings: These recommendations appear to result from the considerations discussed in findings G1, G2, G6 and G8g and concern for the resolution of safety issues early in the initial lifetime of a plant. Considerations include generic issue resolution, reulmaking actions, and timely resolution. The Commission believes that such initiatives would avoid subsequent repetitive considerations of safety issues.

Response: We generally agree with the thrust of the recommendations of the Commission. These are issues that have been discussed extensively

Recommendation A-11 The agency's inspection and enforcement functions must receive increased emphasis and improved management, including the following elements:

a. There should be an improved program for the systematic safety evaluation of currently operating plants, in order to assess compliance with current requirements, to assess the need to make new requirements retroactive to older plants, and to identify new safety issues.

b. There should be a program for the systematic assessment of experience in operating reactors, with special emphasis on discovering patterns in abnormal occurrences. An overall quality assurance measurement and reporting system based on this systematic assessment shall be developed to provide: 1) a measure of the overall improvement or decline in safety, and 2) a base for specific programs aimed at curing deficiencies and improving safety. Licensees must receive clear instructions on reporting requirements and clear communications summarizing the lessons of experience at other reactors.

c. The agency should be authorized and directed to assess substantial penalties for licensee failure to report new "safety-related" information or for violations of rules defining practices or conditions already known to be unsafe.

d. The agency should be directed to require its enforcement personnel to perform improved inspection and auditing of licensee compliance with regulations and to conduct major and unannounced on-site inspections of particular plants.

e. Each operating licensee should be subject periodically to intensive and open review of its performance according to the requirements of its license and applicable regulations.

f. The agency should be directed to adopt criteria for revocation of licenses, sanctions short of revocation such as probationary status, and kinds of safety violations requiring immediate plant shutdown or other operational safeguards.

Findings: These recommendations appear to result from the findings given in sections A and G3, G8h, G9c and G9t. They deal mostly with I&E actions. There are three matters that are discussed here because they interface with NRR.

Response: Item 11a deals with a program for the systematic safety evaluation of currently operating plants. The current SEP program

ENCLOSURE B

ANALYSES OF RECOMMENDATIONS FOR UTILITY AND ITS SUPPLIERS

RECOMMENDATION B1

To the extent that the industrial institutions we have examined are representative of the nuclear industry, the nuclear industry must dramatically change its attitudes toward safety and regulations. The Commission has recommended that the new regulatory agency prescribe strict standards. At the same time, the Commission recognizes that merely meeting the requirements of a government regulation does not guarantee safety. Therefore, the industry must also set and police its own standards of excellence to ensure the effective management and safe operation of nuclear power plants.

(a) The industry should establish a program that specifies appropriate safety standards including those for management, quality assurance, and operating procedures and practices, and that conducts independent evaluations. The recently created Institute of Nuclear Power Operations, or some similar organization, may be an appropriate vehicle for establishing and implementing this program.

(b) There must be a systematic gathering, review, and analysis of operating experience at all nuclear power plants coupled with an industry-wide international communications network to facilitate the speedy flow of this information to affected parties. If such experiences indicate the need for modifications in design or operation, such changes should be implemented according to realistic deadlines.

We concur in part with finding E4. The following are the clarifications and exceptions:

(1) With respect to finding E4a, the emphasis on the condensate polisher as a transient initiator is unwarranted. Other events, such as loss-of-offsite power (which is clearly not subject to complete "safety-related" standards) could also have been an initiator. We agree completely on the role of the PORV, which was addressed in NUREG-0578. In addition to NRC requiring a quality assurance program for safety-related items, NRC also requires under Appendix A to 10 CFR Part 50 that items important to safety be under a quality assurance program (GDC #1).

With regard to independent review, the TMI-2 quality assurance does require the Met Ed's QA organization to concur with implementing procedures including maintenance, repairs, and modifications (Ref. Table 17.2-2 of QA program in docket).

(2) With respect to finding E4b, Regulatory Guide 1.33 and ANSI N18.7-1972 which Met Ed commits to require an independent offsite review of certain operating procedures as described in ANSI N18.7-1972 (Section 4.0).

(3) With respect to finding E4d, we note that the same statement is probably applicable to many, if not all, other utilities.

As discussed in response to finding E4a, NRC does require a quality assurance program to be applied to items important to safety (Ref. Appendix A to 10 CFR Part 50, GDC #1).

We concur with finding E5 with the following comments:

(1) With respect to finding E5a(ii) we note that this matter was addressed in NUREG-0578.

(2) With respect to finding E5b, more extensive review of small-break LOCA guidelines and procedures has been one of the themes of the Bulletins and Orders Task Force since June 1979. This subject is also addressed in NUREG-0578.

With respect to Recommendation 1(b), we agree with this recommendation. Efforts in progress, both by NRC and by the industry, are described below.

Prior to the accident at TMI-2 in March, there were efforts directed at the assessment of reactor operating experience. However, investigations initiated as a result of the TMI-2 accident and other studies show that improvements were necessary in the way that operating experience is collected, analyzed, documented and fedback to strengthen reactor operations and the licensing process.

Some of the actions that have been initiated over the past several months include:

<u>NRC</u>

1. An agency-wide Task Force was appointed in May 1979 to examine the NRC activities directed at the analysis and evaluation of operational data. The Task Force report was completed in June 1979.
2. In July 1979, acting on recommendations from the Task Force, the Commission acted to:
   a) Establish an agency-wide operational Data Analysis and Evaluation Office reporting directly to the EDO. This office is to be staffed with 15-20 senior, experienced professionals. The charter of this office is attached. The office has the stature of a division in a major program office.
   b) Direct that the individual program offices also establish an operational data analysis capability. The program offices were to make input to the agency-wide office, comment on the agency-wide office evaluations, and perform special operational safety data analyses.

Industry

1. A Nuclear Safety Analysis Center (NSAC) under EPRI has been established to systematically review available plant event reports and data. The objectives and program summary of NSAC are attached NSAC's effort is being directed at identifying possible precursor events; identifying trends and problem areas; performing failure analyses efforts; and promoting follow-up with utilities on identified problem areas.
2. The electric utility industry has established an Institute of Nuclear Power Operations (INPO) to ensure a high quality of operation in nuclear power plants. A specific function of INPO is to review nuclear power operating experiences for analysis and feedback to utilities; incorporate lessons learned from such reviews into training programs; and coordinate reporting and analysis with other organizations. INPO intends to sponsor studies and analysis, including human factor studies, in support of reactor operations. INPO will likely have an overall staff of about 200 and an operating budget of about 11 million dollars. Additional information on INPO is attached.
3. Each reactor manufacturer is reviewing his program for the review and feedback of experience to improve operational safety and availability. Discussions with some vendors indicated that programs towards this objective are being [illegible] and integrated into other programs with outside organizations.

Licensees

Each operating power reactor licensee has been required to establish an engineering staff capability to assess and feedback pertinent operating experience by January 1, 1980. One acceptable means of supplying this capability is the creation of a Shift Technical Advisor position on each shift. This well-qualified individual will perform two functions -

Recommendation B.3

Integration of management responsibility at all levels must be achieved consistently throughout this industry. Although there may not be a single optimal management structure for nuclear power plant operation, there must be a single accountable organization with the requisite expertise to take responsibility for the integrated management of the design, construction, operation, and emergency response functions, and the organizational entities that carry them out. Without such demonstrated competence, a power plant operating company should not qualify to receive an operating license.

(a) These goals may be obtained at the design stage by 1) contracting for a "turn-key" plant in which the vendor or architect-engineer contracts to supply a fully operational plant and supervises all planning, construction, and modification; or 2) assembling expertise capable of integrating the design process. In either case, it is critical that the knowledge and expertise gained during design and construction of the plant be effectively transferred to those responsible for operating the plant.

(b) Clearly defined roles and responsibilities for operating procedures and practices must be established to ensure accountability and smooth communication.

(c) Since, under our recommendations, accountability for operations during an emergency would rest on the licensee, the licensee must prepare clear procedures defining management roles and responsibilities in the event of a crisis.

Findings

It appears that findings E2 and E3 led to this recommendation. Our comments on findings E2 and E3 are included in the discussion of recommendation B.1 above.

Response

We agree with Recommendation B.3. Additional discussion of the recommendatio and the work already in progress related to the recommendation is provided below.

Although we agree, we must point out that some utilities must, at some point in time, be designing, constructing, and operating their first plant. Such a utility can obtain expertise elsewhere, but still may merit special attention both by us and by itself. We are currently conducting a study of the resources of each utility with operating power reactors and are developing new criteria by which to judge the acceptability of these utilities. We have contracted with Teknetron Research Inc. to provide an independent input to this study and expect to have new standards by which to judge the utility capability by April 1980. A more detailed description of this program is presented in a memorandum to NRC

RECOMMENDATION B.5

Substantially more attention and care must be devoted to the writing, reviewing, and monitoring of plant procedures.

(a) The wording of procedures must be clear and concise.

(b) The content of procedures must reflect both engineering thinking and operating practicalities.

(c) The format of procedures, particularly those that deal with abnormal conditions and emergencies, must be especially clear, including clear diagnostic instructions for identifying the particular abnormal conditions confronting the operators.

(d) Management of both utilities and suppliers must insist on the early diagnosis and resolution of safety questions that arise in plant operations. They must also establish deadlines, impose sanctions for the failure to observe such deadlines, and make certain that the results of the diagnoses and any proposed procedural changes based on them are disseminated to those who need to know them.

Findings

It appears that findings E1, E4 and E5 led to this recommendation. Our comments of these findings are included in the discussion or recommendation B.1 above.

Response

We agree with Recommendation B.5. Additional discussion of the recommendation and the work already in progress related to the recommendation are provided below.

With respect to Recommendations 5(a), 5(b), and 5(c), we agree. NUREG-0585, TMI Lessons Learned Task Force Final Report, Recommendation 4 would require NRC review of emergency procedures by interdisciplinary review groups for each plant and would require the inclusion of people-oriented sciences (education, training, psychology) in the overall upgrading of emergency procedures.

ENCLOSURE C

ANALYSIS OF RECOMMENDATIONS FOR TRAINING OF OPERATING PERSONNEL

Recommendation C-1

The Commission recommends the establishment of agency-accredited training institutions for operators and immediate supervisors of operators. These institutions should have highly qualified instructors, who will maintain high standards, stress understanding of the fundamentals of nuclear power plants and the possible health effects of nuclear power, and who will train operators to respond to emergencies. (See recommendation A.4.a)

a. These institutions could be national, regional, or specific to individual nuclear steam systems.

b. Reactor operators should be required to graduate from an accredited training institution. Exemption should be made only in cases where there is clear, documentary evidence that the candidate already has the equivalent training.

c. The training institutions should be subject to periodic review and reaccreditation by the restructured NRC.

d. Candidates for the training institute must meet entrance requirements geared to the curriculum.

Findings: This recommendation is apparently based on findings F-1, F-2, F-3, F-4 and F-5. We agree with these findings in that the training was deficient, but do not agree with the characterization that the training was shallow and low quality.

Response: Although the purpose of this recommendation is the same as the purpose of related recommendations by the staff, i.e. to have highly qualified instructors to teach the fundamentals of nuclear engineering and the potential hazards of nuclear power to appropriately qualified candidates for operator and senior operator licenses, the means of achieving the purpose differ. The recommendation would restrict training of operators to a few institutions, separate from the utilities, closely controlled by the regulatory agency and presumably with a uniform and high standard of achievement.

Recommendation C-2

Individual utilities should be responsible for training operators who are graduates of accredited institutions in the specifics of operating a particular plant. These operators should be examined and licensed by the restructured NRC, both at their initial licensing and at the relicensing stage. In order to be licensed, operators must pass every portion of the examination. Supervisors of operators, at a minimum, should have the same training as operators.

Findings. This recommendation is apparently based on finding F-3a, F-3c and F-4c. We agree with these findings,

Response: Utilities are now responsible for training operators in the specifics of a particular plant as recommended. Operators are now only initially examined and licensed by the NRC. The staff has recommended that the NRC examine approximately 10% of the requalifications and three Commissioners have recommended that 100% be done by the NRC. (SECY-330E Option 11 and 12)

The staff has recommended increasing the overall passing grade from 70% to 80% and require at least 70% in each category. The staff now requires supervisors to have at least the same training as operators. Direct supervisors must be licensed, and managers, such as the station superintendent or unit manager, must have equivalent training but not necessarily licensed. The staff has also recommended (LLTF LT 1.6) upgrading the qualifications of supervisors to require a BS degree or equivalent, increase the training to include specific training in accident response and increase the experience in operating a nuclear plant before being considered as a supervisor.

Recommendation C-4

Research and development should be carried out on improving simulation and simulation systems: a) to establish and sustain a higher level of realism in the training of operators, including dealing with transients; and b) to improve the diagnostics and general knowledge of nuclear power plant systems.

Findings: This recommendation is apparently based on finding F-5b.; we agree with this finding.

Response: Research and development work can and should be done to improve simulation systems for training and diagnostic purposes. On a shorter term basis, existing simulators can be programmed to provide effective simulation of sequences like TMI and other situations involving multiple equipment failures and operator errors. This should be done on an interim basis.

In connection with IE Bulletins, the B&W training simulator was programmed to handle the TMI sequence of events, and operators of B&W operating plants have been required to undergo re-training on that simulator.

SECY-79-330E recommends that explicit requirements regarding exercises be included in simulator training. These would cover a broad spectrum of normal and abnormal operations and response to transients involving multiple failures, compound abnormalities and imperfect initialization. Very effective training for abnormal situations can be done on existing simulators if they are programmed properly. Further improvements can be made with more sophisticated simulator systems.

The Office of Nuclear Regulatory Research is initiating a study that will explore the possible use by NRC of a hybrid engineering simulator system.

ENCLOSURE D

ANALYSIS OF RECOMMENDATIONS FOR TECHNICAL ASSESSMENT

Recommendation D-1

Equipment should be reviewed from the point of view of providing information to operators to help them prevent accidents and to cope with accidents when they occur. Included might be instruments that can provide proper warning and diagnostic information; for example, the measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions, and indication of the actual position of valves. Computer technology should be used for the clear display for operators and shift supervisors of key measurements relevant to accident conditions, together with diagnostic warnings of conditions.

In the interim, consideration should be given to requiring, at TMI and similar plants, the grouping of these key measurements, including distinct warning signals on a single panel available to a specified operator and the providing of a duplicate panel of these key measurements and warnings in the shift supervisor's office.

Findings: This recommendation is apparently based on findings A-5, A-7, A-8, G-8e and G-8f We agree with these findings.

Response: The staff has made recommendations for improving the information available to operators. First, (LLTF ST 2.1.3b) licensees are to implement procedures and operator training for prompt recognition of low reactor coolant level and inadequate core cooling using existing reactor instrumentation or any additional instruments they propose. Second, and in the longer term, the LLTF recommended (LLTF LT 7) 1) that licensees conduct a one-year review of their control rooms to evaluate adequacy of the type and arrangement of instrument and controls, 2) that a minimum set of plant safety status parameters be defined and displayed, 3) that the NRC study possible requirements for disturbance analysis systems in nuclear power plants, and 4) that the design requirements for a standard control room be expeditiously completed by IEEE.

Improvements in instruments needed to cope with accidents are also being addressed. Licensees are to install some critical post-accident monitoring

Recommendation D-2

2. Equipment design and maintenance inadequacies noted at TMI should be reviewed from the point of view of mitigating the consequences of accidents. Inadequacies noted in the following should be corrected: iodine filters, the hydrogen recombiner, the vent gas system, containment isolation, reading of water levels in the containment isolation, reading of water levels in the containment area, radiation monitoring in the containment building, and the capability to take and quickly analyze samples of containment atmosphere and water in various places. (See recommendation A.7.)

Findings: This recommendation is apparently based on findings A-11 and E-5n (iodine filters) and A-13 (vent gas system). We agree with these findings. No findings relating to the hydrogen recombiner or containment isolation water level, radiation monitoring or sampling were identified

Response: The following actions for correcting equipment inadequacies have been taken:

Iodine Filters: Safety related filters, which the TMI auxiliary building filters were not, are currently required to be tested. The requirement for safety-related filters on the auxiliary building depends on the adoption of a rule proposed by the LLTF that would require equipment to mitigate the consequences of accidents, such as the TMI-type, that are beyond the current design basis accidents (LLTF LT 10).

Hydrogen Recombiner: The staff has made both short and long term recommendations for hydrogen control. In the short term, licensees who have hydrogen recombiners are to modify, if necessary, the containment penetrations used by the recombiners so as to be single failure proof (LLTF ST 2.1.6a) and to review and modify, if necessary, the shielding and access to recombiners under accident conditions (LLTF ST 2.1.6.b). In the long term, a requirement for recombiners on all plants depends on

Recommendation D-3

Monitoring instruments and recording equipment should be provided to record continuously all critical plant measurements and conditions.

Findings: This recommendation is .pparently based on finding A-9.

Response: The LLTF has recommended that the set of instruments required to monitor and record critical plant parameters be developed by each licensee (LLTF LT 7.2). When defined, these instruments will be required to be installed at an onsite technical support center and be capable of being transmitted, (wholly or in part) to the NRC (LLTF ST 2.2.2.b). A contract between the Office of Research and Sandia Corporation has work underway to develop the generic description of information to be transmitted to the NRC and the associated communications systems.

D-4 continued

a) Licensees were required some months ago to analyze small-break accidents with multiple failures and operator errors. These studies are now being completed and submitted.

b) No finding relating to this recommendation was identified, but the intent is interpreted to be an emphasis on planning. Our response to recommendation A.7, above, treats this subject.

c) The LLTF recommended that the staff systematically assess the safety systems in operating plants (LLTF LT 8) and RES is proposing an Integrated Reliability Evaluation Program (IREP) to accomplish this.

c (i) Licensees have already been required to design and install systems to vent noncondensible gases from the reactor coolant systems (Denton 8/20/79 memo).

c (ii) Licensees have already been required to survey all systems that would or could contain highly radioactive fluid, implement measures to reduce and maintain leakage at as-low-as-practical levels (see recommendation D2) (LLTS ST 2.1.6.a). Enclosing the let-down system with a capability to return gases to the containment could be the result of the LLTF proposed rule to require equipment to mitigate accidents beyond the current design basis accidents (LLTF LT 10).

Recommendation D-7

The Commission recommends that as a part of the formal safety assurance program, every accident or every new abnormal event be carefully screened, and where appropriate be rigorously investigated, to assess its implications for the existing system design, computer models of the system, equipment design and quality, operations, operator training, operator training simulators, plant procedures, safety systems, emergency measures, management, and regulatory requirements.

Findings: This recommendation is apparently based on findings A7, E5c and G3. We agree with these findings.

Response: The NRC has established groups, which are now being organized and staffed, within EDO and each affected program office to assimilate, evaluate, and disseminate operating experience. All licensees have been required to establish similar evaluation groups by January 1, 1980 (LLTF ST 2.2.1.b). The industry is establishing a national center for evaluating experience under NSAC. The LLTF recommended that a nation-wide integrated NRC/utility program to evaluate operating experience be established. The NRC program is described in greater detail in response to recommendation B.1.b, above. The intent is to feedback the results of evaluating operating experience to all of the areas listed by the Presidential Commission in recommendation D-7.

currently participating in an interagency committee effort in this area, stimulated by the "Libassi Report," by Congressional committee reviews on effects of low levels of ionizing radiation, by the issuance of the NAS BEIR III report draft and by NRC's own reorganization (e.g., establishment in November 1978 of the Radiological Health Standards Branch), to result in better interagency coordination of such research and better application of available funding. The Radiological Health Standards Branch has initiated a research program to study the feasibility of an effective and meaningful epidemiological study of (primarily) workers in the nuclear industry. This will be further discussed below.

b. None of the findings of Section B on Health Effects and Section C on Public Health suggest further research on acceptable levels of exposure to ionizing radiation for either the general population or for workers. However, in relation to E.1.a. above, the Libassi Report has recommended collecting more data on worker exposures and on many related worker health parameters, because worker collective exposures represent a greater pool of additional exposure than other organized and monitored communities of people. Even so, this pool only is exposed to about 50,000 person-rem per year for about 100,000 persons working in nuclear related activities, whereas medical exposure results in 20,000,000 person-rem per year in the whole U. S. population.

The EPA, NRC, HEW, DOE, DOD and OSHA have been considering for several years, the acceptable level of radiation exposure and a joint EPA, NRC, OSHA public hearing on this issue is in the final stages of preparation. There is much discussion today as to whether permissible levels of radiation exposure to either the general population or to workers should be changed from a health effects viewpoint. Public hearings on this subject are currently anticipated for late winter 1979-1980.

there are many other mitigation measures. These include evacuation, sheltering, interdiction of foodstuffs, respiratory protection, and protective clothing. There have been studies that have quantified the effectiveness of some of these measures. Nonetheless, the staff agrees that more needs to be done, and in particular we are currently reviewing the pros and cons related to broad KI distribution and re-reviewing the quantitative studies on other measures, as part of the overall emergency preparedness upgrading.

e. None of the findings of the Commission Study of the TMI-2 accident specifically relate to researching genetic or environmental factors that predispose individuals to increased susceptibility to adverse effects. It is well recognized that one of the most difficult aspects of epidemiological studies on radiation effects is separating out other causative factors in cancer induction from the low level radiation effects. The NRC supported study mentioned in C above includes trying to identify all the environmental or genetic factors that ought to be monitored in an effective epidemiological study on effects of radiation exposure.

A Presidential Executive Order is in the process of being issued that directs HHS to coordinate radiation health effects research. NRC has members on an already established interagency health effects of ionizing radiation research coordinating committee that is chaired by HHS.

of Section D. on Emergency Response. The NRC management and staff have recognized the shortcomings of previous emergency preparedness planning in numerous ways since the TMI-2 accident.

Response

At the present time, the most active examples of this recognition are the efforts of a special NRR Task Force in backfitting the emergency preparedness recommendations of Regulatory Guide 1.101 and NUREG-75/111 on all operating plants, and the concommitant effort in the Office of State Programs in making sure that each state with an operating nuclear power plant has an approved emergency preparedness program in accordance with NUREG-75/111. Appropriate training of health professionals and emergency response personnel is a key factor in such programs. The staff is thus in full accord with the Commission in the recommendations of this item.

Recommendation E-4 Utilities must make sufficient advance preparation for the mitigation of emergencies:

a. Radiation monitors should be available for monitoring of routine operations as well as accident levels.

b. The emergency control center for health-physics operations and the analytical laboratory to be used in emergencies should be located in a well-shielded area supplied with uncontaminated air.

c. There must be a sufficient health-related supply of instruments, respirators, and other necessary equipment for both routine and emergency conditions.

d. There should be an adequate maintenance program for all such health-related equipment.

Findings. Recommendations of E.4. regarding utilities making sufficient advance preparation for mitigation of emergencies flow from Findings C8, B2, C9, B1 and perhaps others by implication. The staff agrees with the findings in general recovered (decontaminated) only a few days after the accident, and their contamination with airborne radioactivity did not result in a significant health hazard (radiation exposure of workers). Further staff review of this item has been anticipated and should be accelerated as a result of the Presidential Commission recommendation. The CP review of the design of the plant includes consideration of whether the above mentioned areas are adequately shielded, are located far from sources of radiation, and are ventilated so as to be upstream of potential sources of radioactivity. The accident showed that breaching of some supposed barriers to radioactive material flow occurred. These are the areas that need to be further considered by the NRC staff.

c. The NRC staff members that arrived on the scene shortly after the accident noted the items identified in E.4c in their recommendations to Met Ed management. Recommendations by the staff for corrective measures regarding these items were made to Met Ed management on many occasions after the accident. The staff has given attention and concern to its own regulatory guidance in these regards as well and is currently considering whether specific requirements on numbers of (operational) survey instruments, respirators, etc. should be designated. It is recognized that in the past, the NRC guidance has related more to normal operation, considering that accidents were of very low probability. The staff is now revising its entire radiation protection review process to better accommodate the possibility that accidents will occur and will need different kinds of protective measures and equipment.

ENCLOSURE F

ANALYSIS OF RECOMMENDATIONS FOR EMERGENCY PLANNING AND RESPONSE

Recommendation F-1

Eme gency plans must detail clearly and consistently the actions public officials and utilities should take in the event of off-site radiation doses resulting from release of radioactivity. Therefore, the Commission recommends that:

a. Before a utility is granted an operating license for a new nuclear power plant, the state within which that plant is to be sited must have an emergency response plan reviewed and approved by the Federal Emergency Management Agency (FEMA). The agency should assess the criteria and procedures now used for evaluating state and local government plans and for determining their ability to activate the plans. FEMA must assure adequate provision, where necessary, for multi-state planning.

b. The responsibility at the federal level for radiological emergency planning, including planning for coping with radiological releases, should rest with FEMA. In this process, FEMA should consult with other agencies, including the restructured NRC and the appropriate health and environmental agencies. (See recommendation A.4.)

c. The state must effectively coordinate its planning with the utility and with local officials in the area where the plant is to be located.

d. States with plans already operating must upgrade their plans to the requirements to be set by FEMA. Strict deadlines must be established to accomplish this goal.

Findings

The recommendations flow, in part, from findings D.4 and D.7 with which we agree in substance.

Response

We agree with the recommendation that FEMA should continue to play the lead role at the Federal level for emergency planning.

We agree with the substance of the recommendations in item F.1. and have efforts now underway to accomplish these objectives. These include rule-making to upgrade emergency preparedness standards and efforts to promptly

Recommendation F-2

Plans for protecting the public in the event of off-site radiation releases should be based on technical assessment of various classes of accidents that can take place at a given plant.

a. No single plan based on a fixed set of distances and a fixed set of responses can be adequate. Planning should involve the identification of several different kinds of accidents with different possible radiation consequences. For each such scenario, there should be clearly identified criteria for the appropriate responses at various distances, including instructing individuals to stay indoors for a period of time, providing special medication, or ordering an evacuation.

b. Similarly, response plans should be keyed to various possible scenarios and activated when the nature and potential hazard of a given accident has been identified.

c. Plans should exist for protecting the public at radiation levels lower than those currently used in NRC-prescribed plans.

d. All local communities should have funds and technical support adequate for preparing the kinds of plans described above.

Finding

The recommendation flows, in part, from findings D.10 and D.16 with which we agree. It should be noted that, in practice, NRC staff review of response plans often extended to organizations beyond those within the LPZ.

Response

The basis for emergency response planning has been under examination at the NRC for some time. An NRC/EPA task force published an extensive treatment of this subject in December, 1978 (NUREG-0396; EPA 520/1-78-016) with conclusions consistent with recommendation F.2. The task force concluded that "A spectrum of accidents (not the source term from a single accident

Recommendation F-3

Research should be expanded on medical means of protecting the public against various levels and types of radiation. This research should include exploration of appropriate medications that can protect against or counteract radiation.

Finding

The recommendation flows from finding C.4. with which we agree.

Response

We agree with the recommendation and defer to the Office of Research to take the appropriate NRC staff action and coordinate with HHS on agency leads in this area.

Recommendation F-4

If emergency planning and response to a radiation-related emergency is to be effective, the public must be better informed about nuclear power. The Commission recommends a program to educate the public on how nuclear power plants operate, on radiation and its health effects, and on protective actions against radiation. Those who would be affected by such emergency planning must have clear information on actions they would be required to take in an emergency.

Finding

The recommendation flows from finding H.1. with which we agree.

Response

It is not clear to what agency the recommendation for programs to educate the public on how nuclear power plants operate and on basic radiation health effects is directed. Whether the NRC would be perceived as promotional if it undertook programs in this area will need to be considered.

The NRC efforts underway to promptly upgrade emergency preparedness capabilities will include a requirement for licensees to make the public aware on a continuing basis of the nature of hazards in a radiological emergency and of the actions which they would be expected to take in such an emergency. The performance of periodic response drills by local and State organizations should contribute to this awareness.

ENCLOSURE G

NRR ACTIVITIES TO DATE ON OPERATING PLANTS

Following March 28, 1979, a significant number of NRR actions have been taken. These activities have been directed towards the generation of requirements/positions and their communication and implementing by the industry. The NRC's principal thrust has been towards operating reactors since those are the only plants that have the immediate potential for impacting the health and safety of the public.

The following sections summarize NRR activities and accomplishments over the period since the Three Mile Island accident. These include:

I. Lessons Learned
- Short Term Report and Implementation on Operating Reactors
- Long Term Report

II. Bulletins and Orders

III. Operator Training

IV. Emergency Preparedeness

Of the 125 individual recommendations contained in the President's Commission report, about half were already being considered, evaluated, or implemented prior to the issuance of the report. Implementation of many of these items have been in progress for some time.

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- Nine bulletins were issued to operating plants to assure the licensees understood and implemented the immediate concerns arising from the TMI-2 accident.

- Five Orders were issued to seven B&W operating plants to upgrade AFW system reliability, install anticipatory reactor trips, develop new procedures for small-break LOCA and retrain operators.

- Issued new requirements for AFW systems for nineteen W plants and four C-E plants in 10/79. This effort is continuing for the remaining ten W and C-E operating plants with completion expected in 11/79.

- Reviewed and approved small-break LOCA guidelines for 25 GE plants. Approval expected in early 11/79 on W and C-E operating plants.

- Generic report on B&W plants NUREG-0560 issued 5/79. Generic reports expected on W, C-E, and BWR plants in 11/79 and 12/79.

ENCLOSURE H

SUMMARY OF RELATED NRR ACTIONS

PRESIDENT'S COMMISSION REPORT RECOMMENDATION	RELATED ONGOING OR PLANNED NRR ACTIONS
A.1	None
A.1.a	None
A.1.b	None
A.1.c	None
A.1.d	None
A.2	None
A.2.a	None
A.2.b	None
A.3	None
A.3.a	None
A.3.b	None
A.3.c	None
A.4	. SECY 77-388 . SECY 79-8 . NRR Office Letter No. 16
A.4.a	. SECY 79-330E . LTLL - Rec. 1.2; 1.4; 1.6; 1.8
A.4.b	. STLL - Rec. 2.1.1; 2.1.3.b; 2.1.6.a . LTLL - Rec. 9 . Bulletins & Orders, AFW requirements

PRESIDENT'S COMMISSION REPORT RECOMMENDATION	RELATED ONGOING OR PLANNED NRR ACTIONS
A.9.b	. Section 210 Energy Reorganization Act Requires Planning and Reporting to Congress . LTLL REC.12 (7)
A.9.c	None
A.9.d	. Systematic Evaluation Program . SECY 79-8 . LTLL REC. 11.
A.10	None
A.10.a	. NUREG 292, Recommendation for Improving the Licensing Process
A.10.b	. NUREG 292
A.10.c	None
A.10.d	None
A.10.e	None
A.10.f	. License condition for plant specific
A.11	None
A.11.a	. SEP . IREP
A.11.b	. SECY-79-371

PRESIDENT'S COMMISSION REPORT RECOMMENDATION	RELATED ONGOING OR PLANNED NRR ACTIONS
B.5	. LTLL - Rec. 4
B.5.a	. LTLL - Rec. 4 . Bulletins & Orders work on emergency procedure - NUREG-0611
B.5.b	. LTLL - Rec. 4 NUREG-0611
B.5.c	. NUREG-0611 . STLL - Rec. 2.1.9 . LTLL - Rec. 4
B.5.d	. LTLL - Rec. 6 . SECY 79-371 (NRC Action Only) . STLL - Rec. 2.2.1.b
B.6	None
C.1	. LTLL - Rec. 1 . Contract, RFPA 80-117
C.1.a	. LTLL - Sec. 2.3.1
C.1.b	. LTLL - Sec. 2.3.1
C.1.c	. LTLL - Sec. 2.3.1
C.1.d	. LTLL - Sec. 2.3.1
C.2	. SECY 79-330E, option 11; Rec. 13 . LTLL - Rec. 1.2, 1.4
C.3	. SECY 79-330E, Rec. 7 . LTLL - Rec. 1.4

PRESIDENT'S COMMISSION REPORT RECOMMENDATION	RELATED ONGOING OR PLANNED NRR ACTIONS
D.4.c	. LTLL - Sec. 3.3
D.4.c.(i)	. Denton's Memo 8-20-79 - High Point Vent . LTLL - Rec. 10
D.4.c.(ii)	. STLL - Rec. 2.1.6.a; 2.1.6.b . LTLL - Rec. 10
D.4.d	None
D.5	None - Resolved by prior research
D.6	. TMI-2 Order . TMI Support Task Force
D.7	. LTLL - Rec. 6.1, 6.2 . ERA - Section 208 . SECY 79-371
E.1	. Research User Request - NRR 77-13
E.1.a	None
E.1.b	. EPA Interagency Exposure Limit Task Force . Libassi study participation
E.1.c	. Research User Request NRR 79-3
E.1.d	. Working with FDA
E.1.e	None

PRESIDENT'S COMMISSION REPORT RECOMMENDATION	RELATED ONGOING OR PLANNED NRR ACTIONS
F.2.a	. Emergency Preparedness Review Guidelines No. 3
F.2.b	. Emergency Preparedness Review Guidelines No. 3
F.2.c	None
F.2.d	. NUREG-0553
F.3	None
F.4	. Regulatory Guide 1.101
F.5	None
F.6	. SECY 79-591 . Regulatory Guide 1.101
G.1	None
G.1.a	None
G.1.b	None
G.2	None
G.2.a	None
G.2.b	None

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- MEMO DENTON TO GOSSICK 11/01/79
- PRELIMINARY EVALUATION OF KEMENY COMMISSION RECOMMENDATIONS
- TOTAL OF 44 RECOMMENDATIONS
- TOTAL OF 33 WITHIN THE PURVIEW OF NRR
- INITIATED IMPROVEMENTS RELATING TO ALL OR PART OF 28 OF THE 30 RECOMMENDATIONS APPLICABLE TO ORs
- INITIATED IMPROVEMENTS RELATING TO ALL OR PART OF 31 OF THE 33 RECOMMENDATIONS APPLICABLE TO CPs AND OLs
- REPORT HIGHLIGHTS ONE additional MAJOR SHORTCOMING (A8)
- PLAN TO REQUIRE AN AUGMENTED ONSITE TECHNICAL STAFF FOR OPERATIONAL SAFETY SURVEILLANCE FOR ALL LICENSEES
- WOULD REPORT TO HIGHER MANAGEMENT INDEPENDENT OF PRODUCTION STAFF
- MAJOR DISAGREEMENT WITH FINDING G.12 I.E., NRC UNABLE TO FULFILL ITS RESPONSIBILITY FOR PROVIDING ACCEPTABLE LEVEL OF SAFETY
- ALLOCATION OF NRR RESOURCES TO CONSIDER IMPLEMENTATION OF RECOMMENDATIONS ON ORs WILL RESULT IN FURTHER PAUSE IN LICENSING OF CPs AND OLs

RECOMMENDATION A.1

- ABOLISH PRESENT 5-MAN COMMISSION
- REPLACE WITH SINGLE ADMINISTRATOR
  - APPOINTED BY PRESIDENT
  - SUBSTANTIAL TERM NOT COTERMINOUS WITH THAT OF PRESIDENT
  - MAY BE REMOVED BY PRESIDENT
  - SELECTED FROM OUTSIDE CURRENT AGENCY
- ADMINISTRATOR SHOULD HAVE SUBSTANTIAL AUTHORITY
  - INTERNAL ORGANIZATION
  - MANAGEMENT
  - PERSONNEL TRANSFERS FROM CURRENT AGENCY
  - LOCATED IN SAME BUILDING
- ASSURE OFFICES COMMUNICATE TO AFFECT OVERALL PERFORMANCE

ONGOING OR PLANNED ACTIONS

NOT WITHIN PURVIEW OF NRR

RECOMMENDATION A.2

- ESTABLISH OVERSIGHT COMMITTEE
  - ONGOING EXAMINATION OF AGENCY AND INDUSTRY
  - ASSURE IMPORTANT SAFETY ISSUES ARE ADDRESSED
  - ASSURE OVERALL RISKS OF NUCLEAR POWER ARE EXPLORED
- COMMITTEE MEMBERS SHOULD HAVE DIVERSE BACKGROUND
- REPORT TO PRESIDENT AND CONGRESS AT LEAST ANNUALLY

ONGOING OR PLANNED ACTIONS

NOT WITHIN PURVIEW OF NRR

RECOMMENDATION A.3

- ACRS SHOULD BE RETAINED AND STRENGTHENED
- MEMBERS SHOULD CONTINUE TO BE PART-TIME EMPLOYEES
- FOLLOWING CHANGES PROPOSED
  - INCREASED STAFF FOR INDEPENDENT ANALYSES
  - INCREASED CAPABILITY IN PUBLIC HEALTH FIELD
  - NOT REQUIRED TO REVIEW EACH LICENSE APPLICATION
  - MEMBERS MAY INTERVENE IN HEARINGS
  - MEMBERS AUTHORIZED TO TESTIFY AT HEARINGS
  - EXEMPT FROM SUBPOENA
  - SIMILAR RIGHTS IN RULEMAKING PROCEEDINGS
  - MAY INITIATE RULEMAKING TO RESOLVE ANY GENERIC ISSUES IT IDENTIFIES

ONGOING OR PLANNED ACTIONS

NOT WITHIN PURVIEW OF NRR

RECOMMENDATION A.4

- AGENCY SHOULD BE REQUIRED TO ESTABLISH AND EXPLAIN SAFETY-COST TRADE-OFFS
- IDENTIFY AND REMOVE UNNECESSARY AGENCY RESPONSIBILITIES NOT GERMANE TO SAFETY
- EMPHASIZE RELATIONSHIP OF NEW AGENCY'S SAFETY ACTIVITIES TO RELATED ACTIVITIES OF OTHER AGENCIES
  - (A) AGENCY SHOULD UPGRADE OPERATOR AND SUPERVISOR LICENSING FUNCTIONS
  - (B) AGENCY SHOULD EMPLOY BROADER DEFINITION OF MATTERS RELATED TO SAFETY
  - (C) OTHER SAFETY EMPHASES
    - (I) SYSTEMS INTERACTION
    - (II) CONTROL ROOM DESIGN
    - (III) INCREASED SAFETY RESEARCH CAPACITY

ONGOING OR PLANNED ACTIONS

SECY 77-388, SECY 79-8, NRR OFFICE LETTER NO. 16

- (A) SECY 79-330E, NUREG-0585 (REC. 1.2, 1.4, 1.6, 1.8)
- (B) NUREG-0578 (REC. 2.1.1, 2.1.3B, 2.1.6A), NUREG-0585 (REC. 9), B&OTF ACTIVITIES AFW SYSTEMS
- (C) (I) NUREG-0585 (REC. 8, 12), USI A-17, INTEGRATED RELIABILITY EVALUATION PROGRAM
  - (II) NUREG-0578 (REC. 2.1.3A, 2.1.3B, 2.1.7B, 2.1.8B, 2.1.9), NUREG-0585 (REC.7.1 - 7.5), R.G. 1.97 REVISION
  - (III) REVISED RESEARCH USER REQUEST PROCEDURE; INCREASED USE OF RIL

RECOMMENDATION A.5

ESTABLISH AND ENFORCE HIGHER ORGANIZATIONAL AND MANAGEMENT STANDARDS FOR LICENSEES

ONGOING OR PLANNED ACTIONS

- NUREG-0578 (Rec. 2.2.1 a-c, 2.2.2 a-c, 2.2.3)
- NUREG-0585 (Rec. 2,5,6)
- QAB SURVEY & CRITERIA
- R. G. 1.16

RECOMMENDATION A.6

- LOCATE NEW POWER PLANTS IN REMOTE AREAS
- SITING DETERMINATION SHOULD CONSIDER LOW LEVEL RELEASES

ONGOING OR PLANNED ACTIONS

REPORT OF SITING POLICY TASK FORCE

RECOMMENDATION A.7

- CONSIDER AS PART OF LICENSING REQUIREMENTS CLEANUP AND RECOVERY OF CONTAMINATED PLANT
- REVIEW EXISTING LICENSES; SET DEADLINE FOR MODIFICATIONS

ONGOING OR PLANNED ACTIONS

- NUREG-0578 (REC. 2.1.8A, 2.1.8B, 2.1.8C, 2.2.2E)
- NUREG-0585 (REC. 10, 13)

RECOMMENDATION A.8

BEFORE ISSUING A CP OR OL THE AGENCY SHOULD:

- ASSESS THE NEED TO INTRODUCE NEW SAFETY IMPROVEMENTS IN THIS REPORT AND IN NRC AND INDUSTRY STUDIES
- REVIEW COMPETENCY OF UTILITY TO MANAGE THE PLANT AND THE ADEQUACY OF TRAINING PROGRAM FOR OPERATING PERSONNEL
- CONDITION LICENSING UPON REVIEW AND APPROVAL OF STATE AND LOCAL EMERGENCY PLANS

ONGOING OR PLANNED ACTIONS

MEMO, DENTON TO GOSSICK, DATED 11/01/79

RECOMMENDATION A.9

AGENCY'S AUTHORIZATION TO MAKE GENERAL RULES SHOULD:

(A) REQUIRE PUBLIC AGENDA TO WHICH RULES WILL BE FORMULATED

(B) REQUIRE DEADLINES FOR RESOLVING GENERIC ISSUES

(C) REQUIRE PERIODIC AND SYSTEMATIC REEVALUATION OF RULES

(D) DEFINE IMPROVEMENTS IN RULEMAKING PROCEDURES

ONGOING OR PLANNED ACTIONS

(A) NONE

(B) NUREG-0585 (REC. 12), SECTION 210 ENERGY REORGANIZATION ACT REQUIRES PLANNING AND REPORTING TO CONGRESS

(C) NONE

(D) SEP, SECY 79-8, NUREG-0585 (REC. 11)

RECOMMENDATION A.10

LICENSING PROCEDURES SHOULD FOSTER EARLY AND MEANINGFUL RESOLUTION OF SAFETY ISSUES BEFORE MAJOR FINANCIAL COMMITMENTS IN CONSTRUCTION CAN OCCUR

(A) REDUCE DUPLICATIVE CONSIDERATION OF ISSUES IN ONE PLANT'S LICENSING

(B) RESOLVE RECURRING ISSUES BY RULEMAKING

(C) AUTHORIZE COMBINED CP/OL HEARING

(D) PROVIDE FOR INITIAL ADJUDICATION OF LICENSE APPLICATIONS AND APPEAL TO A BOARD NOT SUBJECT TO FURTHER APPEAL BY THE ADMINISTRATOR

(E) ESTABLISH AN OFFICE OF HEARING COUNSEL TO PARTICIPATE IN HEARINGS AS OBJECTIVE PARTY

(F) SAFETY ISSUES LEFT OPEN IN LICENSING PROCEEDINGS SHOULD BE RESOLVED BY A DEADLINE

ONGOING OR PLANNED ACTIONS

(A) NUREG-0292

(B) NUREG-0292

(C) NONE

(D) NONE

(E) NONE

(F) LICENSE CONDITION FOR PLANT-SPECIFIC ISSUES

RECOMMENDATION A.11

I&E FUNCTIONS TO RECEIVE INCREASED EMPHASIS AND IMPROVED MANAGEMENT INCLUDING:

(A) IMPROVED SYSTEMATIC EVALUATION OF ORs

(B) SYSTEMATIC ASSESSMENT OF OPERATING EXPERIENCE

(C) PENALTIES FOR LICENSEE VIOLATIONS

(D) IMPROVED INSPECTION AND AUDITING

(E) PERIODIC REVIEW OF EACH OPERATING LICENSE PERFORMANCE

(F) ADOPT CRITERIA FOR REVOCATION OF LICENSES

ONGOING OR PLANNED ACTIONS

(A) SEP AND IREP

(B) SECY 79-571

(C) NUREG-0578 (REC. 2.2.3)

(D) NONE

(E) NONE

(F) NUREG-0578 (REC. 2.2.5)

RECOMMENDATION B.1

NUCLEAR INDUSTRY MUST CHANGE ITS ATTITUDES TOWARD SAFETY AND REGULATIONS INCLUDING:

(A) ESTABLISHMENT OF A PROGRAM THAT SPECIFIES STANDARDS AND CONDUCTS EVALUATIONS

(B) GATHERING AND DISSEMINATING OPERATING EXPERIENCE

ONGOING OR PLANNED ACTIONS

NUREG-0585 (SEC. 2.2 AND 2.3.1)

(A) NUREG-0585 (REC. 1.8)

(B) NUREG-0585 (REC. 6.1 AND 6.2) AND SECY 79-371

RECOMMENDATION B.2

EACH UTILITY TO HAVE A SEPARATE SAFETY GROUP:

- REPORTS TO HIGH LEVEL MANAGEMENT
- EVALUATE PROCEDURES AND PLANT OPERATIONS
- ASSESS QA PROGRAMS
- DEVELOP CONTINUING SAFETY PROGRAMS

ONGOING OR PLANNED ACTIONS

MEMO DENTON TO GOSSICK 11/01/79 AND NUREG-0585 (REC. 2.2.1.B)

RECOMMENDATION B.3

SINGLE ACCOUNTABLE ORGANIZATION RESPONSIBLE FOR DESIGN, CONSTRUCTION, OPERATION AND EMERGENCIES

(A) FEEDBACK OF DESIGN AND CONSTRUCTION EXPERIENCE

(B) CLEARLY DEFINED RULES FOR OPERATION

(C) CLEARLY DEFINED RULES IN CRISES

ONGOING OR PLANNED ACTIONS

NUREG-0585 (SEC. 2.2 AND 2.3.1), QAB SURVEY/CRITERIA

(A) NONE

(B) NUREG-0578 (REC. 2.2.1 AND 2.2.2), WASH-1284, 1309 AND 1283, NUREG-0585 REC. 2 THRU 5)

RECOMMENDATION B.4

o ATTRACT HIGHLY QUALIFIED CANDIDATES FOR POSITIONS OF SENIOR OPERATOR AND OPERATIONS SUPERVISOR

o INCREASE PAY SCALES

ONGOING OR PLANNED ACTIONS

NUREG-0585 (REC. 1.1 AND 1.6)

RECOMMENDATION B.5

DEVOTE MORE CARE TO WRITING, REVIEWING, AND MONITORING OF PLANT PROCEDURES

(A) WORDING CLEAR AND CONCISE

(B) REFLECT ENGINEERING THINKING AND OPERATING PRACTICALITIES

(C) CLEAR FORMAT AND DIAGNOSTICS

(D) EARLY DIAGNOSIS AND RESOLUTION OF SAFETY QUESTIONS FROM PLANT OPERATIONS

ONGOING OR PLANNED ACTIONS

NUREG-0585 (REC. 4)

(A) B&OTF ACTIVITIES, NUREG-0585 (REC. 4)

(B) B&OTF ACTIVITIES, NUREG-0585 (REC. 4)

(C) B&OTF ACTIVITIES, NUREG-0585 (REC. 4), NUREG-0578 (REC. 2.1.9)

(D) NUREG-0578 (REC. 2.2.1.B), NUREG-0585 (REC. 6), SECY 79-371

RECOMMENDATION B.6

RATE-MAKING AGENCIES SHOULD GIVE EXPLICIT ATTENTION TO THE SAFETY IMPLICATIONS OF RATE-MAKING

ONGOING OR PLANNED ACTIONS

NOT WITHIN PURVIEW OF NRR

RECOMMENDATION C.1

ESTABLISH AGENCY-ACCREDITED TRAINING INSTITUTIONS FOR OPERATORS AND THEIR IMMEDIATE SUPERVISORS

(A) NATIONAL, REGIONAL OR NSSS-SPECIFIC

(B) GRADUATION REQUIRED FOR OPERATORS

(C) PERIODIC REVIEW AND REACCREDITATION BY AGENCY

(D) CANDIDATES MUST MEET ENTRANCE REQUIREMENTS

ONGOING OR PLANNED ACTIONS

NUREG-0585 (REC. 1)

RECOMMENDATION C.2

- UTILITIES SHOULD BE RESPONSIBLE FOR TRAINING OPERATORS
- OPERATORS SHOULD BE EXAMINED AND LICENSED BY THE AGENCY
- TO BE LICENSED, OPERATORS MUST PASS EVERY PORTION OF EXAM
- SUPERVISORS OF OPERATORS SHOULD HAVE EQUIVALENT TRAINING

ONGOING OR PLANNED ACTIONS

SECY 79-330E (REC. 13), NUREG-0585 (REC. 1.2 AND 1.4)

RECOMMENDATION C.3

TRAINING SHOULD NOT END WHEN OPERATORS RECEIVE LICENSES:

(A) COMPREHENSIVE ONGOING TRAINING

(B) CONTINUOUSLY INTEGRATED WITH OPERATING EXPERIENCE

(C) EMPHASIS ON DIAGNOSING AND CONTROLLING COMPLEX TRANSIENTS AND UNDERSTANDING REACTOR SAFETY

(D) UTILITIES SHOULD HAVE READY ACCESS TO SIMULATOR; REGULAR TRAINING OF OPERATORS AND SUPERVISORS ON SIMULATORS; LICENSES CONDITIONED ON PERFORMANCE ON SIMULATOR

ONGOING OR PLANNED ACTIONS

SECY 79-330E (REC. 7), NUREG-0585 (REC. 1.4)

(A) SECY 79-330E (REC. 7)

(B) SECY 79-330E (REC. 7), NUREG-0585 (REC. 1.4)

(C) SECY 79-330E (REC. 8), NUREG-0578 (REC. 2.1.9), NUREG-0585 (REC. 1.4)

(D) SECY 79-330E (REC. 1.8)

RECOMMENDATION C.4

R & D TO IMPROVE SIMULATION AND SIMULATION SYSTEMS

(A) ESTABLISH AND SUSTAIN A HIGHER LEVEL OF REALISM IN OPERATOR TRAINING

(B) IMPROVE DIAGNOSTICS AND KNOWLEDGE OF PLANT SYSTEMS

ONGOING OR PLANNED ACTIONS

SECY 79-330E (REC. 8 AND 15)

RECOMMENDATION D.1

REVIEW EQUIPMENT FROM STANDPOINT OF PROVIDING INFORMATION TO HELP OPERATORS PREVENT AND COPE WITH ACCIDENTS.

ONGOING OR PLANNED ACTIONS

NUREG-0585 (REC. 7.1 - 7.5)

NUREG-0578 (REC. 2.1.3, A-B, 2.1.9, 2.2.2B)

REVISION TO REGULATORY GUIDE 1.97

RECOMMENDATION D.2

REVIEW EQUIPMENT DESIGN AND MAINTENANCE INADEQUACIES FROM STANDPOINT OF MITIGATING ACCIDENT CONSEQUENCES

ONGOING OR PLANNED ACTIONS

REGULATORY GUIDE 1.52

NUREG-0578 (REC. 2.1.8 A-C, 2.1.4)

SEPTEMBER 13, 1979 EISENHUT LETTER TO UTILITIES (ACRS RECOMMENDATION)

NUREG-0585 (REC. 10)

RECOMMENDATION D.3

CONTINUOUS RECORDING OF ALL CRITICAL PLANT MEASUREMENTS AND CONDITIONS

ONGOING OR PLANNED ACTIONS

REGULATORY GUIDE 1.97

NUREG-0578 (REC. 2.1.2B)

NUREG-0585 (REC. 7.1, 7.2, 7.3)

RECOMMENDATION D.4

IN-DEPTH STUDIES OF PROBABILITIES AND CONSEQUENCES OF ACCIDENTS

(A) SB LOCA AND MULTIPLE-FAILURE ACCIDENTS

(B) RECOVERY AND CLEANUP PLANNING

(C) DESIGN MODIFICIATIONS

  (I) RCS HYDROGEN VENTING

  (II) GAS-TIGHT LETDOWN/MAKEUP

(D) SPONSORED BY NRC; CONDUCTED BY INDUSTRY

ONGOING OR PLANNED ACTIONS

NUREG-0585 (REC. 8, 10)

INTEGRATED RELIABILITY EVALUATION PROGRAM

UNRESOLVED SAFETY ISSUES A-17 AND A-44

(A) B&OTF ACTIVITIES
NUREG-0578 (REC. 2.1.9)

(B) NONE

(C) NUREG-0585 (SEC. 3.3)

  (I) SEPTEMBER 13, 1979 EISENHUT LETTER TO UTILITIES
  NUREG-0585 (REC. 10)

  (II) NUREG-0578 (REC. 2.1.6 A-B)
  NUREG-0585 (REC. 10)

(D) NONE

RECOMMENDATION D.5

CHEMICAL BEHAVIOR AND RETENTION OF RADIOACTIVE IODINE IN WATER

ONGOING OR PLANNED ACTIONS

NONE - RESOLVED BY PRIOR RESEARCH

RECOMMENDATION D.6

MONITOR TMI CLEANUP PROCESS AND RADIOACTIVE MATERIAL TRANSPORTATION AND DISPOSAL

ONGOING OR PLANNED ACTIONS

TMI-2 ORDER

TMI SUPPORT TASK FORCE

RECOMMENDATION D.7

SCREEN AND INVESTIGATE AS APPROPRIATE, EVERY ACCIDENT OR NEW ABNORMAL EVENT.

ONGOING OR PLANNED ACTIONS

NUREG-0585 (Rec. 6.1, 6.2)

ENERGY REORGANIZATION ACT OF 1974 (Sec. 2[illegible]3.

SECY 79-571

RECOMMENDATION E.1

ESTABLISH EXPANDED AND BETTER COORDINATED HEALTH-RELATED RADIATION EFFECTS RESEARCH TO INCLUDE:

(A) BIOLOGICAL EFFECTS OF LOW LEVEL RADIATION

(B) ACCEPTABLE EXPOSURE LEVELS

(C) DEVELOP METHODS TO MONITOR AND DETERMINE CONSEQUENCES OF EXPOSURE TO RADIATION

(D) DEVELOP APPROACHES TO MITIGATE ADVERSE HEALTH EFFECTS

(E) EFFECTS OF GENETIC OR ENVIRONMENTAL FACTORS

ONGOING OR PLANNED ACTIONS

(A) NONE

(B) EPA INTERAGENCY EXPOSURE LIMIT TASK FORCE

(C) RESEARCH USER REQUEST NRR 79-3

(D) WORKING WITH FDA

(E) NONE

RECOMMENDATION E.2

AGENCY POLICY STATEMENTS OR REGULATIONS IN RADIATION-RELATED HEALTH ISSUES (INCLUDING SITING ISSUES) SHOULD BE SUBJECT TO MANDATORY REVIEW BY THE SECRETARY OF THE DEPARTMENT OF HEALTH AND HUMAN SERVICES

ONGOING OR PLANNED ACTIONS

NONE

RECOMMENDATION E.3

STATE AND LOCAL PROGRAMS FOR EDUCATING HEALTH PROFESSIONALS AND EMERGENCY RESPONSE PERSONNEL IN THE VICINITY OF PLANTS

ONGOING OR PLANNED ACTIONS

EMERGENCY PLANNING TASK FORCE ACTIVITIES

RECOMMENDATION E.4

UTILITIES MUST MAKE ADVANCE PREPARATION FOR EMERGENCIES:

(A) RADIATION MONITORS SHOULD BE AVAILABLE

(B) EMERGENCY CONTROL CENTER AND ANALYTICAL LAB TO BE LOCATED IN APPROPRIATE AREA

(C) SUFFICIENT HEALTH-RELATED SUPPLY OF INSTRUMENTS, RESPIRATORS, ETC.

(D) ADEQUATE MAINTENANCE PROGRAM FOR HEALTH-RELATED EQUIPMENT

ONGOING OR PLANNED ACTIONS

(A) NUREG-0578 (REC. 2.1.8B, 2.1.8C), REVISION OF R.G. 1.97

(B) NUREG-0578 (REC. 2.1.6B), REVISION OF SRP 12.3

(C) EMERGENCY PLANNING TASK FORCE ACTIVITIES, SPECIAL NRR PANEL ON RADIATION PROTECTION-TMI-2, REVISION OF SRP SECTION 12

(D) REVISION OF SRP 12.5

RECOMMENDATION E.5

ADEQUATE SUPPLY OF POTASSIUM IODIDE AVAILABLE REGIONALLY FOR RADIOLOGICAL EMERGENCIES

ONGOING OR PLANNED ACTIONS

WORKING WITH FDA

RECOMMENDATION F.1

EMERGENCY PLANS MUST DETAIL ACTIONS PUBLIC OFFICIALS AND UTILITIES SHOULD TAKE IN THE EVENT OF OFFSITE DOSES

(A) EMERGENCY RESPONSE PLAN TO BE APPROVED BY FEMA

(B) FEDERAL LEVEL RESPONSIBILITY FOR RADIOLOGICAL EMERGENCY PLANNING SHOULD REST WITH FEMA

(C) STATE MUST COORDINATE WITH UTILITY AND LOCAL OFFICIALS

(D) STATES WITH PLANS SHOULD UPGRADE TO REQUIREMENTS OF FEMA

ONGOING OR PLANNED ACTIONS

EMERGENCY PLANNING TASK FORCE, REGULATORY GUIDE 1.101

RECOMMENDATION F.2

PLANS FOR PROTECTING PUBLIC TO BE BASED ON VARIOUS CLASSES OF ACCIDENTS

(A) CRITERIA FOR EACH SCENARIO

(B) RESPONSE PLANS ACTIVATED WHEN NATURE AND POTENTIAL HAZARD OF ACCIDENT IDENTIFIED

(C) PLANS TO PROTECT PUBLIC AT LOWER LEVELS OF RADIATION THAN IN CURRENT PLANS

(D) LOCAL COMMUNITIES SHOULD HAVE FUNDS AND TECHNICAL SUPPORT

ONGOING OR PLANNED ACTIONS

DRAFT EMERGENCY ACTION LEVEL GUIDELINES

(A) EMERGENCY PREPAREDNESS REVIEW GUIDELINE NO. 3

(B) EMERGENCY PREPAREDNESS REVIEW GUIDELINE NO. 3

(C) NONE

(D) NUREG-0553

RECOMMENDATION F.3

RESEARCH TO BE EXPANDED ON MEDICAL MEANS TO PROTECT PUBLIC AGAINST RADIATION

ONGOING OR PLANNED ACTIONS

NONE

RECOMMENDATION F.4

PROGRAM TO EDUCATE PUBLIC

- HOW PLANTS OPERATE
- RADIATION AND ITS HEALTH EFFECTS
- PROTECTIVE ACTIONS AGAINST RADIATION

ONGOING OR PLANNED ACTIONS

REGULATORY GUIDE 1.101

RECOMMENDATION F.5

STUDY OF HUMAN COSTS OF RADIATION-RELATED MASS EVACUATION AND EXTENT TO WHICH RISKS DIFFER FROM OTHER EVACUATIONS

ONGOING OR PLANNED ACTIONS

NONE

RECOMMENDATION F.6

- PLANS FOR PROVIDING FEDERAL TECHNICAL SUPPORT SHOULD SPECIFY AGENCY RESPONSIBILITIES AND PROCEDURES
- EXISTING PLANS FOR FEDERAL ASSISTANCE SHOULD BE RE-EXAMINED AND REVISED IN LIGHT OF TMI-2

ONGOING OR PLANNED ACTIONS

- SECY 79-591
- REGULATORY GUIDE 1.101

CLADDING SWELLING AND RUPTURE MODELS FOR LOCA ANALYSIS

November 8, 1979

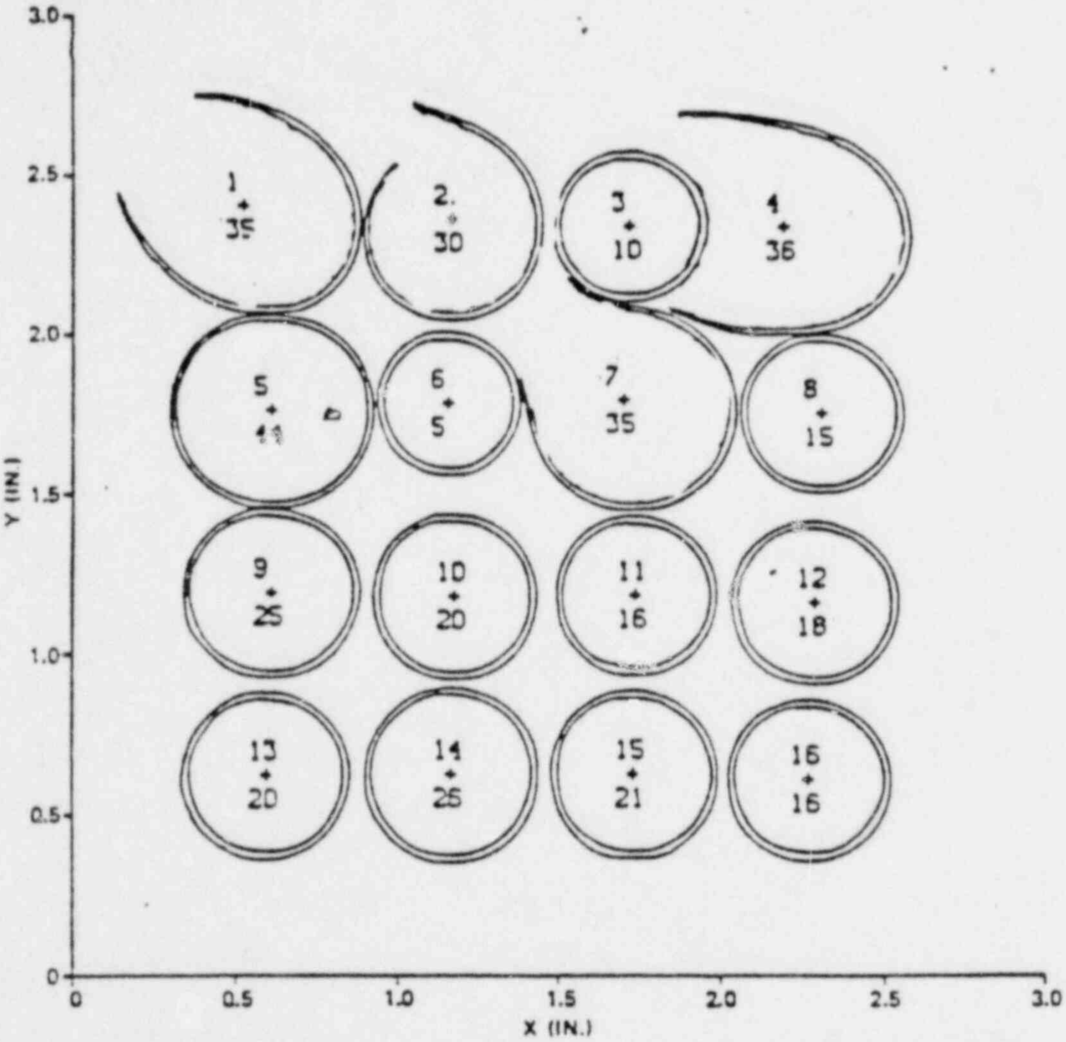
PRESENTATION TO ACRS

by

R. O. Meyer
Reactor Fuels Section
Division of Systems Safety, NRC

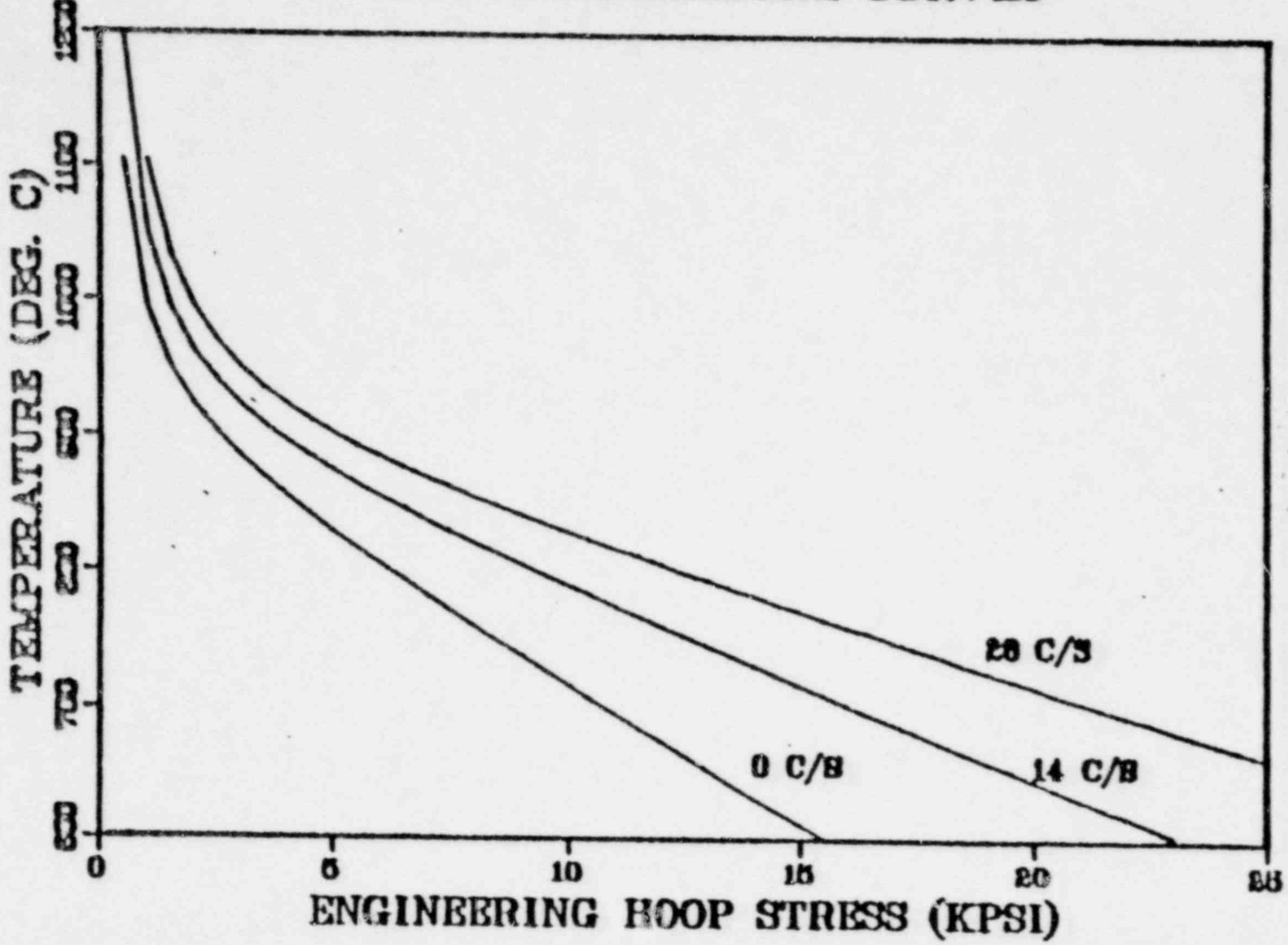
THREE FUEL BEHAVIOR RELATIONS NEEDED FOR ECCS ANALYSIS

1. BURST TEMPERATURE VS STRESS
2. BURST STRAIN VS TEMPERATURE
3. ASSEMBLY FLOW BLOCKAGE VS STRESS

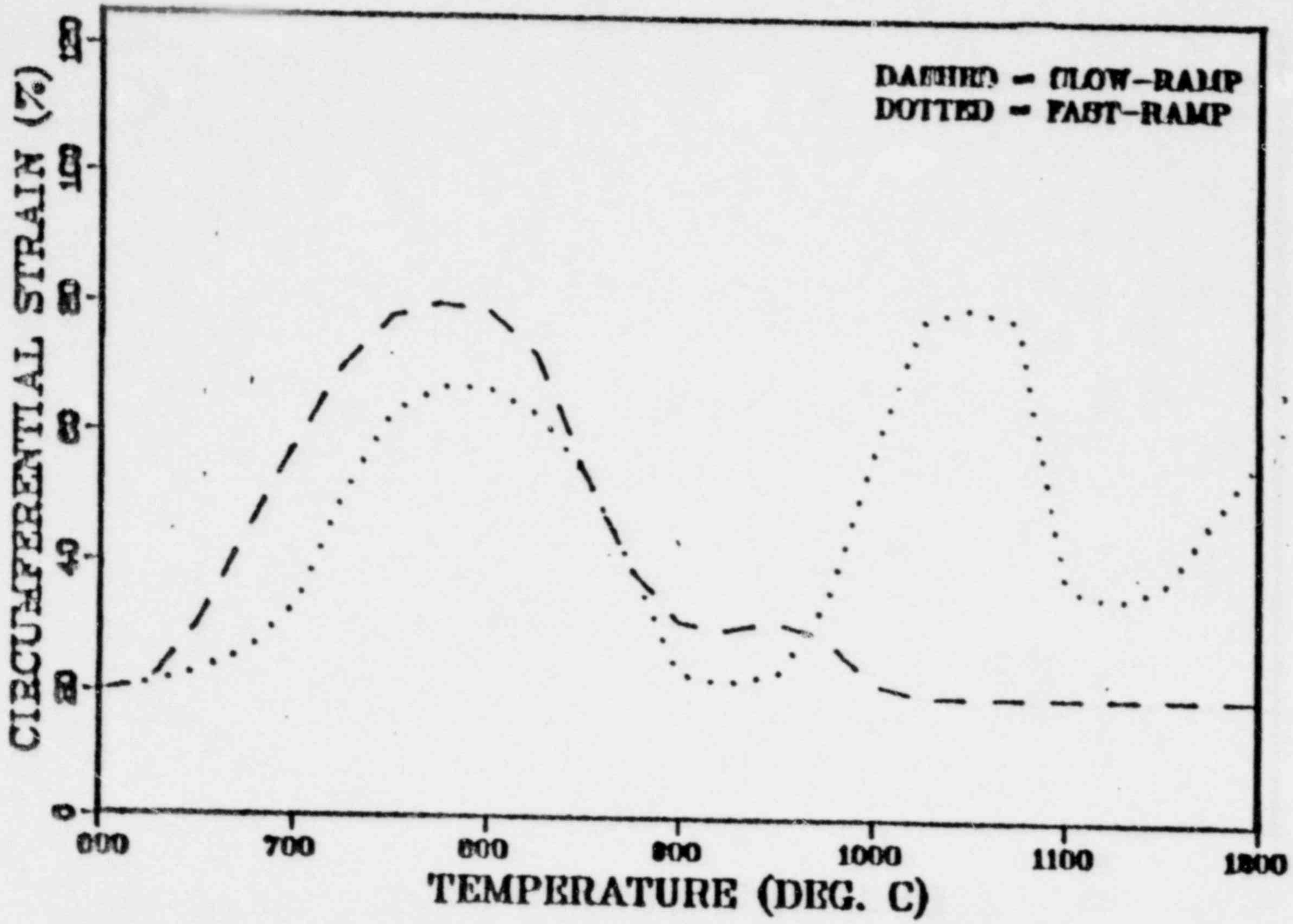


Computer simulation of B-1 section of 76.5-cm elevation showing maximum and minimum flow restriction definitions.

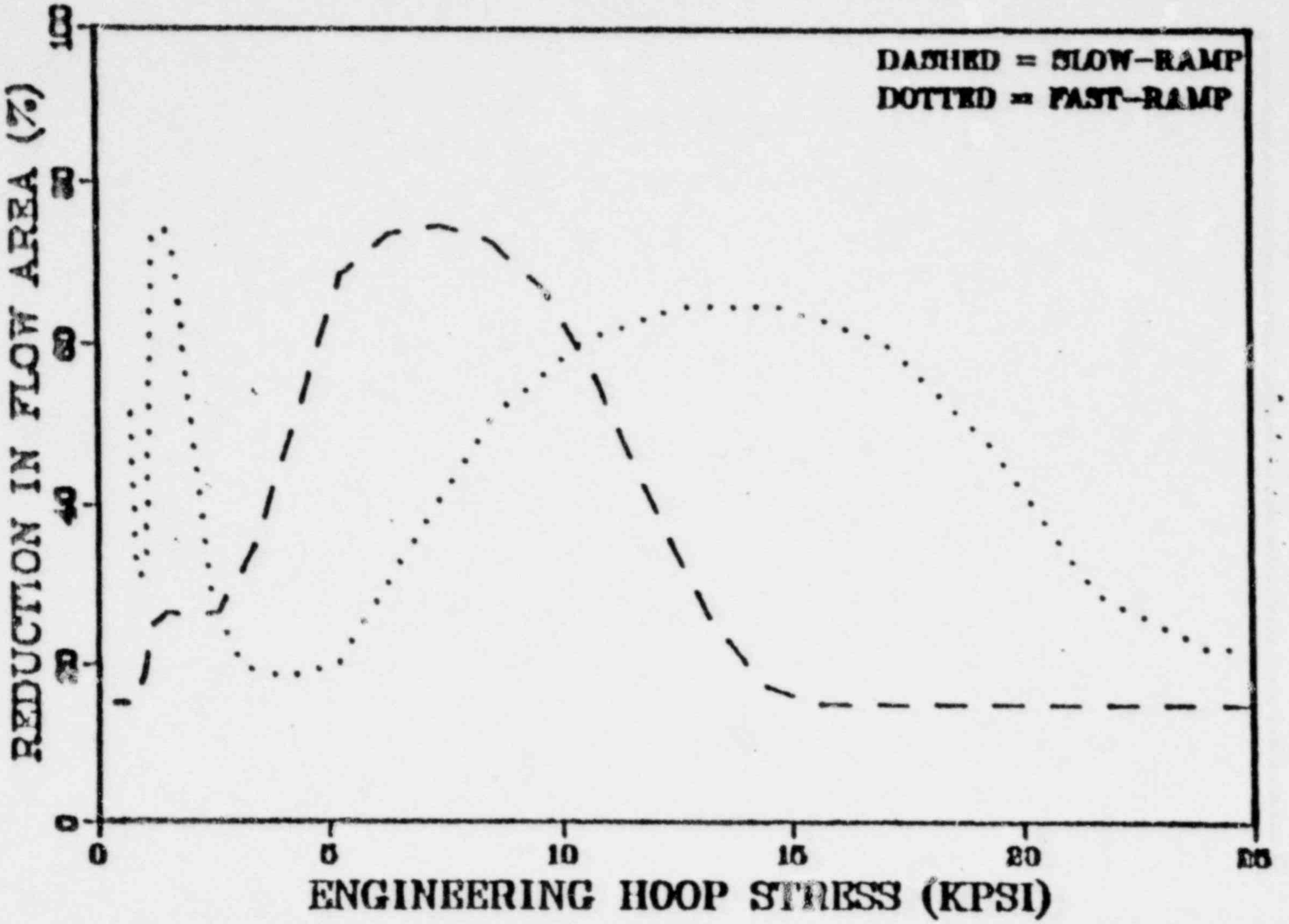
BURST TEMPERATURE CURVES



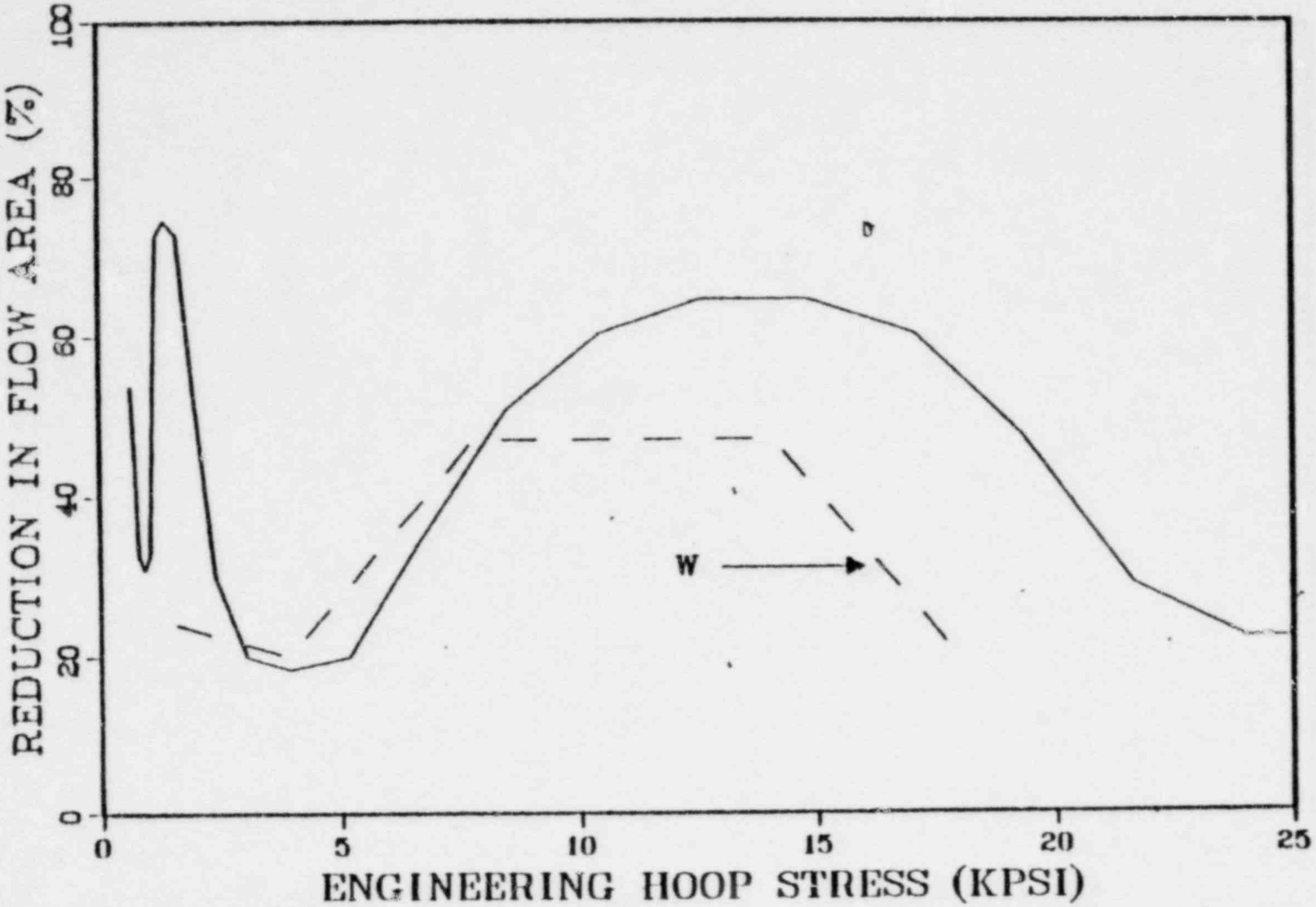
BURST STRAIN CURVES



PWR ASSEMBLY FLOW BLOCKAGE CURVES



W & FAST-RAMP BLOCK. CURVES



B&W (T) & FAST-RAMP BURST STRAIN CURVES

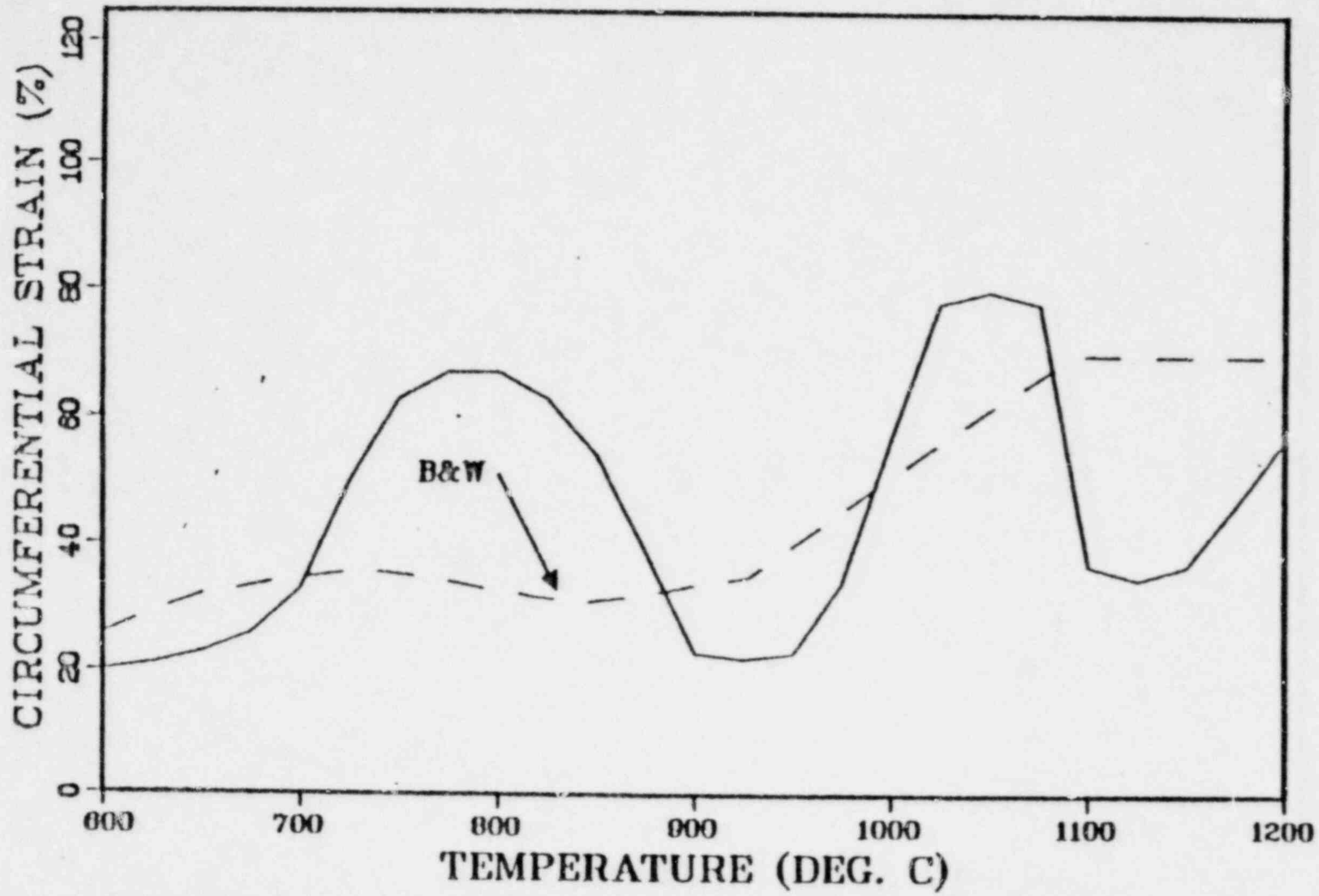


Fig. 25

-38-

A-77

W & FAST-RAMP BLOCK. CURVES

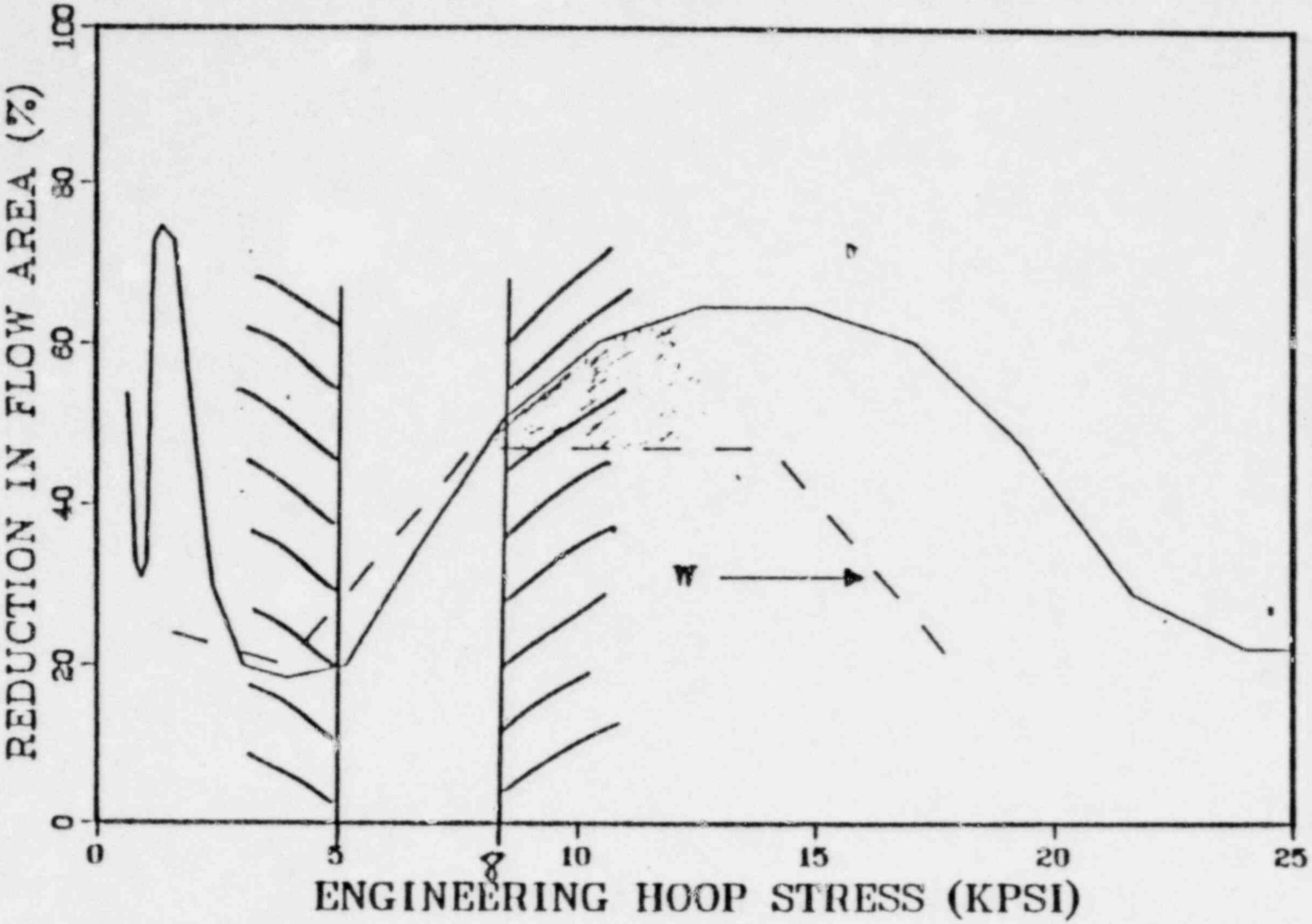
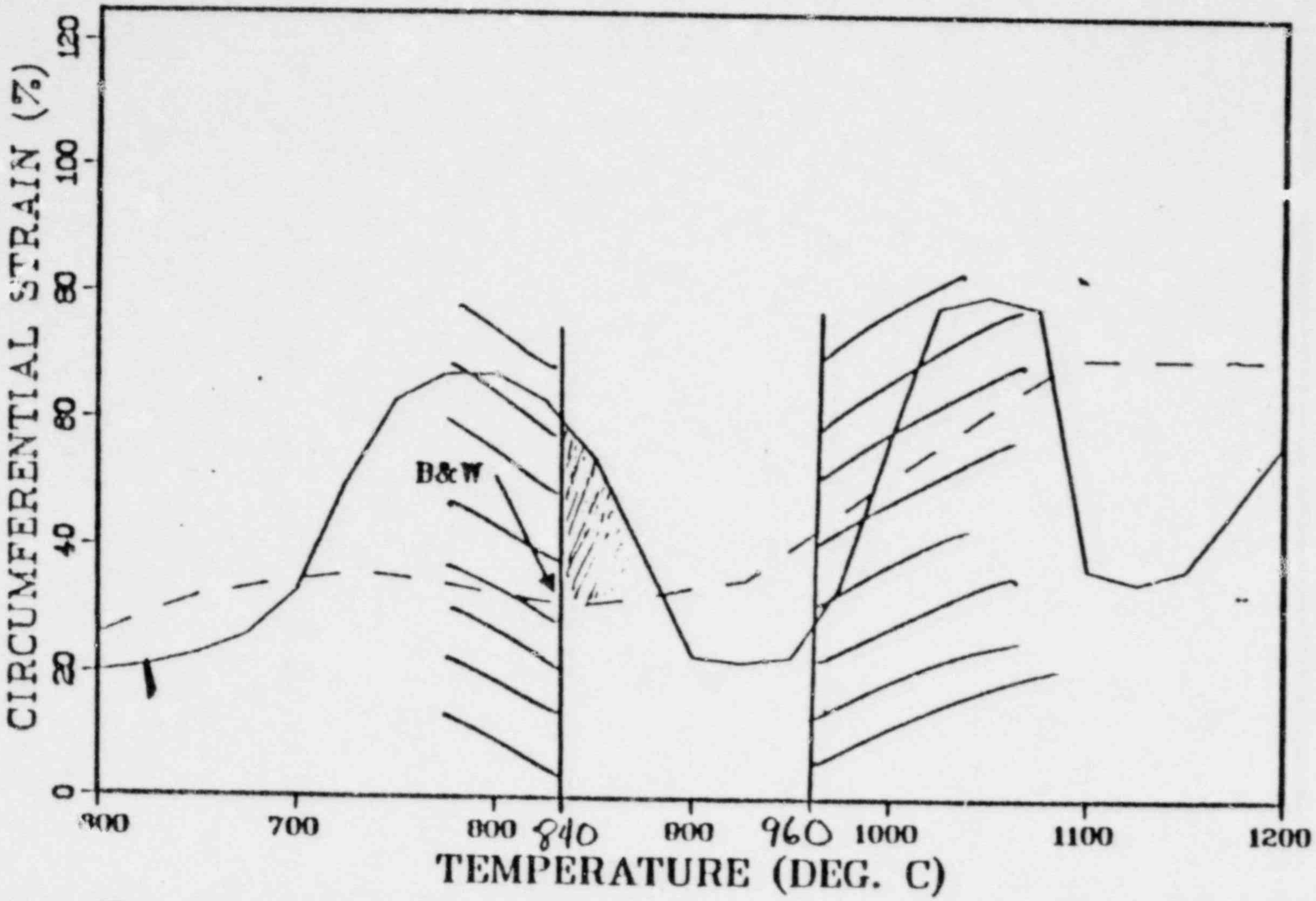


Fig. 52

A-7-78
-99-

B&W (T) & FAST-RAMP BURST STRAIN CURVES



F:g. 25

ACCEPTANCE CRITERIA FOR MARK II PIPING SYSTEMS

MECHANICAL ENGINEERING BRANCH

DIVISION OF SYSTEMS SAFETY

LOAD CASE	$N^{(3)}$	SRV_X	SRV_{ADS}	OBE	SSE	$IBA^{(1\&5)}$	$DBA^{(5)}$	ACCEPTANCE CRITERIA
1	X	X						B
2	X	X		X				B
3	X	X			X			$C^{(4)}$
4	X		X			X		$C^{(4)}$
5	X		X	X		X		$C^{(4)}$
6	X		X		X	$X^{(2)}$		$C^{(4)}$
7	X				X		$X^{(2)}$	$C^{(4)}$
8	X							A
9	X			X				B
10	X	X			X		$X^{(2)}$	$C^{(4)}$

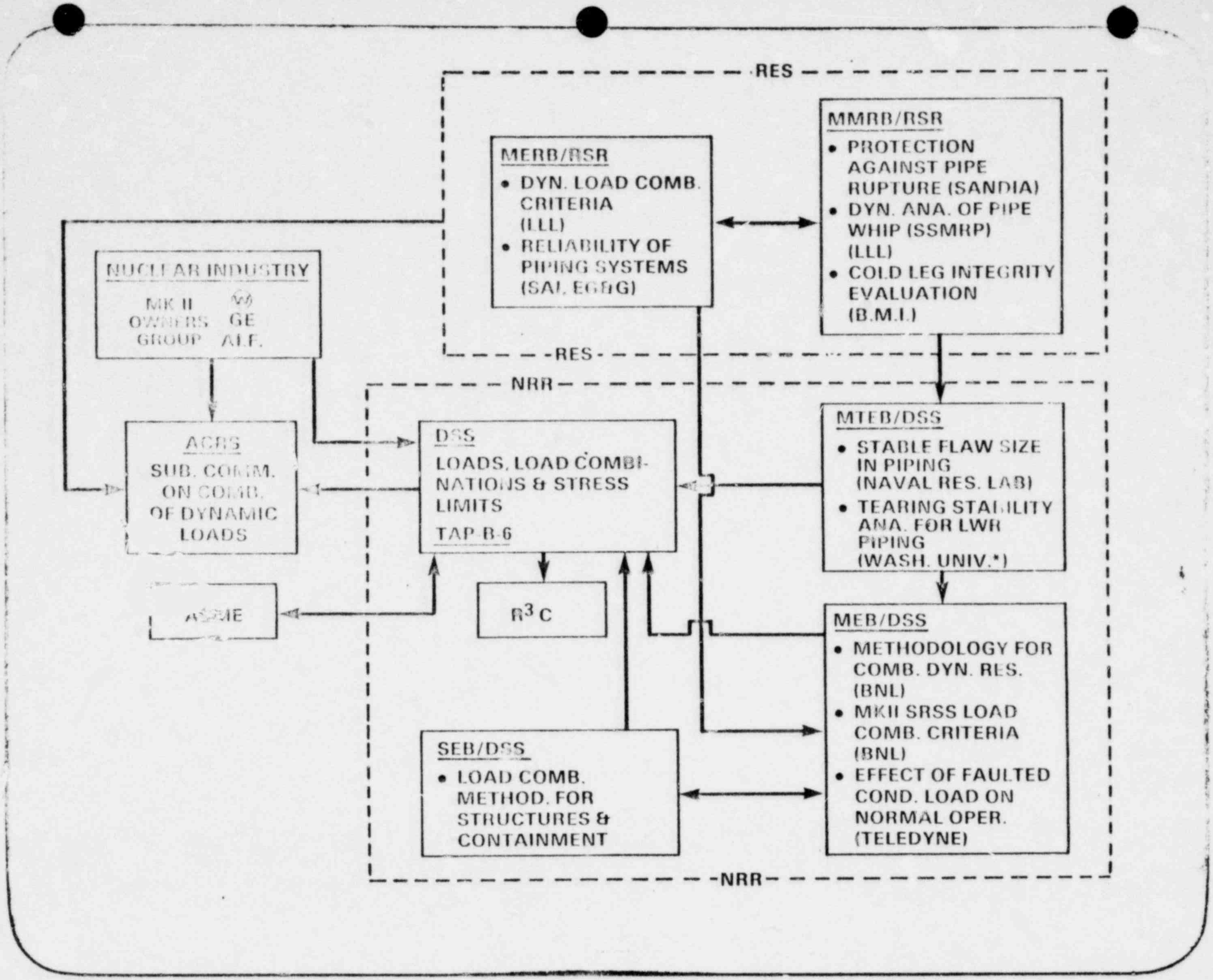
(1) Use SBA or IBA whichever is governing.

(2) Loading due to DBA/SBA/IBA is determined from rated steady state conditions.

(3) N - Normal load consists of pressure, dead weight, thermal & fluid reaction loads.

(4) Piping functional capability should be assured. Service level limits higher than the level specified in this table may be used, provided piping functional capability is demonstrated.

(5) SBA, IBA and DBA shall include all event induced loads whichever are applicable, such as possible annulus pressurization load, pool swell load, condensation oscillation load, chugging load, etc.


RES
MERB/RSR
• DYN. LOAD COMB. CRITERIA (LLL)
• RELIABILITY OF PIPING SYSTEMS (SAI, EG&G)
MMRB/RSR
• PROTECTION AGAINST PIPE RUPTURE (SANDIA)
• DYN. ANA. OF PIPE WHIP (SSMRP) (LLL)
• COLD LEG INTEGRITY EVALUATION (B.M.I.)
-RES-
NUCLEAR INDUSTRY
MK II OWNERS GROUP
GE
A.I.F.
ACRS
SUB. COMM. ON COMB. OF DYNAMIC LOADS
NRR
DSS
LOADS, LOAD COMBINATIONS & STRESS LIMITS
TAP-B-6
MTEB/DSS
• STABLE FLAW SIZE IN PIPING (NAVAL RES. LAB)
• TEARING STABILITY ANA. FOR LWR PIPING (WASH. UNIV.*)
ASME
R^3 C
MEB/DSS
• METHODOLOGY FOR COMB. DYN. RES. (BNL)
• MKII SRSS LOAD COMB. CRITERIA (BNL)
• EFFECT OF FAULTED COND. LOAD ON NORMAL OPER. (TELEDYNE)
SEB/DSS
• LOAD COMB. METHOD. FOR STRUCTURES & CONTAINMENT
-NRR-

DELETION 1

Pages A-82 thru A-161

CHRONOLOGY

Sept 6	W Owners Group Meeting
Sept 9	PSE&G (Salem 1) LER 79-58
Sept 14	IE Information Notice 79-22
Sept 17	Letter To All Licensees - H. Denton
Sept 18-20	Meetings With Licensees
Oct 5-9	Licensee Submittals
Oct 15	Basis For Continued Operation - D. Eisenhut
Oct 19	AIF/NSAC Generic Submittal
Nov 6	Status Report
Nov 8	NRC/Industry Meeting

BASIS FOR CONTINUED OPERATION

1. SAFETY CONCERN BUT NO DEMONSTRATED SAFETY PROBLEM
2. MARGINS IN HELB SAFETY ANALYSES
3. SIMILAR UNRESOLVED SAFETY ISSUES
4. OPERATOR CAN COPE

INITIAL FINDINGS

1. No identified safety problem
2. Concern, however, regarding
   - B&D of systems reviews
   - EQ of equipment
   - Operator action
3. Concur with Rec 9, NUREG 0585

CURRENT RELATED ACTIVITIES

- Fire Protection Reviews
- EQ of Safety Equipment
- Diablo Canyon Seismic Pipe Breaks
- TAP - A 17 - Systems Interaction
- Standards Development For Non-Safety Grade Equipment
- Consequential Control System Failure

NRC/INDUSTRY STEERING COMMITTEE

- develop Task Action Plan
- establish schedule
- oversee performance

APPENDIX X
BACKGROUND MATERIAL FOR DISCUSSION OF
NUREG-0600

HIGHLIGHTS OF TMI-2 SUBCOMMITTEE MEETING
NUREG-0600

1. Jim Allen, Deputy Director Region I, I&E, stated that the report involved 3500 man-days of effort over a four-month period. Seven members of the team examined reactor operations from surveillance testing of the auxiliary FW systems on March 26 until restart of RCP A at 8 P.M., March 28 and seven members. the radiological and emergency response actions of the licensee from the start of the accident until midnight March 30. The investigation did not include:

   (a) Evaluation of actions of NRC or other agencies
   (b) Evaluation of the regulatory process
   (c) Evaluation of the Legislative authority of the NRC
   (d) Evaluation of safety research
   (e) Evaluation of the licensing process, or the inspection and enforcement process
   (f) Review of design, of the systems shortcomings, instrumentation etc.

2. Mr. R. C. Arnold, Sr. V.P. Metropolitan Edison, stated that a final Met. Ed. report on the sequence of events and other accident related matters would be out of the printers about the middle of December.

3. The Staff investigation did not review the adequacy of the procedures. The procedures were admittedly ambiguous. The Office of Inspection and Enforcement reviews operating procedures for adequacy on a sampling basis (about 50%) but does not approve them.

4. Darwin Hunter, NRC Staff stated that operators had, on this occasion and others, throttled high pressure injection in violation of their procedures and training which called for maintaining the HPI whenever the pressure was below 1640 pounds. Dr. Catton noted that the LOCA procedures called for throttling as necessary to maintain level. Also, he pointed out that the symptoms of a major LOCA are rapid pressure decrease <u>and</u> rapid decrease in pressurizer level, (which did not occur).

5. Unidentified leakage had exceeded limits since October 1978, but this was not recognized by the operator because of a calculational error. Also, tailpipe temperatures in excess of procedural limits (135°) had existed since fall of '78. High tailpipe temperatures required closing of the block valve.

6. According to the I&E Staff, the operators were indoctrinated to prevent sodium hydroxide addition if possible, so they tried to limit the amount of injection so that the NaOH valves would not open. (This is not consistent with the fact that the A Makeup pump was manually started 13 seconds into the event as required by procedures.)

7. Dr. Lipinski questioned the wisdom of interconnecting the service and the instrument air lines, if this could be the initiating cause for the TMI-2 event.

8. Both signals on the EMOV indicated it should have been shut. The fact that the valve was stuck open was not known to the operator.

9. The reason for the A Make-up pump failure has not yet been determined (p. 52).

10. Ed Jordan, I&E, indicated (p.57) that information developed after the investigation, will help to better understand the whole event.

11. The Staff feels the operator first deviated from his procedures when he throttled back the make-up to less than 250 gal/min per leg. At this point, the reactor trip and turbine trip were complete (p.61). The operator limited high pressure injection based on pressurizer level alone with no attention to pressure.

12. Mr. Michelson observed that the operator noted the level and that the safety injection was on, and, as had happened before for a non-LOCA case, he noted that water level was recovering. The fact that pressure remained low confused him but did not lead immediately to the obvious conclusion that he had a LOCA. There are procedures for hot and cold leg breaks, but there is no procedure for a break at the top of the pressurizer. Furthermore, part B of the LOCA procedure, which the Staff felt should be followed, called for continuing decrease in pressurizer level and this was not happening. Mr. Hunter indicated this was because the low pressure kept conditions outside the design envelope for which the procedure was written.

13. Mr. Jordan stated that at the existing temperatures and with safety valves, the operator should not have been so concerned with going solid. The Staff did not believe the operator was the sole cause of the accident, but they feel he had an opportunity to prevent it.

14. The operators had been trained to expect saturation only on the secondary side.

15. NUREG-0600 does not go into detail on the operator's thought process. (p. 82).

16. The diesels were manually tripped, according to procedures, but the fuel racks were not reset, which rendered the diesels inoperable, so that they could not respond to a start signal from the ESP or the control room. The Staff was unable to identify the operator who failed to reset the fuel racks. Dr. Lipinski expressed concern that this could have been deliberate, or worse yet, out of ignorance.

17. The duty superintendent, who arrived at 4:50 A.M. was trained on Unit 1 and was unaware of several important differences between the units (p. 86). He left the shift supervisor in charge.

18. There were errors on the part of the technical management as well as on the part of the operators. Errors are attributed to "the licensee," and that can be anyone in the licensee organization.

19. Computer records did little to support the operation because of the time delay but are valuable historical records. They are not necessarily chronologically accurate.

20. There was no clear point at which "non-compliance" terminated, even when the operators "ran out of procedures." Whenever pressure dropped below 1640 again "non-compliance" resumed. Mr. Michelson pointd out that to achieve cold shutdown, at some point, pressure would have to drop below 1640.

21. There is no explanation for the loss of A/C motor control centers. (non-safety-related).

22. Conferences with B&W and GPU, as well as Met. Ed. management resulted in the order to repressurize. (The long period of low pressure operation may have aided in degassing the loops significantly.) Once 400 gpm HPSI flow was established, condenser vacuum was regained, and a heat sink reestablished.

23. Inadequacies in licensee performance were enumerated by the Staff as follows:

    (a) Changes were made to EFW surveillance tests, disabling both EFW trains simultaneously.

    (b) After automatic ESF initiation, the operators reduced HPI flow with "complete disregard for RCS pressure and temperature conditions". Further, licensee training and operating practices contributed significantly to these actions.

    (c) Failure to provide changes to procedures for coping with pressurizer system failures, once routine leaks developed, invalidating existing procedures.

    (d) Failure to recognize saturation conditions.

    (e) Failure to reestablish emergency diesel generator operability.

24. Dr. Lipinski noted that an LER issued on Unit 1 on March 27 indicated that after maintenance on the turbine-driven feed system, a valve was left in the closed position. He inquired if the investigation had looked for a relationship between that LER and the closed AFW valves in Unit 2. Mr. Jordan indicated it had not.

25. Mr. Michelson asked about operation of the letdown system. Mr. Martin indicated it was included in the sequence of events, including its interplay with the makeup system but there was no reason to put it in the body of the report. He pointed out that the operators did not recognize they had a LOCA, so the blocking and bypassing of the ES system allowed them to continue makeup and letdown, deflating the isolation that occurs automatically with safety injection. Mr. Hunter noted that letdown operation is not recorded.

26. Resetting the diesels requires running out to them to reset the fuel racks if they are tripped. The shift foreman did not want the diesels to restart with each ES actuation. They could have been started from the control room and stopped if the racks were not tripped. This was not covered in the procedures or training. Mr. Arnold (Met. Ed.) said he felt his company knew who shut down the diesels and he would pursue finding out why they were not reset.

27. At one point in the accident, licensed personnel were getting advice from non-licensed B&W personnel. The Staff pointed out that the ultimate responsibility rests with the licensed operator, who may, if he wishes, reject the advice. Mr. Jordan agreed that that policy requires clarification. Dr. Carbon observed that the operator started the pumps on orders from higher management, unlicensed, although he did not wish to. Mr. Jordan stated that NRC has not directed criticism to that advice. Mr. Arnold, (Met. Ed.) pointed out that the operator did not believe starting the pumps was unsafe.

28. During 1978, there were seven emergency drills. As in one of the drills, after the TMI-2 accident, an iodine survey instrument taken to Goldsboro, was not operating properly. The drills identified a need to review site emergency criteria. During the accident, declaration of a site emergency was delayed because operations personnel did not realize the criteria had been met.

29. Offsite monitoring team members were not trained in the use of instruments which were used for measurement of radioactive iodine in the environment. Less than half of the portable radiation survey instruments were operable. Respirators were equipped only with particulate filters and were ineffective for iodine protection.

30. A meter reading 1000R/hr pegged on occasion. There were no meters available for higher measurements.

31. The Staff feels that lack of a specific definition of "loss of reactor coolant system pressure" and "high reactor building pressure" in the procedures or emergency plan was a factor in the delay in declaring a site emergency.

32. Several entries were made into high radiation areas without the knowledge of the Supervisor of Radiation Protection and Chemistry. (High airborne radioactivity and whole body rates over 100R/hr.) Two persons received doses over regulatory limits. At times, high range pocket dosimeters could not be located and were not worn. Several cases of head contamination resulted. High radiation levels in the plant were unanticipated in the emergency procedures.

33. An error in reading a meter resulted in an actual level of 400 mr/hr being read as 30,000 mr/hr.

34. No plume measurements were made during several critical periods on March 28 and 29.

35. Required retraining of chemistry technicians had not been done.

36. The Staff has no policy on helicopter flying in close proximity to a plan (especially one undergoing a crisis). Mr. Arnold (Met. Ed.) confirmed Mr. Michelson's concern for the safety of the power lines.

37. Airborne filter samples from early entry into the auxiliary building read off-scale in the TMI laboratory. Mr. Herbein (Met. Ed.) stated that entry was necessary to identify sources of leakage and to restore power to some lube oil pumps needed to start the RCP.

38. Face masks probably were not necessary but were used conservatively because failure of gamma spectroscopy at 9 A.M. rendered the isotope content of the airborne radioactivity unknown. Mr. Arnold observed that the switchboard operator functioned adequately wearing a mask.

39. The Staff observed that there are at present no plans for an integrated report of all aspects of the problem. They agreed it was a good idea.

40. During the Executive Session, the following suggestions were made for matters to be included in an ACRS report:

    (a) A rigorous procedure approval system should be instituted.

    (b) A plant process computer is needed that works well, has proper displays, and is prompt.

6.4

MEMORANDUM

TO: MEMBERS, SUBCOMMITTEE ON ENERGY AND THE ENVIRONMENT

FROM: MORRIS K. UDALL, CHAIRMAN

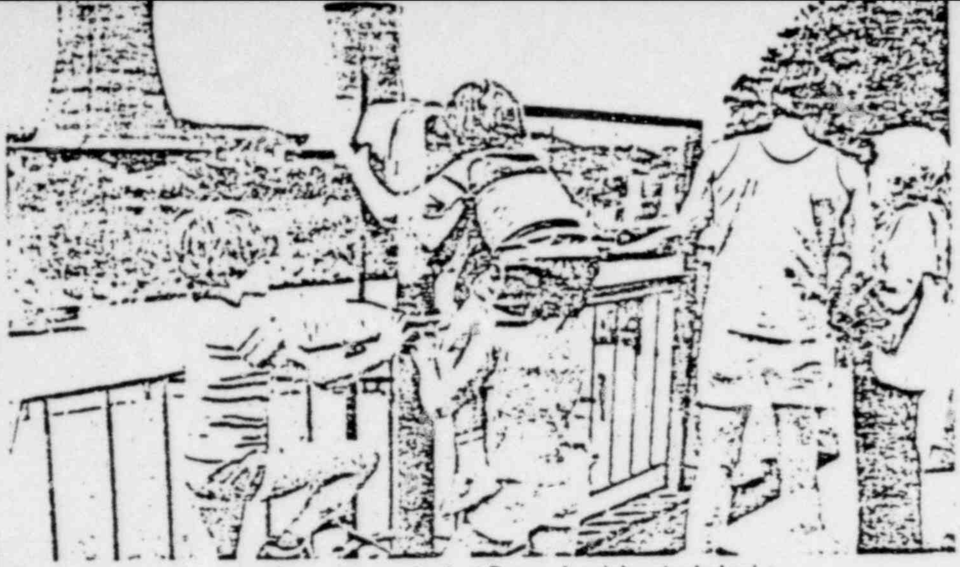
For some time I have been concerned that damage to the Three Mile Island reactor was being ascribed principally and unfairly to the failure of operators to follow established procedures. This impression was reinforced by a report issued by the NRC's Office of Inspection and Enforcement, Investigation into the March 28, 1979 Three Mile Island Accident, NUREG 0600. (See attached clipping from Time Magazine.) A statement in the foreward of NUREG 0600 concludes that, "... emergency procedures were adequate to have prevented the serious consequences of the accident." This statement is inconsistent with statements in the body of NUREG 0600. It is also inconsistent with the conclusion of another NRC report, TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations, NUREG 0578, which states on page 41:

> In the Three Mile Island accident, a loss of feedwater transient led to a small break loss-of-coolant accident when the pilot-operated relief valve failed to close. The emergency procedure for a loss of feedwater did not alert the operators to this possibility, nor did it provide any indication that the opening of the PORV should have been expected. In addition, recent reviews of emergency procedures for the small break loss-of-coolant accident at B&W plants clearly indicate that the procedures were inadequate to provide the operators with needed instructions on actions required to cope with various sizes and locations of small breaks. It is clear from the events at Three Mile Island that operator training and emergency procedures were not adequate for the operators to conclude from the information available that the reactor core was uncovered and inadequately cooled for a long period of time. (Underline added)

A-172

In August, I asked the NRC to explain the basis for the NUREG 0600 conclusion as to operator culpability. This week I received the NRC's response which states, in effect, that reports based on NUREG 0600 had "... placed undue emphasis on the operator deficiencies..." discussed therein. The memorandum states further that the scope of NUREG 0600 was limited and the comprehensive TMI inquiries now in progress are likely to ascribe damage to a variety of causes including inadequacies in equipment, in system design and analysis, and in operator training and operator performance.

In view of the widespread perception that operator error was the principle cause of the damage at TMI, I recommend that you read the enclosed NRC memorandum.



Young tourists clamoring for a close-up look at Pennsylvania's crippled nuke

Science

Three Mile Island Verdict

Human error is to blame

For many of the tourists who are flocking daily to Pennsylvania's crippled Three Mile Island nuclear power station, the 15-minute documentary may have the ring of authority. Prepared by Metropolitan Edison Co., the plant's operator, and being shown daily at the Observation Center across the river from Three Mile Island's cooling towers, the script has a glib explanation for last March's near disaster. It resulted, says the Met Ed film, from "a complex combination of equipment failures, ambiguous instruments and operator failures . . . ." The production also insists that the amount of radiation released into the atmosphere was insignificant.

Unfortunately for the beleaguered utility, its film may now need some editing. For the past four months, the Nuclear Regulatory Commission (NRC), among others, has been looking into the causes and effects of the nation's worst commercial reactor accident. Last week, in a report that is sure to have wide repercussions, NRC staff investigators said that the most serious aspects of the mishap were almost certainly due to human error. And though they acknowledged that the radiation level was low, they said that one burst was greater than any previously revealed.

Some two inches thick and based on many hours of hearings, the NRC report will be some comfort to those who design and build reactors used to generate electricity. It states categorically that although the Pennsylvania plant was not "fail-safe," its equipment and emergency procedures "were adequate to have prevented the serious consequences of the accident, if they had been permitted to function or be carried out as planned." Trouble is, neither the equipment nor the preprogrammed safety procedures built into the Babcock & Wilcox reactor really got a chance.

The investigators confirmed that the plant's operators overrode the automatic safety systems in their attempts to correct the rapidly developing crisis that occurred when an electricity-generator turbine tripped, or shut itself down. Those actions, says the report, turned what should have been a relatively minor glitch into a potential disaster. Instead of letting the reactor's emergency core cooling system perform its safety functions, the operators paid "undue attention" to keeping the coolant from overfilling the reactor and refused to believe instruments indicating that the plant's fuel core was getting perilously hot.

Critical as the investigators may have been of the utility, the NRC itself got a wrist slap from Congress. In a report approved by a 29-to-2 vote, the House Government Operations Committee severely chided the commission for failing "to demonstrate strong constructive leadership" in developing evacuation plans and related emergency procedures for areas surrounding nuclear plants. Of 25 states that have these facilities, the study said, 16 do not have such NRC-approved plans. As one committee staffer summed up: the NRC just "pretended that accidents could not happen." ■

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

CHAIRMAN

October 16, 1979

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

Thank you for your letter of August 8, 1979 concerning NUREG-0600, "Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement."

Inasmuch as the Commission has not made a final determination on the ongoing NRC staff investigations of the accident, we feel that the detailed responses required by your questions and comments would be more appropriately addressed by the staff at this time. At the Commission's request, the staff has prepared the detailed responses which are enclosed.

Please be assured that the Commission will address all of the issues raised in NUREG-0600, as well as those raised in the remaining NRC staff investigations into the accident upon their conclusion.

If I can be of further assistance in obtaining additional information regarding NUREG-0600 or related matters, please let me know.

Sincerely,

Joseph M. Hendrie

Enclosure:
As stated

cc: The Honorable Steven D. Symms

A-175

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

October 4, 1979

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

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

Victor Stello, Director
Office of Inspection and Enforcement

THRU: Lee V. Gossick RB for LVG
Executive Director for Operations

SUBJECT: RESPONSE TO QUESTIONS RAISED BY CONGRESSMAN UDALL

This memorandum provides information in response to Congressman Udall's letter of August 8, 1979, as clarified by additional contacts between members of our staffs. Subsequent to the issuance of NUREG-0600, some statements and reports have suggested, contrary to our intent, that inappropriate operator actions were essentially the sole cause of the TMI-2 accident. In our opinion, some of these statements have placed undue emphasis on the operator deficiencies discussed in NUREG-0600. This may have resulted from a misunderstanding as to the scope of the investigation by our Office of Inspection and Enforcement which is reported in NUREG-0600. This investigation was limited in its scope to the actions of Metropolitan Edison over a specific time frame. There are several other investigations yet to be completed which will examine other possible contributing factors, such as activities of designers, reviewers, builders, vendors and regulatory agencies. It is most likely that the cause of the accident will be a combination of inadequacies that resulted from all of the foregoing. There is a staff consensus that the accident would not have occurred had the high pressure injection pumps not been throttled and eventually turned off. There is also consensus that the operators' actions were the result of inadequacies in equipment performance, transient and accident analyses, operator training and performance, equipment and system design, and information flow.

Congressman Udall's first question requested an analysis to support the conclusion in the report that operators should have been following procedures that pertain to a loss of coolant accident. The answer is given in the next paragraph. The statement in NUREG-0600 (Section I 2.15.1) which indicates that operator actions to limit the high pressure injection flow were influenced by their training was provided as an explanation; it was not intended to imply that this action was in accordance with the licensee's procedures.

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On March 28 the operators were faced initially with a loss of feedwater event, a primary coolant system <u>overpressurization</u> transient. They correctly identified this event. As the event progressed, the PORV opened as expected, but failed to close when the pressure was relieved. The failure of the valve to close initiated a small loss of coolant accident, which is a primary coolant system <u>depressurization</u> event. For a long time the operators did not recognize this change.

Circumstances which possibly contributed to the failure to recognize the loss of coolant accident included the following: (1) indication early in the transient of a full pressurizer, which was known to represent a potential hazard in terms of system overpressurization; (2) a lack of procedural instructions pointing to the possibility of a loss of coolant accident resulting from a loss of feedwater; for example, the loss of feedwater emergency procedures did not mention that the PORV is expected to open and failure of the valve to close would cause a loss of coolant accident; (3) according to the operators' training and Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure," two of the expected symptoms of loss of coolant accidents (low pressurizer level and high radiation alarm in the reactor building) did not exist as far as the operator knew; and (4) the PORV indicator light which gave an indication that the valve was closed. Notwithstanding these difficulties, some time (15 to 30 minutes) into the transient a careful evaluation of all indications, including reactor coolant system pressure and makeup tank level, could have convinced the operators that they were faced with a loss of coolant accident, and led them to the re-initiation of high pressure injection and the prevention of significant core damage. In summary, the operators did not believe they had a loss of coolant accident and did not respond accordingly.

The LOCA procedure, among others, was deficient in that it did not specifically caution the operators that in some circumstances, including a leak from the pressurizer steam space where the PORV is located, the pressurizer level may not be a reliable indicator of the primary system inventory. Had this caution been included, the procedure would have been a better one, and would have better aided the operators in more promptly reaching the proper diagnosis.

In this case, the operators' interpretation of the contradictory persistent symptoms led them to conclude that a loss of coolant had not been experienced. An important symptom, low reactor coolant pressure, was disregarded in view of another important indication, the high pressurizer level. Concern for not overfilling the pressurizer was based on not overpressurizing the primary coolant system.* This should not have been a concern because the inventory in reactor coolant system was low. Better training, particularly about the plant response in small loss of coolant accidents, would have given the operators an improved potential to understand what was happening. This better understanding could have made them less prone to rationalizing the symptoms, and ignoring a significant one. Furthermore, it would have tended to counteract the mind set which gave overwhelming importance to pressurizer level alone.

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*Operator training had emphasized the importance of not overfilling the pressurizer. The reason for this emphasis was principally a concern for assuring that the primary coolant system would not go solid which could result in an overpressurization event of short time constant.

The second question requested clarification as to whether procedures were violated by the operators' failure to trip the reactor coolant pumps fifteen minutes after the accident began. As Congressman Udall's letter suggests, there exists an apparent conflict between the operators' failure to trip the reactor coolant pumps and the requirements of Bulletins 79-05A, 79-05B, 79-06A and 79-06B, which directed that at least one reactor coolant pump be maintained operating. The NRC instructions in the early Bulletins were based on the observation that as long as the reactor coolant pumps were running, the TMI core was cooled and when the pumps were turned off with a large amount of steam in the system, natural circulation did not take place, resulting in core damage. At the same time, it was recognized that insufficient information was available to assess the effect of continuing to run the pumps based on analyses of the spectrum of break sizes. The licensees were requested to generate this information in a timely manner. Safety analyses are now available from all three PWR designers for both cases; i.e., pumps running and pumps turned off. According to these analyses, for a narrow range of small breaks, it is essential that the reactor coolant pumps be shutdown within minutes of the time of reactor trip and ECCS actuation. Continued operation of the pumps during the event increases the rate of mass loss from the break by forcing liquid instead of steam out through the break. If the pumps are subsequently tripped, at a point in the accident when steam volume fractions are high, fuel cladding temperatures could exceed the 2200° F limit specified by the ECCS Acceptance Criteria. It should also be noted that in the case of the B&W design, the pumps must be turned off within 2 or 3 minutes after reactor trip and ECCS initiation. Turning the pumps off later than 3 minutes could be harmful rather than helpful. Since this knowledge was not available prior to the TMI event, it is not surprising to find that the pump shut-off requirement of Procedure 2202-1.3 is deficient and needs to be modified.

Evaluation of the TMI-2 event shows that natural circulation did not take place when the pumps were tripped. Revised procedures, requiring refilling of the secondary side of the steam generators to 95% of the normal operating range when the pumps are tripped, provide reasonable assurance that natural circulation will take place in B&W plants as long as there is sufficient water in the primary coolant system to cover, at least partially, the reactor core and lower portion of the steam generator tubes. Accordingly, the current NRC view is that licensees should be required to shut down the primary coolant pumps immediately when high pressure coolant injection is caused by low primary coolant pressure. This was conveyed through the later set of Bulletins (IE Bulletins 79-05C, 79-06C).

While the operators did not follow the specific requirements of the Emergency Procedure, as is stated above, we now know that the procedure itself was deficient regarding the time of pump shutdown. It should be emphasized that we have not yet decided whether or not to consider this action an item of noncompliance. The coolant pump shutdown was labeled "under consideration as a potential item of noncompliance" in NUREG-0600. The ambiguity which Congressman Udall mentioned in his letter must certainly be a consideration in our final decision on the appropriateness of citing the utility for this action.

Congressman Udall's last question was directed to the availability of the reference documents in the Public Document Room. Most of these documents are now in the Public Document Room. The exceptions are documents to which are presently under review because they may contain proprietary information or because of Privacy Act considerations. We are attempting to clear these few remaining documents as rapidly as possible.

Harold R. Denton
Director
Office of Nuclear Reactor Regulation

Victor Stello
Director
Office of Inspection & Enforcement

News Release

General Public Utilities Corporation
260 Cherry Hill Road
Parsippany New Jersey 07054
201 253-4900

6.4 (NUREG-0600)

Further information

For release p.m.

Date October 30, 1979

WASHINGTON, D. C. -- "The Kemeny Commission conclusions released today lend support to our belief that the Three Mile Island accident involved the entire industrial, technological and regulatory structure of nuclear power," said William G. Kuhns, Chairman of General Public Utilities Corporation.

In emphasizing this point he cited the Kemeny Commission's assessment which states that "the accident occurred as a result of a series of human, institutional, and mechanical failures." He also pointed to a recent NRC staff memorandum which said: "There are several other investigations yet to be completed, which will examine other possible contributing factors, such as activities of designers, reviewers, builders, vendors, and regulatory agencies. It is most likely that the cause of the accident will be a combination of inadequacies that resulted from all of the foregoing."

Kuhns expressed concern that in the Commission's attempt to report an extremely complicated subject involving an equally complex interrelationship between the utility industry, its suppliers and its regulators, the result has been a series of capsulized statements which, of themselves, do not adequately reflect numerous underlying factors or their meaning.

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Kuhns observed that the investigation focused sharply on Met-Ed and made little attempt to evaluate the Company relative to industry practices.

"The accident identified a number of deficiencies that call for improved requirements and performance by all participants. Many of the Commission's broad conclusions are based on criteria which had not been identified prior to the accident or which are not directly related to the accident," he commented. Nevertheless, Kuhns said the Company is committed to addressing each of the Commission's findings and recommendations.

Included among the steps recently undertaken by GPU and Met-Ed, the subsidiary company which operates the Three Mile Island Nuclear Station, are improvements or modifications to the station's equipment, training and operating procedures.

Kuhns noted that the Kemeny Commission report states: 'The TMI training program conformed to the NRC standard for training. Moreover, TMI operator licensee candidates had higher scores than the national average on NRC licensing examinations and operating tests. Nevertheless, the training of the operators proved to be inadequate for responding to the accident.'

"Based upon performance in NRC exams from 1975 through 1978, the TMI control room operators ranked ninth in a group of 30 similar facilities. These facts attest to their skills," Kuhns said. "During that time span, 94% of TMI's applicants passed their license exams reflecting a failure rate one-half that of the industry average."

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"It is interesting to note," Kuhns added, "that all four licensed personnel on duty in the Unit 2 Control Room when the accident occurred had U.S. Navy nuclear program experience and each had roughly five years of TMI operating experience. Also, of the ten senior station personnel who arrived on site within three hours of the initiating events on March 28, seven had degrees in engineering or physics, and of those, two had advanced degrees."

"In fact," he continued, "of the 42 control room operators, shift supervisors and shift foremen assigned to TMI, 26 have Navy nuclear experience and each has a minimum of three to four years of experience at TMI with most having more than six years experience at the facility."

With respect to operator training and support, Kuhns said a graduate engineer would be on site at Three Mile Island at all times during plant operations to provide assistance and advise shift operating personnel. He pointed out that this action has already taken place at the Company's Oyster Creek Nuclear Generating Station, operated by its New Jersey subsidiary, Jersey Central Power and Light Company.

"All reactor operators will undergo extensive retraining and re-examination with increased emphasis on the basic elements of reactor safety that underlie the operating procedures. We have also requested NRC recertification of our operators," he added.

Kuhns re-emphasized that the personnel assigned to the station are among the most qualified in the industry and added that

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the number of personnel and level of operating and maintenance expenditures at GPU's nuclear plants have been well above the industry average.

Specifically, the TMI staff exceeded that of most similarly designed nuclear stations. A 1978 Edison Electric Institute survey of 27 pressurized water reactor nuclear plants showed that TMI had the second largest identified staff.

"According to Federal Energy Regulatory Commission reports for 1975 through 1977, TMI operating and maintenance expenditures were among the highest for similar plants," Kuhns said.

In a major move to further strengthen plant management and technical support, this summer the Company combined technical staffs from Met-Ed and the GPU Service Corporation to form the TMI Generation Group which significantly increased the depth of nuclear experience and more than tripled from 75 to 250 the number of professionals assigned exclusively to TMI activities.

Kuhns believes that the size of the GPU System, its resources, number of employees and years of nuclear experience stand in contrast to the Commission's contention that the Company lacked the knowledge, expertise and personnel to properly operate or maintain TMI.

"Ours is a utility system comprised of 11,000 employees serving 1.5 million customers. As an early participant in the commercial nuclear power program, and consistent with national policy, we constructed an experimental nuclear reactor in the early 1960's, activated the nation's first large-scale nuclear plant in

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1969, and in 1974 placed the first TMI unit in commercial operation. With the addition of TMI-2 late last year, the GPU System operated 2,300 megawatts of nuclear, 7,000 megawatts of coal-fired, and 1,400 megawatts of oil-fired capacity.

"The Oyster Creek and TMI-1 nuclear units have proven to be among the most efficient and productive nuclear facilities in the nation," he emphasized. "Combined, these two units had produced through August of this year over 63 million megawatt hours of electricity. As of February of this year, the GPU System ranked fourth among U.S. utilities in total lifetime production of nuclear generated electricity." As a result of this nuclear program, GPU saved its customers $700 million. These savings have increased rapidly with the continuing escalation of oil prices. But more importantly, the need to reduce our dependence on foreign oil is mandatory.

GPU has also cooperated with the electric utility industry's efforts over the past several months to conduct a searching review of equipment design, operator training and plant procedures.

Efforts began immediately after the accident, as individual utilities undertook a thorough audit of their own nuclear-plant operation and operator training; then quickly led to the formation of a Nuclear Safety Analysis Center (NSAC), to investigate and apply the technical lessons learned at Three Mile Island. In addition, the industry has formed the Institute of Nuclear Power Operations (INPO), a utility-financed organization that will establish benchmarks for excellence in nuclear-power operations, conduct audits to verify that

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these benchmarks are met, and analyze experience with operating reactors in order to share lessons learned with all utilities.

"Looking back upon the accident, and with the benefit of that experience, we have identified a number of elements that require strengthening.

"As an industry, we concentrated our attention on design features, reliability and operating procedures necessary to maintain the system, at all times, in a safe operating mode. One of the things that will be done with all nuclear plants will be to categorize and identify more clearly those major significant tell-tale indicators that allow the operators to more quickly size up the situation, to evaluate the exact level of potential impact on the local public, and to identify optimum emergency responses.

"We have also learned that in order for the public to be able to live with nuclear power, we must do a better job of increasing their understanding of the facts and terms associated with nuclear technology. The public must be able to sort, evaluate and put into perspective what is being said," Kuhns stated.

Kuhns commented that, despite the seriousness of the accident, he was pleased to see the Commission's conclusion that "the radiation doses received by the general population as a result of exposure to the radioactivity released during the accident were so small that there will be no detectable additional cases of cancer, developmental abnormalities or genetic ill-health as a consequence of the accident at TMI.'"

The utility executive believes that there is a need to improve the mechanism for identifying and evaluating operating

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experiences at all plants, interpreting the experiences in terms of their meaning relative to hardware and procedures, and training for safety. Kuhns said "we must make sure that all in-service experience is fed back as quickly and as efficiently as possible to the operators of all plants."

Kuhns expressed hope that potential modifications of the regulatory structure be accomplished without leading to further chaos in an already troubled national energy program.

"It is vital to maintain an effective source of independent public assurance," he commented.

Kuhns said the nuclear option should be preserved - not because it is perceived by some in the long run as less expensive but because of the need for diversified, domestic sources of energy.

"It would be hazardous for the country if we found ourselves totally dependent on any one supply or energy source. We need only look back a few years to the oil embargo and, more recently, to labor interruptions and severe winter weather to see the importance of a diversity of energy sources. We can ill-afford to remain captives of foreign energy supplies, nor can we place all of our hope on our coal reserves which bear a heavy environmental burden. When we examine all energy sources, whichever way we decide to go has risks. The key is to weigh all factors and put them into the proper perspective. The ultimate reason for nuclear power is not simple economics, it is diversity and domestic supply," he stated. "But above all, safety must be the keystone to all energy planning and production."

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NOT ON VYDEC
DRAFT:IC:bjw:11/6/79

TO: R: Muller

FROM: Ivan Catton

SUBJECT: NUREG-0600

NUREG-0600 implies that the reactor operators were at fault for not following the plant procedures as written. It is concluded that the operators were not properly trained and that their re-training was inadequate. It was also implied that the operators should have known that primary system cooling by natural circulation would have been difficult with a voided system. One is left with the belief that if the operators would have been more alert the accident would not have occurred. NUREG-0600 is unsatisfactory in that it does not attempt to go beyond a very legalistic view of the incident.

There were examples of instruments being improperly located (quench tank instruments behind the console), of data not available (in-core T/Cs) and of instruments with insufficient range (hot and cold leg T/Cs) as well as the poor performance of the plant computer that received little or no comment by I&E. If an operator action is incorrect as a result of how information is supplied to him or what information is supplied to him during an emergency, then the operator should not be at fault. To call the incorrect action operator error without determining whether or not the operator was led into the action by poor control room engineering is improper and without it the report is incomplete.

An operator who is considered poorly trained is not at fault for an action he takes as a result of his training. The entire procedure from licensing an operator to his being at control should be suspect. The guide lines set forth by the NSSS vendor, the interpretation by the utility, the training program leading to licensing the operator and his retraining all play a role. An example of training leading to problems is operators being trained to respond to pressurizer level yet expected to do otherwise. Further, the operators did not know to expect saturation on the primary side and as a result only looked at ΔT to determine whether or not they could go to natural circulation. Who is at fault? The weak link can only be found by a critical review of the process and some aspect, or many, should be faulted. It is my opinion that the NRC investigatory branch, I&E, should do so and their report should reflect the results of such a review.

The amount and quality of operator training must be a consideration when deciding whether or not a particular procedure is adequate. The report implies that the operators were at fault for not following the plant procedures. If one keeps the operator training in mind while reading procedures for mitigating a LOCA, one cannot conclude that the operators were at fault.

Loss of coolant was always described by two symptoms connected by an "and". Without the benefit of hindsight, the procedures do not seem to cover the event that occurred. Certain questions need to be considered before one can decide where the problem lies. Knowledge of the pressurizer level being inadequate for RCS status determination had been known by some for two or more years. Why wasn't this information fed into the training program? Another example of inadequacies in the process is operators not knowing that saturation can affect natural circulation. Procedures for going over to natural circulation do not mention avoidance of saturation. Is it the fault of the NSSS vendor, the utility, the NRC licensing process or all three when the operator tries to use natural circulation for cooling under saturated conditions? A proper and complete investigation of the TMI-2 incident should address all facets of an action that is improper.

The accident description is incomplete. It is my belief that the learning process would be enhanced if more detail about actions leading to the early water hammer and subsequent degrading of the secondary side were to be included. The rigid wall connection of air lines leading to air operated valves could not tolerate large amounts of pipe movement. It is not clear whether this was a design error or bad design not uncovered during review. It is, apparently, well known that water hammer is a common event and frequently leads to problems with the secondary side. The interconnection of plant air and instrument air coupled with certain practices for resin removal could have initiated the event. It would be helpful to know if any guidelines are given to a utility in this area and if guidelines exist, are they used.

NUREG-0600 contains a very good description of most of what took place during the TMI-2 incident. For the most part the long wait for its publication did not add substantially to knowledge available a few days following the accident. In depth assessment of where the Vendor-Utility-NRC-operator system was inadequate or in-violation does not seem to have been accomplished. Many of the details of the accident that would help us in the future do not seem to be covered.

The charter of the I&E investigative staff may have been too limited or its staff may have been poorly trained for such a task. If the I&E investigatory staff did not have proper training, experience or manpower for the task then NRC should look within and remedy the problem.

NOT ON VYDEC
DRAFT 2:CM:bjw:11/6/79

TO: Harold Etherington, Chairman, ACRS TMI-2 Subcommittee

FROM: C. Michelson, ACRS Consultant

SUBJECT: ADEQUACY OF TMI-2 EMERGENCY PROCEDURES FOR THE CASE OF A LOSS OF REACTOR COOLANT AT THE TOP OF THE PRESSURIZER

The attached report presents my views concerning this subject.

The conclusions indicate that the applicable procedure for the TMI-2 accident should have been 2202-1.5 (Pressurizer System Failure). However, in my opinion this procedure was unacceptable for that purpose or for any other loss of reactor coolant at the top of the pressurizer.

Emergency Procedure 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure) was also examined and found to provide confusing symptoms and instructions for the case of a loss of reactor coolant at the top of the pressurizer. Therefore, I believe this procedure was not adequate to assure a correct response to the TMI-2 accident. In addition it was not the correct procedure to follow in view of the observed symptoms.

Distribution:

ACRS Members (235th Meeting Folder)
ACRS Technical Staff
I. Catton
W. C. Lipinski
T. Theofanous
E. Jordan, I&E

ADEQUACY OF TMI-2 EMERGENCY PROCEDURES FOR THE CASE
OF A LOSS OF REACTOR COOLANT AT THE TOP OF THE PRESSURIZER

C. Michelson
November 1, 1979

The forward to NUREG-0600 claims that the accident at TMI-2 could have been prevented in spite of certain inadequacies. It states, "The design of the plant, the equipment that was installed, the various accident and transient analyses, and the emergency procedures were adequate to have prevented the serious consequences of the accident, if they had been permitted to function or be carried out as planned."

NUREG-0600 is undoubtedly a comprehensive investigative report of the accident and a credit to the meticulous efforts of many competent people. I have no specific comments or concerns relating to the scope or general content of the report at this time, but I am having some difficulty reconciling the above stated conclusion with my own observations which are, admittedly, based on a more limited viewing of the situation.

There is little doubt that the accident at TMI-2 could have been terminated without significant consequences by a timely closure of the PORV block valve through operator action. However, the plant was designed to be forgiving and it was verified by analysis to be fully capable of handling this lack of action without unacceptable consequences. It was an established design requirement to accommodate a postulated pipe break upstream of the PORV block valve or elsewhere at the top of the pressurizer or a failed open code safety valve. For such cases, termination of the resulting loss of reactor coolant by operator action would not be possible. The equipment required to fulfill this requirement was operable during the accident, but it is not clear to me that the emergency procedures in effect were adequate to assure a proper operation for this specific loss-of-coolant situation and thus prevent serious consequences.

The only TMI-2 emergency procedure which appears to be directly applicable to the accident situation is 2202-1.5 (Pressurizer System Failure). A portion of this procedure deals with a leaking or failed open PORV or code safety valve which was the situation for over two hours. The symptons and automatic actions outlined in this procedure match closely those observed during the accident. However, some of the observed symptons and automatic actions are also indicative of those caused by a loss of reactor coolant, so the procedure to consider might be 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure). This procedure deals with a small leak or rupture which is within the system capability, and a large leak or rupture which leads to the automatic actuation of the engineered safety features. Some of the symptons outlined in this procedure did not match those observed during the accident.

I am not sure which of these procedures the operator thought he was following during the first hours of the accident, so I examined both to determine their applicability and adequacy. My conclusions are based on the following observations which were derived from this examination.

Emergency Procedure 2202-1.5 (Pressurizer System Failure)

This emergency procedure contains a Part B which deals with a failed open PORV and a Part D which deals with a failed open code safety valve. The procedure indicates that both of these conditions lead to symptoms of elevated valve discharge pipe temperature, elevated reactor coolant drain tank pressure and temperature, and the automatic actuation of high pressure injection. The procedure calls for manual closure of the PORV block valve if the PORV fails to close (B.2.B.2.a). For a failed open code safety valve, the procedure instructs the operator to attempt to control pressurizer level using safety injection valve MV-V16B (D.2.B.2.c). It also stipulates to manually initiate additional safety injection if required to maintain pressurizer level (D.2.B.2.d). As a follow-up action, the procedure specifies holding the pressurizer level, if possible, at or greater than 220 inches with safety injection (D.3.2.a).

During the TMI-2 accident, the failed open condition of the PORV was not directly apparent to the operator because the valve position indicating light showed the valve to be closed. The discharge pipe temperature was high on both the PORV and code safety valves. Since the individual valve discharge pipes are joined together, it is usual to experience high temperature on all discharge pipes if any one valve is open. The operator was probably aware of or anticipated that the loss of main feedwater transient would open the PORV and perhaps one or more code safety valves. He could not tell that the code safety valves closed.

The subsequent elevated reactor coolant drain tank pressure and temperature, and the automatic actuation of high pressure injection were additional direct indications of a failed open PORV or code safety valve as opposed to a possible loss of reactor coolant due to a pipe leak or rupture. Since the PORV position light was indicating closed, it would be reasonable to conclude that all of these symptoms were due to a failed open code safety valve. For this case, the applicable procedure is 2202-1.5 which calls for pressurizer level control.

The actual operator response during the accident appeared to follow this procedure. Unfortunately, the procedure is unacceptable for a failed open code safety valve (or a failed open PORV with a defective position indication). For this case, a rapid pressurizer refilling occurs and the level can appear to stabilize even though the core becomes uncovered. The high level in the pressurizer obligates the operator to throttle back on high pressure injection to control level as required by the procedure, and this leads to unacceptable consequences as found out during the TMI-2 accident.

Emergency Procedure 2202-1.5 does not explicitedly warn the operator with a symptom statement that pressurizer level will rise while the reactor coolant system pressure is falling. However, this possibility should be apparent from the requirement to control pressurizer level at or greater than 220 inches by the addition of safety injection while the pressure is falling below 1600 psig. An increasing pressurizer level with decreasing reactor coolant system pressure should not confuse the operator if he believes the event to be a failed open PORV or code safety valve.

Emergency Procedure 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure)

This emergency procedure contains Part A which deals with a, "Leak or Rupture Within Capability of System Operation," and Part B which deals with a, "Leak or Rupture of Significant Size Such That Engineered Safety Features are Automatically Initiated." The procedure indicates that both of these conditions lead to symptoms of decreasing reactor coolant pressure and pressurizer level. For Part A, the level will stabilize with time. For Part B the level will continue to decrease.

At TMI-2, the accident condition of interest was a failed open PORV which remained undetected. This condition was a small break (less than 0.5 ft^2) loss-of-coolant accident until terminated by closure of the upstream block valve. However, the pressurizer level response during this event was not indicative of that predicted by the procedure. For the leak experienced, the pressurizer level soon started to increase instead of stabilizing or continuing to decrease as the system depressurized.

The reason for this difference from predicted behavior is straightforward. A loss of reactor coolant at the top of the pressurizer will produce an increasing pressurizer level response whether the coolant loss is due to a pipe leak or rupture, or a failed open safety or relief valve. A similar loss of reactor coolant from a leak or rupture in a hot or cold lege pipe will produce a decreasing pressurizer level response. The symptoms identified in the emergency procedure are those corresponding to a hot leg or cold leg pipe leak or rupture. These symptoms were not observed during the first two hours of the accident at TMI-2 because the loss of reactor coolant was at the top of the pressurizer.

At this point it should be questioned why the operator would consider further the applicability of this procedure when the observed symptoms directly match those of a failed open code safety vlave (or a failed open PORV which remains undetected) and do not match those of a LOCA. The only significant indicator of a LOCA was the decreasing reactor coolant pressure. The pressurizer level did not behave as predicted and the primary containment response was noticeably delayed. The observed elevation of safety and relief valve discharge pipe temperature and the elevated reactor coolant drain tank conditions were not mentioned in the procedure and are not indicative of a pipe leak or rupture LOCA condition.

From the viewpoint of adequacy, this procedure appears to be acceptable for hot and cold leg pipe leaks or ruptures, but it may be confusing to apply for a loss of reactor coolant at the top of the pressurizer. For this case, the operator would have to ignore the conflicting pressurizer level observations and concentrate on reactor coolant system pressure as the controlling indicator when electing which part of the procedure to use. Guidance concerning the possibility of an increasing pressurizer level with decreasing system pressure is not provided in the procedure.

Conclusions

The early symptoms of the TMI-2 accident were those associated with a failed open code safety valve (or a failed open PORV with a defective position indication). The emergency procedure for a failed open code safety valve is 2202-1.5 (Pressurizer System Failure). This procedure calls for operator actions which closely approximate those performed by the TMI-2 operators during the first two hours of the accident. Unfortunately, this procedure specifies pressurizer level control which is not an acceptable response to this loss of reactor coolant situation. This procedure was, therefore, unacceptable for the TMI-2 accident case.

Emergency Procedure 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure) is not directly applicable to the case of a loss of reactor coolant at the top of the pressurizer. This procedure appears to be based on the reactor coolant system response to a hot or cold leg break. It contains no guidance concerning the unique response of a leak or rupture at the top of the pressurizer. Its use may cause operator confusion whenever the observed pressurizer level is increasing during an emergency because the procedure indicates only a decreasing level. This procedure was, therefore, not adequate for the TMI-2 accident case. In addition, it was not the correct procedure to follow in view of the observed symptoms.

US Department of Energy
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439
Telephone 312/972-4639

November 6, 1979

Mr. Harold Etherington
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Etherington:

Subject: Review of Report No. NUREG-0600, "Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement," August 1979.

The I&E report did not determine conclusively whether water introduced into the instrument air lines contributed to the initiating transient of loss of feedwater at TMI-2. Until such time as nuclear plants have separate dedicated full-pressure decay heat removal systems, the secondary system in a plant plays a very important role.

At the TMI-2 meeting on October 30, 1979, the behavior of the secondary system was discussed with the I&E staff. The I&E inspection did not determine whether the water in the secondary system instrument air lines caused valves to close and result in a water hammer. The water hammer in turn tore loose air connections to control valves which controlled the hot well level. Loss of level control resulted in a full hot well and required that the steam generators be vented to the atmosphere. Under the condition of leaking steam generators, radioactivity is released to the atmosphere. In the case of TMI-2 fortunately, only one steam generator developed a leak and was valved out of the system. The leak in the one steam generator developed as a result of allowing the steam generators to boil dry and then introducing cold feedwater to regain steam generation and heat removal.

The I&E staff stated that on loss of feedwater the turbine is tripped and it is common to have a water hammer event following turbine trip. It is important to determine whether the TMI-2 turbine trip caused the water hammer which in turn caused the air lines to be torn loose. If the turbine trip resulted in tearing loose air lines, there was something basically wrong with the design. Does this condition exist in other plants?

The TMI-2 instrument air system and plant service air system were deliberately interconnected by design to provide a fix because one of the systems did not have sufficient capacity. Does this condition exist in any other plant?

During the course of the TMI-2 accident, the emergency diesel generators started as per design and were subsequently tripped because off-site power available. The diesels are stopped by manually tripping the fuel racks

and the fuel racks must be manually reset to enable the diesels to perform an auto-restart on demand. In the TMI-2 sequence, the diesel fuel racks were tripped and not reset. The I&E staff were not able to determine who tripped the diesels and failed to reset the fuel racks. Had there been a loss of off-site power during the course of the TMI-2 accident after the diesels were disabled, the consequences could have been more severe if there had been a total loss of ac power during the interval that it would have taken to start the diesels. The I&E staff should have clearly established who tripped the diesels and why the fuel racks were not reset. Was it because of improper procedure, deliberate, or because of improper training? The TMI-2 diesel fuel racks were subsequently engaged and manual control over the diesels was provided by manipulation of switches in the control room after an engineer noticed the disabled condition and recommended against continued plant operation with the diesels disabled.

Sincerely,

Walter C. Lipinski /at

Walter C. Lipinski
Senior Electrical Engineer

WCL/at

A-199

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

April 20, 1979

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

THRU: Lee V. Gossick
Executive Director for Operations

FROM: John G. Davis, Acting Director
Office of Inspection and Enforcement

SUBJECT: INVESTIGATION OF THREE MILE ISLAND ACCIDENT BY THE OFFICE OF INSPECTION AND ENFORCEMENT

This is to confirm our discussion of April 13, 1979, concerning the Three Mile Island accident investigation underway by the Office of Inspection and Enforcement (IE).

The investigation performed by IE has two basic goals:

1. To establish, in a comprehensive manner, the facts concerning the Three Mile Island accident. The parameters of this effort are further described in Attachment A.

2. To evaluate the performance of the licensee in association with the Three Mile Island accident as a basis for corrective action or enforcement action as appropriate.

The Office of Inspection and Enforcement investigation does <u>not</u> include, under our current plans, the following:

1. Any evaluation of the actions of the NRC or any of its organizational components during the course of this accident or recovery period.

2. Any evaluation of the actions of other agencies during the course of the accident, or during the recovery period of the accident.

3. Any review and evaluation of the NRC regulatory process as it relates to the Three Mile Island accident for "lessons learned." IE is not collecting information concerning nor evaluating:

   - Legislative authority of the NRC
   - Rules and regulations of the NRC
   - Safety research
   - Licensing process
   - Inspection and enforcement process

The Office of Inspection and Enforcement will expand its investigation as directed by the Commission or the EDO. In the absence of such direction, IE is proceeding with its investigation as outlined in this memorandum.

John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosure:
IE/TMI Conceptual Outline

A-201

OFFICE OF INSPECTION AND ENFORCEMENT
THREE MILE ISLAND ACCIDENT INVESTIGATION

Conceptual Outline

1. The IE investigation of the Three Mile Island accident is directed toward:

   a. Within the time period of the investigation, establishing the facts concerning the immediate causes of the accident and the actions of the plant and the licensee staff during the course of the accident.

   b. Within the time period of the investigation, establishing the facts concerning the actions of the licensee, the NRC, other Federal agencies, and appropriate state agencies.

   c. Evaluating the performance of the licensee during and in response to the Three Mile Island accident as a basis for corrective action or enforcement action as appropriate.

2. The investigation consists of two parts conducted in parallel:

   a. Operational - The inplant, reactor operations situation. This will cover the time period from the closing of the auxiliary Feedwater System valves (or other earlier immediate cause of the accident) to the restart of reactor coolant pump 1A (about 8:00 p.m., March 28, 1979).

   b. Radiological - The inplant and environmental radiological conditions. This will cover the time period from the beginning of the accident until about midnight on March 31, 1979.

3. The Office of Inspection and Enforcement will develop factual information concerning the condition of the plant and the environment and the performance of the licensee for the time periods subsequent to those described in Item 2, above. However, this information will not be included in the IE Three Mile Island Accident Investigation. (Note that this does not include development of information concerning NRC activities. The information will include the licensee's reaction and response to NRC activities but will not develop, for example, how an NRC position or recommendation was formulated.)

4. The investigation will include:

a. Sequence of Events

(1) Operational

Identify the sequence of events in relation to facility operation. Determine licensee actions or lack of actions related to operation of the facility within licensed parameters, constraints and limits. Determine the participation of licensee supervision, management and engineering support. Determine causes of the event.

(2) Radiological

Identify the sequence of events in relation to licensee activities in radiation control on site and off site. Determine licensee actions or lack of actions related to controlling and monitoring on site exposures, protection of workers, and control and monitoring of off site releases.

b. Immediate Cause of Accident

(1) Equipment

Trace the performance and maintenance history of important equipment which malfunctioned at the beginning or during the early phases of the incident. Define the signals or other intelligence provided to operators concerning serviceability and availability of equipment. Identify the serviceability of equipment required for operation.

(2) Procedures

Determine the requirements contained in surveillance and maintenance procedures and the appropriateness f procedures. Identify failures to follow procedures. Determine appropriateness of communications between plant groups on plant and equipment status. Determine whether emergency operating procedures were appropriate and were followed.

(3) Staff Performance

Describe the performance of the operators and other licensee personnel during the accident. Review training of operators particularly training concerning response to off-normal limits.

c. Licensee Management of Accident

Describe the response of the licensee from the standpoint of management and supervision of the accident. Identify the engineering and radiological support requested and received and its source. Determine the licensee's reaction to and analysis of the accident as it unfolded with particular emphasis on engineering analysis of alternative modes of accident recovery.

d. Emergency Plan Activation

Examine the licensee's emergency plan implementation to include preplanning and tests. Develop a detailed chronology of the implementation of the emergency plan with particular emphasis on timeliness of notification.

5. This investigation is being managed by the NRC Region I Office.

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

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

THRU: Lee V. Gossick
Executive Director for Operations

FROM: John G. Davis, Acting Director
Office of Inspection and Enforcement

SUBJECT: INVESTIGATION OF THREE MILE ISLAND ACCIDENT BY THE OFFICE OF INSPECTION AND ENFORCEMENT

Reference is made to my memorandum of above subject dated April 20, 1979.

The investigation by the Office of Inspection and Enforcement of the accident and of the licensee's actions during the course of the accident has been underway for approximately seven weeks. During the conduct of this investigation, IE has interviewed approximately 110 people and has examined other information sources in its efforts to determine what transpired, the proximate causes, and licensee actions before and during the incident proper.

As a result of a current review of the progress of the investigation, I believe the following matters should be specifically called to your attention:

1. The investigation is complex, involving extensive interplay of mechanical and control systems with human actions. This complexity has been further compounded by the need to release information prior to the completion of the investigation, by other concurrent investigations and by the need to reply to specific questions prior to completion of the IE investigation. This <u>may</u> lead to IE being unable to meet its projected August 1, 1979 date for the investigation report. We are currently examining our schedules to account for anticipated additional investigatory work. We will inform you by June 25, 1979, of any necessary rescheduling of the projected date for the investigation report.

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2. The thrust of the IE investigation is to establish facts and to evaluate the performance of the licensee. In performing its investigation, IE has attempted to focus its investigation in these areas. Consequently, it should be clear that IE's investigation concerning the facts about actions of the NRC, other Federal agencies, and appropriate state agencies will only establish facts (within the time periods of the IE investigation) and only insofar as those actions influenced the actions of the licensee.

3. The time period for the operations part of the IE investigation is from the surveillance testing of the auxiliary feedwater system (March 26) to the restart of the reactor coolant pump 1A (about 8:00 p.m., March 28, 1979).

4. The time period for the radiological part of the IE investigation is from the beginning of the accident until about midnight on March 30, 1979. This previously had been stated as "about midnight on March 31, 1979." The investigation thus far apparently has determined that the significant radiological actions under the licensee's control occurred prior to midnight, March 30. In addition, after that date we believe the influence of the NRC was sufficient to make separation of the licensee's independent actions difficult. Consequently, the current plans are to close the period of the IE radiological investigation at about midnight, March 30, 1979.

5. As discussed with you, the initial round of interviews has been aimed at determining facts. Subsequent interviews will aim at establishing causes. For these subsequent interviews, IE believes the authority to administer oaths to the interviewees may be of value. Consequently, IE will be requesting by an action paper delegation of authority to IE investigators to administer oaths.

Should you desire to discuss these matters, I shall be happy to do so.

John G. Davis
Acting Director
Office of Inspection
and Enforcement

cc: See page 3

A-206

cc: E. Kevin Cornell, EDO
Harold R. Denton, NRR
Saul Levine, RES
Robert B. Minogue, SD
William J. Dircks, NMSS
James J. Cummings, OIA
Robert G. Ryan, SP
James R. Shea, IP
Harold D. Thornburg, IE
Norman C. Moseley, IE
E. Morris Howard, IE
James H. Sniezek, IE
Boyce H. Grier, IE
James P. O'Reilly, IE
James G. Keppler, IE
Karl V. Seyfrit, IE
Robert H. Engelken, IE
Dudley Thompson, IE
Leonard I. Cobb, IE
James M. Allan, IE

DISTRIBUTION:
L. V. Gossick
T. A Rehm
J. G. Davis
EDO RF
IE RF
SECY (3)
MPA

OFFICE	IE:AD	ELD	EDO			
SURNAME	JGDavis:sc		LVGossick			
DATE	6/6/79	6/ /79	6/ /79			

OPERATIONAL ASPECTS

GENERAL SUMMARY

A. CONDITIONS PRIOR TO TURBINE TRIP

1. 97% POWER WITH ICS IN FULL AUTO
2. NORMAL MAKEUP AND LETDOWN
3. ONE IDENTIFIED ACTION STATEMENT
4. RCS UNIDENTIFIED LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATIONS (PROCEDURAL ERROR)
5. WATER ADDITIONS TO RCS INCREASED IN HOURS PRIOR TO TRIP
6. EMOV AND SAFETY VALVE TAILPIPE TEMPERATURES ABOVE PROCEDURAL LIMITS (NEW PROCEDURAL GUIDANCE NOT PROVIDED OPERATORS)
7. STAFF ON DUTY MET TECHNICAL SPECIFICATION REQUIREMENTS (ADDITIONAL SHIFT SUPERVISOR DUE TO UNIT 1 RESTART)

8. TWO AUXILIARY OPERATORS AND FOREMAN WORKING TO TRANSFER RESIN IN CONDENSATE POLISHER

B. TURBINE TRIP

1. TURBINE TRIP OCCURRED 04:00:37 (MARCH 28, 1979)

2. CAUSED BY LOSS OF ALL FEEDWATER

3. CAUSE OF FEEDWATER LOSS NOT DEFINITELY DETERMINED

(PROBABLE CAUSE RELATED TO CONDENSATE POLISHER OPERATIONS)

C. SEQUENCE OF EVENTS

1. DETAILED SEQUENCE
CONTAINED IN APPENDIX I-A

2. SELECTED ASPECTS OF SEQUENCE

A. EMOV FAILED TO RECLOSE AFTER OPENING

B. LARGE LOSS OF RCS INVENTORY

C. PRESSURIZER LEVEL REMAINED HIGH DESPITE INVENTORY LOSS -RCS PRESSURE DROPPED

D. REACTOR COOLANT PUMPS TRIPPED AT 74 AND 101 MINUTES - NATURAL CIRCULATION WAS NOT ESTABLISHED.

E. OPEN EMOV ISOLATED AT 2.3 HOURS AFTER TRIP

F. BY 2½ HOURS, CORE HAD BEEN UNCOVERED TO SOME DEGREE, FISSION PRODUCTS RELEASED, AND HYDROGEN GENERATED.

G. REMAINDER OF SEQUENCE:

(1) REPRESSURIZE TO FILL LOOPS TO ESTABLISH NATURAL CONVECTION

(2) DEPRESSURIZE TO USE DECAY HEAT SYSTEM

(3) REPRESSURIZE TO FILL LOOPS AND START RCP.

H. RCP STARTED AND FORCED CIRCULATION ESTABLISHED APPROXIMATELY 16 HOURS AFTER TURBINE TRIP.

D. SHIFT CREW ACTIONS

1. BACKGROUND

   - OPERATORS TRAINED TO AVOID SOLID PRESSURIZER
   - OPERATORS TRAINED THAT ANY RCS INVENTORY LOSS WILL BE SEEN AS A LOW PRESSURIZER
   - OPERATING EXPERIENCE WITH PRIOR TRIPS SHOWED THAT PRESSURE RECOVERED WHEN LEVEL REESTABLISHED

2. REVIEW OF ACTIONS

   A. MOST SIGNIFICANT ACTIONS

      - THROTTLING HPI TO MINIMUM
      - FAILURE TO ISOLATE EMOV

   B. ACTIONS THAT DID NOT CONTRIBUTE TO ACCIDENT AS IT EVOLVED.

      - AUTOMATIC START CAPABILITY OF EMERGENCY DIESELS DISABLED AFTER FIRST HIGH PRESSURE INJECTION

- CORE FLOOD TANKS ISOLATED DURING FIRST LOW PRESSURE PERIOD

E. <u>MANAGEMENT ACTIONS</u>

1. PLANT PARAMETERS UTILIZED IN PLANNING ACTIONS

2. EXCEPTIONS

   - DISBELIEF OF HIGH SYSTEM AND INCORE TEMPERATURES

   - CORE FLOOD TANK BEHAVIOR MISUNDERSTOOD

   - PRESSURE SPIKE IN BUILDING NOT PURSUED

3. INVESTIGATION DID NOT ATTEMPT TO CONCLUDE WHAT OUTCOME WOULD HAVE BEEN UNDER DIFFERENT CONDITIONS

F. <u>OFFSITE INTERFACES</u>

1. LICENSEE ENGINEERING STAFF

2. BABCOCK AND WILCOX

3. BURNS AND ROE

4. NRC

OBJECTIVE:

To Determine Facts and Assess Licensee Performance Regarding:

(1) Emergency Preparedness Prior to the March 28 Incident and,

(2) Response to In-Plant and Environmental Radiological Conditions Following the Incident.

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SCOPE:

(1) Investigation of Emergency Preparedness Includes Review of Emergency and Health Physics Training Conducted by the Licensee Prior to March 28 and Review of Equipment, Supplies and Procedures Needed for Implementation of the Site Emergency Plan.

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SCOPE: (CONT'D)

(2) Emergency Response Actions From the Time of the Incident Until Midnight on March 30 are Being Investigated Including Actions to Detect and Classify the Incident, Activate the Emergency Organization, Monitor and Control Effluents, Control Occupational Radiation Exposure, and Assess Environmental Conditions. Actions Taken by the Commonwealth of Pennsylvania, the NRC, or Other Government Agencies Will Not be Included in This Investigation.

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RADIOLOGICAL TEAM ORGANIZATION

A.F. Gibson — Team Leader

D.M. Collins — Technical Assistant

D.E. Donaldson — Emergency Planning

T.H. Essig — Environmental Monitoring

L.L. Jackson — Effluent Control

G.P. Yuhas — In-Plant Radiation Protection

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SOURCES OF INFORMATION

Licensee Logs

Licensee Records

Transcribed Telephone Communications

Discussions

Interviews

A-217

PREACCIDENT CONDITIONS

- RADIATION PROTECTION & CHEMISTRY STAFF
- EMERGENCY DRILLS
- EMERGENCY PLAN TRAINING
- ROUTINE RADIATION PROTECTION TRAINING
- RADIATION PROTECTION EQUIPMENT & SUPPLIES
- EMERGENCY EQUIPMENT
- ROUTINE ENVIRONMENTAL MONITORING PROGRAM
- RADWASTE SYSTEMS

INITIAL EMERGENCY RESPONSE

- DETECTION AND CLASSIFICATION
- ORGANIZATION ACTIVATION
- NOTIFICATIONS

TMI EMERGENCY PLAN
TABLE 1

Site Emergency — Condition (c)

"c) Loss of Primary Coolant Pressure, Coincident With High Reactor Building Pressure and/or High Reactor Building Sump Level."

General Emergency — Condition (a)

"a) Reactor Building High Range Gamma Monitor High Alarm."

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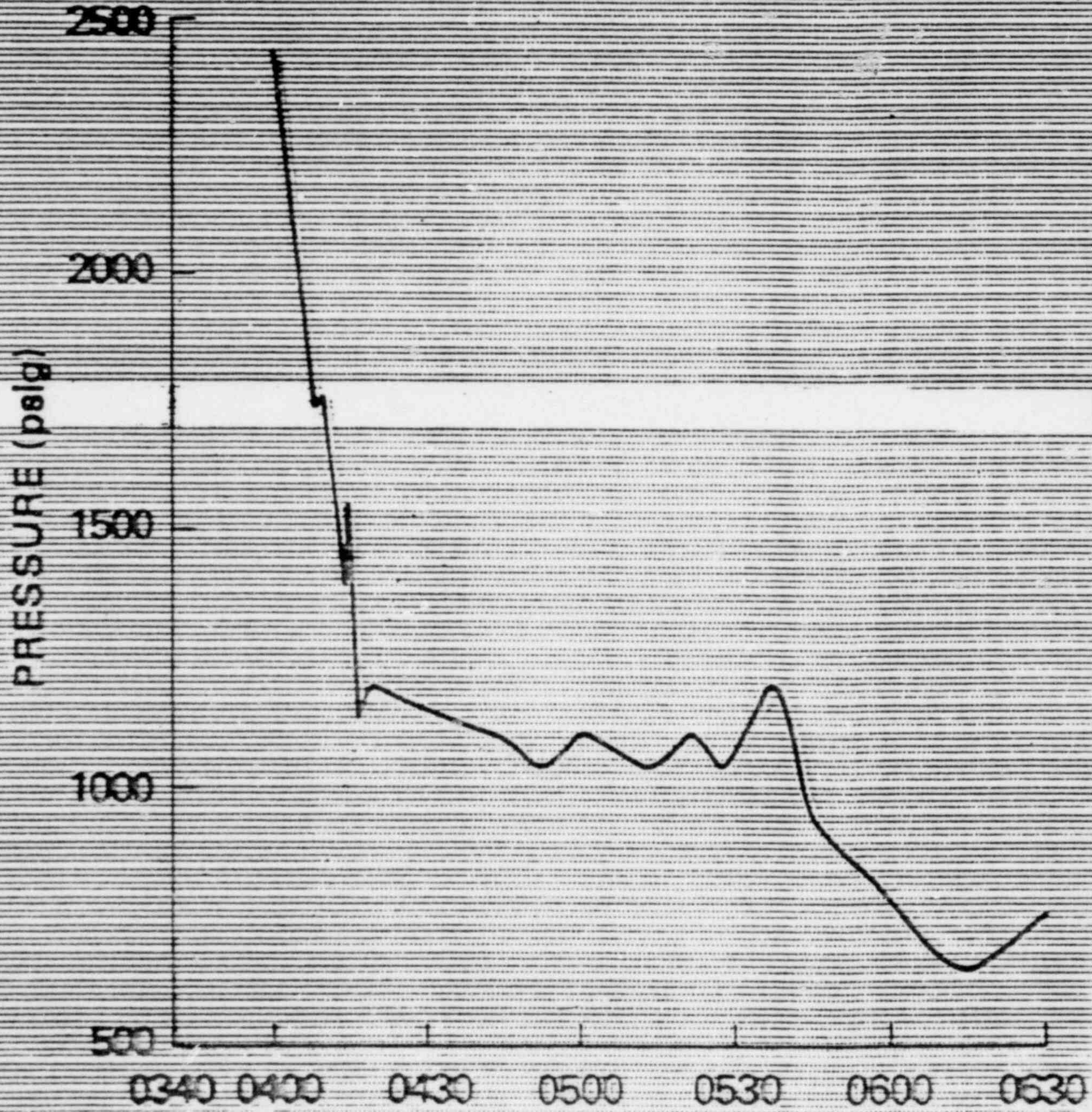
REACTOR COOLANT PRESSURE VS. TIME ON MARCH 28, 1979, AS DISPLAYED ON THE WIDE RANGE PRESSURE RECORDER



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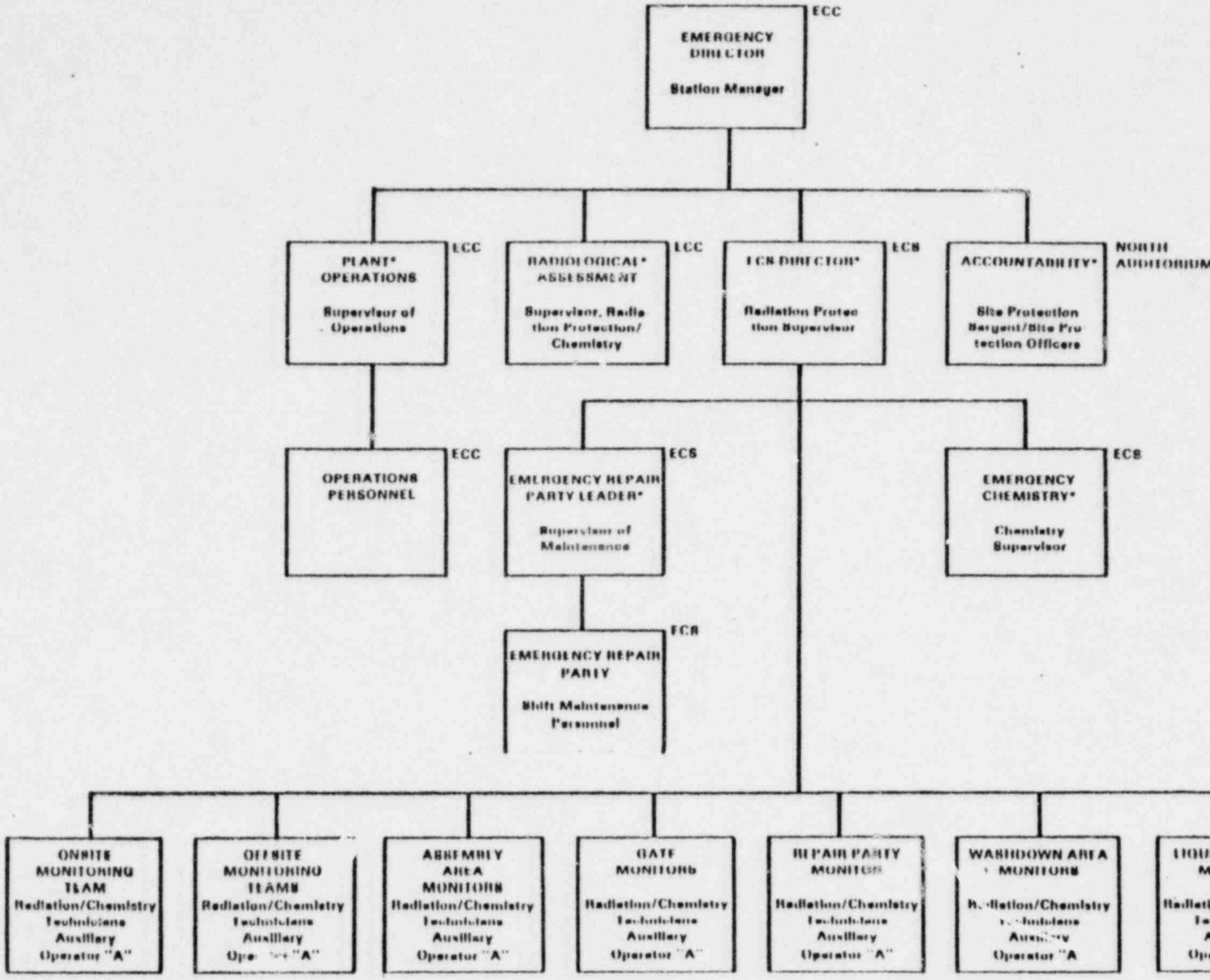
"NORMAL" EMERGENCY ORGANIZATION



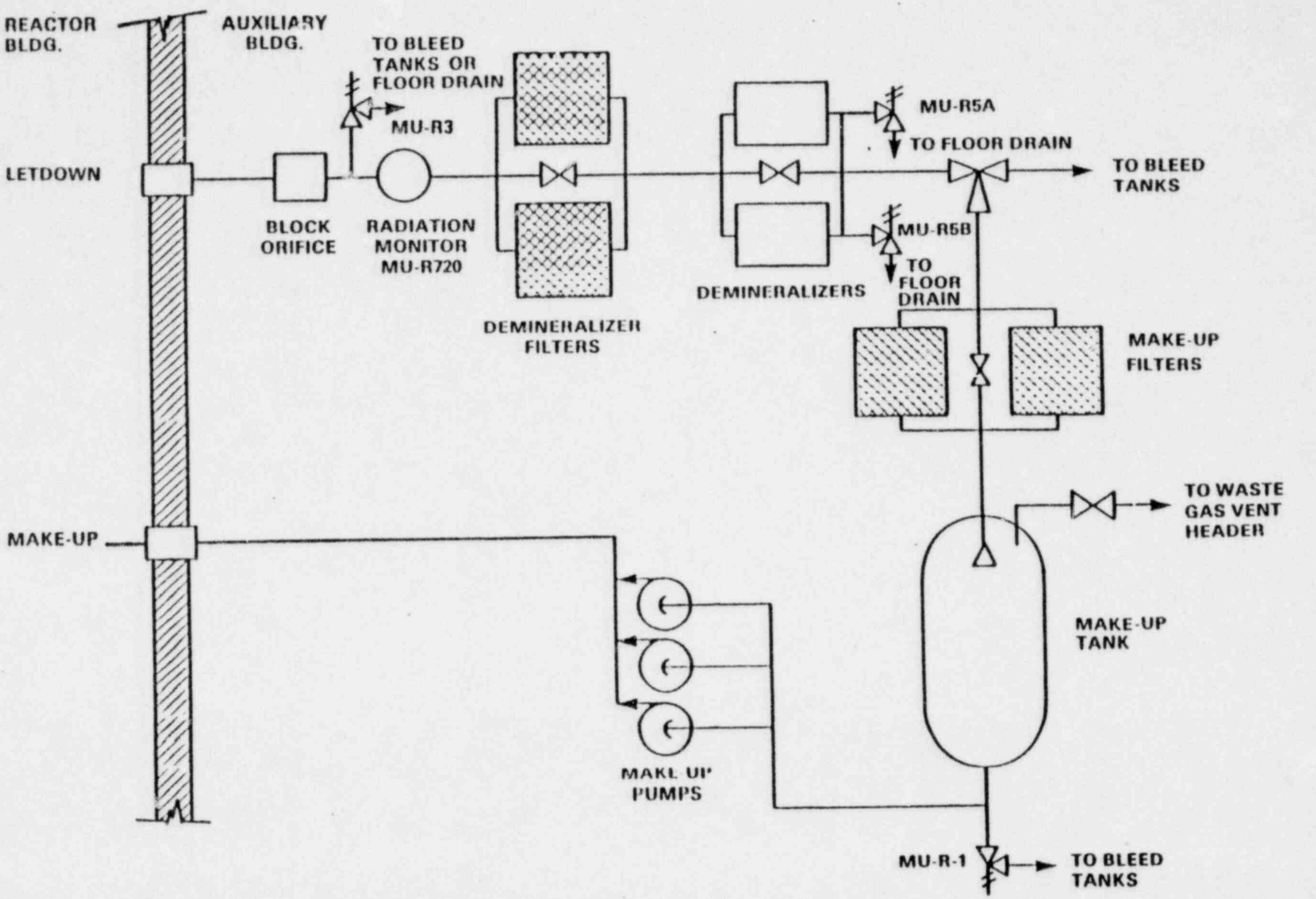
A-222

ECC EMERGENCY CONTROL CENTER, UNIT 2 CONTROL ROOM
ECS EMERGENCY CONTROL STATION, UNIT 1 CHEMISTRY/HEALTH PHYSICS LAB AREA
* FUNCTIONAL TITLE ADDED FOR CLARITY, LICENSEE'S PLAN LISTS NORMAL DUTY TITLES

RADIOACTIVE EFFLUENTS

- PATHWAYS
- MONITORING
- QUANTIFICATION

REACTOR COOLANT MAKE-UP AND PURIFICATION SYSTEM
(SIMPLIFIED DRAWING)



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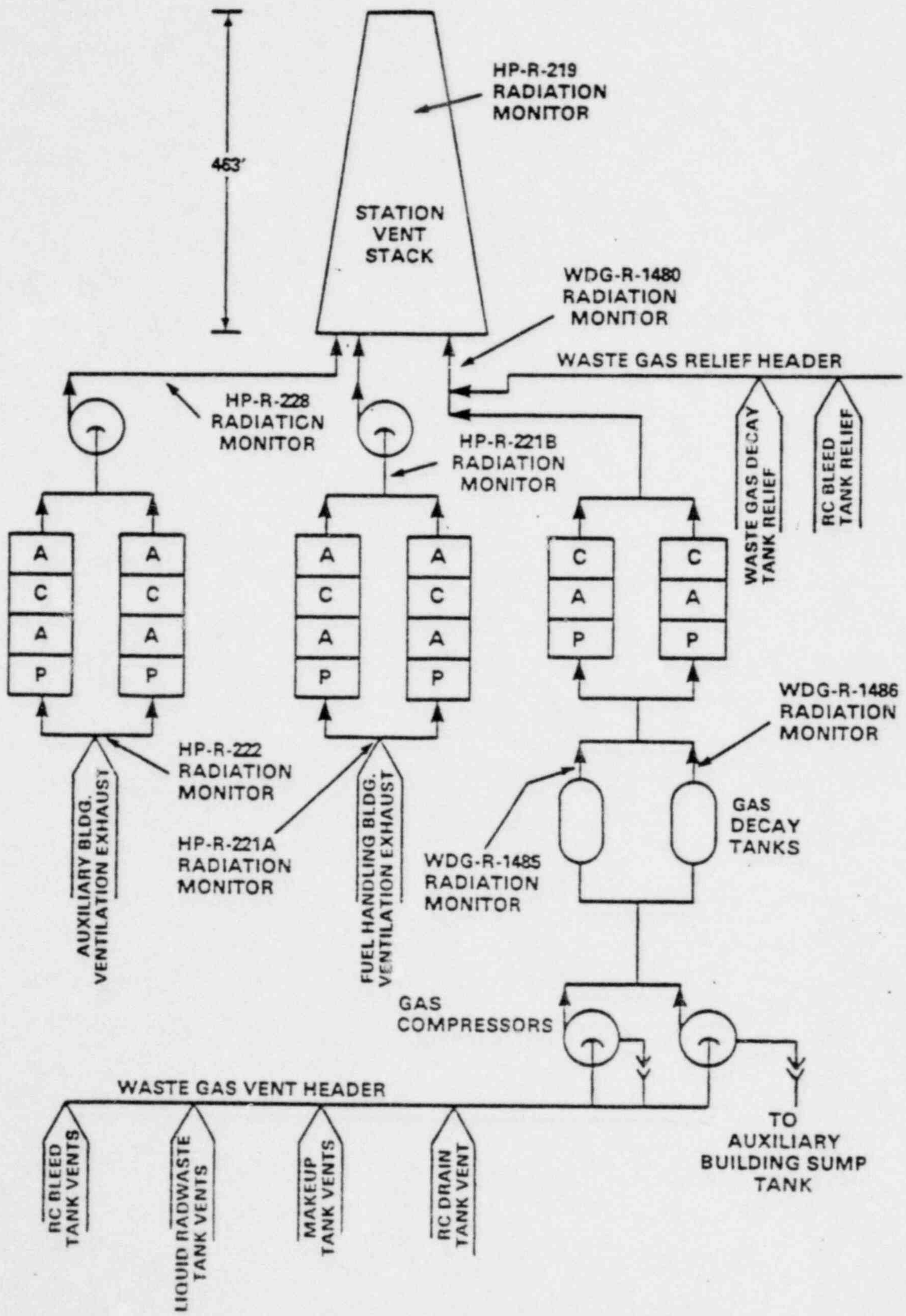
INPUTS TO STATION VENT
(SIMPLIFIED ILLUSTRATION)



PLANT RADIATION PROTECTION

- IMPLEMENTATION OF EMERGENCY PLAN
- ASSESSMENT OF IN-PLANT RADIOLOGICAL CONDITIONS
- ACCESS CONTROL
- AIR SAMPLING
- RADIATION SURVEYS
- PERSONNEL DOSIMETRY
- RESPIRATORY PROTECTION

ENVIRONMENTAL ASSESSMENT

- INITIAL OFFSITE DOSE CALCULATION
- INITIAL CONFIRMATORY SURVEYS
- OFFSITE RADIATION MEASUREMENTS

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

OCT .. .. ..

APPENDIX XV
PROPOSED FINE OF METROPOLITAN EDISON CO.
FOR TMI-2 ACCIDENT

Docket Nos. 50-289
and 50-320

Metropolitan Edison Company
ATTN: Mr. R. C. Arnold
Sr. Vice President
260 Cherry Hill Road
Parsippany, New Jersey 07054

Gentlemen:

SUBJECT: INVESTIGATION REPORT NUMBER 50-320/79-10

This refers to the investigation conducted by the Special Investigation Team from the NRC's Office of Inspection and Enforcement of activities authorized by NRC License Number DPR-73 and specifically of your activities preceding, during and immediately following the nuclear accident that occurred at the Three Mile Island Nuclear Power Station, Unit Number 2, on March 28, 1979. Because of the similarity of Units 1 and 2 and commonality of management of the two units, corrective actions taken in response to this letter and its enclosures must be equally applicable to Units 1 & 2. Further, the NRC staff will consider the effectiveness of actions taken in response to this correspondence in developing its position on readiness for restart before the Atomic Safety and Licensing Board constituted to consider the restart of Unit 1. Copies of this correspondence and your response will be furnished to this Board.

Areas examined during this investigation are described in the Office of Inspection and Enforcement Investigation Report Number 50-320/79-10, published also as NUREG-0600. Numerous potential items of noncompliance were identified during the investigation and are described in the report. As a result of additional NRC review and because of mitigating circumstances, not all of the potential items identified in the report were cited in Appendix A.

Based on the results of this investigation and additional consideration of the potential items of noncompliance identified in Investigation Report Number 50-320/79-10, it appears that certain of your activities were not conducted in full compliance with NRC requirements as set forth in the Notice of Violation, enclosed as Appendix A. The nature and number of the significant alleged items of noncompliance found during the investigation demonstrate serious weaknesses in your management controls.

We have identified six alleged violations, the most severe of the NRC noncompliance categories, four of which contributed to the severity of the accident on March 28, 1979. We believe the course of the accident would have been altered, if not prevented entirely, had compliance with NRC requirements been achieved.

These noncompliances demonstrate serious weaknesses in your ability to maintain an effective health physics program, control maintenance activities, develop and review procedures, adhere to approved procedures and conduct your audit activities.

Failure to follow procedural requirements for operation with the electromatic relief valve and safety valve discharge line temperature within your procedural requirements had a significant impact on the course of the accident on March 28, 1979. Following this procedure would have resulted in closure of the block valve which would have isolated the relief valve and preven[illegible] the accident. Furthermore, this elevated temperature condition had been in existence for several months and apparently conditioned your operating staff such that the abnormality on March 28 was obscured or rationalized away resulting in delayed closure of the isolation valve until after fuel damage had occurred. This failure is considered to be one of the more significant issues. Other examples of failure to follow procedures, cited in Appendix A, that occurred prior to and during the accident reveal weaknesses in controls which are mandatory for safe nuclear power plant operation.

Crucial to nuclear safety is the determination by your review of procedures and approval authority that operations identified in the operating procedures are in accordance with the facility technical specifications. Your Plant Operations Review Committee reviewed and your plant manager authorized a surveillance procedure which placed valves in a condition that resulted in emergency feedwater header isolation. Further, on three occasions identified in this investigation, the header was isolated. The training of the operating staff should have made this condition apparent to them. This condition, leading to temporary defeat of emergency feedwater, should have been immediately identified on the first occasion of isolation and a revision of procedures should have been initiated. The plant staff performing this operation should have been imbued with the philosophy of not proceeding with operations that defeat safety systems, but of stopping operations, revising procedures, and proceeding with reviews to properly authorize the correct procedural actions.

We also identified inadequacies in your training of personnel who were designated to fill emergency job categories as defined in your Emergency Plan. Further, your retraining program for radiation protection and chemistry personnel failed to include the required topics. Training and retraining are essential for the continued proficiency of the staff and nuclear safety.

During the course of the accident there was a significant departure from normal health physics procedures and practices. It is recognized that in the interest of overall safety during an accident of this magnitude there may be circumstances justifying departure from stringent health physics practices. Nevertheless, we believe that insufficient measures were taken to control health physics actions and decisions during the course of the accident.

Records were missing for maintenance for emergency feedwater isolation valves in January 1979. The control of equipment for purposes of maintenance is essential for continued safe operation of a nuclear power plant. Records showing the status of such equipment are an essential ingredient for safety. Without this status documentation, the continuity of the work is lost, and more important, the operators and maintenance crew are unable to tell that nuclear safety has been established, the equipment maintenance may be performed, and the equipment has been tested and properly returned to service. These principles of equipment control also apply to surveillance testing. We also found, although the reasons are not fully understood, that the isolation valves in this system were closed at the time of the accident on March 28, 1979. Again, a failure of management control for equipment and surveillance testing is evident.

You have committed to a QA/QC inspection program which includes observation of operations and functional testing. Our investigation could find no information to indicate that a QA/QC inspection program ever existed at your facility for the observation of operations and functional testing.

These matters and other noncompliances taken together leave little doubt that your management controls for the operation of the Three Mile Island facilities are inadequate. Each of these inadequacies must be resolved.

In light of the seriousness of these alleged noncompliances and in view of the significance and nature of our inspection findings, we propose to impose civil penalties. The total civil penalties for all items cited in Attachment A are $725,000. The Atomic Energy Act limits the total civil penalty within any thirty day period to $25,000. Limiting the penalties for those items cited from October 1978 until March 1979 to $25,000 for each thirty day period, results in subtraction of $570,000. Therefore, a total penalty of $155,000 is proposed. Appendix B of this letter is the Notice of Proposed Imposition of Civil Penalties.

In determining the amount of the penalties assigned the staff took into account the severity and duration of the noncompliance, including the relationship of the items of noncompliance to the accident itself and the relationship of the noncompliance to other items of noncompliance.

The influence of NRC on your actions during the accident and preceding it has also been evaluated by this office both in determining noncompliance and in the penalty assessed. The Presidential Commission, the special NRC investigation and other investigative bodies are examining further the role of the NRC as well as the activities of other organizations in connection with the accident at Three Mile Island. The finding and recommendation of these other investigations will be evaluated in determining whether any further action is appropriate.

You are required to respond to this letter; in preparing your response you should follow the instructions in Appendices A and B. In addition, your response should address the steps taken to assure that your activities are in compliance with all Commission requirements since the noncompliances described in Appendix A, which are limited to the scope of our investigation, indicate t[illegible]ures of your overall management control. In this regard we expect that y[illegible] will conduct a comprehensive audit of all administrative and management controls to establish needed actions to assure full compliance.

Your written reply to this letter and Notice of Violation and the findings of our continuing inspections of your activities and further consideration of these matters may lead to further enforcement action, such as additional civil penalties or orders to suspend, modify or revoke the license. Among other things, additional enforcement action is under review with regard to the reportability of several items of information following the onset of the accident, including specifically the calculated dose rate of 10-40 R/hr in Goldsboro, the elevated in-core thermocouple indications and the pressure spike in the containment vessel. Further, we have already suspended the license to operate Unit 2. The public will be informed of any proposal to operate Unit 2, and any proposal to operate Unit 2 would be subject to a hearing.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosures will be placed in the NRC's Public Document Room.

Sincerely,

Victor Stello, Jr.
Director
Office of Inspection
and Enforcement

Enclosures:
1. Appendix A
2. Appendix B

Distribution

	MAIL STOP
PDR	
NSIC	
LPDR	
TIC	
State of Pennsylvania	H-1016
ACRS (3)	H-1149
SECY	
D. Thompson, IE	H-1057
CA (3)	MNBB-3709
F. Ingram, PA	MNBB-9604
J. P. Murray, ELD	MNBB-9604
J. Lieberman, ELD	MNBB-9604
M. Grossman, ELD	MNBB-9604
B. Brodenick, ELD	
DOR	MNBB-12105
J. Crooks, OMPA	Landow-1200
J. J. Cummings, OIA	
Enforcement Coordinators: Regions I, II, III, IV, V	
E. M. Howard, DSI:IE	
T. W. Brockett, IE	
IE Files	
Central Files	
Civil Penalty Book	L-316
CON	
XOOS Reading File	
EDO Reading File	
IE Reading File	

APPENDIX A
NOTICE OF VIOLATION

Metropolitan Edison Company   Docket No. 50-320

This refers to the investigation conducted by an Office of Inspection and Enforcement Investigation Team at the Three Mile Island Nuclear Generating Station, Middletown, Pennsylvania, of activities authorized by NRC License No. DPR-73.

During the investigation conducted on March 28, 1979 through July 31, 1979 (Investigation No. 50-320/79-10), the following apparent items of noncompliance were identified:

1. Technical Specification 3/4.7.1, "Turbine Cycle," requires in Section 3.7.1.2, that three independent steam generator emergency feedwater pumps and associated flow paths shall be operable during power operations, except: if one emergency feedwater system is inoperable it must be restored to operable status within 72 hours or the plant must be in Hot Shutdown within the next 12 hours.

   Contrary to the above, for an undetermined period just prior to the reactor trip at approximately 0400 hours on March 28, 1979, the flow paths to both steam generators were made inoperable by feedwater header isolation valve closure. (In addition, on January 3, February 26 and March 26, 1979, the flow paths from all three emergency feedwater pumps were simultaneously made inoperable by feedwater header isolation valve closure during the performance of, and in accordance with, an improper surveillance test procedure.)

   This violation contributed to an accident. (Civil Penalty $5,000)

2. The severity and uniqueness of the accident which occurred at Three Mile Island resulted in a marked reduction in the normal good health physics practices which are mandated by the NRC Regulations. Under the circumstances of an accident of this magnitude the NRC recognizes that in the interest of reactor safety a departure from normal health physics practices and standards may sometimes be mandated by the exigencies that exist during such conditions. However, the NRC also believes that the licensee, with the resources available and taking into account the time frame available for conduct of safety-related functions, could have taken additional measures to better control the overall health physics actions and decisions which were made during the course of the accident. The following items of noncompliance exemplify unacceptable degradation from health physics practices pertaining to control of access to high radiation areas, conduct of radiation surveys, and personnel radiation exposure monitoring.

   10 CFR 20.201, "Surveys," requires in Section (b) that each licensee shall make or cause to be made such surveys as may be necessary to comply with the regulations in 10 CFR 20.

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10 CFR 20.202, "Personnel Monitoring," requires that the licensee supply appropriate personnel monitoring equipment and requires its use for each individual who enters a restricted area and is likely to receive a dose in excess of 25 percent of the applicable value specified in 10 CFR 20.101.

Technical Specification 6.12, "High Radiation Area," requires that each area in which the intensity of radiation is greater than 1000 mrem/hr be provided with locked doors to prevent unauthorized entry into the area and that any individual entering the area be equipped with a continuously indicating dose rate monitoring device.

10 CFR 20.103, "Exposure of individuals to concentrations of radioactive materials in air in restricted areas," requires in Section (a)(3) that the licensee make suitable measurements of the concentrations of radioactive materials in air for detecting and evaluating airborne radioactivity in restricted areas for the purposes of determining compliance with the regulation in 10 CFR 20.103(a)(1).

10 CFR 20.101, "Exposure of individuals to radiation in restricted areas," requires that no licensee possess, use or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter a dose in excess of three rem to the whole body, or 18 3/4 rem to the hands and forearms, or 7 1/2 rem to the skin of the whole body.

Contrary to the above:

A. From 1100 hours on March 28, 1979 until the afternoon of March 30, 1979, the doors to the auxiliary building were not locked and access was not otherwise controlled even though the building was known to be a high radiation area with radiation levels much greater than 1000 mrem/hr during this period;

B. From the evening of March 28, 1979 until the evening of March 29, 1979, at least two entries into the auxiliary building were made by individuals who were not equipped with a radiation monitoring device which continuously indicated the dose rate;

C. No measurements were made of the concentrations of airborne radioactive materials in the Unit 2 auxiliary building for periods during which individuals were exposed from 1100 hours on March 28, 1979 through midnight March 30, 1979, nor in the Unit 2 nuclear sample room and primary chemistry laboratory for periods during which individuals were exposed from 0400 hours on March 28 through 0800 hours on March 30, 1979;

D. On March 29, 1979, an Auxiliary Operator was permitted to enter areas of the auxiliary building where exposure rates of up to 100 R/hr existed. Radiation survey information and appropriate personnel monitoring were

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not provided to the operator for this entry. This contributed to the operator receiving a whole body dose of 3.170 rems. When this dose was added to the operator's previous dose for the quarter, the operator's quarterly whole body dose was 3.870 rems as measured by personnel dosimetry devices;

E. On March 29, 1979, a Nuclear Engineer entered an area of the auxiliary building where the radiation level was greater than that which could be measured by his portable survey instrument (2R/hr). Failure to perform a survey of the exposure rate in this area contributed to the individual receiving a whole body dose of 3.14 rems for this entry. When this dose was added to the engineer's previous dose for the quarter, the engineer's quarterly whole body dose was 4.175 rems as measured by personnel dosimetry devices;

F. On March 29, 1979, a Chemistry Foreman was permitted to repeatedly enter high radiation areas and handle samples of highly radioactive reactor coolant. This contributed to the Foreman receiving a whole body dose of 4.100 rems. When this dose was added to the Foreman's previous dose for the quarter, the Foreman's quarterly whole body dose was 4.115 rems as measured by personnel dosimetry devices;

G. On March 29, 1979, a Chemistry Foreman and a Radiation Protection Foreman were permitted to handle a highly radioactive reactor coolant sample without adequate personnel monitoring and without first performing a survey of hand and forearm exposure rates. Handling of this sample resulted in a calculated dose to the hands and forearms of the Chemistry Foreman of about 147 rems and a calculated dose to the hands and forearms of the Radiation Protection Foreman in the range of 44 to 54 rems; and

H. On March 28, 1979 and March 29, 1979, several individuals received skin contamination of the hand and other parts of the body sufficient to cause exposure rates in the range of 20-100 mR/hr when measured with a hand-held survey instrument and no evaluation of the dose to the skin of these individuals was made.

Each day constitutes a separate violation [March 28 (A, B, C, and H), March 29 (A, B, C, D, E, F, G, and H), and March 30 (A and C)]; a civil penalty of $5,000 is imposed for each. (Cumulative Civil Penalty $15,000)

3. Technical Specification 6.5.1, "Plant Operations Review Committee," requires: in Section 6.5.1.6.a, that the Plant Operations Review Committee (PORC) review all procedures (and changes thereto) required by Technical Specification 6.8 and any other procedure (or change) determined to affect nuclear safety.

Contrary to the above, inadequate reviews were performed on both Procedure Change Request No. 2-78-707, Revision 4 to Surveillance Procedure 2303-M27A/B, and Procedure Change Request No. 2-78-895, Revision 8 to Surveillance Procedure 2303-M14A/B/C/D/E; both were reviewed and approved by the PORC (November 9, 1978 and August 15, 1978 respectively). Each approved change included a valve lineup which resulted in emergency feedwater header isolation, contrary to Technical Specification 3/4.7.1 requirements.

Each of these inadequate reviews constitutes a separate violation which contributed to an accident; a civil penalty of $5,000 is imposed for each.
(Cumulative Civil Penalty $10,000)

4. Technical Specification 6.8, "Procedures," requires in Section 6.8.1 that procedures be established, implemented and maintained covering identified activities.

   A. Emergency Procedure 2202-1.5, "Pressurizer System Failure," Revision 3, requires in Section A.2.B.1 that electromatic relief isolation valve RC-R2 be closed if, among other things, the valve discharge line temperature exceeds the normal 130°F.

      Contrary to the above, the electromatic relief valve discharge line temperature had been in the range of 180°-200°F since October of 1978 and isolation valve RC-R2 was not closed as of 0400 hours on March 28, 1979. Additionally, on March 28, 1979, the discharge line temperature of 283°F was noted at 0521 hours, but the isolation valve RC-R2 was not closed until 0619 hours, allowing a significant loss of RC inventory.

      Each day the plant operated in noncompliance with this procedure constitutes a separate violation, a civil penalty of $5,000 is imposed for each. (Cumulative Civil Penalty $630,000)

   B.1 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure," Revision 11, requires in Sections B.2.2.3, B.3.6.2 and A.3.2.5: that high pressure injection is initiated on low RCS pressure (1600 psig), and that the operator verify high pressure injection is operating properly as evidenced by flow in all four legs (250 gpm); that flows be maintained at this rate by throttling as RCS pressure drops; and that high pressure injection not be terminated until RCS pressure can be maintained above the reset point (1640 psig) or until low pressure injection flow is established at 3000 gpm.

      Contrary to the above:

      1. At about 0405 on March 28, 1979, high pressure injection flow was throttled to minimum conditions even though RCS pressure was less than 1600 psi and falling; and without low pressure injection flow established.

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2. At various times throughout the day of March 28, 1979, the high pressure injection system was modified such that the required flow rates were not maintained during continuing low pressure conditions within the RCS following the period when the reactor coolant pumps were stopped and the high pressure injection system was the only mode available for the removal of core decay heat.

B.2 Emergency Procedure 2202-1.3, "Loss of Reactor Coolant/Reactor Coolant System Pressure," Revision 11, requires certain actions to be taken following the automatic initiation of high pressure injection, including in Section B.3.1, that all ESF equipment is verified to be in its ESF position (capable of performing its intended function).

Contrary to the above, during the period of approximately 0600 hours until 1300 hours on March 28, 1979, during continuing low pressure conditions within the RCS, the Core Flood System was removed from its ESF position (rendered inoperable) by closing both tank isolation valves. [This portion of the ESF was inactivated during a period when reduction of Reactor Coolant System pressure was not the immediate goal. This removed from service this safety feature during a period when it could have been called upon. In the course of the accident while attempting to depressurize to activate the decay heat removal system NRC recognized that it was necessary to isolate the core flood system and encouraged this action. This citation does not apply to isolation during this attempt].

This violation contributed to an accident.
(Civil Penalty $5,000)

C. Operating Procedure 2104-6.2, "Emergency Diesels and Auxiliaries," Revision 9, establishes the procedures for the control of the emergency diesel generators:

1. Section 4.10, "Diesel Generator - Automatic Start Upon Engineered Safety Features Actuation," states in the closing step, 4.10.6, that the unit can be shutdown after the Engineered Safeguards Feature actuation has been cleared.

2. Section 4.6, "Diesel Generator 1A(1B) Shutdown to Emergency Standby," states in the closing step, 4.6.6, to place the diesel generator on standby in accordance with Section 4.2; and

3. Section 4.2, when completed, establishes conditions for automatically starting the diesels upon actuation of an Engineered Safeguards Feature (ESF) including requirements

to place the "Emergency Standby/Maintenance Exercise" switch in the Emergency Standby position and resetting the fuel racks.

Contrary to the above, at about 0430 hours on March 28, 1979, both the 1A and 1B diesel generator fuel racks were manually tripped, thereby preventing an automatic start of the diesel generators upon ESF actuation and manual start from the control until 0949 hours.

This violation had the potential to contribute to an accident. (Civil Penalty $4,000)

D. Emergency Procedure 2202-2.2 "Loss of Feedwater," Revision 3, requires in Section 2.B.2.d that the operator adjust feed flow to control steam generator levels at 30 inches.

Contrary to the above, from approximately 0532 hours until 0543 hours, the level in A steam generator decreased to 10 inches (the minimum level indication) while the A steam generator level was being controlled manually.

This is an infraction. (Civil Penalty $3,000)

E. Three Mile Island Nuclear Station Administrative Procedure 1004, "Three Mile Island Emergency Plan 1004," Revision 2, dated February 15, 1978:

1. Requires in Section 2.1, that the "Station Superintendent/Senior Unit Superintendent, Unit Supt./Shift Supervisor/Unit Supt. - Technical Support in the Control Room will, after reviewing the emergency conditions, classify the emergency as one of the following:

"a. Personnel or Local Emergency,

"b. Site Emergency, and

"c. General Emergency

"He will make this classification according to the condition of Table 1 of this Plan, and initiate actions according to the Emergency Plan Implementing Procedures, and according to his own best judgment;" and

2. States in Table 1 of Section 2.1 that a Site Emergency exists when there is a reactor building high range gamma monitor alert alarm (Condition No. e).

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Contrary to the above:

1. Adequate written procedures were not established and implemented in that Section 2.1 of Procedure 1004 for implementing the Emergency Plan lacked sufficient specificity and failed to result in a Site Emergency being declared at approximately 0430 on March 28, 1979, even though primary system pressure had decreased to the point where safety injection was automatically initiated and a reactor building sump high level alarm existed; and

2. A site emergency was not declared at 0635 hours on March 28, 1979, at which time Condition "e" of Three Mile Island Emergency Plan 1004 had occurred.

This is an infraction. (Civil Penalty $4,000)

F. Three Mile Island Nuclear Station Health Physics Procedure 1670.9, "Emergency Training and Emergency Drills," Revision 4, dated January 16, 1978:

1. Identifies in Section 3.1, the on-site emergency job categories and requires that training programs for these categories will be conducted on an annual (calendar year) basis; and

2. Describes in Section 3.1.1 through 3.1.9, the training program for all on-site emergency job categories.

Contrary to the above, during calendar year 1978, not all individuals having emergency responsibilities were trained in that two Emergency Directors, one Accident Assessment individual, eight Radiological Monitoring Team Members, and 37 Repair Party Team Members had not received the specified training. In addition on March 28, 1979, during an emergency, at least four individuals who were assigned as required members of a Radiological Monitoring Team and seven individuals who were assigned as required members of a Repair Party Team performed emergency duties for which they were not trained.

This is an infraction. (Civil Penalty $4,000)

G. Station Administrative Procedure 1002, "Rules for the Protection of Employees Working on Electrical and Mechanical Apparatus," Revision 14, requires in Section 4.3, 4.4 and 4.5 that on restoration of equipment to service, removed tags will have all required information entered thereon and then be suitably stored, and that the shift foreman shall approve equipment operation by signing the original tagging application. Additionally, Station Corrective Maintenance Procedure 1407-1, Revision 0, specifies in Section 5.0, "Job Ticket (Work Request) Flow," the step-by-step process for initiating, processing, obtaining approvals and ultimate filing of the "Job Package" which will include, among other things, documentation of corrective action taken (resolution description and certification of satisfactory post maintenance testing) and Station Preventative Maintenance Procedure E-2, "Dielectric Check of Insulation, Motors and Cables," specifies how to make the measurements and contains data sheets for recording the values measured.

Contrary to the above, when inspected on June 20, 1979, the tagging application could not be found for maintenance performed in January, 1979, on Emergency Feedwater isolation valves (EF-V12A, 12B, 32A, 32B, 33A, and 33B). No suitable documentation to determine whether the maintenance work had been completed, tags removed, acceptance criteria met, or valves approved for operation could be found. The TMI-2 maintenance log lists this work request as being in an open status as of June 20, 1979.

This is a deficiency. (Civil Penalty $2,000)

5. Technical Specification 6.8, "Procedures," requires in Section 6.8.2 that changes to procedures which implement the Emergency Plan shall be reviewed by the Plant Operations Review Committee and approved by the Unit Superintendent prior to implementation.

   Contrary to the above, a change to Station Health Physics Procedure 1670.7, "Emergency Assembly, Accountability and Evaluation," was made without the required review and approval. An additional assembly area was designated and the method used to perform accountability was modified by a memorandum dated October 13, 1978, from the Radiation Protection Supervisor to all departments. As a result, on March 28, 1979, in response to an emergency, some licensee personnel followed the approved procedure while others followed the guidance in the October 13, 1978 memorandum, creating some confusion and delaying prompt attainment of full accountability.

   This is an infraction. (Civil Penalty $4,000)

6. Environmental Technical Specification 5.7 requires that detailed written procedures for instrument calibration be prepared and followed.

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Three Mile Island Nuclear Station Surveillance Procedure 1302-5.24, Revision 3, dated December 19, 1974, specifies the method of calibration and requires that it be performed annually.

Contrary to the above, as of March 29, 1979, eight environmental samplers had not been calibrated since 1974.

This is an infraction. (Civil Penalty $4,000)

7. Technical Specification 6.2, "Organization," states in Section 6.2.1 and 6.2.2 that the unit organization and the organization of the corporate technical support staff shall be as shown on Figure 6.2-1.

   Contrary to the above, on March 28, 1979, the organization of the unit and corporate technical support staff was different from that specified in Figure 6.2-1 in that:

   A. A position titled, "Superintendent of Administration and Technical Support" was added to the organization on September 18, 1978 and filled on March 1, 1979, such that the "Supervisor, Radiation Protection and Chemistry," reported to this new position rather than directly to the "Station Superintendent/Senior Unit Superintendent;" and

   B. There were two "Supervisor of Maintenance" positions, one for each unit, rather than one; and

   C. A position titled "Superintendent of Maintenance" had been added such that the "Supervisors of Maintenance" report to this new position rather than directly to the "Station Superintendent (Station Manager)/Senior Unit Superintendent;" and

   D. The position of "Chemical Supervisor" had been vacant since the issuance of the Technical Specifications.

   On March 28, 1979 through March 30, 1979, the above organizational discrepancies decreased the effectiveness of the licensee's response to the accident.

   This is an infraction. (Civil Penalty $3,000)

8. Technical Specification 6.4 "Training," requires that a retraining and replacement training program for the unit staff be maintained that meets or exceeds the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

Contrary to the above, as of March 28, 1979, a retraining program meeting or exceeding ANSI N18.1-1971 recommendations had not been maintained for members of the radiation protection and chemistry staff in that only 2 of the 10 topics recommended were included in the program.

This is an infraction. (Civil Penalty $4,000)

9. Technical Specification 3/4.4.6, "Reactor Coolant System Leakage," requires in Section 3.4.6.2, that Reactor Coolant System (RCS) leakage be limited to 1 gallon per minute (GPM) of "Unidentified Leakage," and that unless rates above this limit are reduced to within the limit within four hours, the plant must be placed in "Hot Standby" in the next six hours and in "Cold Shutdown" in the next thirty hours.

Contrary to the above, from March 22 until March 28, 1979, RCS "Unidentified Leakage" remained above 1 gpm, and the plant was not placed in "Cold Shutdown."

Each day constitutes a separate infraction; a civil penalty of $3,000 is imposed for each. (Cumulative Civil Penalty $21,000)

10. 10 CFR 20.401, "Records of surveys, radiation monitoring, and disposal," requires in Section (a) that each licensee maintain records showing the radiation exposure for all individuals for whom personnel monitoring is required on a Form NRC-5 or equivalent and in Section (b) requires that each licensee maintain records of the results of surveys required by 10 CFR 20.201(b).

Contrary to the above:

A. The results of approximately 500 ground level radiation surveys conducted during March 28-30, 1979 in offsite areas bordering the Three Mile Island site were not documented in a manner which permitted a precise evaluation of the type of radiation (Beta/Gamma) which existed in the environs. Pertinent information such as the type of instrumentation used and whether the end window on the probe was open or closed was not recorded.

B. The records of the radiation exposure for at least 5 individuals exposed during the period March 1 to 31, 1979 had not been recorded or maintained on a form NRC-5 or equivalent as of July 5, 1979. Furthermore, as of July 5, 1979 the assessment of their doses had not been completed.

This is an infraction. (Civil Penalty $4,000)

11. 10 CFR 50, Appendix B, Criterion X, "Inspection," requires that a program for inspection of activities affecting quality shall be established and executed to verify conformance with documented instructions, procedures and drawings for accomplishing the activity.

Three Mile Island Nuclear Station - Unit 2, Final Safety Analysis Report, Chapter 17.2.15, Section X, requires that the inspection program include random observation of operations and functional testing by individuals independent of the activity being performed.

Procedure GP 4014, "QQA Surveillance Program," Revision 0, requires independent observation of activities affecting quality to verify conformance with established requirements utilizing both inspection and auditing techniques...for compliance with written procedures and the Technical Specifications.

Contrary to the above, as of March 28, 1979, the normal operations surveillance testing activities had not been made subject to random and/or routine inspections by independent methods.

This is an infraction. (Civil Penalty $3,000)

This Notice of Violation is sent to Metropolitan Edison Company pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Metropolitan Edison Company is hereby required to submit to this office within twenty (20) days of the receipt of this Notice, a written statement or explanation in reply, including for each item of noncompliance: (1) admission or denial of the alleged items of noncompliance; (2) the reasons for the items of noncompliance if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further items of noncompliance; and, (5) the date when full compliance will be achieved.

The total civil penalties for all items cited is $725,000. However, pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (42 USC 2282), the total of civil penalties for any thirty day period cannot exceed $25,000. Consequently $570,000 has been subtracted to reduce the total penalties to $25,000 for each 30 day period resulting in the total civil penalty herein proposed of $155,000.

APPENDIX B
NOTICE OF PROPOSED IMPOSITION OF CIVIL PENALTIES

Metropolitan Edison Company

Docket No. 50-320
License No. DPR-73

This office has considered the enforcement options available to the NRC including administrative actions in the form of written Notices of Violation, Civil Monetary Penalties, and Orders pertaining to the modification, suspension or revocation of a license. Based on these considerations we propose to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (42 USC 2282), and to 10 CFR 2.205 in the cumulative amount of One Hundred and Fifty-Five Thousand Dollars ($155,000) for the specific items of noncompliance set forth in Appendix A to the cover letter. In proposing to impose civil penalties pursuant to this section of the Act and in fixing the proposed amount of the penalties, the factors identified in the Statements of Consideration published in the Federal Register with the rulemaking action which adopted 10 CFR 2.205 (36 FR 16394) August 26, 1971, and the "Criteria for Determining Enforcement Action," which was sent to NRC licensees on December 31, 1974, have been taken into account.

Metropolitan Edison Company may, within twenty (20) days of receipt of this Notice pay the civil penalties in the cumulative amount or may protest the imposition o the civil penalties in whole or in part by a written answer. Should Metropolitan Edison Company fail to answer within the time specified, this office will issue an Order imposing the civil penalties in the amount proposed above. Should Metropolitan Edison Company elect to file an answer protesting the civil penalties, such answer may (a) deny the items of noncompliance listed in the Notice of Violation in whole or in part, (b) demonstrate extenuating circumstances, (c) show error in the Notice of Violation, or (d) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties in whole or in part, such answer may request remission or mitigation of the penalties. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate by specific reference (e.g., giving page and paragraph numbers) to avoid repetition.

Metropolitan Edison Company's attention is directed to the other provisions of 10 CFR 2.205 regarding, in particular, failure to answer and ensuing orders; answer, consideration by this office, and ensuing orders; requests for hearings, hearings and ensuing orders; compromise; and collection.

Upon failure to pay any civil penalties due which have been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, the matter may be referred to the Attorney General, and the penalties, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Atomic Energy Act of 1954, as amended (42 USC 2282).

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

November 8, 1979

ACRS Members
ACRS Technical Staff

COMMENTS ON REGULATORY GUIDE 1.97, REVISION 2, "INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT"

Enclosed are some comments received by the Regulatory Activities Subcommittee on the subject Regulatory Guide. All these comments have been transmitted to the NRC Staff for consideration. At the November 7, 1979 meeting, the Regulatory Activities Subcommittee instructed that the NRC Staff should resolve these comments along with other public comments which may be received during the public comment period of this Guide. Resolution of these comments and other public comments will be reviewed by the Regulatory Activities Subcommittee during the final review of this Guide, subsequent to the public comment period and prior to recommending it to the ACRS full Committee for concurrence with its Regulatory Positions

Sam Duraiswamy

Sam Duraiswamy
Reactor Engineer

Enclosures:
1. Comments from Dr. Okrent.
2. Comments from Mr. Bender.
3. Comments from Dr. Catton.
4. Comments from Dr. Zudans.
5. Comments from the General Electric Company.
6. Comments from the Babcock & Wilcox Company.

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

COPY

RECEIVED

October 22, 1979

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U.S. NUCLEAR REG. COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

TO: Sam Duraiswamy

FROM: D. Okrent

SUBJECT: REG GUIDE 1.97, REVISION 2

1. Does the proposed guide respond directly to the ACRS recommendation for continuous readout of hydrogen concentration? Is the proposed range on H_2 up to 10%? Is this limit set by the available instrumentation. If not, why not a higher value than 10%?

2. Should the guide indicate a forthcoming need or interest in a better means of predicting what isotopes were released at what rate, when, if a serious accident leads to a loss of containment integrity? What capability is expected for the 100 channel gamma ray spectrometer for the containment? Is it possible to know how much Cesium-137 is in the atmosphere? How much iodine? Will the gamma ray spectrum measurement be able to do this? Will it need a sampling procedure? If so, will it be automated, tested, reliable, etc. Could one get continuous or very frequent measurements, so that with a relatively simple computing program, one would use containment pressure and the detailed radiation component measurements to say what leaked and when?

3. Will radiation level in coolant be measured using the regular let-down cooling line? Does this leave containment? May it be isolated when you want to know about this parameter?

4. Is there adequate information provided on the specifics of any radiation getting into the control room or other vital areas for operator action (e.g., Diesel building)?

5. In high range radiation in containment, how many positions are needed?

ENCLOSURE 1

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

November 5, 1979

FROM: M. Bender

SUBJECT: COMMENTS ON REGULATORY GUIDE 1.97

The Regulary Guide still does not have the proper approach to accident information. The requirements are too pervasive and the purpose of the instrumentation appears too similar to that of normal plant instrumentation. The TMI-2 experience showed that a few carefully selected instruments were all that was required to respond to the emergency.

The guide should differentiate between instrumentation needed promptly at the time of an accident and that which can be provided later if available on a standby basis. For example, sampling taps that will permit sampling of primary coolant, containment, filter effluents should be permanently installed but the measuring instruments need not be permanently installed if the accident analysis shows that the types of accidents to be monitored would allow ample time for such instruments to be installed once the accident event is identified. Instruments intended to show early progression of an accident need special attention. Fuel temperatures and outlet coolant temperatures are exemplary of this capability. Core coolant voiding is another. The guide should require a list of these accident parameters first and then establish how they are to be measured.

It is unwise to specify accident instrumentation that has performance capability unique to a specific accident unless that accident has a high probability of imposing the performance demands. For example, primary coolant pressure ought to be measurable to some level above the pressure relief settings but it is unnecessary to measure the pressure level up to bursting pressure. If bursting pressure conditions develop, they will occur so fast that the operators would not be able to respond to the event. Slower events would be relieved by the pressure relief valves. If we were interested in a pressure near bursting, it would be only because we might want to reuse the vessel. A means of determining structural strain would be valuable in such a circumstance but that is not an instrument to follow the course of an accident. The pressure measuring capability for containment is an entirely different problem. Loss of in-containment cooling could cause overheating and consequent containment overpressure over a long period of time justifying a monitoring capability, but it too might be installed sometime after the accident had been initiated.

It is important that this guide not become a set of requirements covering the monitoring of every minor accident. The instruments of interest are those to help the operator in emergencies when he needs real help in diagnosing an accident on a timely basis.

ENCLOSURE 2

The following is a logical approach:

1. Establish the classes of accidents that need to be monitored with special instruments e.g., gross fuel damage, radio-acitivity release to containment, primary system rupture, ATWS type reactivity excursions.

2. Examine the plant to determine what is normally provided as instrumentation of use in accident circumstances.

3. Determine whether any important accident parameters cannot be measured with existing instruments and determine if such instruments should be permanently installed.

4. Determine what contingent provisions are needed to allow for instument and sampling capability in the event of an accident.

5. Determine whether provision to add instruments in the event of an accident is appropriate and if so how?

6. Use some type of probabilitic approach to determine how to qualify instrumentation for this purpose (e.g. how often would steam environments be associated with serious accidents?)

The use of normal instrumentation to follow the course of accidents is a desirable capability but we should not automatically make a useful instrument a part of this accident monitoring capability because it exists. Too much monitoring capability is confusing to the operator. We want to specify the minimum need and we want the instrument signals to be easily interpretable with respect to accident mitigation during emergencies.

A conscious decision needs to be made as to where to draw the line between instrumentation that could routinely monitor an accident and that which is specifically intended to follow the accident over a specified period of time as a basis for operational guidance and emergency response. It is important that emergency monitoring information be related to symptoms of the accident. Routine monitoring might be important subsequent to the accident to show the condition of equipment needed during accident recovery. Core outlet temperatures need to be measured because they tell whether the coolant is superheated and thus the fuel is not being adequately cooled. Coolant overtemperature concurrent with high coolant pressure may indicate that there is no secondary coolant or that primary coolant flow through the core is blocked. Differentiating between these two possibilities should be possible with other diagnostic instrumentation, for example, core pressure drop.

In establishing the qualifications for instrumentation, care should be take to avoid imposing requirements that result in a highly sophisticated measuring

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device to satisfy circumstances that have low probability of occurring simultaneously with the type of event to be monitored. For example, a fission counter that is called upon to work because of a reactivity excursion from a 0.6g seismic event is an unlikely need, but one that will respond to a demand at 0.2g might be a more likely requirement and might be easily met with existing technology.

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

November 2, 1979

TO: Sam Duraiswamy

FROM: Ivan Catton

SUBJECT: REGULATORY GUIDE 1.97, REVISION 2

The guide as written is not responsive to the Lessons Learned from TMI. Some specific areas I would like to have discussed are as follows:

1. Show how a timely heat balance can be obtained?

2. What are measurement tolerances and how are they arrived at?

3. Display of the measurements with proper processing is the single most important factor in timely following the course of an accident. I would like to hear a discussion of factors relating to how an operator is going to receive the results of various measurements listed. It should include the process computer and any requirements that should be imposed on it, as well as location of instrument displays. Questions such as, should strip charts be used?; where and for what measurements, should also be considered.

4. Ranges on many temperature measurements do not cover ranges experienced at TMI. How were those listed choosen?

5. The containment instrumentation section does not indicate locations of pressure, temperature and hydrogen sensors. Some discussion of the impact of location would help.

6. Why aren't feedwater system temperatures measured?

7. Existing plant instrumentation supplies a large amount of the needed information. How much of the listed data in 1.97 is additional?

8. What are the requirements for measuring reactor coolant inventory?

ENCLOSURE - 3

U.S. DEPARTMENT of ENERGY
ARGONNE NATIONAL LABORATORY
9700 SOUTH CASS AVENUE, ARGONNE, ILLINOIS 60439 TELEPHONE 312/972-4639

October 29, 1979

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

Attention: Mr. S. Duraiswamy

Subject: Review of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"

Dear Mr. Duraiswamy:

The guide has effectively listed those parameters which should be measured to assess plant and environs conditions during and following an accident, but like standard ANS-4.5 is weak in precisely defining what is to be done with this measured information. The time-history responses of the measurements are essential to accident assessment and operator response. For changing conditions, the operator will need current values as well as past values to assess rates of change of parameters. Table 1, p. 1.97-12, under "Criteria," includes "10. Display method" with footnotes: (15) Where trend or transient information is essential to planned operator actions, (16) Recording, and (17) Dial or digital indication.

One of the lessons learned from TMI was that the information being displayed to the operators was inadequate (i.e., data logger). This regulatory guide does not address this issue. There are two solutions: (1) Improve this guide to address the issue of data presentation to the operator, or (2) Prepare a separate regulatory guide which defines the display requirements for the measurements of R.G. 1.97.

In addressing accident transient response and the recording of the measured variables, the rate of change is important not only with respect to the sensor and signal conditioning equipment performance, but also with respect to the recording equipment performance. If faster recording speeds are required, is the measured value to be displayed and recorded on a continuous basis? This presents a problem on available length of recording paper. Should the recorder have two speeds -- a slow speed for normal conditions and a fast speed for accident conditions which is triggered by a measured value corresponding to accident conditions? If a triggered signal is used to initiate a fast recorder speed, what is the minimum chart paper length that should be available within the recorder to ensure that the chart does not run out at the beginning of a transient. Can more than one variable be recorded on a single recorder? The correlation of events to a single time frame is important. If individual recorders are used, time synchronization must be provided by event marks, etc. Is digitizing data

ENCLOSURE - 4

and recording acceptable? Should a special time sequence of events recorder be provided to simultaneously record all important measurements on a useable time scale and total time period? How should all of the measured information be grouped and presented to the operator. Should these measurements be displayed in a second location other than the control room (backup control area)? Should any of these variables be transmitted to an NRC accident response center?

Unless the above questions are addressed and covered by a regulatory guide, the provision of measurements in a plant will not in itself aid an operator in responding to an accident.

The guide as prepared has not considered the recommendations of TMI-Lessons Learned, i.e., vent valves in primary system high points (need valve positions) and PWR pressure vessel level.

In Tables 2 and 3, the Measured Variable column includes measurements with an "or", i.e., Primary System Safety Relief Valve Positions or Flow Through or Pressure in Relief Valve Lines. "Flow" and "Pressure" will not be good measures of low flows. It is recommended that all measurements with "ors" be re-evaluated. In other cases the "or" should not be "or" but should be two separate measurements, i.e., Secondary Safety/Relief Valve Positions or Main Steam Flow should be two separate measurements.

Table 1 should include quench tank measurements of pressure, temperature, and level to indicate opening of safety/relief valves.

In allowing "ors" the guide does not consider the benefit of separate diverse measurements. If the primary measurement is in error, the diverse measurement will give the operator hopefully correct information.

The guide does not address the requirements for measuring the effluents from containment for radioactivity, i.e., which containment penetrations should be monitored for radioactivity.

Standard ANS-4.5 lists operator sampling of reactor coolant to verify alarm limits on fuel failure. The regulatory guide is silent on this measurement. Continuous monitoring is preferred, but if the guide accepts the ANS position, what is the minimum acceptable time required to determine that fuel failure has occurred?

Table 2. Core exit temperatures should be included to assess degraded core conditions.

Sincerely,

Walter C. Lipinski

Walter C. Lipinski
Senior Electrical Engineer

WCL/at

C7-1183

Franklin Research Center RECEIVED
A Division of The Franklin Institute

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1 Copy

U.S. NUCLEAR REG. COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

October 30, 1979

Mr. Sam Duraiswamy
Reactor Engineer
U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, D.C. 20555

Re: Comments to Regulatory Guide~~s 1.141~~, 1.97 ~~and Snubber Qualific...~~

Dear Sam:

The following comments and suggestions are offered for discussion at 7 November 1979 ACRS meeting.

~~I. Regulatory Guide 1.141, Revision 1, "Containment Isolation Provisions for Fluid Systems"~~

~~B. Discussion, C. Regulatory Position. Revision as shown acceptable. I also believe that it would be a good practice to annunciate to the control room whenever a system penetrating containment is actually penetrating the containment. For example, if containment sump pumps are actually transferring water to auxiliary building, operators will have a signal on that. Motor operated value position indicators will serve that purpose, however, sealed closed isolation valves under administrative control do not require position indicators (see ANSI N271-1976, 4.2.3), hence status of this isolation will not be indicated.~~

~~D. Implementation.~~

~~Where are terms Type 1 (FDA-1), Type 2 (FDA-2), etc. defined?~~

II. Proposed Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"

B. Discussion.

Third paragraph of Page 1.97-3 states that some instrumentation components need no special qualification under Regulatory Guide 1.97 (if the environment for accident and normal operating conditions remain the same). I feel that at this location reference should be made to documents controlling qualification of instruments under this set of conditions.

ENCLOSURE - 5

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The last paragraph of Page 1.97-3 should explicitly indicate that design should consider effects of non-Seismic Category I building on instrumentation components.

Discussion of Design Criteria, Page 1.97-12 is recommended. In particular, comments are requested on how these criteria take care of non-Seismic Category I building effects on instruments.

In Table 2, Page 1.97-14 all primary coolant temperature gages should be provided with secondary scale indicating saturation pressure, and all primary coolant pressure gages should have secondary scale indicating saturation temperature. However, the same purpose may be obtainable with the instruments measuring subcooling. Comments from staff in this matter should be requested.

III. ~~Proposed Regulatory Guide,~~ "Qualification and Production Tests for Safety-Related Snubbers"

The prescribed tests and qualification are very extensive. As stated in the text, hydraulic snubbers will not function without hydraulic fluid. How can tests be designed to predict leak rate during operation in plant? To what degree the fact that leak rates during the test do not exceed the specification, defines the expected behavior during the operation? I believe the subject of leakage should be discussed in greater detail at the meeting.

Is the relaxation of requirements for units larger than of 150,000 pound rated capacity justifiable for reasons indicated? Is it not true that these large units are in particular important to the plant safety? Is there some special operational surveillance planned for these units to assure their availability when needed?

Very truly yours,

Zenons Zudans
Senior Vice President,
Engineering

ces

A-254

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

November 6, 1979

APPENDIX XVIII
REGULATORY GUIDE 1.97 (REV. 2): VENDORS' COMMENTS

Dr. Chester Seiss
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Draft Regulatory Guide 1.97, Rev. 2.

Dear Dr. Seiss:

B&W has reviewed draft Regulatory Guide 1.97, Revision 2, and offers the attached comments for your Subcommittee consideration. Due to the short time available for review, these comments may not be complete and they are meant to be only the major comments B&W has at this time.

Your consideration would be appreciated.

Very truly yours,

Edward Kane
for James H. Taylor
Manager, Licensing

JHT:dsf

Attach.

cc: R. B. Borsum (B&W)

ENCLOSURE 7

B&W Comments on the Proposed Draft 2
of Reg. Guide 1.97

Reference: Proposed Revision 2 to Regulatory Guide 1.97 Draft 1, October 15, 1979

1. Comment: We believe that a systematic approach should be taken to identification of accident monitoring variables and associated instrumentation design criteria such as that presented in Sections 5 and 6 of the draft ANS 4.5 Standard. Tables 2 and 3 of the reference are not AMI function oriented and as a result it is difficult to establish the necessity and sufficiency of the variables in these tables.

   Recommendation: Reverse the format of Tables 2 and 3 so that "purpose" appears first; then identify (or better yet, require the licensee to identify) variables necessary and sufficient to satisfy the purpose and meet applicable design criteria.

2. Comment: In order to assist the control room operator to clearly understand plant status during an accident, a minimum set of plant variables and instrumentation should be identified as "accident monitoring instrumentation". The inclusion of Type D and Type E variables by Regulatory Position C (4) greatly expands the list of variables to be addressed as "accident monitoring instrumentation". The expanded list dilutes the benefit of having a minimum set of clearly identified and qualified instrumentation to monitor plant status during accident conditions. Type D instrumentation, instrumentation to monitor safety system operation (versus monitoring safety function accomplishment), is best (and more completely) addressed in the context of safety system design criteria. The definition of Type E instrumentation is loosely constructed and open ended.

   Recommendation: Delete the variable types "D" and "E" from the scope of Reg. Guide 1.97.

3. Comment: Other areas of disagreement between the proposed Rev 2 to RG 1.97 and the draft standard ANS 4.5 should be carefully reviewed. These include:

   a) Regulatory Position C (1) expands the scope to include information for off-site emergency planning, without supplying criteria sources for selecting variables.

   b) Reg. Position C (2) expands the Type C definition to also include potential breach of fuel clad and reactor coolant boundary. "Potential" breach is an open ended and difficult concept to implement.

   c) Reg. Position C (3) expands the events to be addressed to include transients as well as accidents.

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d) Reg. Positions C (6), C (7), C (13), and C (14) increase design and qualification requirements for Type C instrumentation. Event analysis is required, whereby Type C definition, no event is defined.

e) Reg. Position C (9) arbitrarily increases the duration of qualification from 100 days to 200 days without supplying adequate justification.

Recommendation: Revise these areas to agree with ANS Standard 4.5 unless a sufficient engineering basis is established for more stringent requirements.

COPY

NUCLEAR ENERGY

PROJECTS DIVISION

GENERAL ELECTRIC

General Electric Company, 178 Curtner Ave., San Jose
California 95125

MFN-270-79

November 5, 79

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: G. R. Quittschreiber

Gentlemen:

Subject: Draft Revision 2 Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"

General Electric Nuclear Energy Group has reviewed Draft 1, dated October 15, 1979, of Revision 2 to Regulatory Guide 1.97 and offers the following comments.

1. The draft of Regulatory Guide 1.97 (hereafter referred to as 1.97) recognizes draft ANS Standard 4.5, "Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station" (hereafter referred to as 4.5). However, 1.97 imposes requirements far beyond those set forth in 4.5. The 4.5 requirements are necessary and sufficient themselves and the additional requirements in 1.97 are unjustified. Further, some of the additional requirements in 1.97 may result in less safety if they are implemented. The following comments expand upon this concern.

2. The criteria and safety functional requirements for all instrument types should be clearly specified in 1.97. For example, 1.97 states, "With regard to the discussion of Type D Variables, Type D Variables and Instruments are within the scope of Accident Monitoring Instrumentation, although they are not addressed in Draft Standard ANS 4.5. They are, however, along with an additional type, Type E, included in this Regulatory Guide". The foregoing statement does not include criteria or safety functional requirements for Type D and E instruments.

   Without criteria, the Type D and E instruments cannot be properly assessed for compliance. Lack of criteria will lead to inconsistent application, particularly because the 1.97 list is not uniformly applicable to all plants.

3. The instrument quality specified in 1.97 should be related to the importance of the safety function to be monitored. There is no discussion of the relative importance of safety functions in 1.97. A much more graded set of quality requirements should be specified so that the quality of those instruments is commensurate with their importance to safety.

ENCLOSURE. 6

4. The requirement for redundancy or diversity of monitored variables should be specified at the safety function level only. Specification of redundancy or diversity at the variable or instrument level is unnecessary and unjustifiable. Redundancy or diversity will be provided by the designer at the variable or instrument level as necessary to satisfy the requirement imposed at the safety function level.

5. The guide 1.97 should focus on providing information for control room operator action and avoid emergency planning instrumentation. The latter belong more appropriately in the emergency planning Regulatory Guide 1.101. The provisions in 1.97 may duplicate or conflict with those in the emergency planning Regulatory Guide or approved state emergency plans.

6. Radiation and radioactivity concentration instrument ranges should be set taking credit for plant unique features such as radionuclide holdup in suppression pools, and secondary containment.

7. Seismic Class I qualification should be specified only for Type A Instruments because requiring Seismic Class I for Types B through E instruments is inconsistent with seismic qualification requirements in Regulatory Guide 1.29.

8. Rod position indication should not be included in 1.97 as it is not needed for either redundant or diverse verification of subcriticality. Two channels of source range monitors (SRM) (neutron monitors) satisfy the redundancy requirement. The SRMs are the primary and secondary means of demonstrating subcriticality.

   Instantaneous display of rod insertion provides a third indication of initial subcriticality which is sufficient. The operator can, within a few minutes after scram, note which rods (if any) are not fully in and drive them in individually. (Verifying scram has always been a priority operator action taught in operator training.) In addition, qualifying rod position indication electronics for some severe environments at times after a few minutes into an event is not technically feasible. Also, relocation of rod position instrumentation outside containment is not practical.

We thank you for the opportunity to comment on 1.97. We will make specific and detailed comments on the revision of 1.97 when it is issued for public comment.

Very truly yours,
R. B. Buchholz

RHB:ph
(retyped: da 11/7/79)

BWR IN-CORE TEMPERATURE MONITORING

Summary

The BWR operates under saturated conditions. Therefore the primary method to assess adequate core cooling is by means of reactor water level instrumentation, as has been done on all BWRs. In-core temperature monitoring would provide information of negligible value. It will not help already planned operator actions (and may confuse the operator). Emergency plans should be based on other more reliable parameters, such as water samples and containment and effluent radiation levels.

BWR Design & Operation

BWR fluid is essentially saturated. Transition from a subcooled to a saturated condition (which might be sensed in a PWR in an approach to inadequate core cooling) is not available as a useful measurement parameter in a BWR. Analyses and measurements (e.g. Reference 1 & 2) have confirmed natural circulation capability is an inherent BWR feature. As long as there is adequate water level there is no question of the adequacy of core cooling.

BWR Instrumentation

Thus reactor vessel water level is the BWR parameter that monitors adequacy of core cooling. The BWR provides multiple, redundant, single-failure proof safety-grade water level indication. The reactor control, reactor protection, and operator safety actions are all keyed to this safety-grade water level indication. In addition there is safety grade main steam line radiation monitoring which will isolate the reactor and cause a scram upon reading 3 times normal radiation levels.

BWR Response to Events Which Threaten Core Cooling

All events which threaten the ability to provide adequate core cooling have a common factor: water level decreases. This is true whether the event is the loss of makeup water as in Loss-of-Feedwater transients, a sustained imbalance between feedwater flow and steam flow as in Feedwater Control Failure transients, or an excessive loss of liquid or steam inventory as in postulated Loss-of-Coolant accidents. Automatic action is taken to scram the reactor at about 15 feet above active fuel. Should water level continue to decrease, high pressure injection systems (High Pressure Coolant Injection and Reactor Core Isolation Cooling) are initiated at a water level about 10 feet above active fuel. Finally, if necessary, a number of safety/relief valves open automatically (Automatic Depressurization System) at 18 inches above active fuel to cause a sufficient reactor pressure reduction and low pressure injection systems (Low Pressure Coolent Injection and Low Pressure Core Spray) are initiated to restore water level before any significant heatup.

Operators are trained to back up the above automatic actions, if necessary. Long term decay heat removal is also assured via redundant low pressure Residual Heat Removal systems.

Purposes of In-Core Temperature Monitors

In order to assess the need for in-core temperature monitoring in BWRs, it is necessary to first establish what purposes or functions could be served by having such an instrumentation. These functions are outlined as follows:

A. During the early stages of an event to sense the approach to inadequate core cooling and provide input for operator action to prevent damage.

B. During the course of an event, to indicate the approach to cladding failure and provide input for operator action to limit the damage.

C. After an event to assess the extent of core damage and provide input for operator action to assure safe shutdown.

D. During the entire event are to provide input for emergency planning.

Comparison to Purposes

For function A, in-core temperature monitoring would be of no value, because during this time the reactor water/steam will be at thermodynamic saturation. Water level is the key to automatic protection and operator action as discussed previously.

For function B, appropriate automatic and operator action is assured by water level indication. With the multiple diverse means the BWR has to inject water into the reactor in an emergency, inadequate core cooling is an incredible scenario. However, even if automatic ECCS initiation did not occur, the operator would be taking action to provide water injection manually because he knows that water level is low and decreasing and no ECC systems are running. The operator backup is particularly assured with the new emphasis in operator training on core cooling. In fact in-core temperature monitoring might confuse operator action, due to such causes as thermocouple partial failure.

For function C, as stated above, depressurization is always possible in a BWR. No input from in-core temperature monitors should change the course of operator actions to achieve adequate core cooling and safe shutdown. Assessment of a change in core damage is more accurately determined from water samples. As part of the Post Accident Monitoring regulation guide conformance, as well as conformance to paragraph 2.1.8a of NUREG0578 the ability to rapidly take and analyze a water sample after an accident is now a requirement. In-core temperature monitors are not a reliable indication of the extent of core damage and may lead to incorrect conclusions.

For function D, the only case in which off-site emergency actions would be considered is the complete loss of ECCS. However, public protection should be primarily based on fission product release to the reactor coolent or containment, as observed in water samples and containment and effluent radiation monitoring, not core temperature.

References

1. General Electric report NEDO-10174, "Consequences of a Flow Blockage Incident in a Boiling Water Reactor".

2. General Electric report NEDO-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K".

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BWR IN-CORE TEMPERATURE

POSSIBLE USES	BWR CAPABILITY	COMMENTS
• APPROACH TO INADEQUATE CORE COOLING	• REDUNDANT SAFETY-GRADE RPV WATER LEVEL	• SATURATED FLUID - NO INDICATION
• LIMIT CORE DAMAGE	• REDUNDANT SAFETY-GRADE RPV WATER LEVEL	• MAY PROVIDE A CONFUSING SIGNAL TO OPERATOR
• ASSURE SAFE SHUTDOWN	• DEPRESSURIZATION ALWAYS POSSIBLE • STRONG INHERENT NATURAL CIRCULATION	• MAY PROVIDE A CONFUSING SIGNAL TO OPERATOR
• ASSESS CORE DAMAGE	• CONTAINMENT AND EFFLUENT RADIATION MONITORS • WATER SAMPLES	• NOT A GOOD INDICATION OF CORE DAMAGE
• INPUT TO EMERGENCY PLANS	• CONTAINMENT AND EFFLUENT RADIATION MONITORS • WATER SAMPLES	• SHOULD NOT BASE EMERGENCY PLANS ON THIS PARAMETER

IN-CORE TEMPERATURE MONITORS OF NEGLIGIBLE VALUE

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ANS-4.5

Draft 4
November, 1979

APPENDIX XIX
INSTRUMENTATION TO FOLLOW AN ACCIDENT
ANS STANDARD 4.5 (DRAFT 4)

DRAFT

CAUTION NOTICE: This Standard is being prepared or reviewed and has not been approved by ANS. It is subject to revision or withdrawal before issue.

DRAFT

American National Standard

ACCIDENT MONITORING CAPABILITY FOR THE CONTROL ROOM OPERATOR IN A NUCLEAR POWER GENERATING STATION

Assigned Correspondent

T. F. Timmons
Westinghouse Electric Corporation
Power Systems Company
P.O. Box 355, MNC-410
Pittsburgh, PA 15230

Writing Group ANS 4.5
Standards Committee NUPPSCO
Secretariat ANS

TABLE OF CONTENTS

FOREWORD

ANS 4 established Working Group 4.5 in late July 1979 to prepare a draft standard on Accident Monitoring Instrumentation which would complement other standards. Two primary objectives were 1) to address that instrumentation which permits the operator to monitor expected parameter changes in the accident period, and 2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

This draft standard provides:

1. a list of functions to be performed (design basis section 5.0)

2. a framework to determine those variables to be monitored (design basis section 5.0)

3. an identification of three time periods of interest (definitions 3.0)

4. an identification of four variable types (definitions 3.0)

5. a delineation of applicable design criteria for the variables to be monitored (design criteria section 6.0)

The significant issues in the development of this standard have been:

1. the scope of the document in terms of applicability to the control room operator or the plant operator (licensee). The work group chose a control room operator scope.

2. the pre-planned operator actions designated by the accident analyses in Chapter 15 of a plant's FSAR and the not previously planned operator action that may be required during unforeseen events. The Working Group established Type A instrumentation for the former, and Type B or C instrumentation for the latter.

3. The monitoring of fission product barrier integrity and the potential for breach of a given barrier. The work group chose monitoring of actual breach for the fuel, reactor coolant system, and containment barrier, and the potential for breach of the containment barrier.

4. The degree of alignment of accident monitoring instrumentation with IEEE Class 1E (ANS Class EC-3) and whether an intermediate class is needed between 1E and non-1E. The work group chose to define specific design criteria for each variable type in lieu of applying a blanket classification such as 1E.

5. Whether a list of variables should be included as an appendix to the standard:

   a. a list of only Type C variables
   b. a list of Type A, B, C and D variables. The work group chose to include Type B and Type C variables recommendations in the standard.

6. The definitions of instrument Types B and D and whether these types should be included in the standard. The work group chose to include Type B and to exclude Type D variables in this standard. Type D variables should be addressed by Safety System Standards, i.e., IEEE-603.

The membership of the Working Group is as follows:

L. Stanley, Chairman

T. Timmons, Vice Chairman and Correspondent

D. Sommers

E. Wenzinger

D. Lambert

R. Bauerle

J. Castanes

M. Wolpert

H. Mumford

X. Polanski

E. Dowling

Additional input has been provided to the Working Group by industry, university, and government participants throughout the meetings. The Work Group is very appreciative of this assistance.

1.0 Introduction

The Code of Federal Regulations requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to assure adequate safety. The purpose of this standard is to establish criteria for the selection of variables and instrumentation. These criteria are based on the sequence and duration of the phases through which an accident progresses. The control room operator may have different information requirements for each phase of an accident.

This standard presents criteria for monitoring the response of the plant to design basis events. It also presents criteria for monitoring the integrity of fission product barriers under conditions which have degraded beyond the design bases. This fission product barrier monitoring is considered to be an extra set of requirements imposed on the instrumentation beyond that required for satisfactorily monitoring accident scenarios postulated in the plant safety analysis.

Throughout these criteria, three verbs have been used to indicate the degree of rigor intended by the specific criterion. The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

2.0 SCOPE

This standard contains criteria for determining the variables to be monitored by the control room operator of a light water reactor, as required for safety, during the course of an accident and during the long-term stable shutdown phase following the accident. Also included are criteria for determining the requirements for the equipment used to monitor those variables.

The scope of the standard is limited to onsite environment and process monitoring. Emergency preparedness planning is, or will be, covered by other standards.

3.0 DEFINITIONS

Phase I — That portion of the accident extending from the initiation of the accident to that point at which the plant is in a controlled condition.

Phase II — That portion of the accident extending from the time at which the plant is in a controlled condition to the time at which personnel access to that part of the plant which requires inspection, repair or replacement is possible.

Phase III — That portion of the accident extending from the end of Phase II to the time at which the plant has returned to operating status or has been decommissioned.

Type A — Those variables to be monitored that provide the information required to permit the control room operator to take the pre-planned manual actions to accomplish safe plant shutdown for design basis accident events and to maintain long term plant stability.

Type B — Those variables to be monitored that provide to the control room operator information to monitor the process of accomplishing critical safety functions, i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, maintaining primary containment integrity and radioactive effluent control.

Type C — Those variables to be monitored that provide to the control room operator information to monitor (1) the extent to which parameters, which have the potential for causing a breach of the final fission product barrier (i.e., the containment), have exceeded the design basis values, or (2) that a fission product barrier (i.e., fuel clad, reactor coolant pressure boundary or the containment) may have been breached.

Type D Those variables to be monitored that provide to the control room operator information to monitor the operation of individual safety systems.

Design Basis Accident Events

Those events postulated in the plant safety analyses, any one of which may occur during the lifetime of a particular plant, excluding those events which are expected to occur during a calendar year for a particular plant; and those events that are not expected to occur but are postulated in the plant safety analyses because their consequences would include the potential for the release of significant amounts of radioactive material.

4.0 DISCUSSION

It is the philosophy of this Standard that instrumentation is required to monitor plant performance during and after an accident. The purposes of the accident monitoring instrumentation are enumerated in Section 5.0, Design Basis. This Standard specifies the criteria to be used by the designer in selecting the variables to be monitored.

Certain concepts have been established to aid the system designer in the selection of variables to monitor the course of an accident and to arrive at appropriate design criteria for instruments to monitor these variables.

4.1 Planned Versus Unplanned Operator Actions

The plant safety analysis defines the accident scenarios from which the safety system design bases and the planned or anticipated operator actions are derived. Accident monitoring instrumentation shall be provided to permit the operator to take required actions to address these analyzed situations. However, instrumentation shall also be provided for unplanned situations, (i.e., to ensure that, should plant conditions evolve differently than predicted by the safety analysis, the operator has sufficient information to monitor the course of the event). Instrumentation shall also be provided to indicate to the operator if fission product barrier integrity has degraded beyond the prescribed limits of the Safety Analysis.

4.2 Variable Types

Four classifications of variables have been identified. Operator manual actions during accidents included in the plant safety analysis are anticipated or pre-planned. Those variables that provide information needed by the operator to perform these manual actions are designated

Type A. No identification of specific Type A monitored variables is provided because they are plant unique. The process for selecting Type A variables is given in Section 5.1.1. Those variables needed to assess that the plant safety functions are being accomplished, as identified in the plant safety analysis, are designated Type B. Variables used to monitor for the gross breach of one of the fission product barriers or the potential breach of the final fission product barrier (containment) are designated Type C. Type C variables used to monitor the potential breach of containment have an arbitrarily-determined, extended range. The extended range shall be chosen to minimize the probability of instrument saturation even if conditions exceed those predicted by the safety analysis. The fourth classification, Type D, consists of those variables monitored to ascertain that the safety systems are performing as designed. Type D variables are less important than Types A, B and C for accident monitoring since safety system performance only implies safety function accomplishment. Type D variables and instruments are not considered to be within the scope of Accident Monitoring Instrumentation. Guidance on the selection of Type D variables and the specification of appropriate design criteria is not given in this standard. This guidance should be provided in standards for design of safety systems (e.g. IEEE-603, ANSI N18.2, etc). The four classifications are not mutually exclusive in that a given variable (or instrument) may be included in one or more types. This differentiation by variable type is intended only to guide the designer in his selection of accident monitoring variables and applicable criteria.

4.3 Accident Phases

The typical accident sequence has been subdivided into three phases: Phase I covers the initial portion of the accident, Phase II covers the stable long-term cooling portion of the accident up to the time where personnel access is possible, and Phase III addresses the period following personnel access to the accident area. This sub-division has been made so that variable selection and design criteria application can reflect the differing conditions which characterize these three phases. For example, Phase I can be anticipated to be of relatively short duration, having relatively severe plant conditions, and allowing no personnel access to the accident area. Phase II is expected to be of longer duration, to require a significant number of operator actions, under milder plant conditions, but with still no personnel access to the accident area. Phase III is expected to be of even longer duration, but during this phase personnel access is possible. Different design criteria are then appropriate for each of the three phases. In this Standard, guidance and criteria are provided for Phases I and II. Phase III is not addressed by this standard.

5.0 Design Basis

The plant designer shall perform and document an evaluation to select accident monitoring variables and instruments. He shall identify instruments required by his design to enable the control room operator to:

A. Perform pre-planned manual actions.

B. Monitor:

(1) Reactivity control
(2) Reactor core cooling
(3) Reactor coolant system integrity
(4) Primary containment integrity and
(5) Radioactive effluent control

C. Ascertain (1) the extent to which variables that indicate the potential for causing a breach of the primary containment, have exceeded the design basis values, and (2) a fission product barrier (i.e. fuel clad, reactor coolant system pressure boundary or the primary containment) may have been breached.

5.1 Variable Selection for Phases I and II

The process for selection of the Accident Monitoring Instrumentation variables shall include:

5.1.1 For Type A

1) Identification of the design basis accident events for which manual action is required.

2) Identification of planned operator actions

3) Identification of the monitored variables needed for planned operator actions.

5.1.2 For Type B

1) Identification of the monitored variables that provide the most direct indication needed to assess the accomplishments of:

a. Reactivity Control
b. Reactor Core Cooling
c. Reactor Coolant System Integrity
d. Primary containment Integrity
e. Radioactive Effluent Control

Guidance on the selection of these variables is provided in Section 6.0.

5.1.3 For Type C

1) Identification of the monitored variables that provide the most direct indication of a gross breach of a fission product barrier or of an approach to breach of the containment. These instruments may have extended ranges. Guidance on the selection of these variables is provided in Section 6.0.

5.1.4 Phase II Termination

Prior to the termination of Phase II, the ability to gain access to that part of the plant that requires inspection, repair, or replacement shall be determined. Instrumentation that indicates when conditions are acceptable for personnel access shall be identified.

5.2 PERFORMANCE REQUIREMENTS FOR PHASES I AND II

The determination of performance requirements for Accident Monitoring Instrumentation shall include, as a minimum, the following considerations:

1) Identification of the range of the process variable.
2) Identification of the required accuracy of measurement.
3) Identification of the required response characteristics.
4) Identification of the time interval during which the measurement is needed.
5) Identification of the local environment(s) in which the instrumentation components must operate.

For Types A and B these performance requirements shall be derived from the plant safety analysis. For Type C, guidance on performance requirements is provided in Section 6.3. These requirements for Type C are based on engineering judgment.

6.0 DESIGN CRITERIA

6.1 GENERAL DESIGN CRITERIA

The following General Design Criteria apply to Accident Monitoring Instrumentation Type A, B, and C Variables unless otherwise noted.

6.1.1 EQUIPMENT QUALIFICATION~~S~~

Accident monitoring instrumentation that is to be environmentally qualified shall be qualified to IEEE Standard 323-1974.

Accident monitoring instrumentation that is to be seismically qualified shall be qualified to IEEE Standard 344-1975. The instrumentation shall be qualified to continue to function within the required accuracy following, but not necessarily during, a safe shutdown earthquake.

6.1.2 DURATION

Accident monitoring instrumentation shall be qualified for the length of time its function is required. Unless other times can be justified, Phase II instrumentation shall be qualified to function for not less than 100 days. A shorter time may be acceptable if instrumentation equipment replacement or repair can be accomplished within an acceptable out-of-service time, taking into consideration the environment where the equipment is located.

6.1.3 DIRECT MEASUREMENT

To the extent practical, accident monitoring instrumentation inputs shall be from sensors that directly measure the desired variables.

6.1.4 MINIMIZING MEASUREMENTS

To the extent practical, the same instruments shall be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during an accident situation, instruments with which he is most familiar. However, where the required range of accident monitoring instrumentation results in a loss of required instrumentation sensitivity in the normal operating range, separate instruments shall be used.

6.1.5 INSTALLATION

Permanently installed instrumentation shall be used for those instruments required to function during Phase I. Permanently installed instrumentation need not be provided for those functions required only for Phases II and III providing it can be demonstrated that the instrument components can be installed when required, considering the local environment.

6.1.6 DISPLAY LOCATION AND IDENTIFICATION

Accident monitoring instrumentation displays shall be located accessible to the operator and be distinguishable from other displays so that in an accident situation, the operator can rapidly identify the accident monitoring instrumentation.

6.1.7 EQUIPMENT REPAIR

The accident monitoring instrumentation shall be designed to facilitate timely recognition, location, replacement, and repair or adjustment of malfunctioning equipment.

6.1.8 TEST AND CALIBRATION

6.1.8.1 Test

Capability shall be provided for testing, with a high degree of confidence, the operational availability of each instrument channel during plant operation. This may be accomplished in various ways, for example:

1. By observing the effect of perturbing the monitored variable.

2. By observing the effect of introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable.

3. By cross-checking between channels that bear a known relationship to each other.

Where testing during reactor operation is not possible, it shall be shown that there is no practical way of implementing such a requirement without adversely affecting plant safety or operability. In addition, it shall be shown that the probability of a failure of the component which is not periodically tested during plant operation is acceptably low and that such testing can be routinely performed when the reactor is shut down.

6.1.8.2 Calibration

Capability shall be provided for calibration of each instrument channel during normal plant operation or during shutdown as determined by the required interval between calibrations. Equipment that does not require periodic calibration is exempt from this requirement.

6.1.9 SINGLE FAILURE CRITERIA

That accident monitoring instrumentation ~~that is required to meet the~~ single falure criterion (see Table 6.4-1) shall ~~be~~ use independent redundation or diverse instruments. Diversity, the use of different variables to provide the required information, is preferred.

6.1.10 REDUNDANT READOUT AMBIGUITY

Where a disagreement between redundant displays could lead the operator to defeat or fail to accomplish a required safety function, additional information shall be provided to allow the operator to deduce the actual conditions that are required for him to perform his role. This may be accomplished by providing an independent channel which monitors a different variable bearing a known relationship to the redundant channel or by providing an additional independent channel of instrumentation of the same variable or by providing the capability for the operator to perturb the measured variable and determine by observation of the response which instrumentation display has failed.

6.2 TYPE B INSTRUMENTS

6.2.1 GENERAL REQUIREMENTS

The number of instruments required shall be only that minimum set needed to adequately monitor the accomplishment of the following functions:

a. Reactivity Control
b. Reactor Core Cooling
c. Reactor Coolant System Integrity
d. Primary containment Integrity
e. Radioactive Effluent Control

Type B instruments provide control room indication beyond that which may be required for any preplanned operator action.

6.2.2 VARIABLES FOR REACTIVITY CONTROL MONITORING

The measured variable shall indicate the accomplishment of control of reactivity in the core. The measured variable should be neutron flux. The range of measurement should extend from one count per second on the source range instrument to the intermediate range instrument value corresponding to 1.% of full reactor power. This range is intended to encompass all neutron flux levels at which the core can be subcritical.

6.2.3 VARIABLES FOR CORE COOLING MONITORING

The measured variables shall indicate the accomplishment of core cooling. For the PWR, the measured variables should be hot leg temperature, cold leg temperature, pressurizer level, and reactor coolant system pressure. For the BWR, the measured variable should be reactor vessel water level. Core channel outlet temperature monitoring should be considered for inclusion as a desireable variable to ascertain cooling for PWR's.

6.2.4 VARIABLES FOR REACTOR COOLANT SYSTEM INTEGRITY

The measured variable shall indicate the accomplishment of RCS Integrity. The measured variable should be primary system pressure.

6.2.5 VARIABLES FOR PRIMARY CONTAINMENT INTEGRITY

The measured variables shall indicate the accomplishment of primary containment integrity. The measured variables should be primary containment hydrogen (or oxygen for inerted containments) concentration, primary containment pressure and primary containment isolation valve positions.

6.2.6 VARIABLES FOR RADIOACTIVE EFFLUENT CONTROL

The measured variables shall indicate the accomplishment of radioactive effluent control. The measured variables should be noble gas monitoring of the identified plant release points.

6.3 TYPE C INSTRUMENTS

6.3.1 Type C instruments shall meet the following criteria:

6.3.1.1 The number of instruments used shall be only that minimum set needed to adequately monitor the three barriers;

6.3.1.2 Each measurement shall be as direct as possible;

6.3.1.3 Any chosen measurement(s) shall detect the possibility of a gross breach of one or more barriers (i.e., > 1 percent fuel clad failure, a RCS pressure boundary breach producing a loss of reactor coolant inventory in excess of the normal makeup capability, a containment breach capable of producing radiation releases in excess of 10 CFR 100 at the site boundary using TID-14844 source terms); the ranges established for Type C instruments are not mechanistically related to a postulated accident scenario.

6.3.1.4 During the period of need for Type C instruments, no other failures shall be assumed beyond the breach of a barrier coincident with loss of off-site power;

6.3.2 <u>Fuel Clad Barrier Monitoring</u>

6.3.2.1 The measured variable(s) shall detect and alarm the breach of the fuel clad barrier (i.e., > 1 percent fuel clad failure);

6.3.2.2 Operator sampling of reactor coolant should be used as the means to verify the measured variable.

6.3.2.3 The measured variable should be reactor coolant system radioactivity. The instrument range should be equivalent to the fuel clad gap activity corresponding to 0.5% to 5% failed fuel. A narrow accuracy band for this measured variable is not significant in achieving this function; for example, ±50% to ±100% accuracy of reading should be acceptable.

6.3.3 <u>Reactor Coolant System Pressure Boundary Monitoring</u>

6.3.3.1 The measured variable(s) shall detect and alarm a breach of the reactor coolant system that produces a loss of coolant inventory in excess of normal makeup capability. The spectrum of RCS pressure boundary breaches extends up to and includes the largest double-ended pipe break.

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6.3.3.2 The means used to detect RCS pressure boundary breach should include one RCS pressure boundary variable and one containment variable over the full spectrum of break sizes.

6.3.3.3 The measured PWR variables should be RCS pressure, primary containment sump water level and ~~primary~~ containment pressure. The instrument range should be the design value plus a specified margin ($\geq$ 10%).

6.3.3.4 The measured BWR variables should be drywell pressure and containment sump water level. The instrument range should be the design value plus a specified margin ($\geq$ 10%).

6.3.4 Containment Pressure Boundary Monitoring

6.3.4.1 The measured variable(s) shall detect and alarm a breach of the containment pressure boundary that is capable of producing radiation releases in excess of 10 CFR 100 at the site boundary using TID-14844 source terms.

6.3.4.2 The means used to detect containment pressure boundary breach should include containment pressure (BWR and PWR), environs radiation monitoring for gross gamma (PWR), and secondary containment air space radiation monitoring for gross gamma (BWR).

6.3.4.3 The instrument range for containment pressure should be design pressure plus a specified margin ($\geq$ 10%).

6.3.4.4 The instrument range for environs radiation monitoring should be 10^{-3} to 10^{2} R/hr. The instrument range for secondary containment air space radiation monitoring should correspond to the 10 CFR 100 value for off-site doses. Instrument accuracy should be $\pm$ 1/2 decade (100 Kev-3 Mev).

6.3.5 Potential Breach of the Final Fission Product Barrier

6.3.5.1 The measured variables should be containment pressure, containment hydrogen concentration, and RCS pressure for indicating the potential for causing a breach of the final fission product barrier (i.e., containment).

6.3.5.2 An arbitrary range of 3 times design pressure for concrete and 4 times design pressure for steel should be used for containment pressure. Instrument accuracy should be ± 10% of span.

6.3.5.3 An arbitrary range of 0-10 volume percent hydrogen should be used for containment hydrogen concentration. Instrument accuracy should be ± 10% of span.

6.3.5.4 An arbitrary range of 1.5 times design pressure should be used for RCS pressure. Instrument accuracy should be ± 10% of span.

6.3.6 ENVIRONMENTAL QUALIFICATION

6.3.6.1 Type C instruments shall be environmentally qualified ~~in accordance with Section 6.1.1 except~~, the assumed maximum value of the monitored parameter shall be the value equal to the maximum range for the instrument. The monitored parameter shall be assumed to approach this peak by extrapolating the most severe initial ramp associated with the Design Basis Accidents. The decay for this parameter shall be considered proportional to the decay for this parameter associated with the Design Basis Accidents. No additional qualification margin needs to be added to the extended range parameter. See figure 6.3-1. All environmental envelopes except that pertaining to the parameter measured by the instrument shall be those associated with the Design Basis Accidents.

6.4 SPECIFIC DESIGN CRITERIA

Design Criteria specific to particular accident phases and variable types are presented in Table 6.4-1.

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TABLE 6.4.1

DESIGN CRITERIA

CRITERION	PHASE 1 VARIABLE TYPE A	PHASE 1 VARIABLE TYPE B	PHASE 1 VARIABLE TYPE C	PHASE II VARIABLE TYPE A	PHASE II VARIABLE TYPE B	PHASE II VARIABLE TYPE C
1. Qualify seismically to IEEE 344-75 (operate after SSE)	Yes	Yes	No	Yes	No	No
2. Meet single failure per IEEE 379-77	Yes	Yes	No	Yes	Yes	No
3. Qualify environmentally to IEEE 323-74	Yes	Yes	Yes(1)	Yes	Yes	Yes(1)
4. Consider loss of off-site power	Yes	Yes	Yes	Yes	No	No
5. Power source	Emergency	Emerg.	Emerg.	Emerg.	Normal(6)	Normal(6)
6. Out of service interval - prior to accident	(2)	(2)	≤72 Hr(3)	(2)	(2)	≤72 Hrs(3)
7. Out of service interval - during accident	None	None	≤2 Hr	(2)	(2)	≤2 Hrs

TABLE 6.4-1 (Continued)

DESIGN CRITERIA

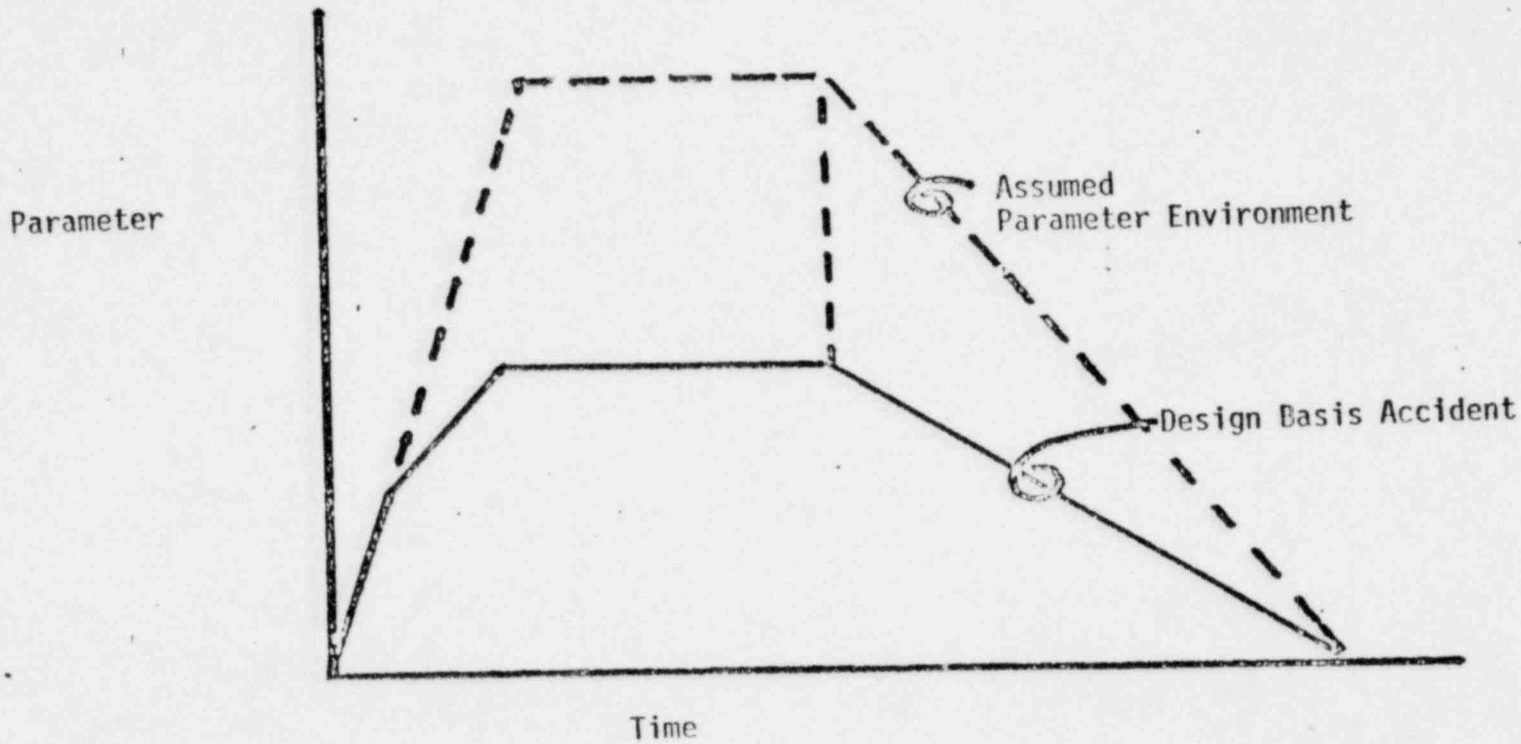
CRITERION	PHASE 1 VARIABLE TYPE			PHASE II VARIABLE TYPE		
	A	B	C	A	B	C
8. Portable instrumentation	No	No	No(7)	Yes	Yes	Yes
9. Level of quality assurance	NQA-1-79	NQA-1-79	NQA-1-79	NQA-1-79	NQA-1-79	NQA-1-79
10. Display type(4)	Continuous	Continuous	Continuous	Continuous	Continuous	On demand
11. Display method	Recording(5)	Recording	Indicator	Recording(5)	Indicator	Indicator
12. Identification as accident monitoring type	Yes	Yes	Yes	Yes	Yes	Yes
13. Periodic Test per IEEE-338-1977	Yes	Yes	Yes	Yes	Yes	Yes

NOTES: (1) See Paragraph 6.3.6 of this Standard.
(2) IEEE 279-1971 Paragraph 4.11 Exemption

NOTES TO TABLE 6.4-1 (Continued)

(3) Based on normal tech spec requirements on out-of-service safety systems.
(4) Continuous indication or recording displays a given variable at all times; intermittent indication or recording displays a given variable periodically; on demand indication or recording displays a given variable only when requested.
(5) Where trend or transient information is essential to planned operator actions.
(6) May be manually connected to emergency buss
(7) Radiation monitoring outside containment may be portable.

Figure 6.3-1. Typical Environmental Qualification Envelope for Type C Instruments



ANS 4.5

- STANDARD BEGUN 7/30/79
- COMPLETE DRAFT ISSUED 9/19/79
- BALLOT BY ANS-4 AND REVIEW BY NUPPSCO INDICATE NEED FOR FURTHER REVISION OF ANS 4.5
- REVIEW AND RESOLUTION OF COMMENTS ARE PROCEEDING ON AN ACCELERATED SCHEDULE
- CONCENSUS STANDARD WILL BE AVAILABLE IN APRIL - MAY 1980

AMI OBJECTIVES

1. CHARACTERIZE STATUS OF PLANT DURING AN ACCIDENT

2. CLEAR AND UNDERSTANDABLE

   - MINIMUM INSTRUMENT SET

   - UNIQUELY IDENTIFIED

3. ASSURANCE OF AVAILABILITY

ANS 4.5 APPROACH

1. DEFINED ACCIDENT PHASES

2. DEFINED FUNCTIONS TO BE PERFORMED

3. DEFINED PROCESS FOR VARIABLE SELECTION

4. DEFINED CRITERIA TO BE APPLIED TO VARIABLES (BASED ON INTENDED USE)

5. DESIGNER SELECTS VARIABLES BY APPLYING CRITERIA

MONITORING FUNCTIONS

TYPE A - PREPLANNED MANUAL ACTION

TYPE B - CRITICAL SAFETY FUNCTIONS

REACTIVITY CONTROL

CORE COOLING

REACTOR COOLANT SYSTEM INTEGRITY

PRIMARY CONTAINMENT INTEGRITY

RADIOACTIVE EFFLUENT CONTROL

TYPE C - BARRIER INTEGRITY

FUEL FAILURE

REACTOR COOLANT SYSTEM BREACH

PRIMARY CONTAINMENT BREACH

POTENTIAL FOR PRIMARY CONTAINMENT BREACH

RG 1.97 CONCERNS

1. SYSTEMATIC APPROACH TO DESIGN NOT FOLLOWED

   TABLE 2/3 VARIABLES ARE NOT DERIVED FROM AMI CRITERIA

   SUFFICIENCY AND NECESSITY OF TABLE 2/3 VARIABLES TO MEET AMI FUNCTIONS NOT ESTABLISHED

2. ADDITION OF "D" AND "E" VARIABLE TYPES BLURS AMI FOCUS

   TYPE "D" VARIABLES SHOULD BE ADDRESSED WITH SAFETY SYSTEM DESIGN

   TYPE "E" DEFINITION IS AN OPEN-ENDED CATCHALL

3. OTHER DIFFERENCES WITH ANS 4.5

   SCOPE EXPANDED TO EMERGENCY PLANNING WITHOUT PROVIDING CRITERIA

   EXPANDED TYPE "C" DEFINITION FOR BARRIER BREACH

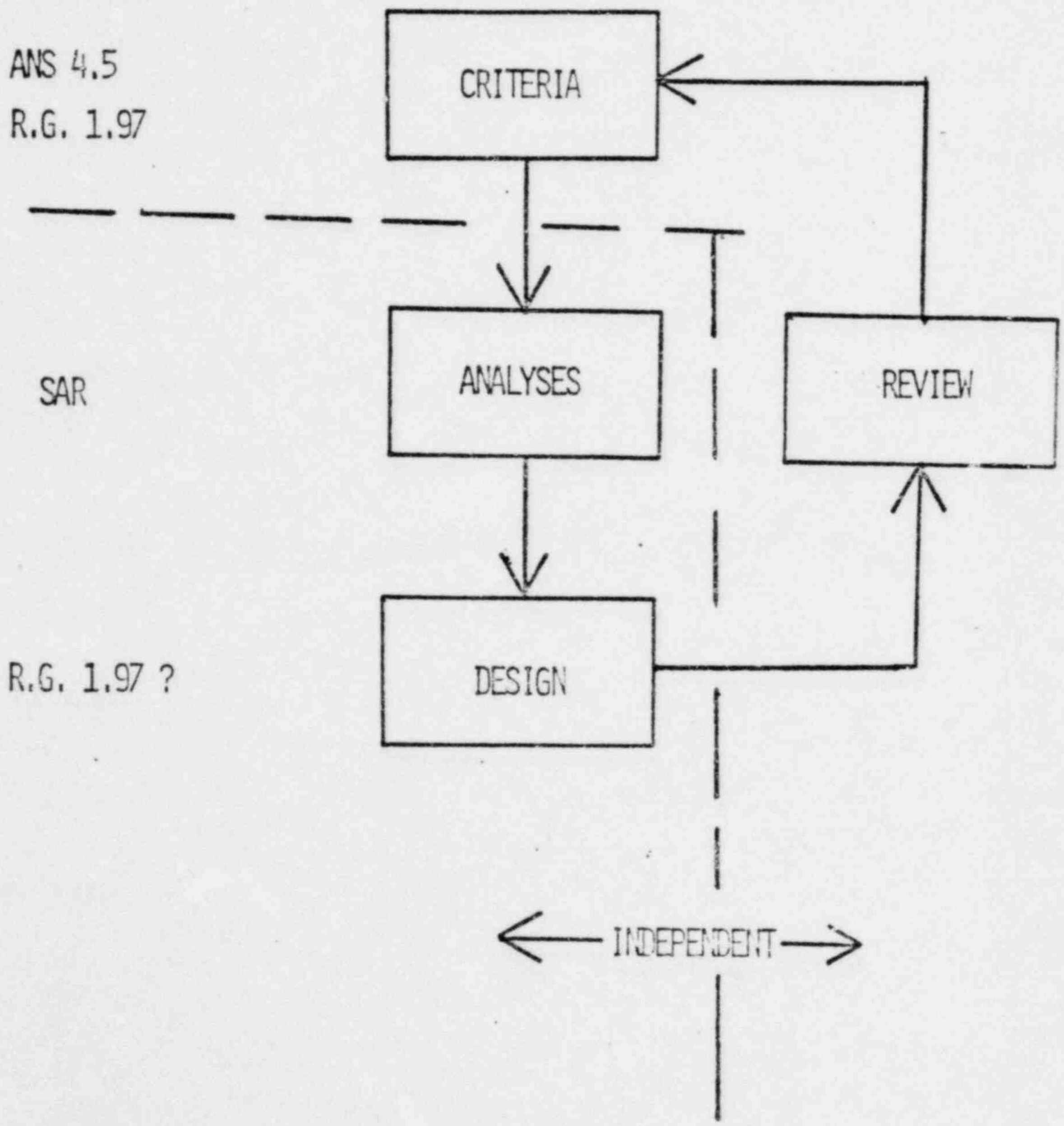
   TRANSIENTS AS WELL AS ACCIDENTS ADDRESSED

   TYPE "C" QUALIFICATION LEVEL INCREASED

   PHASE II QUALIFICATION DURATION INCREASED

   ANALYSIS REQUIRED FOR TYPE "C" NOT POSSIBLE

DESIGN PROCESS



6.1

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

November 1, 1979

APPENDIX XXI
BACKGROUND MATERIAL FOR DISCUSSION OF RESOLUTION OF NRC CATEGORY A SAFETY RELATED TASKS (NUREG-0606)

M. Bender, Chairman, Generic Items Subcommittee

SUBJECT: COMPARISON OF ITEMS BETWEEN NUREG-0606, "UNRESOLVED SAFETY ISSUES," SUMMARY AQUA BOOK AND THE ACRS GENERIC ITEMS

As you requested, I went through the Aqua book and compared the 19 unresolved safety issues listed there with the ACRS generic items contained in Report No. 7. The results of the cross check are contained in the attached table. The majority of the items appear to have a direct counter part from one list to the other. In a few cases there are no items directly related from one list to the other, although somewhat similar items or responsible subcommittees were cited.

1

Attachment:
As stated

cc: ACRS Members
R. Fraley
M. Libarkin
J. McKinley

TABLE

"UNRESOLVED SAFETY ISSUES" - NUREG-0606 AQUA BOOK	ACRS GENERIC ITEMS - REPORT NO. 7 MARCH 21, 1979
A-1 Water Hammer	74. Water Hammer
A-2 Asymmetric Blowdown Loads	73. Vessel Support Structures
A-3, A-4, A-5 Steam Generator Tube Integrity	64. Steam Generator Tube Leakage
A-7 Mark I Long Term Program	75. Behavior of BWR Mark I Containments
A-8 Mark II Programs	No ACRS Generic Item on BWR Mark II Containments although two generic items on BWR Containments: 67. Behavior of BWR Mark III Containments 75. Behavior of BWR Mark I Containments Fluid Dynamics Subcommittee looks at BWR containment programs.
A-9 ATWS	29. Anticipated Transients Without Scram
A-10 BWR Feedwater Nozzle Cracking	68. Stress Corrosion Cracking in BWR Piping
A-11 Reactor Vessel Materials Toughness	15. Pressure Vessel Surveillance of Fluence and NDT Shift 16. Nil-ductility Properties of Pressure Vessel Materials 55. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock
A-12 Fracture Toughness of Steam Generator and Reactor Coolant and Pump Supports (This item came up during the North Anna licensing process - questions were raised as to the potential for lamillar tearing and low fracture toughness of the support materials used. Similar material used at other PWRs made the issue generic.	No one-to-one relation with ACRS generic items, although some relation to 73. Vessel Support Structures, however, item 73 is basically blowdown loads. This issue is being followed the the Metal Components Subcommittee A-298

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

September 23, 1979

SECY-79-409A

INFORMATION REPORT

For: The Commissioners

Thru: L. V. Gossick, Executive Director for Operations

From: H. R. Denton, Director
Office of Nuclear Reactor Regulation

Subject: STAFF PROGRESS ON UNRESOLVED SAFETY ISSUES

Purpose: To inform the Commission of recent actions related to NRR's Unresolved Safety Issues Program and to provide the Commission with current schedules for resolving these issues.

Discussion: In SECY 79-409 dated June 21, 1979, I described the actions I had taken to assure timely staff action on Unresolved Safety Issues. The approach described included the assignment of a full time Dire[illegible] Deputy and to the maximum extent possible, named reviewer[illegible]icated to the Unresolved Safety Issues Program within the NRR Divisions. I also provided a summary of Unresolved Safety Issue schedules. I indicated that I intended to provide another progress report to the Commission in September 1979. This information report fulfills that commitment.

Enclosure 1 is a program overview providing schedules as now projected for Unresolved Safety Issues. This program overview was taken from NUREG-0606, "Unresolved Safety Issues Summary - Aqua Book" published on September 4, 1979. The current schedules are not significantly changed from those provided last June. The only significant schedule slip is one of about 7 months for completion of Task A-1, Water Hammer, from December 1980 to August 1981. The principal reason for this slip was the identification of additional work required to assess the safety significance of certain water hammer scenarios.

As indicated in Enclosure 1, four NUREG reports were scheduled for issuance this month. They are reports related to Tasks A-2, A-12, A-24 and A-42. None have been issued to date. All, however, are

Contact:
[illegible]. Aycock
492-7754

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in the final stages of preparation and publication. The NUREG report for Task A-12 has been sent to the printer and will likely be issued by the projected date of September 28, 1979. Issuance of the NUREG reports for the other three tasks may be delayed until October. I intend to transmit an information report to the Commission that provides each USI NUREG report when it is published and describes our plans for implementing the results of the staff evaluation described in the report. The description will include such items as our plans for public notice and/or comments, RRRC review, ACRS review, standards and/or SRP development and implementation on operating plants.

In addition to the program overview in Enclosure 1, the Aqua Book provides detailed management information for each of the Unresolved Safety Issue tasks including schedule logic networks.

I will continue to keep you advised of progress on Unresolved Safety Issues. I intend to provide another progress report to the Commission in January 1980.

SEP 27 1979

H. R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Program Overview

DISTRIBUTION:
Commissioners
Commission Staff Offices
Exec. Dir. for Opers.
ACRS
Secretariat

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Taken from NUREG-0606,
Unresolved Safety Issue
Summary - Aqua Book

PROGRAM OVERVIEW - PROJECTED COMPLETION DATES

TITLE	1978 NRC ANNUAL REPORT	POST TMI-2 WORK PLAN (SECY 79-409)	CURRENT	REMARKS
A-1. Water Hammer	NRR Tech Report - 1979 Remainder of A-1 - 1980	July 1979 December 1980	Complete August 1981	NUREG-0582 Issued July 1979. Expanded task scope and TMI-2 manpower impacts resulted in slip.
A-2. Asymmetric Blowdown Loads	Early 1979	October 1979	September 30, 1979	Manpower impacts of Seismic reviews of Shutdown plants resulted in slip.
A-3, A-4, A-5 - Steam Generator Tube Integrity	Early 1980	September 1980	May 30, 1980	Better definition of necessary evaluation of accidents with tube failures has made it possible to recover some time.
A-7. Mark I Long Term Program	October 1979	December 1979	December 1, 1979	
A-8. Mark II Programs	October 1980	Lead Plant Program - 1979 Long Term Program - 1980	LPP - November 30, 1979 LTP - November 15, 1980	
A-9. ATWS	Recommendations to the Commission in Early 1979	Probably 1980	Recommendations to the Commission on April 15, 1980 - ATWS Rule Issued on December 1, 1980	TMI-2 manpower impacts on staff/Industry and the need to address a number of TMI related questions in ATWS evaluations resulted in slips
A-10. BWR Nozzle Cracking	Late 1979	October 1979	November 30, 1979	
A-11. Reactor Vessel Materials Toughness	July 1979	December 1979	December 31, 1979	TMI-2 manpower impacts & & delays in development of elastic-plastic fracture test methods resulted in slip.
A-12. Steam Generator & Reactor Coolant Pump Support	August 1979	September 1979	September 28, 1979	
A-17. Systems Interactions	Phase I - September 1979 Phase II - September 1980	Phase I - Early 1980 Phase II - Late 1980 or early 1981	Phase I - March 1, 1980 Phase II - March 1981	Sandia is behind schedule in completing Phase I report. Underestimated level of effort required
A-24. Qualifications of Class IE Safety-Related Equipment	Phase I (Interim position) - 1979 Phase II - N/S	Phase I - Aug 30, 1980 Phase II - N/S	Phase I - September 15, 1979 Phase II - Deleted from Task Scope	

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PROGRAM OVERVIEW - PROJECTED COMPLETION DATES (CONTINUED)

TITLE	1978 NRC ANNUAL REPORT	POST TMI-2 WORK PLAN (SECY 79-409)	CURRENT	REMARKS
A-36. Control of Heavy Loads Near Spent Fuel	Early 1979	December 1979	November 14, 1979	TMI-2 Manpower Impacts resulted in slip.
A-39. SRV Pool Dynamic Loads	October 1979	Mark I - Dec. 1979 Mark II - Dec. 1979 Mark III - March 1980	Mark I - December 1979 Mark II - December 1979 Mark III - March 1980	
A-40. Seismic Design Criteria	Phase I - 1979 Phase II - 1981	Phase I - Late 1979 Phase II - 1981	Phase I - December 1979 Phase II - March 30, 1981	
A-42. Pipe Cracks in Boiling Water Reactors	N/S	August 1979	September 15, 1979	Task will be complete with issuance of NUREG-0313, Rev. 1.
A-43. Containment Emergency Sump Performance	N/S	N/S - Task Action Plan - July 1979	N/S	Management of Tasks A-43 and A-44 was transferred to RES in July 1979 to return some NRR manpower to casework. Task Action Plans are under development.
A-44. Station Blackout	N/S	N/S - Task Action Plan - July 1979	N/S	

A-303

PROGRAM OVERVIEW   PROJECTED COMPLETION DATES

ACRS Generic Item No.

TITLE	1978 NRC ANNUAL REPORT	POST TMI-2 WORK PLAN (SLCY 79-409)	CURRENT	REMARKS	ACRS Generic Item No.
A-1, Water Hammer	NRR Tech Report - 1979 Remainder of A-1 - 1980	July 1979 December 1980	Complete August 1981	NUREG-0582 Issued July 1979. Expanded task scope and TMI-2 manpower impacts resulted in slip.	74
A-2, Asymmetric Blowdown Loads	Early 1979	October 1979	September 30, 1979	Manpower impacts of Seismic reviews of Shutdown plants resulted in slip.	73
A-3, A-4, A-5 - Steam Generator Tube Integrity	Early 1980	September 1980	May 30, 1980	Better definition of necessary evaluation of accidents with tube failures has made it possible to recover some time.	64
A-7, Mark I Long Term Program	October 1979	December 1979	December 1, 1979		75
X A-8, Mark II Programs	October 1980	Lead Plant Program - 1979 Long Term Program - 1980	LPP - November 30, 1979 LTP - November 15, 1980		—
A-9, ATWS	Recommendations to the Commission in Early 1979	Probably 1980	Recommendations to the Commission on April 15, 1980 - ATWS Rule Issued on December 1, 1980	TMI-2 manpower impacts on staff/industry and the need to address a number of TMI related questions in ATWS evaluations resulted in slips	29
A-10, BWR Nozzle Cracking	Late 1979	October 1979	November 30, 1979		68
A-11, Reactor Vessel Materials Toughness	July 1979	December 1979	December 31, 1979	TMI-2 manpower impacts & & delays in development of elastic-plastic fracture test methods resulted in slip.	55 16
X A-12, Steam Generator & Reactor Coolant Pump Support	August 1979	September 1979	September 28, 1979		
A-17, Systems Interactions	Phase I - September 1979 Phase II - September 1980	Phase I - Early 1980 Phase II - Late 1980 or early 1981	Phase I - March 1, 1980 Phase II - March 1981	Sandia is behind schedule in completing Phase I report. Underestimated level of effort required	52
A-24, Qualifications of Class IE Safety-Related Equipment	Phase I (interim position) - 1979 Phase II - N/S	Phase I - Aug 30, 1980 Phase II - N/S	Phase I - September 15, 1979 Phase II - Deleted from Task Scope		33

X = ACRS interest but no generic item listed in ACRS list
X = Not on ACRS list and little ACRS interest

A-30A

TITLE	1978 NRC ANNUAL REPORT	POST TMI-2 WORK PLAN (SECY 79-409)	CURRENT	REMARKS
X A-36. Control of Heavy Loads Near Spent Fuel	Early 1979	December 1979	November 14, 1979	TMI-2 Manpower Impacts resulted in slip.
A-39. SRV Pool Dynamic Loads	October 1979	Mark I - Dec. 1979 Mark II - Dec. 1979 Mark III - March 1980	Mark I - December 1979 Mark II - December 1979 Mark III - March 1980	75 67
X A-40. Seismic Design Criteria	Phase I - 1979 Phase II - 1981	Phase I - Late 1979 Phase II - 1981	Phase I - December 1979 Phase II - March 30, 1981	
✓ A-42. Pipe Cracks In Boiling Water Reactors	N/S	August 1979	September 15, 1979	Task will be complete with issuance of NUREG-0313, Rev. 1. 68
X A-43. Containment Emergency Sump Performance	N/S	N/S - Task Action Plan - July 1979	N/S	Management of Tasks A-43 and A-44 was transferred to RES in July 1979 to return some NRR manpower to casework. Task Action Plans are under development.
A-44. Station Blackout	N/S	N/S - Task Action Plan - July 1979	N/S	2

A-30G

Outline for S. Hanauer PAR Briefing
November 9 - USI

I. Unresolved Safety Issues Program

A. Definition of Unresolved Safety Issue

B. Generic Issues Program

1. Pre-TMI
2. Sources of New Issues
3. Effect of TMI

C. 1978 NRC Annual Report

1. 17 Issues

D. USI Task Force

1. Director, Deputy, dedicated reviewers, RES assistance

E. How a Issue is Resolved

1. 6 steps
2. Products

II. Status

A. Schedules

B. Accomplishments

1. So Far in 1979
2. Planned Near Term

C. Problems

1. A-9
2. A-11
3. A-17
4. A-42
5. A-44

III. New Unresolved Safety Issues

   A. How identified

   B. How decided

IV. Long Range Program Projections

   A. FY81-83 Budget

A-307

"UNRESOLVED SAFETY ISSUES PLAN"

"SECTION 210. THE COMMISSION SHALL DEVELOP A PLAN FOR PROVIDING FOR SPECIFICIATION AND ANALYSIS OF UNRESOLVED SAFETY ISSUES RELATING TO NUCLEAR REACTORS AND SHALL TAKE SUCH ACTION AS MAY BE NECESSARY TO IMPLEMENT CORRECTIVE MEASURES WITH RESPECT TO SAFETY ISSUES. SUCH PLANS SHALL BE SUBMITTED TO THE CONGRESS ON OR BEFORE JANUARY 1, 1978 AND PROGRESS REPORTS SHALL BE INCLUDED IN THE ANNUAL REPORT TO THE COMMISSION THEREAFTER."

A-308

- PLAN REQUIRED
- SPECIFICATION OF USI
- ANALYSIS
- CORRECTIVE MEASURES
- REPORT TO CONGRESS

DEFINITION OF AN UNRESOLVED SAFETY ISSUE - 1978 NRC ANNUAL REPORT

"AN UNRESOLVED SAFETY ISSUE IS A MATTER AFFECTING A NUMBER OF NUCLEAR POWER PLANTS THAT POSES IMPORTANT QUESTIONS CONCERNING THE ADEQUACY OF EXISTING SAFETY REQUIREMENTS FOR WHICH A FINAL RESOLUTION HAS NOT YET BEEN DEVELOPED AND THAT INVOLVES CONDITIONS NOT LIKELY TO BE ACCEPTABLE OVER THE LIFETIME OF THE PLANTS AFFECTED."

A-309

- GENERIC - SEVERAL PLANTS
- IMPORTANT
- ADEQUACY OF SAFETY REQUIREMENTS
- NOT LIKELY TO BE ACCEPTABLE OVER LIFE OF PLANT

SELECTION PROCESS FOR UNRESOLVED SAFETY ISSUES

INITIAL GENERIC ISSUES IDENTIFICATION

- ACRS REPORTS ON GENERIC ISSUES
- TASC DETERMINATION OF CATEGORIES A, B, C, D

PRELIMINARY RISK-BASED EVALUATION OF GENERIC ISSUES

ABNORMAL OCCURRENCES

POINT SYSTEM FOR ESTIMATING PRIORITY OF GENERIC TASKS

- SAFETY SIGNIFICANCE
- ENVIRONMENTAL SIGNIFICANCE
- EFFECTIVENESS OR EFFICIENCY
- URGENCY
- PROMISES OR PUBLIC INTEREST
- BROADNESS OF APPLICATION

A-310

GENERIC ISSUES PROGRAM

1. PRE-TMI PRIORITY

   A. TOP 20

      19 - USI
      1 - HIGH PRIORITY B-6

   B. SECOND 24

   C. 78 OTHERS

2. SOURCES OF NEW ISSUES

   STAFF LICENSING REVIEW
   ACRS
   EVENT IN OPERATING PLANT - LER
   DEFICIENCY IDENTIFIED IN DESIGN OR CONSTRUCTION
   RESEARCH RESULT

3. EFFECT OF TMI

   BULLETINS AND ORDERS
   LESSONS LEARNED
   KEMENY
   OTHER REVIEWS

UNRESOLVED SAFETY ISSUES
1978 NRC ANNUAL REPORT

1. WATER HAMMER
2. ASYMMETRIC BLOWDOWN LOADS
3. PWR STEAM GENERATOR TUBE INTEGRITY
4. BWR MARK I AND MARK II PRESSURE SUPPRESSION CONTAINMENTS
5. ANTICIPATED TRANSIENTS WITHOUT SCRAM
6. BWR NOZZLE CRACKING
7. REACTOR VESSEL MATERIALS TOUGHNESS
8. STEAM GENERATOR AND REACTOR COOLANT PRESSURE SUPPORTS
9. SYSTEMS INTERACTIONS
10. QUALIFICATION OF CLASS IE SAFETY-RELATED EQUIPMENT
11. REACTOR VESSEL PRESSURE TRANSIENT
12. RHR SHUTDOWN REQUIREMENTS
13. CONTROL OF HEAVY LOADS NEAR SPENT FUEL
14. SEISMIC DESIGN CRITERIA
15. PIPE CRACKS IN BOILING WATER REACTORS
16. CONTAINMENT EMERGENCY SUMP PERFORMANCE
17. STATION BLACKOUT

USI TASK FORCE IN NRR
JUNE 1979

DIRECTOR	S. H. HANAUER
DEPUTY DIRECTOR	M. B. AYCOCK
TASK MANAGERS AND REVIEWERS	CURRENTLY ABOUT 30 DEDICATED AND 20 PART-TIME (ABOUT 35 EQUIVALENT PERSONS)

A-313

STEPS IN RESOLVING AN ISSUE

STEP	PRODUCT
1. Description of Problem and Technical Approach	Task Action Plan
2. Generate and Assemble Necessary Technical Information	Technical Reports
3. Analyze and Develop Licensing Requirements for Public Safety	Staff NUREG Report
4. Peer and Public Review	Comments
5. Promulgate the Requirements	Rules, Guides, Standard Review Plans
6. Implementation	Changes in Design, Hardware, Procedures

SCHEDULES FOR UNRESOLVED SAFETY ISSUES

UNRESOLVED SAFETY ISSUE		AQUA BOOK SCHEDULE
1. WATER HAMMER	A-1	8/81
2. ASYMMETRIC BLOWDOWN LOADS	A-2	9/79
3. PWR STEAM GENERATOR TUBE INTEGRITY	A-3 A-4 A-5	5/80
4. BWR MARK I AND MARK II PRESSURE SUPPRESSION CONTAINMENTS	A-6 A-7 A-8 A-39	12/77C 12/79 11/80 3/80
5. ANTICIPATED TRANSIENTS WITHOUT SCRAM	A-9	12/80
6. BWR NOZZLE CRACKING	A-10	11/79
7. REACTOR VESSEL MATERIALS TOUGHNESS	A-11	12/79
8. STEAM GENERATOR AND REACTOR COOLANT PRESSURE SUPPORTS	A-12	9/79
9. SYSTEMS INTERACTIONS	A-17	3/80 (PH I) 3/81 (PH II)
10. QUALIFICATION OF CLASS IE SAFETY-RELATED EQUIPMENT	A-24	9/79
11. REACTOR VESSEL PRESSURE TRANSIENT	A-26	9/78C
12. RHR SHUTDOWN REQUIREMENTS	A-31	5/78C
13. CONTROL OF HEAVY LOADS NEAR SPENT FUEL	A-36	11/79
14. SEISMIC DESIGN CRITERIA	A-40	12/79 (PH I) 3/81 (PH II)
15. PIPE CRACKS IN BWR	A-42	9/79
16. CONTAINMENT EMERGNCY SUMP PERFORMANCE	A-43	N/S
17. STATION BLACKOUT	A-44	N/S

COMPLETED AND IMMEDIATELY FORTHCOMING

	CURRENT ESTIMATE
	12/77C
	11/79
	11/79
	9/78C
	5/78C
	11/79

ON TIME OR MINOR SLIPS

CURRENT ESTIMATE
SAME
11/79
SAME
SAME
SAME
SAME
SAME
SAME
SAME
SAME
SAME
11/79
1982

PROBLEMS

CURRENT
<u>ESTIMATE</u>

RESOURCES/APPROACH?

CONTRACTOR SCHEDULE?

SCOPE?

NEW TASKS?

APPROACH?

A-318

MAJOR ACCOMPLISHMENTS SO FAR IN 1979

- A-1 - NUREG (Subtask) Issued 7/79
- A-7 - Mark I Acceptenace Criteria Issued 11/79

A-319

REPORTS IN FINAL PUBLICATION

- A-12 - NUREG
- A-42 - NUREG
- A-24 - NUREG

PLANNED MAJOR ACCOMPLISHMENTS IN THE REMAINDER OF 1979

A-2 - NUREG

A-7 - NUREG (SER)

A-10 - NUREG

A-12 - NUREG

A-24 - NUREG

A-36 - NUREG

A-40 - FINAL PHASE I CONTRACTOR REPORT

A-42 - NUREG

A-320

PROBLEMS

A-9 - ATWS

RESOURCES NOT AVAILABLE

TECHNICAL APPROACH

DESIGN BASIS ACCIDENT

VS. SPECIFIED HARDWARE CHANGES

EARLY VERIFICATION

GENERIC VERIFICATION

CREDIT FOR OPERATOR ACTION

DISPUTE OVER COST/BENEFIT

RESULT: DELAY IN DEVELOPING PROPOSED RULE

PROBLEMS

A-11 Reactor Vessel Material Toughness

T/A Contractor work late and difficult to interpret

Result: Few months' delay

PROBLEMS

A-17 Systems Interactions

Lessons from TMI

Changing ACRS Emphasis

How to Scope Phase II

Result: Enlargement of Phase II and likely delay in resolution of issue

PROBLEM

A-42 PIPE CRACKS IN BWR

REPORT IN PRESS

NEW CRACKING

ACRS CONCERNS

NEW PWR PROBLEMS

RESULT: ADDITIONAL TASKS

ENLARGED SCOPE

PROBLEM

A-44 - STATION BLACKOUT

NO PROGRESS SO FAR

SHORT TERM CONCERNS

INVOLVEMENT WITH IREP

A-325

NEW USI FOR 1979 ANNUAL REPORT

- ABBREVIATED PROCESS
- REVIEW CANDIDATES

    TMI RECOMMENDATIONS

    ACRS REPORTS

    OTHERS

- SELECT OBVIOUS USIs (TASC/DENTON)
- ACRS REVIEW
- COMMISSION DECISION
- MORE DETAILED REVIEW OF CANDIDATES IN 1980
- SPECIAL REPORT TO CONGRESS LATER IN 1980

A-326

LONG RANGE PROGRAM PROJECTIONS

. FY81-83 BUDGET ASSUMPTIONS MADE IN SUMMER 1979

7 NEW USI IN FY80

2 NEW USI EACH YEAR AFTER FY80

OTHER GENERIC ISSUES - AVAILABLE RESOURCES

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

OFFICE OF THE
COMMISSIONER

October 9, 1979

Dr. Max W. Carbon
Chairman
Advisory Committee
on Reactor Safeguards
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

SUBJECT: SHORT-TERM RECOMMENDATIONS OF TMI-2 LESSONS LEARNED TASK FORCE

Dear Dr. Carbon:

In the Committee's August 13, 1979 letter to Chairman Hendrie providing the Committee's views on the short-term recommendations of TMI-2 Lessons Learned Task Force, the Committee stated that it believes that the "orderly and effective implementation and the appropriate level of review and approval by the NRC staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by NUREG-0578." It would be helpful for me if you would identify in more detail which of the scheduled items the Committee believes should be extended and the basis for those recommendations.

Sincerely,

Peter A. Bradford
Commissioner

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Ahearne
Samuel J. Chilk

DELETION 1

pages A-329 thru 31

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

JANUARY 8 1979

Docket No. 50-293

Mr. J. E. Howard
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Howard:

Attached for your information is a copy of NUREG 0460, Volume 3 which details our current view related to ATWS. In this supplement a variety of options are considered regarding ATWS. We intend to select one of the ATWS options in the near future and to pursue it to adoption.

However, it is important to note that all of the options under serious consideration by the NRC staff (options #2, 3, and 4 in Volume 3 of NUREG 0460) regarding resolution of the ATWS issue for BWRs require installation of an RPT. While you have committed to install a RPT on your facility, Pilgrim Nuclear Power Station, you have not yet begun to take steps toward such installation, on the grounds that you were awaiting firmer requirements by NRC. The NRC staff now has a firm position that RPT is required for your facility. Therefore, we see no bases for any further delay in implementing an RPT for your facility. The RPT designs discussed in this letter are compatible with ATWS requirements.

To expedite your installation of an approved RPT, the staff is providing a modified description (Appendix A, attached) of design requirements which provide some additional flexibility over those previously provided (May, 1978), but which the staff has found acceptable for RPT systems to be installed in the near future.

For all operating plants, the Monticello RPT design described in NEDO 25016 and summarized in Appendix B has been accepted by the staff as meeting the Appendix A criteria. Sections of NEDO 25016 related to ARI should be ignored as that system is not addressed by this letter. Some operating plants have already installed the "BWR/4" or "Hatch" RPT, and the staff also accepts that design as meeting the Appendix A criteria provided the changes specified in Appendix B, or equivalent changes, are incorporated.

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Both the Monticello design and the modified "BWR/4" or "Hatch" design utilize generator field breakers which have been modified so that they are provided with two trip coils. One coil for each breaker is actuated only by reactor pressure and water level sensors in RPT division A, and the other coil is actuated by pressure and level sensors in RPT division B, thereby providing redundancy of power supplies available to the overall system and increasing trip reliability.

Either the Monticello or modified "BWR/4" or "Hatch" design, would be an acceptable RPT design provided diverse final trip relays of a different type are used, or obtained from a different manufacturer than the primary scram relays used in the RPS.

The staff has not reviewed the specific design of the time delay circuitry recently proposed for the Monticello RPT design for low-level initiated pump trips. We agree that time delays on the order of 10 seconds are desirable to avoid making the consequences of a postulated LOCA more severe, and we agree that such delays of around 10 seconds have insignificant effect on ATWS consequences (for low-level initiated ATWS pump trips only). Therefore, we find incorporation of such circuitry on either RPT design discussed above to be acceptable, provided:

1. The time delay is realized only for low-level initiated pump trips; and,

2. The circuitry is incorporated in such a way that it does not significantly affect the overall reliability of the RPT; that is, that no single failure in the timing circuit(s) can cause failure of the pump trip to occur. This could be accomplished, for example, by use of a separate, independent timing (delay) circuit with <u>each</u> low-level sensor, or equivalent.

Implementation as soon as possible of an RPT in accordance with the attached design criteria will provide an increased level of safety over the lifetime of the plant and should be installed as promptly as is reasonable.

The staff has given careful consideration to the concern expressed by some licensees that RPT design requirements may change in the future. We have concluded that the design criteria outlined in this letter (Appendix A) are, for operating plants, equivalent to those enclosed with the May, 1978 letters to all BWR licensees, and we intend to effect no changes to those criteria in the future.

We believe that RPT design, procurement, and installation can be accomplished within a two year period without requiring additional outage time beyond refueling outages.

We have given consideration to steps that can be taken at present, in order to reduce the risk from ATWS events during the interim period before recirculation pump trip circuitry and any other necessary plant modifications are completed. We have determined that many of the following steps are practicable and appropriate for your facility for this interim period. We therefore, request that you inform us within 90 days that you have done the following:

1. Developed emergency procedures to enable operators to recognize an ATWS event, including consideration of scram indicators, rod position indicators, flux monitors, vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure, and radiation indicators.

2. Train operators to take actions in the event of an ATWS including consideration of manually tripping the recirculation pumps and scramming the reactor by using the manual scram buttons, changing individual rod scram switches to the scram position, stripping the feeder breakers on the reactor protection system power distribution buses, opening the scram discharge volume drain valve, prompt actuation of the standby liquid control system, and prompt placement of the RHR in the pool cooling mode to reduce the severity of the containment conditions.

Early operator action as described above would provide significant protection from those ATWS events which occur at low power levels where the rise in the vessel pressure and the containment temperature is limited to acceptable values by manual recirculation pump trip and actuation of the existing standby liquid control system. If the operator were to promptly (in a few seconds) trip the recirculation pumps to assure that the short term rise in vessel pressure is not excessive, protection will also be provided for those ATWS events where the common mode failure occurs in either the electrical portion of the scram system or in some portions of the drive system.

Mr. J. E. Howard

Within 90 days inform us of your schedule for implementation of your commitment to install an RPT system for your plant. Such system should conform to the acceptable systems described in this letter and your schedule should be consistent with the staff's overall objective of assuring that an acceptable RPT system is installed at your facility within two years.

Sincerely,

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. NUREG 0460, Volume 3
2. Appendices A and B

cc w/enclosure No. 2:
see next page

Boston Edison Company

cc
Mr. Paul J. McGuire
Pilgrim Station Acting Manager
Boston Edison Company
RFD #1, Rocky Hill Road
Plymouth, Massachusetts 02360

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Henry Herrmann, Esquire
Massachusetts Wildlife Federation
151 Tremont Street
Boston, Massachusetts 02111

Plymouth Public Library
North Street
Plymouth, Massachusetts 02360

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APPENDIX A

CRITERIA FOR HIGH PRESSURE-LOW LEVEL INITIATED RECIRCULATION PUMP TRIP (RPT) TO BE INSTALLED IN OPERATING BWRs BEFORE NOVEMBER 1, 1979*

A. General Functional Requirement

The RPT system shall automatically initiate the appropriate action whenever the conditions monitored by the system reach a preset level.

B. Independence and Integrity

The RPT system and components shall be independent and separate from components and/or systems that initiate anticipated transient(s) being analyzed and diverse from the normal scram system to minimize the probability of disabling the operation of the mitigating system. Diversity can be achieved by incorporating as many of the following methods as is practicable:

1. Use of RPT final trip relays from different manufacturers (required).

2. Use of energized versus de-energized trip status.

3. Use of AC versus DC power sources.

It shall be demonstrated that the function of the RPT system and components will not be disabled as a consequence of events being analyzed.

Diversity of the RPT pressure and level sensing devices (including relays used in such sensing devices) from similar or identical devices used on the RPS is not required, since failure of those devices on both the RPT and the RPS is not likely to cause an ATWS due to the presence of other diverse trips on the RPS (high flux, valve position, etc.).

---

*The NRC staff has reviewed the Monticello RPT design and the "Hatch" RPT design, and finds that they meet these criteria (provided the changes specified in the cover letter are made to the "Hatch" design). Plant specific reviews will be conducted only as necessary to ascertain that the plant design is the same as, or equivalent to, one of the approved designs.

C. Equipment Qualification

The RPT system equipment and components shall be tested to verify that the system will provide, on a continuing basis, its functional capability under conditions relevant to postulated ATWS events, including extremes of conditions (as applicable) relating to environment, which are expected to occur in the lifetime of a plant.

D. Periodic Surveillance and Preventative Maintenance Testing and Calibration

Periodic surveillance and preventative maintenance tests and calibration requirements shall be identified to provide continuing assurance that the RPT system, including sensors and actuated equipment, is capable of functioning as designed and that system accuracy and performance have not deteriorated with time and usage. These requirements shall be particularly directed toward the detection of those failures or degradation of accuracy and performance which would not otherwise be likely to be detected during the course of normal operations. Integrated system testing shall also be performed to verify overall system performance.

E. Quality Assurance

A quality assurance program in conformance with the requirements of 10 CFR 50 Appendix B shall be applied to the RPT system design and equipment.

F. Administrative Controls

Administrative controls shall be established to control the access to all set point adjustments, calibration and test points.

G. Information Readout

The RPT system shall be designed to provide the operator with accurate, complete and timely information regarding its status. For those functions, including operations, test or maintenance, and calibration, which require direct operator interaction, human engineering factors such as information displays (e.g., display formats, layout and controls) and functional controls (e.g., methods, location and identification) shall be included in the design.

H. Maintainability

The design shall include measures which enhance maintainability to reduce mean-time-to-repair and to assure the continued availability and reliability of the system for the life of the plant. The system design shall include features which facilitate the recognition, location, replacement, repair and/or adjustment of malfunctioning equipment and components or modules.

Appendix B

Acceptable RPT Designs

Monticello RPT Design

The Monticello design simultaneously trips both MG sets "A" and "B" generator field breakers upon receipt of either reactor high pressure or low-low water level control logic input signals. The logic to each breaker is two-out-of-two (pressure) or two-out-of-two (level) (2/2 or 2/2), i.e., contacts "A" and "C" or contacts "B" and "D" must close to trip the breaker. The Monticello design employs diversity, testability, separation and redundancy.

Modified BWR/4 or Hatch RPT Design

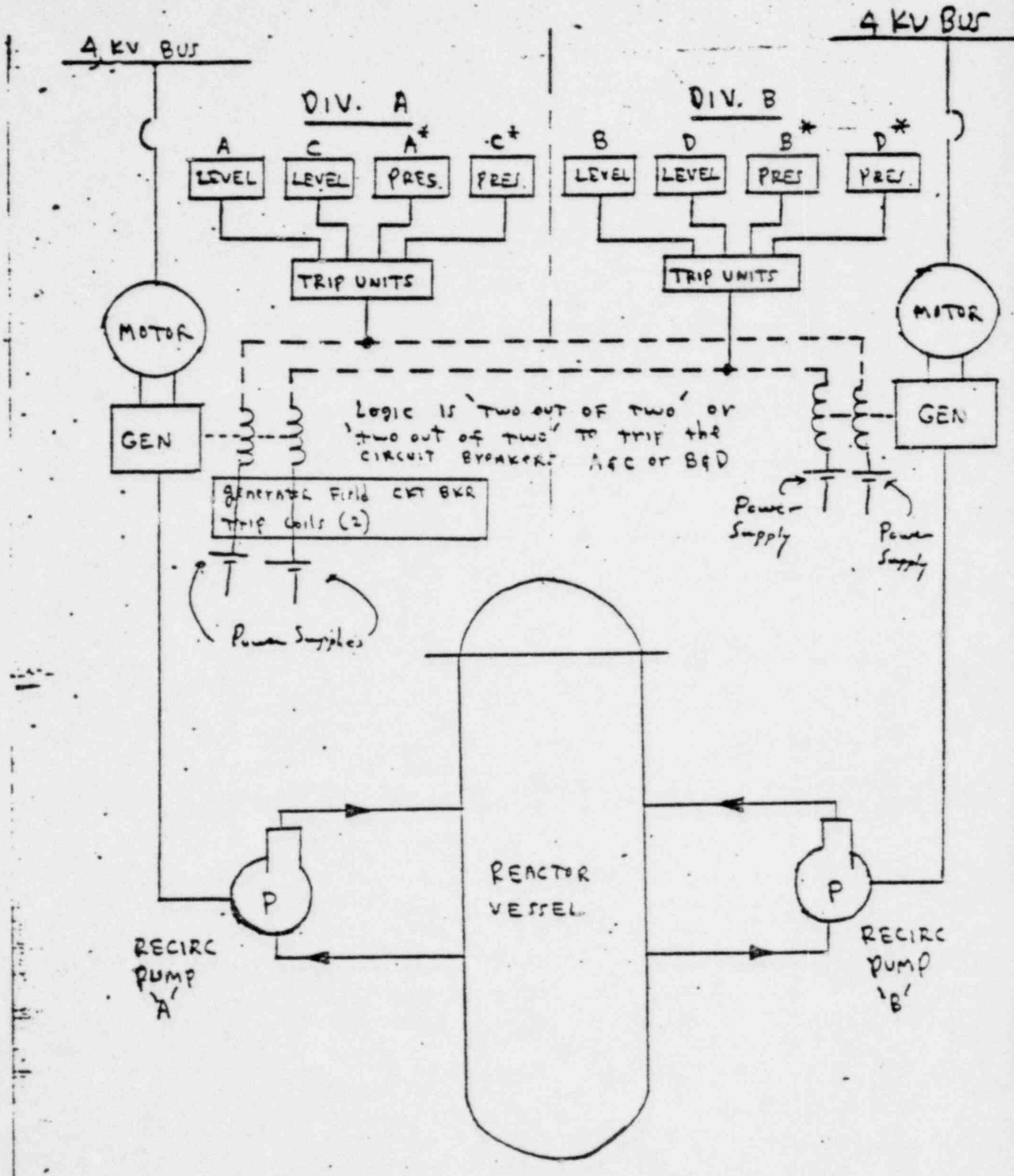
The modified "BWR/4" or "Hatch" design results in the independent (separate) trip of each of the two recirculation pumps upon receipt of either one reactor high pressure signal or one low-low water level signal. The logic to each MG set "A" and "B" generator field breaker is one-out-of-two (level) or one-out-of-two (pressure) (1/2 or 1/2). The modified "BWR/4" or "Hatch" design employs diversity, testability, separation, and redundancy.

The modification to the existing "Hatch" design which makes it acceptable is accomplished as follows:

1) Add a second trip coil to each recirculation loop's M-G set generator field breaker, as per the identical modification made to Monticello.

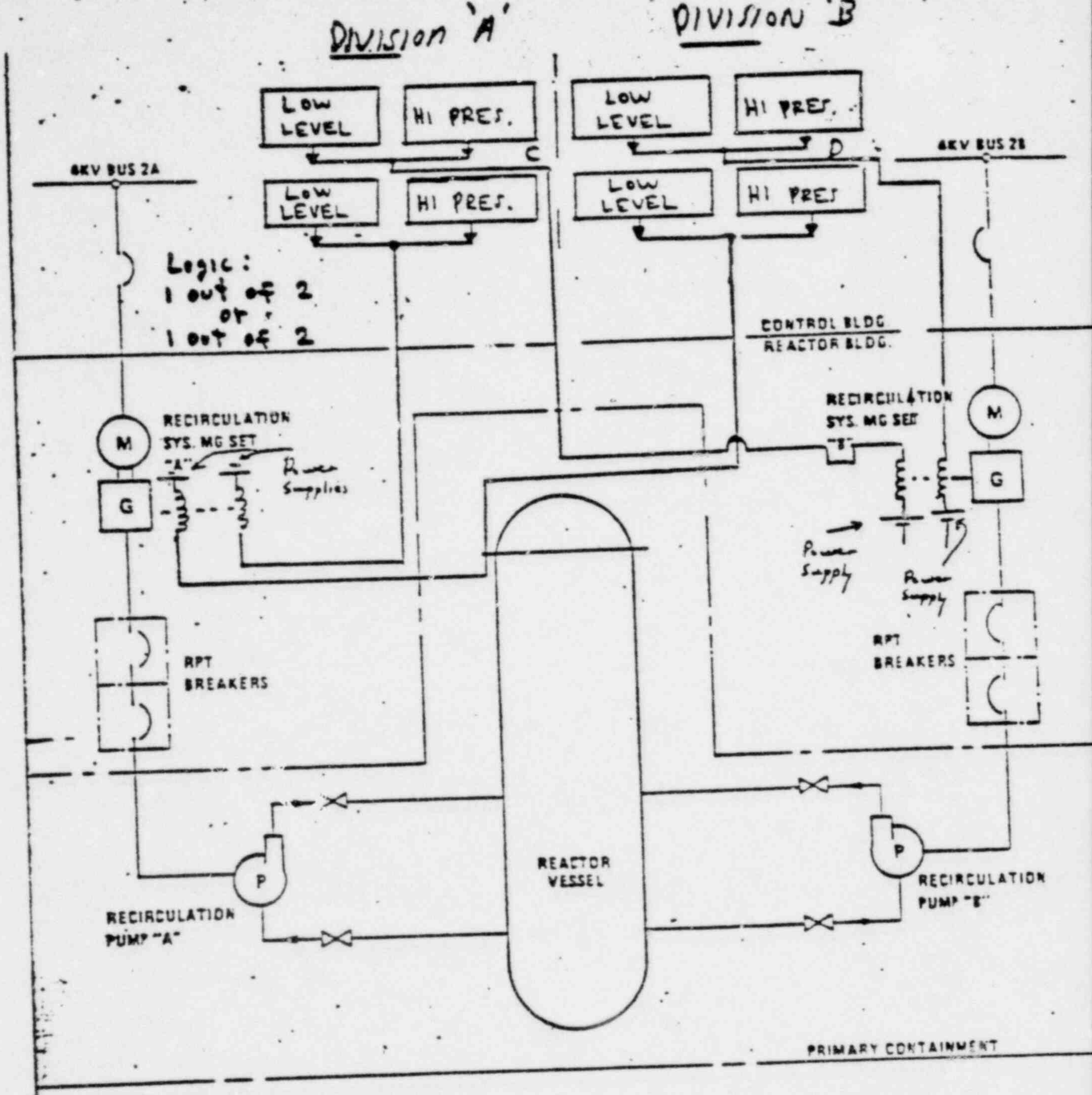
2) Connect one of the pressure sensors and one of the low level sensors in RPT train A to the old (existing) trip coil in the recirculation loop A M-G set generator field breaker. Connect one of the pressure sensors and one of the low level sensors in RPT train B to the new trip coil in the recirculation loop A M-G set generator field breaker.

3) Connect the other pressure sensor and the other low level sensor in RPT train A to the new trip coil in the recirculation loop B M-G set generator field breaker. Connect the other pressure sensor and the other low level sensor in RPT train B to the old (existing) trip coil in the recirculation loop B M-G set generator field breaker.



MONTICELLO ATWS RPT DESIGN

 ATTACHMENT III



"Modified Hatch" RPT

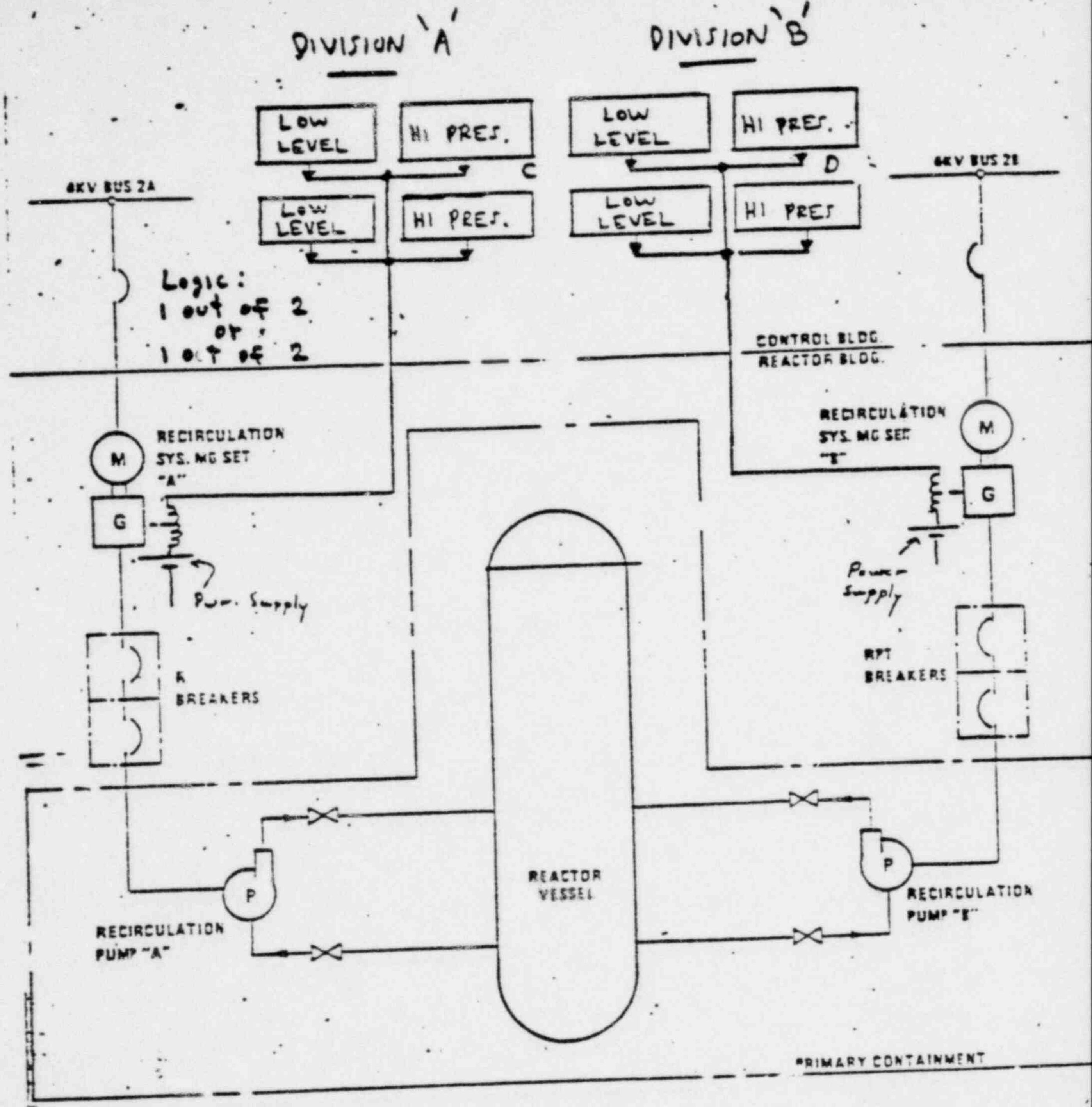
AMEND. 33 3/77

GEORGIA POWER COMPANY
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION SYSTEM WITH RECIRCULATION PUMP TRIP

FIGURE 5.5-15

 ATTACHMENT IV



AMEND. 33 3/77

GEORGIA POWER COMPANY
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
FINAL SAFETY ANALYSIS REPORT

RECIRCULATION SYSTEM WITH
RECIRCULATION PUMP TRIP

FIGURE 5.5-15

"Hatch" RPT

A-343

ATTACHMENT V

D. Tondi

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

SEPTEMBER 4 1979

MEMORANDUM FOR: D. Eisenhut, Acting Director, Division of Operating Reactors

THRU: WG W. Gammill, Acting Assistant Director for Operating Reactor Project, DOR

FROM: T. A. Ippolito, Chief, ORB #3, DOR

ATWS RECIRCULATION PUMP TRIPS FOR OPERATING REACTORS

On August 30, you asked me the status of the ATWS recirculation pump trip (RPT) generic task. I asked Vern Rooney, the DOR Lead Engineer for ATWS, to prepare the following status report.

By letters dated January 8, 1979 we requested all operating BWR licensees that did not have ATWS RPT to commit to installation of an acceptable RPT within two years. All licensee responses have now been reviewed and found acceptable (enclosure 1). Most licensees chose to install "Monticello" type RPT's, with all installation scheduled to be completed by or before the Fall 1981 refueling outages. All licensees have also informed us that they have developed emergency procedures to enable operators to recognize an ATWS and have trained operators to take action in the event of an ATWS. A summary of licensee schedules for RPT installation appears in Table 1.

Thirteen BWRs installed ATWS RPTs prior to the time that acceptable criteria were defined in our letter of January 8, 1979 (most of these were installed before operating licenses were issued). A listing of these BWRs appears in Table 2. Review for acceptability of ATWS RPT design for these plants is planned to be included in the plant-by-plant review for total ATWS acceptability which will take place following generic determination of the final NRC ATWS requirements.

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Memo, Lainas to Ippolito 8/20/79
2. Tables 1 and 2

See Table 2

A-344 ATTACHMENT VI

Table 1. Summary of BWR Licensees' schedules for proposed RPT system installation.

PLANT	RPT SYSTEM DESIGN TO BE INSTALLED: MONTICELLO	RPT SYSTEM DESIGN TO BE INSTALLED: HATCH	EMERGENCY PROCEDURES	OPERATOR TRAINING
Big Rock Point	Installation to begin Feb. 1981		Procedures completed and available for NRC review	Training to be completed during current refueling outage
Dresden 1	Installation during Jan. 1981 (modified)		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Dresden 2	Installation during Fall 1980 refueling outage		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Dresden 3	Installation during Spring 1980 refueling outage		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Lacrosse	Installation complete by January 1981; RPT type not identified		Interim procedures completed March 1979	Training completed March 1979
Millstone 1	Installation during Summer 1980; RPT type not identified		Interim procedures completed	Training completed
Nine Mile Point 1	Installation during Mar. 1981 refueling outage		Procedures to be completed June 1, 1979	Training to begin June 1979 (12 weeks)
Pilgrim 1	Completed in May 1980		Procedures completed	Training completed

Table 1. Summary of BWR Licensees' schedules for proposed RPT system installation. (Continued)

PLANT	RPT SYSTEM DESIGN TO BE INSTALLED		EMERGENCY PROCEDURES	OPERATOR TRAINING
	MONTICELLO	HATCH		
Quad Cities 1	Installation during Fall 1980 refueling outage		Interim procedures completed	Training completed
Quad Cities 2	Installation during Fall 1981 refueling outage		Interim procedures completed	Training completed
Vermont Yankee	Installation during Fall 1980 refueling outage		Interim procedures completed	Training completed

TABLE 2

BWRs with ATWS RPTs Installed Before 1/8/79

Browns Ferry 1
Browns Farry 2
Browns Ferry 3
Peach Bottom 2
Peach Bottom 3
Cooper
Duane Arnold
Hatch 1
Hatch 2
FitzPatrick
Brunswick 1
Brunswick 2
Oyster Creek

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

AUG 20 1979

MEMORANDUM FOR: T. Ippolito, Chief, Operating Reactors Branch #3
Division of Operating Reactors

FROM: G. Lainas, Chief, Plant Systems Branch
Division of Operating Reactors

SUBJECT: STATUS OF BWR LICENSEE COMMITMENTS TO INSTALL ATWS RECIRCULATION PUMP TRIPS (TAC 6342)

By letters dated January 8, 1979 to all BWR licensees that currently do not have ATWS recirculation pump trips (RPT's) installed, DOR requested installation of RPT's. At the request of V. Rooney (ORB#3), the Plant Systems Branch and our consultant, Lawrence Livermore Laboratory, evaluated the licensee responses to determine compliance with our January 8, 1979 letter. A summary of the BWR licensees' commitments to install recirculation pump trips is enclosed. All of the licensees have committed to the overall DOR objective of ensuring that an acceptable RPT system be installed at each BWR facility within two years of receipt of the January 1979 letter. We find this acceptable.

G. Lainas, Chief
Plant Systems Branch
Division of Operating Reactors

Enclosure:
As stated

cc: D. Eisenhut
B. Grimes
W. Gammill
R. Vollmer
D. Ziemann
~~V.~~ Rooney
V. Stello
S. Hanauer
[illegible]
[illegible]
[illegible]
[illegible]
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[illegible]
[illegible]

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FOREWORD

This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues Technical Assistance Program being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Operating Reactors, by the Lawrence Li ermore Lab ratory.

The NRC funded the work under an authorization titled "Electrical, Instrumentation and Control System Support", B&R 20 19 04 031, FIN A-0231.

The work was performed by EG&G, Energy Measurements Group, San Ramon Operations, for the Lawrence Livermore Laboratory under the U.S. Department of Energy contract number DE-AC08-76NV01183.

ABSTRACT

This report summarizes the commitments of boiling water reactor Licensees to a request by the U. S. Nuclear Regulatory Commission for the installation of recirculation pump trip systems in those nuclear power plants that do not currently have such systems. The Licensee's schedules for system implementation, emergency procedures development, and operator training are reviewed. This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues Support Program being conducted for the U. S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory.

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1. INTRODUCTION

By letters dated January 8, 1979[1] to all boiling water reactor (BWR) Licensees that currently do not have recirculation pump trips (RPT's) installed, the U. S. Nuclear Regulatory Commission (NRC) requested the installation of RPT's. The Licensees were informed that the Monticello RPT design described in Reference 2 has been accepted by the NRC staff as meeting the required criteria. The NRC staff has also accepted the Hatch RPT design provided that those changes specified in Appendix B of the January 8, 1979 letter are incorporated. The Licensees were asked to respond within 90 days with their schedules for implementation of an RPT system of either the Monticello or the modified-Hatch design. This schedule was to be consistent with the NRC staff's overall objective of ensuring that an acceptable RPT system be installed at each BWR facility within two years.

The Licensees were also requested to inform the NRC staff within 90 days that they had completed the following:

> "1. Developed emergency procedures to enable operators to recognize an ATWS [anticipated transient without scram] event, including consideration of scram indicators, rod position indicators, flux monitors, vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure, and radiation indicators.
>
> "2. Train[ed] operators to take actions in the event of an ATWS including consideration of manually tripping the recirculation pumps and scramming the reactor by using the manual scram buttons, changing individual rod scram switches to the scram position, stripping the feeder breakers on the reactor protection system power distribution buses, opening the scram discharge volume drain valve, prompt actuation of the standby liquid control system, and prompt placement of the RHR [reactor heat removal system] in the pool cooling mode to reduce the severity of the containment conditions."[1]

Licensees of the following BWR's were asked to install RPT systems and to respond accordingly to the NRC:

(1) Big Rock Point
(2) Dresden 1
(3) Dresden 2
(4) Dresden 3
(5) Lacrosse
(6) Millstone 1
(7) Nine Mile Point 1
(8) Pilgrim 1
(9) Quad Cities 1
(10) Quad Cities 2
(11) Vermont Yankee.

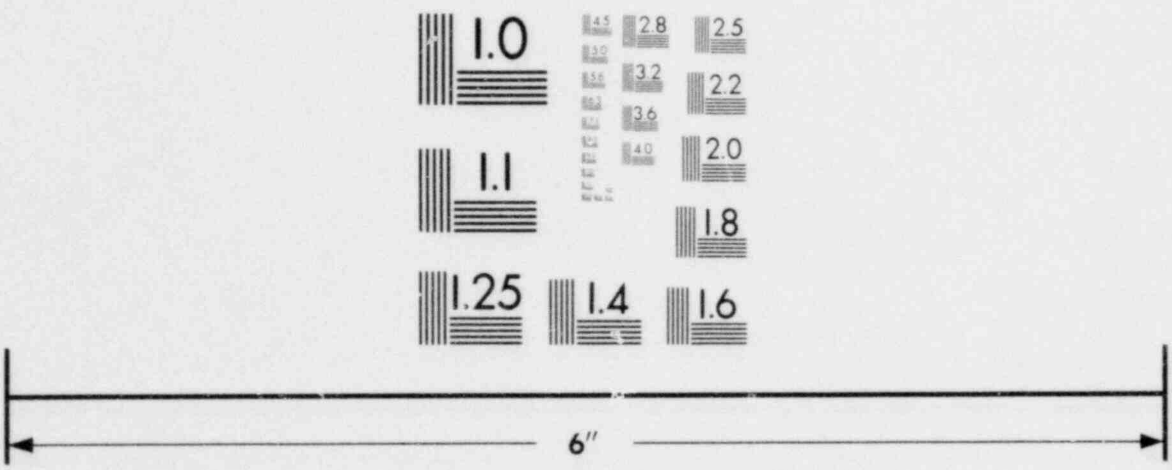
The results of these reviews are summarized in Table 1. A more detailed review of each Licensee's submittal is given in Section 2.

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART

Table 1. Summary of BWR Licensees' schedules for proposed RPT system installation.

PLANT	RPT SYSTEM DESIGN TO BE INSTALLED: MONTICELLO	RPT SYSTEM DESIGN TO BE INSTALLED: HATCH	EMERGENCY PROCEDURES	OPERATOR TRAINING
Big Rock Point	Installation to begin Feb. 1981		Procedures completed and available for NRC review	Training to be completed during current refueling outage
Dresden 1	Installation during Jan. 1981 (modified)		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Dresden 2	Installation during Fall 1980 refueling outage		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Dresden 3	Installation during Spring 1980 refueling outage		Interim procedures completed	Unit in outage; training to be completed before unit is returned to service
Lacrosse	Installation complete by January 1981; RPT type not identified		Interim procedures completed March 1979	Training completed March 1979
Millstone 1	Installation during Summer 1980; RPT type not identified		Interim procedures completed	Training completed
Nine Mile Point 1	Installation during Mar. 1981 refueling outage		Procedures to be completed June 1, 1979	Training to begin June 1979 (12 weeks)
Pilgrim 1	Completed [illegible] May 1980		Procedures completed	Training completed

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Table 1. Summary of BWR Licensees' schedules for proposed RPT system installation. (Continued)

PLANT	RPT SYSTEM DESIGN TO BE INSTALLED MONTICELLO	HATCH	EMERGENCY PROCEDURES	OPERATOR TRAINING
Quad Cities 1	Installation during Fall 1980 refueling outage		Interim procedures completed	Training completed
Quad Cities 2	Installation during Fall 1981 refueling outage		Interim procedures completed	Training completed
Vermont Yankee	Installation during Fall 1980 refueling outage		Interim procedures completed	Training completed

2. SUMMARY OF THE RESPONSES TO THE NRC REQUEST FOR THE BWR LICENSEE'S COMMITMENT TO INSTALL RPT SYSTEMS WITHIN TWO YEARS

2.1 BIG ROCK POINT

2.1.1 RPT Installation

In response to the NRC letter dated January 8, 1979, Consumers Power Company stated in its letter[3] dated April 9, 1979 that:

> "Consumers Power Company will install an RPT at Big Rock Point as requested. This installation will be completed during the 1981 refueling outage currently scheduled to begin in late February 1981. The RPT to be installed will be based on the Monticello design. Significant differences in plant design exist between Big Rock Point and Monticello and thus changes to the Monticello design will be required. A detailed description of the Big Rock Point RPT design will be submitted for NRC approval when the extent of these changes is finalized."

Also, Consumers Power Company has determined that auxiliary rod injection (ARI) can be installed at Big Rock Point in conjunction with the RPT at a small incremental cost. Accordingly, ARI will be included as part of the Big Rock Point RPT installation.

2.1.2 Emergency Procedures

Consumers Power Company stated in its letter[3] that:

> "The NRC letter of January 8, 1979 also required the development and implementation of interim emergency procedures for response to an ATWS event. Operator training in these procedures was required to be conducted within 90 days. An emergency procedure for responding to an ATWS event has been in effect at Big Rock Point for some time. This procedure has been reviewed and upgraded. Operator training in this upgraded procedure will be completed prior to plant start-up following the current refueling outage."

2.1.3 Operator Training

Consumers Power Company stated in its letter[3] that:

"The training provided to Big Rock Point operations personnel emphasizes fast recognition of an ATWS event. This facilitates prompt manual tripping of the recirculating pumps in the interim period before RPT is installed. The procedure incorporates operator actions intended to manually trip the reactor. In addition, the procedure provides the operator with specific indications which will be sufficient to determine that actuation of the liquid poison system is necessary and requires that this action be taken.

"The emergency procedures and training program are available for on-site review by NRC pesonnel. A summary of the contents of the program is provided below:

1. Definition of an ATWS event.
2. Operator training to recognize an ATWS event including:
   - Scram Indicators
   - Rod Position Indicators
   - Flux Monitors
   - Reactor Pressure Indicators
3. Operator training to take mitigating action including:
   - Recognition of an ATWS Event
   - Scramming the Reactor by Using Manual Scram Button
   - Manual Tripping of the Recirculating Pumps
   - Actuation of the Liquid Poison System.

"Both the training program and the interim procedures are designed so that they will continue to be effective after installation of RPT."

2.2 DRESDEN 1

2.2.1 RPT Installation

In response to the NRC letter of January 8, 1979, Commonwealth Edison Company stated in its letter[4] dated March 29, 1979 that the design of the RPT system to be installed on Dresden 1 will be equivalent to the Monticello design with the following differences:

> "1. The four D-1 recirculation pumps are single speed direct drive. The RPT signals will trip the appropriate 4 kV circuit breakers rather than the field breaker of a motor generator set.
>
> "2. The present design of the D-1 recirculation pump control system includes a pump trip initiated by low steam drum level. This feature will be retained as is. The equipment necessary to provide a reactor high pressure RPT signal will be added.
>
> "The modifications will be reviewed and implemented in conformance with the requirements of 10 CFR 50.59."

The Dresden 1 RPT system will be installed in January 1981.

2.2.2 Emergency Procedures

The Dresden 1 interim emergency procedures have been prepared and designed to provide the operator with sufficient information so that (s)he will know if and when (s)he must initiate the standby liquid control (SBLC) system and will not hesitate to proceed accordingly.

2.2.3 Operator Training

Commonwealth Edison Company stated in its letter[4] that the:

> ...training program, which has not been initiated, emphasizes prompt recognition of an ATWS event so that the decision to manually trip the recirculation pumps can be made as soon as possible and also defines a specific logical sequence of operator actions to insert the control rods....

"A summary of the content of the program is provided below:

1. Define the ATWS event to all licensed personnel.

2. Operator training to recognize an ATWS event includes:
   (a) scram indicators
   (b) rod position indicators
   (c) flux monitors
   (d) reactor/containment indicators

3. Operator training to take mitigating action without automatic RPT includes:
   (a) recognition of an ATWS event
   (b) manual scram of the reactor (manual scram button)
   (c) manual trip of the recirculation pumps
   (d) manual scram of the reactor by alternate means
   (e) actuation of the standby liquid control system
   (f) manual initiation of RHR system (pool cooling mode)

"Both the training program and the interim procedures are designed to result in consistent operator action before and after the RPT modification is incorporated."

2.3 DRESDEN 2

2.3.1 RPT Installation

In response to the NRC letter of January 8, 1979, Commonwealth Edison Company stated in its letter[4] dated March 29, 1979 that the design of the RPT.system to be installed on Dresden 2 will be equivalent to the Monticello design. The RPT system will be installed during the Fall 1980 refueling outage.

2.3.2 Emergency Procedures

The emergency procedures for Dresden 2 will be the same as for Dresden [illegible]. Refer to Section [illegible].

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2.3.3 Operator Training

The operator training for Dresden 2 will be the same as for Dresden 1 (refer to Section 2.2.3).

2.4 DRESDEN 3

2.4.1 RPT Installation

In response to the NRC letter of January 8, 1979, Commonwealth Edison Company stated in its letter[4] of March 29, 1979 that the design of the RPT system to be installed on Dresden 3 will be equivalent to the Monticello design. The RPT system will be installed during the Spring 1980 refueling outage.

2.4.2 Emergency Procedures

The emergency procedures for Dresden 3 will be the same as for Dresden 1 (refer to Section 2.2.2).

2.4.3 Operator Training

The operator training for Dresden 3 will be the same as for Dresden 1 (refer to Section 2.2.3).

2.5 LACROSSE

2.5.1 RPT Installation

In response to the NRC letter of January 8, 1979, Dairyland Power Cooperative stated in its letter[5] dated April 30, 1979 that:

> "RPT design effort is currently in progress. Our principal technical consultant, Nuclear Energy Services, has completed a preliminary design report which indicates that the RPT system will conform to an acceptable system as described in your letter. Installation of this system would be accomplished within two years from the date of your letter."

It is not obvious which of the RPT types (Monticello or Hatch) will be installed by the Licensee.

2.5.2 Emergency Procedures

Dairyland Power Cooperative stated in its letter[5] that:

"The necessary revisions to operating procedures were accomplished on March 2, 1979 by the addition of Section 3.12, Anticipated Transient Without Scram (ATWS) to the LACBWR Operating Manual."

2.5.3 Operator Training

Dairyland Power Cooperative stated in its letter[5] that:

"The necessary operator training was accomplished and completed during the period March 6-23, 1979."

2.6 MILLSTONE 1

2.6.1 RPT Installation

In response to the NRC letter of January 8, 1979, Northeast Nuclear Engineering Company stated in its letter[6] dated March 29, 1979 that:

"...Northeast Nuclear Energy Company (NNECO) will install and make operational an ATWS RPT of acceptable design during the refueling outage scheduled for the Summer of 1980."

It is not obvious which of the RPT types (Monticello or Hatch) will be installed by the Licensee.

2.6.2 Emergency Procedures

Northeast Nuclear Engineering Company stated in its letter[6] that the interim emergency procedure is designed to provide the operator with sufficient information so that (s)he knows if and when (s)he must initiate the SBLC and will not hesitate to proceed. These interim emergency procedures are available for on-site review by NRC staff members.

2.6.3 Operator Training

Northeast Nuclear Engineering Company stated in its letter[6] that:

> "...the training program emphasizes fast recognition of an ATWS event so that the decision to trip the pumps can be made as soon as possible, followed by a specific logical sequence of operator action to scram the rods.
>
> "...the training program along with records of completion are available for on-site review by NRC staff members. A summary of the contents of the program is provided below:
>
> 1. Defines the ATWS event to all licensed personnel.
>
> 2. Operator training necessary to recognize an ATWS event includes:
>   - scram indicators
>   - rod position indicators
>   - flux monitors
>   - vessel level and pressure indicators
>   - containment temperature and pressure
>   - noise level from suppression pool.
>
> 3. Operator training necessary to take mitigating action without automatic RPT includes:
>   - recognition of an ATWS event
>   - manual tripping of the recirculation pumps
>   - scramming the reactor by using manual scram buttons
>   - scramming the reactor by switching mode switch out of RUN
>   - changing individual rod scram switches to the scram position
>   - exhaust the scram air header
>   - prompt actuation of the standby liquid control system (SLC)
>   - insert [illegible] rods by normal rod motion [illegible]

- prompt initiation of the containment cooling subsystem to reduce severity of the containment conditions
- initiate iso-condenser.

"The training program places special emphasis on the identification of the ATWS event and the initiation of SLC because an operator decision must be made at a specific point in the process. In addition, both the training program and the interim procedures are designed such that when ATWS RPT is incorporated, the training program and interim procedures will still result in consistent operator actions."

2.7 NINE MILE POINT 1

2.7.1 RPT Installation

In response to the NRC letter of January 8, 1979, the Niagara Mohawk Power Corporation stated in its letter[7] dated April 6, 1979 that:

"A recirculation pump trip similar to the 'Monticello' design discussed in your letter will be installed during the next scheduled refueling outage planned for March 1981. After installation of this trip, procedures will be appropriately modified to reflect the revised system performances. Training will support such revised procedures."

2.7.2 Emergency Procedures

Niagara Mohawk Power Corporation stated in its letter[7] that:

"An operating procedure to respond to an Anticipated Transient Without Scram event is now under development and will be completed by June 1, 1979."

2.7.3 Operator Training

Niagara Mohawk Power Corporation stated in its letter[7] that:

"The five shifts of licensed operators will begin training immediately after June 1, 1979. This training will be completed within 12 weeks after initiation."

2.8 OYSTER CREEK

A response from Jersey Central Power and Light was not received because they were not sent a letter, having already committed to a RPT.

2.9 PILGRIM 1

2.9.1 RPT Installation

In response to the NRC letter of January 8, 1979, Boston Edison Company stated in its letter[8] dated April 10, 1979 that:

"...Boston Edison Company commits to install under the provisions of 10 CFR 50.59 and make operational an ATWS RPT of the Monticello design by May 1980. Further, under the provisions of 10 CRF 50.59, Boston Edison Company will at the same time install and place in service an ARI, i.e., an ATWS Rod Insertion System utilizing the existing spare trip output contacts from the ATWS RPT Logic Cabinets. This feature will provide the capability to diversely and independently remove air pressure from the CRD [control rod drive] units. This installation will be treated as a Class 1E modification."

2.9.2 Emergency Procedures

The development and implementation of interim emergency procedures has been completed.

2.9.3 Operator Training

All licensed personnel have completed the required operator training.

2.10 QUAD CITIES 1

2.10.1 RPT Installation

In resp[illegible] to the NRC letter of January 8, 1979, Commonwealth Edison Company stated in its letter[4] of March 29, 1979 that the design of the RPT system to be installed at Quad Cities 1 will be equivalent to the Monticello design. The RPT system will be installed during the Fall 1980 refueling outage.

2.10.2 Emergency Procedures

The emergency procedures for Quad Cities 1 will be the same as for Dresden 1 (refer to Section 2.2.2).

2.10.3 Operator Training

The operator training for Quad Cities 1 will be the same as for Dresden 1 (refer to Section 2.2.3).

2.11 QUAD CITIES 2

2.11.1 RPT Installation

In response to the NRC letter of January 8, 1979, Commonwealth Edison Company stated in its letter[4] of March 29, 1979 that the design of the RPT system to be installed at Quad Cities 2 will be equivalent to the Monticello design. The RPT system will be installed during the Fall 1981 refueling outage.

2.11.2 Emergency Procedures

The emergency procedures for Quad Cities 2 will be the same as for Dresden 1 (refer to Section 2.2.2).

2.11.3 Operator Training

The operator training for Quad Cities 2 will be the same as for Dresden 1 (refer to Section 2.2.3).

2.12 VERMONT YANKEE

2.12.1 RPT Installation

In response to the NRC letter of January 8, 1979, Vermont Yankee Nuclear Power Corporation stated in its letter[9] of April 4, 1979 that:

> "...Vermont Yankee feels confident that installation of an RPT similar to the Monticello RPT design can be completed during the 1980 Fall refueling outage."

2.12.2 Emergency Procedures

The interim emergency procedures for Vermont Yankee will be available for on-site review by NRC staff members by April 8, 1979.

2.12.3 Operator Training

Vermont Yankee records documenting operator training completion will be available for on-site review by NRC staff members by April 8, 1979.

REFERENCES

1. NRC letters (Denton) BWR Licensees (Boston Edison Company, Commonwealth Edison Company, Consumers Power Company, Dairyland Power Cooperative, Jersey Central Power and Light, Niagara Mohawk Power Corporation, Northeast Nuclear Engineering Company, Vermont Yankee Nuclear Power Corporation) dated January 8, 1979.

2. N. S. Sheng, Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant, General Electric Company, Boili Water Reactor Projects Department, San Jose, California, NEDO-25016 Class I (September 1976).

3. Consumers Power Company letter (Bixel) to NRC (Ziemann) dated April 9, 1979.

4. Commonwealth Edison Company letter (Reed) to NRC (Denton) dated March 29, 1979.

5. Dairyland Power Cooperative letter (Linder) to NRC (Denton) dated April 30, 1979.

6. Northeast Nuclear Engineering Company letter (Counsil) to NRC (Denton) dated Marach 29, 1979.

7. Niagara Mohawk Power Corporation letter (Dise) to NRC (Denton) dated April 6, 1979.

8. Boston Edison Company letter [illegible] to NRC (Denton) dated April [illegible]

9. [illegible] Nuclear Power Corporation letter [illegible] to NRC [illegible] dated April [illegible]

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

November 2, 1979

W. Kerr, Chairman
ATWS Subcommittee

NRC LETTER ON ATWS

Attached for your information is a letter from H. Denton to all operating licensees, and applicants for a license, that expresses concern with the pace of resolution of the ATWS issue. Specifically, Dr. Denton notes that some of the vendors responses to Dr. Mattson's February 15, 1979 early verification letter are incomplete. The letter states that the Staff will submit a proposed ATWS rule to the Commission in early 1980, regardless of whether or not Industry responses to the Mattson letter have been received.

Paul Boehnert
Paul Boehnert
Reactor Engineer

Attachment: as stated

cc: ACRS Members
ATWS Consultants:
C. Bennett
S. Ditto
E. Epler
J. Lee
W. Lipinski
S. Saunders

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ATWS0010034

PAB

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

RECEIVED

OCT 17 1979

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U.S. NUCLEAR REG. COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

DISTRIBUTED TO ACRS MEMBERS

All Power Reactor Licensees
All Applicants With Applications for a License

Gentlemen:

This past March, the NRC transmitted to you a copy of Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors" (ATWS) and a copy of an NRC letter that was sent this past February to each of the four nuclear reactor vendors. The letters to the vendors contained requests for information needed to perform generic analyses related to ATWS.

As we pointed out in our March letters, the generic analyses we requested were intended to confirm that the modifications proposed by the NRC staff for various classes of LWR designs would in fact accomplish the degree of ATWS prevention and mitigation described by the staff in its report. We also pointed out that we had chosen to work directly with the vendors in obtaining this information in an effort to conserve both NRC and industry resources. We requested that utilities cooperate with the vendors in performing the requested analyses.

Shortly after sending the letters to the vendors, the NRC Staff met with representatives of each of the NSSS vendors and many Utility representatives in Bethesda on March 1, 1979. The meeting was called to discuss the "early verification" approach in which we planned to use generic analyses as the basis for rulemaking. We hoped thereby to avoid costly and unnecessary repetitive analysis for individual plants. At the meeting, a tentative schedule was agreed to for generic analyses for each class of plants to be provided in three separate packages to be submitted May 1, September 1, and December 1, 1979.

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US NRC
ACRS
EXECUTIVE DIRECTOR
H-1016
WASHINGTON DC 20555

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ATWS-1

Immediately following the March 1 meeting, the NRC staff met separately with each of the NSSS vendors and agreements were made as to the minimum information to be supplied in the May 1 package. Also, as noted above, copies of the ATWS staff report and the generic analyses questions were transmitted to the Utilities.

On March 28, 1979 the Three Mile Island accident occurred. Because of the heavy expenditure of NRC resources required for Three Mile Island related activities, essentially no staff effort was applied to the ATWS issue for three months or so following the accident. There was also a substantial reduction in effort on the part of the PWR industry during that period, and some reduction for BWRs.

In June, 1979, the NRC Office of Nuclear Reactor Regulation was temporarily reorganized. Within this interim organization a group was assgned under the direction of S. Hanauer to work on the 19 Unresolved Safety Issues as designated by the Commission and reported to Congress this past January in NUREG-0510. ATWS is one of these 19 issues.

A preliminary NRR Staff review suggested that, for PWRs, the Three Mile Island accident raised new questions with regard to the appropriateness and adequacy of the resolution of ATWS as proposed by the Staff in Volume 3 of NUREG-0460. For BWRs, the staff has concluded that the technical impact of Three Mile Island was minimal and that the completion and review of the generic analyses for BWRs as specified in March should proceed as expeditiously as possible.

A meeting was held in Bethesda on July 25, 1979 to discuss, with representatives of PWR utilities and designers, considerations arising from the Three Mile Island accident that might be relevant to ATWS. For your information, a copy of the staff minutes of that meeting is attached as Enclosure 1. As can be seen from the minutes, at the meeting the staff:

a) Reiterated that ATWS is still believed by the staff to be a serious safety concern and that future protection should be provided. We stated that we are unwilling to wait another year to make progress on ATWS.

b) Expressed some general and specific technical concerns raised by the Three Mile Island accident with regard to the ATWS resolution proposed in Volume 3 of NUREG-0460.

c) Asked the industry to provide in writing, within 30 days of the meeting date, its preliminary assessment of the Three Mile Island impact on ATWS, the scope of effort now foreseen to resolve TMI issues, and a realistic schedule for providing the needed ATWS information. This would include both the March request and the TMI-related analyses.

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OCT 1 7 1979

Subsequent to the July 25 meeting, we have met with representatives of the four NSSS vendors and of some Utility/Owners. We have met with GE to discuss the scope of the remaining generic analysis information to be supplied for BWR 4/5/6's. We have also met with representatives of the GE BWR/3 Owners, B&W, B&W ATWS Owners Group, W, W ATWS Owners Group, and CE. At all these meetings, we considered further the required information and the schedule for its submittal.

We have now received letters (see the list in Enclosure 2, attached) from the various groups describing the information to be furnished and projected schedules. On the basis of our review of these letters and meetings with the industry representatives, we perceive that the projected responses in several cases would not address several questions in our February 15 letter. In particular, several items are lacking that we will need to justify acceptance of the hardware approaches of NUREG 0460 Vol 3 rather than using the design basis accident approach.

I am determined to submit a proposed ATWS rule to the Commission for both PWRs and BWRs early in 1980. The type and content of the rule we will propose will depend critically upon the types and content of the information available to the staff. This will, of course, include whatever responses are actually provided by the industry in response to the questions attached to the February 15 staff letter, the March meetings, and the Three Mile Island related concerns as discussed in the July 25 and subsequent meetings.

I still believe that it is possible for the early verification generic analysis program to provide an acceptable resolution of the ATWS issue and that this is the way to achieve resolution with the least possible expenditure of NRC and industry resources. However, I want to reiterate that the success of this approach depends on whether or not all of the information necessary for the staff to confirm that its proposed ATWS modifications provide an acceptable level of protection, for all plants, is provided by the industry.

I strongly encourage you to join or form Utility/Owners Groups, if you have not already done so, and provide the resources necessary to supply the needed technical information pertaining to your plants, either operating or under construction. It would further reduce the impact on the industry as well as the staff resources if the ATWS effort coordination and the review role is performed by one industry group.

If you have additional questions on the generic analysis early verification program discussed in this letter, please contact Mr. Ashok Thadani, (301-492-7341).

Sincerely,

H. R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC-Industry ATWS Meeting Summary dtd 7/25/79
2. List of letters from Industry on Content of Report Submittals

A-374

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

JUL 27 1979

Task Action Plan A-9

MEMORANDUM FOR: S. H. Hanauer

FROM: A. Thadani

SUBJECT: NRC-INDUSTRY ATWS MEETING SUMMARY

The staff met with the PWR vendors, the Atomic Industrial Forum (AIF) and several utility representatives to discuss the impact of TMI-2 events on the ATWS resolution plan described in Volume 3 of NUREG-0460.

The staff made the following initial remarks:

1) ATWS is still a safety concern and protection from these events must be provided. Although plants need not be shutdown immediately because of relatively low likelihood of a severe ATWS in a PWR in the next couple of years, ATWS resolution with suitable speed is necessary to permit an implementation plan which would assure an acceptably low risk from ATWS over the life of nuclear plants.

2) The staff would like to receive industry views on the impact of TMI-2 on ATWS and how to proceed from now on to resolve ATWS. The staff noted that they intend to propose an ATWS solution to the Commission preferably with but if necessary without the industry input.

3) In view of TMI-2 accident, the staff expressed the following general concerns with the Vol. 3 proposed resolution and asked for industry comments.

   a) What assurance do we have that the excessive calculated pressures for some designs modified per Alternative #3 would not result in loss of integrity of reactor coolant pressure boundary. (Note - Some designs may experience peak pressures - 4000 psi).

   b) Would increasing the number of safety valves as per Alternative #4 result in insufficient overall risk reduction? Would the primary system integrity be maintained? Would it be better to have larger capacity valves?

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1) Analyses indicate the sensitivity of peak pressure to AFWS design and actuation time for some plants.

Why should auxiliary feedwater actuation not be delayed beyond technical specification values? What bases are available to assume AFWS actuation earlier than the technical specification value? How do the analyses take into consideration the limits on AFWS injection rate due to water hammer considerations? How is the impact of flow restrictors on some AFWS designs considered in the ATWS analyses? How are the significant plant specific features of AFWS treated in the analyses?

2) As in question 1 above how are the differences in ECCS designs evaluated? For example, for some ATWS events, the pressure and the pressurizer level remain high enough such that either the HPSI cannot be actuated (because of shut off head considerations) or the operator may fail to actuate HPSI because of insufficient available information.

3) Would single failure cause all PORVs to fail to open? If so, then analyses must be based on all PORVs failing to open. Further, several plants are operating today with PORVs isolated. For these plants credit cannot be taken for relieving capability of these valves.

4) What assurance do we have that the ATWS events with a stuck open safety valve have been correctly analyzed? What is the potential for core uncovering under this scenario? What is the importance of ECCS actuation, reactor coolant pumps operation, and the pressurizer safety/relief valve discharge model on the potential for uncovering of the core? Further, why should more valves not be assumed to stick open following discharge of subcooled water.

5) For long term shutdown, discuss the following:

   a) ailable equipment, instrumentation and their qualification. (Must consider the effect of water discharged to the containment via ruptured quench tank).

   b) impact of loss of offsite power

   c) continued operation of reactor coolant pumps. Also consider tripping of reactor coolant pumps.

   d) Describe natural circulation, including effects of non-condensables. Is reflux boiling mode of operation anticipated? If so, justify.

B&W

1) Basically agrees with the staff concerns. Industry has longer list of items that could impact ATWS.

2) Stress analyses should be completed.

3) Likelihood of additional failures beyond ATWS should be considered.

4) Prevention is better than mitigation.

B&W Owners Group

1) ATWS is not a safety problem.

2) Even if ATWS occurs, no significant risk to public health and safety.

3) TMI-2 suggests a desirability for realistic analyses. TMI-2 suggests a need to assure that analyses bound the facilities.

4) Wait until "Lessons Learned" and "Bulletins and Orders" issues are resolved before pushing ahead with ATWS.

After the above industry comments, the staff made the following concluding remarks.

1) We don't intend to go too fast on ATWS.

2) If Early Verification is to be pursued then there is a need to assure that earlier ATWS analyses are correct and review the industry TMI-2 related list. In this regard the industry was invited to meet with the staff to discuss the technical issues which impact ATWS. The staff asked the industry to provide their assessment of TMI-2 impact on ATWS, the scope of effort to resolve these issues, and the schedule for performing this effort within 30 days.

3) We cannot wait another year to make progress in ATWS.

The list of attendees is in the enclosure.

A. Thadani

Enclosure:
As stated

cc: See next page

ENCLOSURE

ATWS Meeting with Vendors & AIF

July 25, 1979

Ashok Thadani	NRC/DSS
Arthur McBride	B&W
Alan Hosler	WPPSS
Samir K. Sarkar	FP&L
Alan E. Ladieu	YAEC
Fred T. Stetson	AIF
Richard G. Rateick	DECO
Andrew J. Rushnok	OEC
M. Srinivasan	NRC/DSS
F. Akstulewicz	NRC/DSE
G. Sorensen	WPPSS/AIF
T. Speis	NRC/DSS
F. C. Cherny	NRC/DSS
J. A. Norberg	NRC/OSD
Stuart Thickman	TVA - EN DES
Karl O. Layer	BBR
J. Ted Enos	AP&L
Ted Myers	TECo
Robert Dieterick	SMUD
Michael J. Salerno	CPCo
S. Hardy Duerson	B&W
Bob Steither	W
Gary Augustine	W
P. M Abraham	Duke Power
Mark Wisenburg	USTVA - Office of Power
Michael Tokar	NRC/DSS
Paul Boehnert	NRC/ACRS
David Bessette	NRC/ACRS
Steven Traisman	Pacific Gas & Electric
Sam Miranda	W
Pat Loftus	W
Fred Mosby	Wyle Laboratory
Roger Newton	Wisconsin Electric Power
Craig Grochmal	Stone & Webster
Charles A. Daverid	Long Island Lighting Co.
Robert L. Stright	SNUPPS
Joseph M. Weiss	GE
Joseph A. Gonyeau	Northern States Power

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Letter from R. H. Bucholz (GE) to S. Hanauer, "ATWS Generic Analyses - Content of December 1979 Submittal", dated September 5, 1979.

Letter from J. H. Taylor (B&W) to S. Hanauer, "B&W Commitments for ATWS", dated September 13, 1979.

Letter A. E. Scherer (CE) to S. Hanauer, "NRC Request for Generic ATWS Information", dated August 31, 1979.

Letter L. O. DelGeorge (BWR 3 Owners representative) to S. Hanauer, "ATWS BWR/3 Plants and Vermont Yankee - Generic Analysis Supplement", dated August 28, 1979.

Letter T. M. Anderson (W) to S. Hanauer, "ATWS", dated August 24, 1979.

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

OCT 29 1979

MEMORANDUM FOR: Robert E. Jackson, Chief
Geosciences Branch, DSS

THRU: Leon Reiter, Section Leader
Geology and Seismology Section
Geosciences Branch, DSS

FROM: Harold E. Lefevre, Geologist
Geology and Seismology Section
Geosciences Branch, DSS

SUBJECT: GEOLOGIC FEATURES IN THE SERVICE SPILLWAY AREA OF THE MAIN DAM - WOLF CREEK GENERATING STATION UNIT I (STN 50-482)

On Tuesday afternoon, September 25, 1979, C. R. Oberg (NRC I&E Region IV) notified H. Lefevre (NRC staff Geologist, Wolf Creek) that faulting had been identified in the excavation for the Service Spillway of the Main Dam for the Wolf Creek Generating Station. The Service Spillway (a non-safety related structure) is located approximately 3 miles south of the main plant area. Mr. Oberg had been informed by the applicant's (Kansas Gas and Electric Company's (KG&E) geologic consultants, Dames and Moore, that the faulting was similar in age (at least 280 million years old) and mode of deformation (during or shortly after deposition of the original sediments) to that previous observed and reported upon by NRC staff geologists in 1977 and 1978. The earlier faulting had been observed in the Power Block area and in the Outlet Tunnel area of the Main Dam. Mr. Oberg was further informed by Dames and Moore personnel that Mr. Frank W. Wilson of the Kansas Geological Survey had been contacted by Dames and Moore and would visit the site the following day, September 26, 1979. H. Lefevre contacted Mr. Wilson later in the afternoon of September 25, and confirmed that he would be making a site visit on September 26.

It should be noted that Mr. Wilson is quite familar with the site and regional geology and has visited the Wolf Creek facility on several occasions both on State matters as well as with NRC geologists. Mr. Wilson is considered by the NRC staff to be well qualified to comment upon geologic conditions observed at the site.

OCT 29 1979

Discussion with F. Wi[illegible]on, Kansas Geological Survey, Sept. 27, 1979

At approximately 1:00 a.m., Thursday, September 27, 1979, H. Lefevre (NRC staff) contacted F[illegible]ank Wilson regarding his observations of the previous day at the Service Spillway area of the Wolf Creek site. Mr. Wilson indicated that he had examined a shale exposure containing a thin (8") coal bed where two types of features were visible - (1) shear zones with minor displacements (about 2 inches) and (2) zones where shale appeared to have been "injected" through the coal bed. Mr. Wilson stated that the features are localized, die out within a few inches of their ends, and are overlain and underlain by undisturbed strata. According to Mr. Wilson these features, like those observed in the power block area in 1977 (NRC-Kansas Geological Survey site visit) and in the Main Dam area in 1978 (observed both by the NRC and F. Wilson) are penecontemporaneous (formed during the Pennsylvanian time more than 280 million years ago) and were not caused by tectonic activity. Mr. Wilson indicated that he would submit the report of his observations to the NRC. This September 27, 1979 report has been received and is appended to this memorandum.

Mr. C. Oberg of NRC Region IV was informed on the same day at 11:30 a.m. regarding F. Wilson's observations and conclusions.

Discussions with Applicant's Geologic Consultants

F. Wilson's September 27, 1979 letter mentions a "pop-up" relief structure in the non-Category I Service Spillway excavation. According to Mr. Wilson this feature has an amplitude and width of 2-3 inches, an estimated length of 15-20 feet and a strike of about N50°W. Mr. Wilson is uncertain of the origin of the "pop-up" but mentions three possible causes: (1) air slaking of the uncovered shale, (2) local stress, or (3) regional stress. Subsequent staff conversations with Dames and Moore (geologic consultants to the applicant) on October 12, 17, 18 and 19 revealed the following:

1. The "pop-up" is 12 feet long and is confined to the central portion of the Service Spillway. The amplitude and width are as described by F. Wilson. No further movement was detected following Mr. Wilson's observations.

2. The "pop-up" occurs along a portion of the 60 ft. long 0.1" wide sub-vertical sandstone dike within the Ireland Member of the Pennsylvanian Lawrence Formation.

3. The "pop-up" occurred sometime between opening of the excavation on September 21 and September 26, the time of F. Wilson's site visit.

4. The Ireland Member projected northward underlies the power block at a depth of at least 140 ft.

5. Undisturbed sandy shale beds were traced across the sand dike where exposed in the excavation walls to the northwest and southeast of the "pop-up" with no disruption or offset.

6. Neither "pop-ups" nor sand dikes have been observed elsewhere within the excavations at the Wolf Creek site.

7. Vertical holes drilled at various locations throughout the site area (Power Block, the ESWS Pump House, and in the Main Dam area) for the purpose of pre-splitting of an excavation face prior to blasting remained undeformed after opening of the excavation, thus indicating an absence of observable stress affecting the bedrock.

Conclusions

The shear and injection features exposed in the Service Spillway area of the Main Dam, some 3 miles south of the power block area, are not capable within the maning of Appendix A to 10 CFR Part 100 and consequently pose no hazard to the Wolf Creek facility. The Service Spillway features, like those observed in 1977 and 1978 by NRC geologists, are clearly old (greater than 280 million years) and are confined to small zones within the Pennsylvanian age Lawrence Formation. The features are clearly localized since neither overlying nor underlying strata have been disturbed.

The small pop-up observed in the floor of the Service Spillway excavation occurred within a few days of opening of the excavation. No similar features have been observed in other excavations within the Wolf Creek site. No wall dislocations resulting from excessive stress have been detected at the site. Based upon the localization of the "pop-up" at a non-safety-related location about 3 miles from the Power Block area, confinement to a geologic

OCT 29 1979

unit stratigraphically at least 140 feet below the Power Block area, and an absence of similar structure within other portions of the site area, the NRC staff concludes that the mechanism causing the "pop up" poses no hazard to the safety-related structures at the Wolf Creek Generating Station. The NRC staff concludes that the feature is most likely the result of stress relief (rebound) resulting from excavation within the Ireland Member of the Pennsylvanian Lawrence Formation combined with the discontinuity caused by the thin sand dike within an otherwise essentially homogeneous shale medium.

Harold E. Lefevre

Harold E. Lefevre, Geologist
Geology and Seismology Section
Geosciences Branch
Division of Systems Safety

Attachment:
As stated

cc: w/attachment
D. Eisenhut
W. Gammill
J. Knight
O. Parr
E. Licitra
E. O'Donnell
L. Reiter
H. Lefevre
J. Greeves
F. Wilson, Kansas Geol. Survey
C. Oberg, Region IV
S. Lewis
R. G. Ryan
PDR
Local PDR
ACRS (17)

KANSAS GEOLOGICAL SURVEY
Environmental Geology Section

1930 Avenue "A", Campus West
The University of Kansas
Lawrence, Kansas 66044
913-864-4991

September 27, 1979

Mr. Harold LeFevre
U.S. Nuclear Regulatory Commission
Geosciences Branch
Div. of Systems Safety
Washington, D.C. 20555

Dear Mr. LeFevre:

On September 26, I visited the excavation site of the service spillway at the main dam of the Wolf Creek Nuclear Power Station near Burlington, Kansas with engineers of the Water Resources Division of the State Board of Agriculture. Their principal responsibility is to inspect various phases of dam construction to assure that it is being carried out according to the plans and specifications of the permit issued by their department. I advise them on the engineering geology aspects of the site

Dave Fenster and Charles Bandoian of Dames and Moore Engineering Company - the geotechnical consulting firm for the project - had also requested that I view some minor shear zones and other structures which had been uncovered in the walls of the excavation.

The formation involved is called the Ireland Member of the Lawrence Formation by the project geologists but which is an unnamed shale above the Ireland sandstone according to KGS nomenclature. It is a gray, laminated, silty to sandy shale containing an impure coal about 8 inches thick. About 15 to 16 minor structures were exposed in an interval from the base of the coal to about 10 feet into the shale above it. These were of two types: 45° shear zones with minor displacement, and vertical discontinuities in the coal bed a few inches wide where either overlying or underlying shale appeared to have been "injected" through the coal. Only one of these discontinuities was associated with a shear zone. Neither the shear zones nor the discontinuities appeared to have extended across the width of the excavation which is perhaps 50 feet. It was interesting to note that at one place a 45° shear zone had an apparent dip to the north (approx., my orientation may have been off) and another about 10 feet away had an apparent dip to the south. The two nearly intersected at the top.

In all instances, both the shear zones and the coal discontinuities died out within a few inches of their ends and strata above and below are undisturbed, demonstrating that they were developed shortly after deposition and are not now active.

My tentative opinion and the opinions of the site geologists are that the structures are related to post-burial compaction of the sediments while they were in a semi-indurated state and that they are not caused by tectonic activity.

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This observation, however, should not be taken to mean that tectonic features may not eventually be uncovered at the site. I noted, for example, a miniature "pop-up" relief structure in the shale on the bottom of the excavation which had occurred very recently. It had an amplitude and width of 2-3 inches and an estimated length of 15-20 feet. The orientation (abbreviated to north half of compass) was N50°W. It is not certain whether this structure is related to local stress, air slaking of the uncovered shale, or to actual regional ambient stress but I think the latter should at least be considered.

Sincerely,

Frank W. Wilson

Frank W. Wilson, Chief
Environmental Geology Section

FWW:elp

cc: Ray Seiple
Dames & Moore
Box 585
New Strawn, KS 66839

JOHN C. MAXWELL
GEOLOGIST
122 WESTERN HILLS DRIVE
AUSTIN, TEXAS 78731
512-454-1921

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, 1979, 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 - 0666, 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

Mr. Ward's contention 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. 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.

RM

Rondout Associates, Incorporated

P.O. Box 224, Stone Ridge, New York 12484

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October 29, 1979

U.S. NUCLEAR REG. COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

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. The size of the April 24, 1867, earthquake. 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, 1877, 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. I. 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 to change the earthquake intensity on the basis of that isolated report alone is not justified.

Conclusion: The earthquake should still be considered as Intensity VII.

2. The location of the April 24, 1867, event. 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.

Conclusion:

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. Microearthquakes associated with the Humboldt fault and "the development of a seismic gap"

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 in-

terpreted 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.

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. Pomeroy

Appendix I

Documents Utilized in this Study

BIBLIOGRAPHY

1. Earthquake History of the Un[illegible] States, Publ. 41-1 Revised Edition (Through 1970) U.S. Dept. of Comm[illegible]ce, NOAA, EDS, Boulder, Colorado, 1973.

2. Docekal, J. Earthquakes o[illegible] the Stable Interior, with Emphasis on the Midcontinent, Unpublis[illegible] [illegible]nesis, 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, 1967

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.

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Locality	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)

Locality	Assigned MM Intensity	Earthquake Effects
12. Lawrence, KS (Douglas Co.)	VI	Three shocks felt over a period of 30 seconds (17)
(a) 2:57 p.m. or 2:45 p.m.		Earth trembled & vibrated Doors & windows violently shaken (8,43)
(b) 2 or 3 shocks		*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 (Leavenworth Co.)	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)
(a) 2:30 p.m. (b) 3 shocks felt, 30 sec. duration (c) W → E		*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)

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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 "over-topping 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)

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Locality	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 Republican 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)

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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)

Questionable 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.

ISOSEISMAL MAP OF THE APRIL 24, 1867 EARTHQUAKE IN KANSAS

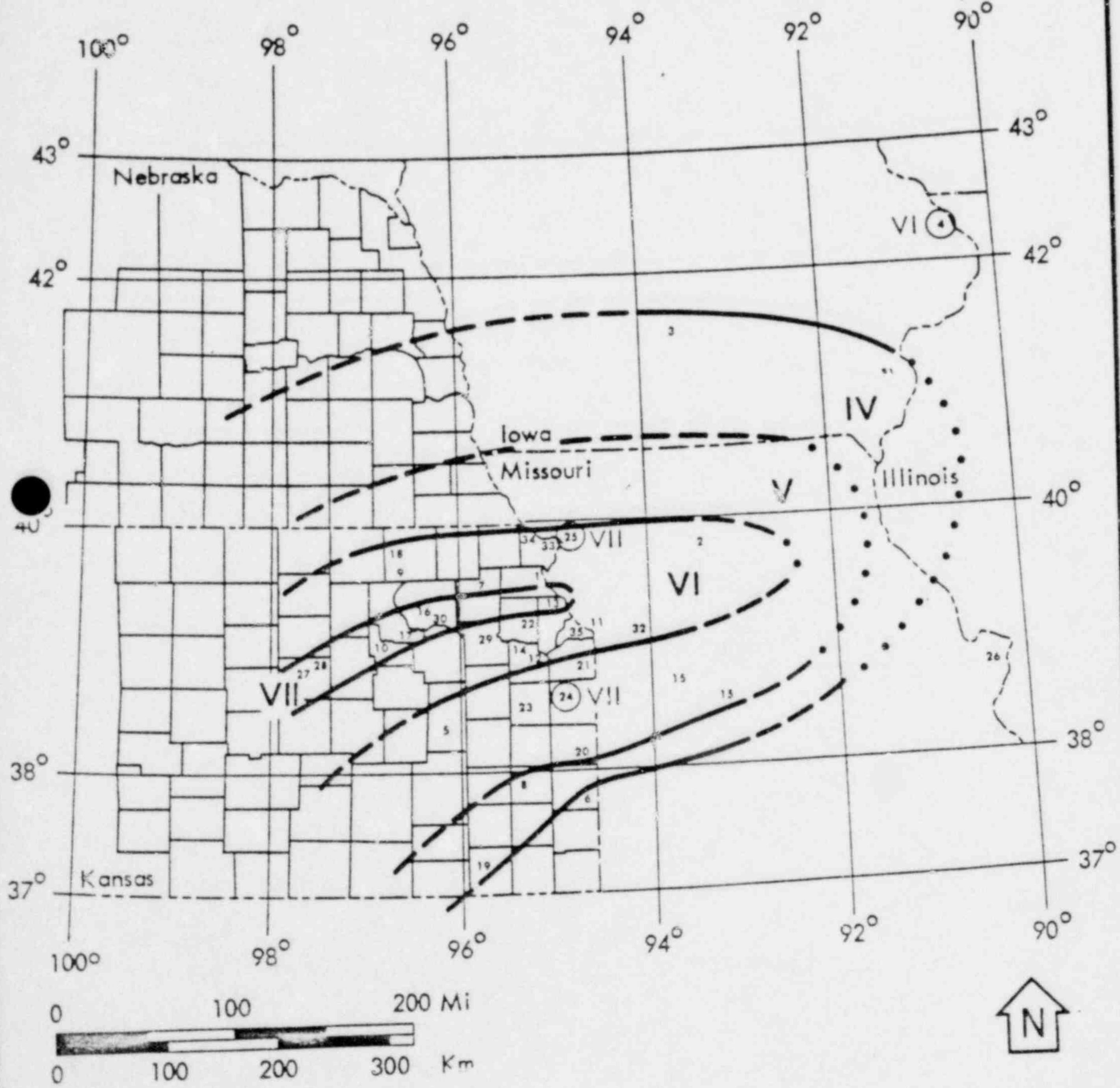

Fig. 3

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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.

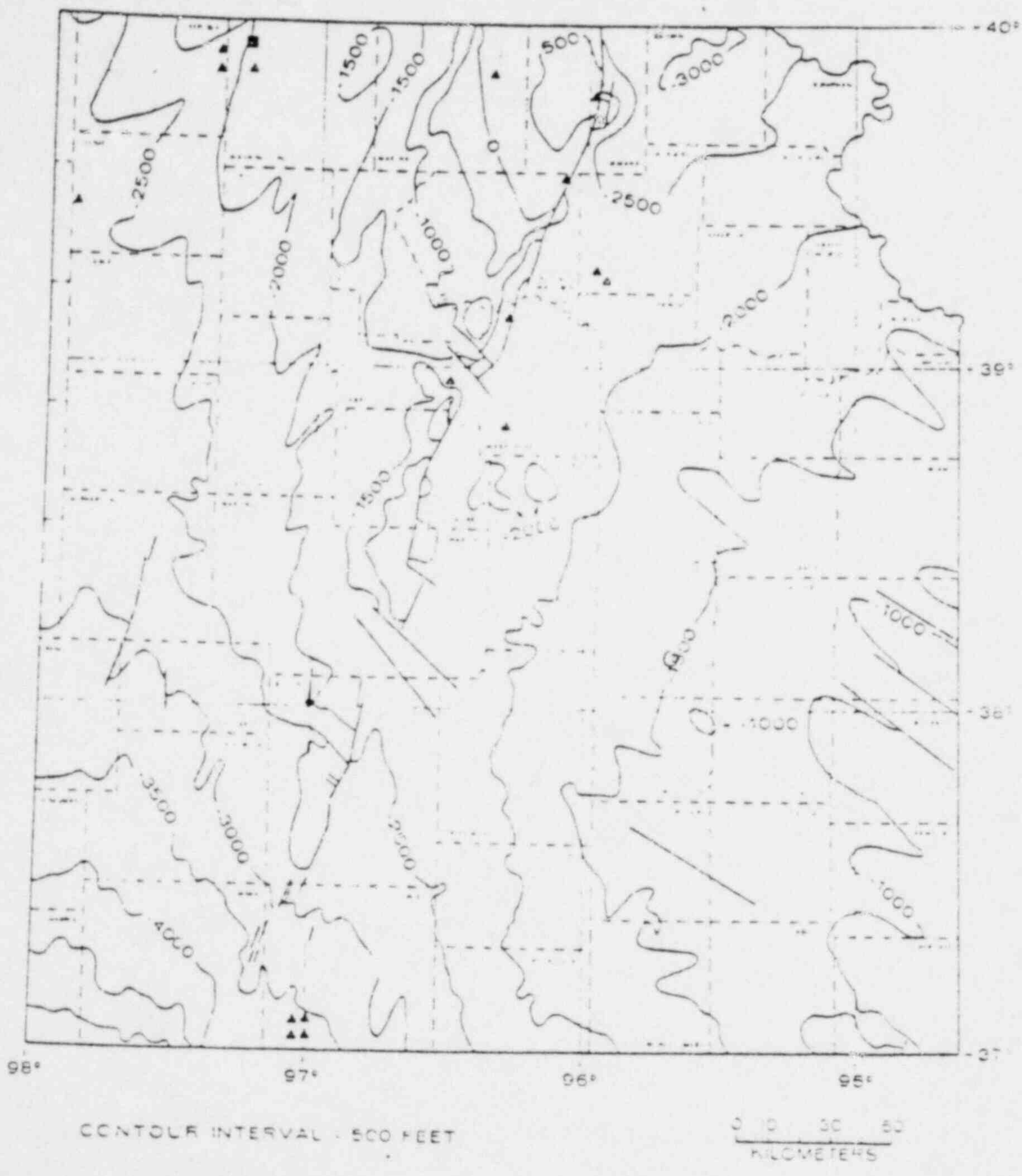

FIGURE 1. GENERALIZED CONTOUR MAP OF THE TOP OF PRECAMBRIAN BASEMENT ROCKS IN EASTERN KANSAS.

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APPENDIX XXVII
RECOMMENDED ACRS ACTION CONCERNING
GENERIC ITEMS

Recommended ACRS Action Concerning Generic Items Agreed at 233rd ACRS Meeting

Resolved Items	Follow-up by
1. NPSH for ECCS Pumps — Reactor Operations SC. This is covered by Reg. Guide 1.1. The Reactor Operations Subcommittee could review this with the Division of Operating Reactors to determine whether all plants are in compliance. Potential for vortex problems should be considered.	HE/RKM MTG. Dec 3
2. Emergency Power — Joint Power and Electrical Systems and Reactor Operations SCs Reg. Guide 1.6, 1.9, and 1.32 in conjunction with portions of IEEE-308 (1971) covers this matter. However, the question concerning loss of DC power or combined loss-of-offsite- and -onsite-AC power are presently of concern from a risk standpoint. The Power and Electrical Systems Subcommittee and the Reactor Operations Subcommittee should jointly review the status of emergency power requirements. The question of grandfathering older plants should also be considered regarding emergency power.	WK/GRO has lead Mtg Dec. 13
3. Hydrogen Control After Loss-of-Cooling Accident -- TMI-2 Implications SC. The present hydrogen control requirements are based primarily on the concern for hydrogen build-up in containment following a LOCA where the fuel temperature rises to the level at which zirconium-water reaction proceeds rapidly, leading to hydrogen generation sufficient to cause burning or explosion. The Reg. Guide limits in 1.97 presume an oxidiation rate that is a function of surface area and a termination point related to ECCS capability. The Three Mile Island Accident displayed high hydrogen generation because the ECCS was not permitted to do its job. The TMI-2 Implication Subcommittee should recommend actions for reevaluation of this generic item.	DO/RKM Mtg. Dec. 4 to Review TMI-2 Lessons Learned
4. Instrument Lines Penetrating Containment -- No action required Reg. Guide 1.11 and its Supplement adequately cover this point and no further action is needed.	NA*
5. Strong Motion Seismic Instrumentation -- No action required This is covered in Reg. Guide 1.12 and there does not appear to be the need for further action.	NA
6. Fuel Storage Pool Design Bases -- Joint Plant Arrangements and Safeguards and Security SCs. This is covered by Reg. Guide 1.13, however, the committee has frequently raised questions concerning the location of the fuel storage pool because of industrial sabotage questions. The Plant Arrangements and Safeguards and Security Subcommittee should review this matter and make recommendations to the full committee concerning the need for further action, especially regarding the location of the fuel pool with respect to grade.	MB/RKM and JCM/RKM Future Joint Meeting Planned

*NA = no action

	Follow-Up By
7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles -- No action required This is covered by Reg. Guide 1.14 supported by knowledge developed in the Safety Research Program. Based on the staff evaluation of the R&D work, this matter appears to be adequately covered.	NA
8. Protection Against Industrial Sabotage -- Joint Plant Arrangements and Safeguards and Security SCs. Reg. Guide 1.17 covers this matter, but since the issuance of Reg. Guide 1.17, committee letters have continued to raise questions about the adequacy of industrial sabotage protection. This matter should be addressed by joint effort of the Plant Arrangements Subcommittee and the Safeguards and Security Subcommittee.	MB/RKM and JCM/RKM Same as item #5
9. Vibration Monitoring of Reactor Internals and Primary System -- No action required Reg. Guide 1.20 covers these matters and the recent review of the loose parts monitoring technology indicated that current interpretations of Reg. Guide 1.20 by the NRC Staff serve the situation adequately.	NA
10. In-Service Inspection of Reactor Coolant Pressure Boundary -- Metal Components SC. This is covered by Section XI of the ASME Boiler and Pressure Vessel Code and Reg. Guide 1.,65 along with other modifications of the Code recently evaluated by the Reg. Guide Subcommittee. Questions remain as a result of Duane Arnold piping problems and various PWR feedwater line problems. This matter is under active review by the Metal Componnts Subcommitt e and an update of recommendations concerning this matter should be provided from that Subcommittee.	PGS/EGI On-going review will keep under surveillance
11. Quality Assurance During Design, Construction, and Operation -- Reactor Operations SC. Requirements of 10 CFR 50, Appendix B, ASME Boiler and Presure Vessel Code, Section III, ANSI-N45.2 (1971), Reg. Guides 1.28, 1.33, 1.64, 1.70.6, and proposed standard ANS-3.2, all address these matters. The NRC staff should be asked for a collective appraisal concerning the coverage in these documents. The Reactor Operations Subcommittee should then reassess the adequacy of this coverage. Recent experiences at Three Mile Island and concerns about the seismic restraints justify a determination concerning QA control adequacy.	HE/RKM Mtg. Dec 3
12. Inspection of BWR Steam Lines Beyond Isolation Valves -- No action required This adequately covered by ASME Boiler and Pressure Vessel Code, Section XI.	NA
13. Independent Check of Primary System Stress Analysis -- No action required This is adequately covered by ASME Boiler and Pressure Vessel Code, Section III.	NA
Operational Stability of Jet Pumps -- No action required The work on Dresden-2 and -3 installations and other operating experiences adequately satisfy the ACRS concern.	NA

	Follow-Up by
15. Pressure Vessel Surveillance of Fluence and NDT Shift - Metal Components SC (Review together with Item 16) This is covered by 10 CFR 50, Appendix A and ASTM Standard E-185. The NRC staff has recently recommended and the ACRS has approved the use of surveillance specimens from multiple reactor installations as satisfying the intent of the regulatory requirements. 10 CFR 50 will be modified accordingly under rulemaking proceedings.	PGS/EGI Mtg. Jan 16
16. Nil-ductility Properties of Pressure Vessel Materials -- Metal Components SC. This is covered by 10 CFR 50, Appendix A and Appendix G, ASME Boiler and Pressure Vessel Code, Section III and was addressed in the ACRS 1970 Report on Light Water Reactor Pressure Vessel Integrity, WASH-1285. The situation still appears to be adequate from a safety standpoint, but the ACRS Metal Components Subcommittee should reexamine the nil-ductility problem as a function of temperature for some of the older vessels nearing the end of their specified life and any new questions that have arisen concerning the upper shelf properties of materials.	PGS/EGI Same as item #15
17. Operation of Reactor with Less Than All Loops in Service -- No action required Standard Review Plan, Appendix 7A and Branch Technical Position EICSB-12 cover this matter adequately.	NA
18. Criteria for Preoperaitonal Testing -- Reactor Operations SC. This is covered by the most recent revision to Reg. Guide 1.68 but the uniformity of the preoperational testing program at various sites is unclear. The present concerns about plant operating skills suggests a need to have the Reactor Operations Subcommittee examine the nature of preoperational test programs in order to determine whether the requirements of Reg. Guide 1.68 really satisfy regulatory needs.	HE/RKM Mtg. Dec. 3
19. Diesel Fuel Capacity -- No action required Standard Review Plan 9.4 covers this matter adequately.	NA
20. Capability of biological shield withstanding double-ended pipe break at safe ends. Regulatory review practices cover this matter adequately. It may be appropriate to have one of the ACRS consultants examine a few examples of the design treatment to ascertain whether the approach is based on correct safety criteria.	MCG/Zudans review by Mar.
21. Operation of One Plant While Others are Under Construction -- Have Fellows review The coverage under Reg. Guide 1.17; 1.70; Sections 13.62; 1.101; ANSI N-18, 1.7; and Standard Review Plan 13.3, Appendix A; and 13.6 are all relevant to this question. One of the ACRS Fellows should be asked to review these documents to determine whether they treat all of the ACRS questions that have been raised and whether any other matters deserve attention. The potential for a Three Mile Island type of accident is particularly relevant to this matter. LERs should also be reviewed. Report by J. Bickel to M. Bender dtd. 10/3/79=major problem is security background checks and maintenance procedures for the operating plants.	Bickel report completed MB has follow-up

	Follow-Up By
22. Seismic Design of Steam Line -- Combination of Dynamic Loads SC. This is covered by Reg. Guide 1.29 but the Combination of Dynamic Loads Subcommittee is reexamining the design bases. Recommended changes to Reg. Guide 1.29 may evolve from the combination of dynamic loads review.	MB/EGI Mtg held in Sept. Plan another for Feb/ Mar
23. Quality Group Classification for Pressure Retaining Components -- Plant Arrangements SC (include analysis of secondary system (eg steam lines piping failures). Reg. Guide 1.26 covers this matter but questions arising from the interactive effect of non-safety grade equipment as seen in the Three Mile Island-2 accident may lead to changes in these classifications. The Plant Arrangement Subcommittee should review this matter.	MB/RKM Dec 5 SC Mtg (Deferred)
24. Ultimate Heat Sink -- No action required Reg. Guide 1.27 covers this matter satisfactorily.	NA
25. Instrumentation to Detect Stresses in Containment Walls -- No action required Reg. Guide 1.18 covers this matter but there are some controversial questions associated with grouted tendons. Current Staff interpretations provide adequate controls.	NA
26. Use of Furnace Sensitized Stainless Steel -- Reg. Guide 1.44 may need an update to better define "rapid-cooling". Bring to NRC Staffs attention but do not reopen consideration of Reg. Guide.	RFF will inform NRC Staff
27. Primary System Detection and Location of Leaks -- reassign to Metal Components SC Reg. Guide 1.45 addresses this matter and experiences at Duane Arnold and other plants indicate that the procedures are suitable. Exploring the use of TV cameras to find leaks could be explored.	PGS/EGI Jan 9 ACRS Staff review EPRI program
28. Protection Against Pipewhip -- Combination of Dynamic Loads SC. This is covered by Reg. Guide 1.46 but the Combination of Dynamic Loads Subcommittee will be reviewing these requirements as they are being influenced by combined load considerations. The question of whether the more elaborate requirements of combined loads introduce undesirable requirements should be examined.	MB/EGI Mtg. in Feb or Mar.
29. Anticipated Transients Without Scram -- ATWS SC Although this matter was covered by WASH-1270, issued in September 1973, the NRC has not yet established an implementation plan nor are the technical bases fully established. The ACRS ATWS Subcommittee should continue to review this matter and recommend actions to the full Committee.	WK/PAB Committee concurred with plan proposed by S.H. Hanauer in NUREG-0600
30. ECCS Capability of Current and Older Plants (small LOCA needs attention) -- ECCS The status should be updated through review by the ECCS Subcommittee, possibly with some support form the Plant Arrangements Subcommittee. Concerns about the oldest installations, e.g., Indian Point 1, have been resolved by NRC licensing action over the past several years.	MSP/ALB

	Follow-Up By
31. Positive Moderator Coefficient -- No action required PWR's presently follow a practice that satisfies the concerns about moderator coefficients under normal conditions. The transient questions associated with LOCA and the uncertainties associated with ATWS effects are under review.	NA
32. Fixed In-Core Detectors on High-Power PWR's -- No action required In-core monitoring needs to be re-reviewed in the light of TMI-2 experience, but it is unlikely that fixed in-core detector needs would change because of such a review. This item seems O.K.	NA
33. Performance of Critical Components (Pumps, Valves, etc.) in Post LOCA Environment -- Power and Electrical Systems SC. The qualification requirements in Reg. Guide 1.40, 1.63, 1.73, 1.89, and IEEE Standards 382 (1972), 383 (1974), 317 (1972), and 323 (1974), all address these matters. However, the experience at Three Mile Island-2 might alter some of these requirements. The Power and Electrical Systems Subcommittee should examine the need for new requirements.	WK/GRO Keeping Under Surveillance
34. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containment -- ACRS Fellow The NRC staff requirements for Mark II and Mark III containments address these matters adequately. A review of actual experience with Mark II design might be useful for updating our knowledge. One of the ACRS Fellows might be assigned to make such a review. LERs should also be considered. G. Young report to M. Bender 9/24/79. Most failures occurred during testing.	G. Young report =problem resolved
35. Emergency Power for Two or More Reactors at the Same Site -- Power and Electrical Systems SC. Reg. Guide 1.81 covers this matter. Shared diesels at older plants should be examined. Will consider all shared systems and components.	WK/GRO Future Mtg
36. Effluents from Light Water cooled Nuclear Power Reactors -- No action required This environmental question is resolved by the requirements of Appendix I of 10 CFR 50.	NA
37. Control Rod Ejection Accident -- No action required This is covered adequately by the requirements of Reg. Guide 1.77.	NA
38. Main Steam Isolation Valve Leakage of PWR -- No action required Reg. Guide 1.96 covers this adequately.	NA
39. Fuel Densification -- No action required Requirements of 10 CFR 50, Appendix K and case-by-case review of vendor fuel models covers this matter satisfactorily.	NA
40. Rod Sequence Control Systems -- No action required The practices of the NRC staff, including those established by GE NEDO 10527 cover this matter satisfactorily.	NA

		Follow-Up By
41.	Seismic Category 1 Requirements for Auxilary Systems -- Combination of Dynamic Loads SC. This is covered by Reg. Guide 1.26 and 1.29, but may be reexamined if new questions of interpretation arise out of a Combination of Dynamic Loads Subcommittee review.	MB/EGI Mtg. Feb.
42.	Instruments to Detect Limited Fuel Failures -- Joint Power and Electrical Systems and Reactor Fuel SCs. Although this has been addressed in an NRC document entitled "Fuel Failure Detection in Operating Reactors" by Siegal and Hagan, June 1976, the experience of Three Mile Island warrants further review of this matter. The Power and Electrical Systems Subcommittee should evaluate this question in combination with the Reactor Fuel Subcommittee. Call to attention of NRC Staff. Resolved. Will keep under surveillance.	WK/GRO and PGS/PAB
43.	Instrumentation to Follow the Course of an Accident -- Power and Electrical Systems SC. Reg. Guide 1.97, Revision 1, addresses this matter but the requirements have never been recognized. The Power and Electrical Systems Subcommittee should reexamine the requirements of 1.97 to determine whether they realistically define the need and whether a more definitive Reg. Guide should be provided based on TMI-2 experience.	CPS/SD Reg. Guide out for public comment
44.	Pressure in Containment Following LOCA's -- TMI-2 Implications SC. TMI-2 experience suggests the need to review this matter for low pressure containment. Will be considered during review of long-term lessons learned report	DO/RKM
45.	Fire Protection -- Fire Protection SC. Branch Technical Position 9.5.1 provides a staisfactory review process. Reg. Guide 1.120 whose development has been suspended because of ACRS concerns should now be reinitiated with attention being addressed to the requirements found acceptable for current Standard Plant Designs.	MB/PST Mtg. Dec 5
46.	Control Rod Drop Accidents (BWRs) -- Core Performance SC. This had been adequtely covered by NRC review practices. However, LERs have raised questions, short period scram concern raised by E. Epler. Low probability event	WK/PAB Will follow up
47.	Rupture of High Pressure Lines Outside Containment -- No action required Standard Review Plan Sections 3.6.1 and 3.6.2 cover this matter adequately.	NA
48.	Isolation of Low Pressure from High Pressure Systems -- Reactor Operations SC. Standard Review Plan 5.4.7 addresses this matter. A few LERs have been identified which may have reopened concern for this question.	HE/RKM Mtg. Dec. 3
49.	Monitoring for Loose Parts Inside the Reactor Pressure Vessel -- No action required Reg. Guide 1.133 covers this matter.	NA
50.	Qualification of New Fuel Geometry -- No action required Standard Review Plan 4.2, Revision 1, satisfies ACRS interest.	NA

	Follow-Up By
51. Maintenance and Inspection of Plants -- Reactor Operations SC. The ACRS originally accepted the postion that recent attention f the staff to these matters was adequate. The experience at TMI-2 reopens the question. The Reactor Operations Subcommittee should determine whether this matter needs additional effort.	HE/RKM Mtg/ Dec. 3
52. Safety Related Interfaces Between Reactor Island and Balance of Plant -- Plant Arrangements SC. Standard Review Plan 1.8 covers the matter in an administrative sense, but systems interaction questions from the TMI-2 accident experience warrent reexamination by the Plant Arrangements Subcommittee.	MB/RKM Will address at next SC Mtg

Resolution of Pending Items

	Follow-Up By
53. Turbine Missiles -- Get update from S. H. Bush.Nothing new to update. Particular attention given to older plants.	MWL/SHB
54. Effective Operation of containment Sprays in a LOCA -- Generic Items SC will follow at an appropriate time. This matter should be reexamined by the Generic Item Subcommittee. The selection of chemical additives is still under [illegible] by the NRC Staff.	MB/PST Waiting for NRC Staff Report
55. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock -- Metal Components SC. Reg. Guide 1.2 covers current practice satisfactorily. The situation with respect to old plants is still unclear and the LERs display some events where thermal shocks have exceeded Tech. Spec. limits. The implications of the LERs need more attention. The Metal Components Subcommittee should address this. Special concern for repressurization after or during cooldown.	PGS/EGI Mtg. Jan 9
56. Instruments to Detect (Severe) Fuel FAilures -- Power and Electrical Systems SC. The Three Mile Island experience justifies reexamination of this question.	WK/GRQ Keeping Under Surveillance
57. Monitoring for Excess Vibration Inside the Reactor Pressure Vessel -- Power and Electrical Systems SC. Methodology exists to address this matter in the pressure vessel, but the quality of its sensitivity has been related to actual safety needs. The capability seems to be adequate but the matter should be kept under surveillance by the Power and Electrical Systems Subcommittee.	WK/GRQ Keeping Under Surveillance Have ACRS Fellow review
58. Non-Random Multiple Failures -- Single Failure Criterion SC. Items 58.a, Reactor Scram Systems; 58.b, Current Sources; and 58.c, DC Sources, are matters of concern. The systems interaction work is now under active review by the Plant Arrangements Subcommittee and it should continue to assess this question. The single-failure criterion is relevant. Sandia is reviewing	MB/RKM Keeping Under Surveillance

	Follow-Up By
59. Behavior of Reactor Fuel Under Abnormal Conditions -- Reactor Fuel SC. Recent experience at Three Mile Island-2 should be evaluated to determine what is needed in this area. The ACRS Research Report has suggested that the PBF program be reoriented to address the question of intermediate level fuel degradation where fuel cladding has been significantly damaged and some fuel melting may have occurred.	PGS/PAB Study TMI-2 core performance when available
60. BWR and PWR Primary Coolant Pump Overspeed During LOCA — Joint ECCS and Plant Arrangements SC. Requires review by ECCS and/or Plant Arrangements Subcommittees.	MSP/ALB and MB/RKM will reexamine problem
61. Advisability of Seismic Scram — Extreme External Phenomena SC. Information is available from the Japanese and from the Canadians with respect to seismic scram . The Extreme External Phenomena Subcommittee should evaluate whether this new information provides sufficient background to make a judgment about when seismic scrams may be desirable in nuclear plants.	DO/RPS Will develop proposed Committee position
62. Emergency Core Cooling System Capability for Future Plants --Joint ECCS and Plant Arrangements SC. The requirements of 10 CFR 50, Section 50.3.4 (a)(4), 50.3.4 (b)(4), 50.4.6, and Appendix K, establish fuel performance requirements that have enhanced the emergency core cooling system capability of plants since this generic item was identified. All of the LOCA evaluation models have now been completed. The need for other cooling approaches to improved ECCS capability needs to be reviewed by the ACRS. The ECCS and Plant Arrangements Subcommittees should jointly attempt to determine whether this generic matter is adequately resolved, and if not, what actions are needed.	MSP/ALB and MB/RKM Will reexamine problem
63. Ice Condenser Containment -- Reassign to TMI-2 Implications The ECCS Subcommittee should determine whether adequate design margin exists during LOCA for ice condenser containments. If design margins are of importance, the action required to establish design margins should be identified.	DO/RKM Review effects of large H_2 generation
64. Steam Generator Tube Leakage -- Metal Components SC. Regulatory Guide 1.83 establishes a safe operating mode, but the leakage frequency is still of concern. The Metal Components Subcommittee should review this matter and establish the path of action for generic resolution. Reg. Guide handles plugging. Question is how to prevent SG tube failure	PGS/EGI Mtg. Jan 16
65. ACRS/NRC Periodic Ten-Year Review of All Power Reactors -- Reactor Operations SC. The Three Mile Island accident reemphasizes the need to establish a policy concerning this matter. The NRC Staff presently has a program to review the older licensed reactor systems as a basis for defining periodic review policy. The ACRS Reactor Operations Subcommittee should evaluate this activity on a continuing basis until the NRC has established an acceptable policy.	HE/RKM Mtg. Dec 3

	Follow-Up By
66. Computer Reactor Protection System -- Power and Electrical Systems SC. This system continues to be reviewed by the Power and Electrical Systems Subcommittee and a periodic status report on the progress represents adequate action for the present.	WK/GRQ Keeping under surveillance
67. Behavior of BWR Mark III Containments -- Fluid Dynamics SC. The experimental programs to verify Mark III containment behavior are in progress and the Fluid Dynamics Subcommittee is maintaining an overview of this work and reporting regularly to the full Committee. These actions seem appropriate.	MSP/ALB Keeping under surveillance
68. Stress Corrosion Cracking in BWR Piping -- Metal Components SC. This matter is under active review by the ACRS Subcommittee on Metal Components. R&D work is underway under Industry sponsorship as well as by DOE and NRC. The problem is still of concern but the actions underway meet the present need. Will report to Committee periodically.	PGS/EGI Keeping under surveillance
69. Locking Out of ECCS Power Operated Valves -- Reactor Operations SC. This matter should be examined by the Reactor Operations Subcommittee and appropriate action suggested.	HE/RKM Mtg. Dec. 3
70. Design Features to Control Sabotage -- <u>Joint</u> Safeguards and Security <u>and</u> Plant Arrangements SCs. This applies only to newly designed plants. The Committee's intent is unclear. The Safeguards and Security Subcommittee should reexamine this question in conjunction with the Plant Arrangements Subcommittee for the purpose of establishing a direction for resolution.	JCM/RKM and MB/RKM Future SC mtg planned
71. Decontamination of Reactors -- Joint Metal Components <u>and</u> Reactor Radiological Effects SCs. The Three Mile Island accident shows the importance of this question but the original intent was primarily to address the decontamination of reactors to reduce operator exposure during in-service inspection and other circumstances. The status of the experimental work sponsored by Industry needs to be reviewed by either the Reactor Operations Subcommittee or the Metals Components Subcommittee. NOTE: Reactor Radiological Effects Subcommittee will consider occupational exposure aspects, and Waste Management Subcommittee will consider waste disposal.	DWM/RM has lead Future SC mtg. planned
72. Decommissioning of Reactors -- Reactor Radiological Effects Subcommittee. This is an active NRC program of long duration and the status should be reported periodically by the Waste Management Subcommittee.	DWM/RM Keeping under surveillance
73. Vessel Support Structures -- Combination of Dynamic Loads SC. The problem here is primarily asymmetric load questions and load combinations. This matter should probably be addressed on a probabalistic basis and should be reviewed by the Combination of Dynamic Loads Subcommittee. BNWL is studying.	MB/EGI Keeping under surveillance

	Follow-Up By
74. Water Hammer -- Fluid Dynamics SC. The NRC staff is actively studying this matter but the problem should be addressed on a case-by-case basis. An ACRS Subcommittee with competent personnel to address the fluid mechanics questions should be assigned to review the status. Will review NRC Staff report.	MSP/ALB
75. Behavior of BWR Mark I Containment -- Fluid Dynamics SC. This matter is being addressed through R&D programs by the Mark I owners group and all of the open questions are nearing resolution. The ACRS needs an update of the status of this work. The Fluid Dynamics Subcommittee should be requested to summarize current status and establish the actions ultimately needed to resolve open questions.	MSP/ALB Will report to Committee at Dec. Mtg
76. Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment -- Power and Electrical Systems SC. The TMI-2 accident reemphasizes the importance of this type of question and perhaps related ones. The Power and Electrical Systems Subcommittee should review this matter with the Regulatory Staff and Industry representatives to establish whether current practice is satisfactory, and if not, what actions might be appropriate to improve current practice.	WK/GRQ Future SC mtg planned
77. Soil Structure Interaction -- Extreme External Phenomena SC. The technology for evaluating soil structure interactions is developing rapidly. The ACRS should request one or more of its consultants who are not actively pursuing personal interest in this question to summarize the current status of technology in order to determine whether the current situation satisfies the generic concerns. The Extreme External Phenomena Subcommittee could undertake to sponsor such a review.	DO/RPS ACRS Consultants are reviewing

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

NOV 2 1979

MEMORANDUM FOR: Raymond F. Fraley, Executive Director, Advisory Committee on Reactor Safeguards

THRU: Lee V. Gossick, Executive Director for Operations TAR for L.V.G.

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

SUBJECT: NRC PROCEDURES FOR THE CONTROL OF ACRS REQUESTS AND ACRS CONSULTANT REPORTS

The response to Commissioner Kennedy from Mr. Gossick, dated September 21, 1979, requested that AEOD review the handling of (a) ACRS requests requiring staff action and (b) ACRS consultant reports. This memorandum provides a response to this request and identifies immediate steps that can be taken to improve the control and processing of these items. Other aspects of the September 21, 1979 memorandum will be addressed in future correspondence.

Enclosure 1 discusses the proposed plan with regard to ACRS correspondence requesting or requiring staff action. The handling of ACRS consultant reports is discussed in Enclosure 2. In both cases, the objectives of our review were to assure that the proper tracking, control, and follow-up actions were taken using existing systems to a maximum practical extent. These proposed procedures reflect discussions with Dr. Moeller (ACRS), Mr. Fraley, Mr. Libarkin (ACRS staff), the Director's office of NRR, and individuals in other affected offices. We would appreciate your comments on these proposals.

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

Enclosures:
1. ACRS Correspondence Requesting Staff Action
2. ACRS Consultant Reports

cc w/enclosures:
H. Denton, NRR
W. Russell, NRR
W. Dircks, NMSS
ELD
R. Minogue, SD
S. Levine, RES

ENCLOSURE 1

ACRS Correspondence Requesting Staff Action

Background

In the past, requests by the ACRS, requiring staff action such as the review of particular subjects, have been made in a variety of ways. Requests made by the full Committee are generally forwarded in memoranda from the ACRS Chairman or Exec[illegible]e Director to the EDO or a specific Office Director such as the Director, NR[illegible], or are included in the ACRS letter reports to the Chairman, NRC. In addition, requests for information may be formally sent to various staff levels and, frequently, are made orally during meetings. The Committee semiannually summarizes all requests made at ACRS full Committee meetings, including oral, and the staff provides a written response or status report to the Committee.

Informal requests are sometimes made by individual members, or subcommittees, and are usually forwarded (in writing or orally) by the ACRS member or an ACRS staff member to project managers, individual staff members, or supervisors. These informal requests do not normally receive widespread distribution, particularly to line management, and appropriate follow-up actions are not assured unless a formal request is handled as controlled correspondence. Such informal requests, e.g., in support of ACRS subcommittee activities, although requiring some specific staff action, will not be considered a formal request in the context of this document.

Based on discussions with the ACRS Executive Director, the ACRS desires a written response to all formal transmittals to the agency which reflect action at the Committee level. The response, at a minimum, would be an acknowledgment of receipt of the material and an indication that a written or an oral response is planned and noting a point of contact within the staff and an estimated date of response.

A review of the existing situation has indicated the need for several actions: (1) that formal ACRS requests be transmitted by the Chairman or Executive Director; (2) that since more than one NRC office deals with the ACRS, the addressee for ACRS requests should be the EDO level or above; and (3) that formal ACRS requests be controlled and followed-up via existing principal correspondence systems.

Proposed ACRS Actions

1. All formal correspondence requiring staff action be signed by either the Chairman of ACRS or the Executive Director and addressed either to the Chairman, NRC, or the Executive Director for Operations (EDO).

2. Copies of such correspondence be sent directly to the Office Director and key individual (if known) directly involved with the subject material.

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2. Following each main Committee meeting, the ACRS staff writeup containing the meeting commitments, decisions, and requests for staff action be formally sent to the staff as indicated in (1) above.

Proposed NRC Actions

1. ACRS letters requesting staff action be handled as principal correspondence.

   (a) If received at Commission level, the Secretariat assigns a yellow ticket indicating action is assigned to the EDO and noting any special instructions.

   (b) If received at EDO level (or forwarded by Secretariat), the Administration and Correspondence Branch assigns a green ticket indicating action responsibility, completion deadline, information routing, special instructions, etc. The target completion date of 30 days from receipt would normally be assigned.

2. The responsible office, in turn, assigns action to a specific individual and uses the normal control and follow-up actions appropriate to principal correspondence.

3. Should it not be possible to complete action within a 30 day period, an interim reply be sent indicating the reasons for delay and when a response can be expected. The completion date specified in the green ticket would be revised accordingly.

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ENCLOSURE 2

ACRS Consultant Reports

Background

Reports by ACRS consultants are generated on a variety of subjects at a rate of about 100 per year. In the past, copies of these reports were informally sent to NRR, but a system has not been used within NRR to formally and systematically distribute, analyze, and document the impact of these reports on the licensing process or whether licensing board notification is appropriate.

The ACRS believes the NRC staff should receive these consultant reports and assess the safety significance of the information they contain. However, at the same time, the ACRS notes that these reports are forwarded to the staff without endorsement and, in many cases, are even unreviewed by the ACRS. Thus, the ACRS cannot verify the completeness, accuracy, or relevancy of the information.

NRR has recently defined a formal program for dealing with the review of reports generated by NRC supported research. This program, now in the process of being documented and forwarded to the Commission as an Information Paper, will also be applied to the handling of ACRS consultant reports. In the short-term, this program includes the following NRR actions applicable to ACRS consultant reports:

a) Assign responsibility to a specific NRR division and branch.

b) Define responsibility of NRR reviewers in the context of the review.

c) Establish a follow-up system to assure that reviews are completed by NRR reviewers in a timely manner.

d) Establish a system to assess the results of the review for possible licensing impact, including interface with licensing board notification procedures.

e) Set up a system so that all reports not reviewed within 60 days are automatically sent to appropriate branch.

A longer-term program is to be finalized before the end of CY 1979 which will assure that the results of each review are tracked, cataloged, and retrievable. The longer-term program is being developed in conjunction with a request for additional resources. In this regard, it is noted that using an average of one man-week of review time per report, and an average of 100 ACRS consultant reports per year, a total resource commitment of approximately two man-years would be involved in the systematic review outlined above.

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Proposed ACRS Actions

Promptly forward all ACRS consultant reports for information to the Director, NRR with a copy to the Director, AEOD.

Proposed NRC Actions

For NRR to assess the significance of the ACRS consultant reports to the reactor licensing process and to board notification requirements. Initially, this assessment should use the short-term program and, subsequently, the longer-term program.

Note: Should any of the consultant reports be outside the scope of NRR, NRR will forward them to the responsive NRC office for review and follow-up action.

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

November 9, 1979

APPENDIX XXIX
REVISION TO PARA. 6.b., NUREG-0567

N. Haller, Director
Office of Management and Program Analysis

Subject: PROPOSED POLICY AND PROCEDURES FOR DIFFERING PROFESSIONAL OPINIONS, NUREG-0567

In accordance with the memorandum from Chairman Hendrie requesting comments on the subject document, the following are provided regarding the participation of the ACRS in this process.

These comments are based on discussion with the Committee.

Revise paragraph 6.b. to read as follows:

b. The ACRS

If the differing professional opinion relates to a 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.

R F Fraley

R. F. Fraley
Executive Director

cc: ACRS Members
H. H. E. Plaine

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

November 14, 1979

APPENDIX XXX
ACRS REPORT ON NUREG-0600, INVESTIGATION INTO THE MARCH 28. 1979 THREE-MILE-ISLAND ACCIDENT BY IE

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

SUBJECT: NUREG-0600 "INVESTIGATION INTO THE MARCH 28, 1979 THREE MILE ISLAND ACCIDENT BY OFFICE OF INSPECTION AND ENFORCEMENT"

Dear Dr. Hendrie:

During its 235th meeting, November 8-10, 1979, in accordance with the Commission's request, the Advisory Committee on Reactor Safeguards completed its review of NUREG-0600. The report was also discussed at a Subcommittee meeting in Washington, D. C. on October 30, 1979. During its review the Committee had the benefit of discussions with the Nuclear Regulatory Commission (NRC) Inspection and Enforcement (I&E) Staff, and of comments from the licensee.

The stated scope of NUREG-0600 is limited to investigation of the licensee's operational actions prior to and during the course of the accident, and his actions to control release of radioactive materials and to implement his emergency plan during the course of the accident. Consistent with this limitation, emphasis is placed on departure from Technical Specifications prior to the accident and departure from the licensee's procedures during the course of the accident, with little consideration of other factors.

Other investigations and other NRC task force studies have considered not only the actions taken by the licensee, but also other facets of the accident, including peculiarities of the nuclear steam supply system that tended to inhibit recovery or to confuse the operators by leading to pressure and level conditions not anticipated by the written procedures, and deficiencies of the control room and system design that degraded the quality of information available to the operator. Additional details not in NUREG-0600 can be found, for example, in a report entitled "Analysis of Three Mile Island Unit 2 Accident" (NSAC-1, July 1979) prepared by the Electric Power Research Institute, Nuclear Safety Analysis Center.

NUREG-0600 includes a factual chronology with event descriptions, and a finding of operational and administrative shortcomings and errors. It concludes (Appendices IB and IIF) that a total of 36 items of potential operational or administrative noncompliance existed. The Office of Inspection and Enforcement subsequently, by letter of October 25, 1979 to Metropolitan Edison Company, imposed fines for seventeen violations, infractions and deficiencies, many of them multiple occurrences.

Because the limited scope of the report tends to lead to a catalog of violations with only limited recognition of other factors that contributed to errors by the operators, the Committee has some concern that it may be concluded from the charges of failure to follow accident procedures that such failure is automatically a violation.

Accident procedures are prepared by the licensee and are not approved by NRC, but the licensee is required to follow them. The Committee believes that an accident procedure cannot be sufficiently detailed to encompass every possible sequence of events, and that it must be based on the assumption that a particular set of conditions exists; a deviation from this set of conditions may make it necessary to depart from the procedure. As an example, TMI-2 Emergency Procedure 2202-1.3 (Loss of Reactor Coolant/Reactor Coolant System Pressure) which is referred to in NUREG-0600, is believed by the Committee to include confusing symptoms and instructions for the case of a loss of reactor coolant at the top of the pressurizer. Likewise TMI-2 Emergency Procedure 2202-1.5 (Pressurizer System Failure) which calls for pressurizer level control is believed to be unacceptable for the TMI-2 accident or for any other loss of reactor coolant at the top of the pressurizer. The question, therefore, arises whether an operator, using his best judgment, is guilty of a violation if he consciously takes an action that is at variance with procedures which in themselves may contain confusing or incorrect guidance. The Committee believes that, if so, this is the wrong approach to protecting the health and safety of the public during an emergency and that the operator, guided by the written procedures, his training, and available technical advice, should be allowed to use his best judgment to deal with the problem. His judgment will obviously be subject to post-factum appraisal.

The Committee has found this report less than satisfactory, and its title misleading, chiefly because of limitations in its predefined scope. For this reason, the Committee recommends the preparation and issuance of a summary report that consolidates and integrates the findings of the several NRC Task Forces that have investigated and reported on this accident.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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

November 14, 1979

APPENDIX XXXI
CLARIFICATION OF ACRS REPORT OF
AUG. 13, 1979 in NUREG-0578

Honorable Peter A. Bradford
Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Bradford:

In your letter of October 9, 1979 to the Advisory Committee on Reactor Safeguards you referred to the Committee's letter to Chairman Hendrie of August 13, 1979 concerning "Short-Term Recommendations of TMI-2 Lessons Learned Task Force" and noted the ACRS statement that "orderly and effective implementation and the appropriate level of review and approval by the NRC Staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by NUREG-0578." You asked that the ACRS "identify in more detail which of the scheduled items the Committee believes should be extended and the basis for those recommendations."

The ACRS comment was intended as a general observation. The Committee was not favoring any unnecessary delays. However, the Committee anticipated that exceptions to the original schedule might be desirable or even necessary. For example, with regard to the Shift Technical Advisor, the Committee anticipated that not all licensees would be able to obtain within the time specified the services of sufficiently qualified personnel for three-shift, seven-days-a-week duty, including provisions for the ongoing training which is called for and appropriate to the task. In this respect, the Committee believes that, where licensees are not able to comply with the NRC requirements on schedule, they should be required to submit temporary alternative proposals for approval by the Staff.

Other items, such as the establishment of an onsite technical support or operational support center may also be difficult to achieve at all operating reactors by the scheduled time. In addition, some items of equipment or instrumentation may not be available on the time schedule proposed.

Furthermore, some of the changes will require shutdown of the reactor. Some grouping of such changes is likely to be desirable to limit the number of transients associated with shutdowns that are required for this purpose.

The ACRS does not believe public safety will be unduly jeopardized by extending the implementation schedule for some reasonable period.

Sincerely yours,

Max W. Carbon

Max W. Carbon
Chairman

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Ahearne
Samauel Chilk

bcc: ACRS Members
R. Fraley
M. Libarkin
J. McKinley
T. McCreless
R. Major
J. Jacobs
H. Voress

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
November 14, 1979

APPENDIX XXXII
ACRS ACTIONS ON PROPOSAL REVISIONS OF REGULATORY GUIDES

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

SUBJECT: ACRS ACTION ON PROPOSED REVISIONS OF REGULATORY GUIDES

Dear Mr. Gossick:

During its 235th meeting, November 8-10, 1979, the ACRS concurred in the regulatory position of Regulatory Guide 1.141, Revision 1, "Containment Isolation Provisions for Fluid Systems", with the condition that the implementation section of this Guide be revised consistent with the TMI-2 Lessons Learned Task Force recommendations (NUREG-0578).

In addition, the ACRS agreed with the NRC Staff's plan to issue Proposed Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", for public comment.

Sincerely,

Max W Carbon

Max Carbon
Chairman

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

OFFICE OF THE COMMISSIONER

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

September 28, 1979

APPENDIX XXXIII
IDENTIFICATION OF NRC REGULATIONS
WHICH NEED CHANGES

MEMO TO: Max W. Carbon, Chairman
Advisory Committee on
Reactor Safeguards

FROM: Peter A. Bradford

SUBJECT: IDENTIFICATION OF NRC REGULATIONS WHICH NEED CHANGES

In an April 20, 1978 memorandum, the Commission requested the ACRS and the staff to take certain actions to implement the recommendations of the report, "Follow-up on ACRS Letters." The Commission later considered the need for additional procedural guidelines relative to Commission involvement in ACRS advice items on which Commission action might be appropriate. The guidance drafted by the Office of Policy Evaluation included the following item:

2. "When ACRS advice indicates a need or desirability of changes in regulations, or in any procedures requiring Commission-level consideration, the matter should be brought to the Commission's timely attention, together with any staff views and recommendations with respect to appropriate Commission action."

The Commission decided that specific guidance was unnecessary, in part because the staff has to identify any significant regulatory changes for Commission approval. Recently, however, the question has been raised whether the lack of a formal procedure for obtaining ACRS views on NRC regulations needing changes has inhibited Committee recommendations in this area. Therefore, I would appreciate the Advisory Committee on Reactor Safeguards' views on whether or not the lack of a specific procedure for identifying rules and regulations which need revision has inhibited the Committee. In addition, please identify any rules and regulations which you believe need to be addressed promptly in order to ensure public health and safety.

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Ahearne
Lee V. Gossick
Samuel J. Chilk
Al Kenneke
Len Bickwit

Additional Documents Provided for ACRS' Use

1. Draft 1 of Regulatory Guide 1.97 (Rev. 2), Instrumentation for Light-Water-Cooled Nuclear Power Plant and Environs Conditions During and Following an Accident

2. Memorandum, Proposed Visit/Meeting of ACRS with the RSK and GPR During May 1980, dtd Nov. 10, 1979

3. NUREG-0585, TMI-2 Lessons Learned Task Force Final Report, Oct. 1979

4. Draft 2, Regulatory Guide 1.141 (Rev. 1), Containment Isolation Provisions for Fluid Systems

5. Memorandum, J. G. Davis, IE, to NRC Commissioners, Investigation of the Three Mile Island Accident by the Office of Inspection and Enforcement, dtd Apr. 20, 1979

6. Memorandum, J. G. Davis, IE, to NRC Commissioners, Investigation of Three Mile Island Accident by the Office of Inspection and Enforcement, dtd June 8, 1979

7. Memorandum, S. Levine, NRC Staff, to R. F. Fraley, ACRS, Response to Recommendations by ACRS on Safety Research, dtd Oct. 26, 1979

8. Memorandum, R. J. Mattson, NRC Staff, to R. F. Fraley, ACRS, ACRS PWR Question Regarding Effect of Pressurizer Heater Uncovery on Pressurizer Pressure Boundary Integrity, dtd Nov. 5, 1979

9. Teknekron Research, Inc., McLean, VA, Draft Report, Analysis of the First Eighteen Months of Licensed Operation of Babcock and Wilcox Plants, Sept. 27, 1979

10. Docket 50-255, License DP-20, Palisades Plant, License Event Report 79-037, Open Containment Exhaust Valves Bypass Line, dtd Sept. 28, 1979