ACRS -1710

Issue Date:

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MINUTES OF THE
237TH ACRS MEETING
JANUARY 10-12, 1980
WASHINGTON, DC

CERTIFIED

The 237th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC, was convened at 8:30 a.m., Thursday, January 10, 1980.

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

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

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

I. Chairman's Report (Open to Public)

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

A. Reviewers

The Chairman named Messrs. Mathis and Okrent as reviewers for the 237th ACRS Meeting.

B. TVA Proposal for Low Power Operation

The Chairman noted receipt of correspondence between the NRC and TVA regarding proposed licensing for special low-power testing of the Sequoyah Nuclear Power Plant (see Appendix IV).

II. Meeting With NRC Staff on NRC Reactor Research Budget (Closed to Public)

[Note: This meeting was closed in accordance with Section 9(b) of GISA. Thomas G. McCreless was the Designated Federal Employee for this portion of the meeting.]

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Mr. Seiss, Reactor Safety Research Subcommittee Acting Chairman, noted that when the Committee prepared NUREG-0603, Comments on the NRC Safety Research Program Budget in July 1975, it was agreed that it would delay the publication of its annual report to Congress on the NRC Reactor Safety Research Program until February. In view of that agreement, the Committee should complete its 1979 annual report to Congress by February 15. He suggested that the Committee consider a proposed draft of the report at this meeting, concerning itself primarily with arriving at its consensus positions, and that the final editorial work could be done at the 238th ACRS Meeting (February). (For background material, see Appendix V, not available to public.)

The Chairman noted that he and Messrs. Carbon and Fraley had met with H. R. Myers, of the staff of the House Subcommittee on Energy and Environment of the Committee on Interior and Insular Affairs, who had requested that he be briefed on the Committee's views regarding improved reactor safety research. It was the consensus of the Committee that R. F. Fraley should brief H. Myers and his staff on this matter as a follow-up to the previous meeting.

The Committee developed its positions regarding the appropriate level of research needed in the NRC's Reactor Safety Research Program, noted certain shifts in emphasis from the administration position, and recommended restoration of cut funds in certain areas (the details of these positions will be contained in the Committee's 1979 annual report to the Congress on the NRC's Reactor Safety Research Program).

III. Meeting With Members of the NRC Staff on the Proposed NRC Action Plan to Implement the Recommendations of the President's Commission and Other Studies on the TMI-2 Accident (Open to Public)

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

A. Subcommittee Meeting

Mr. Etherington, Chairman of the Subcommittee on the TMI-2 Accident Action Plan, noted that the Subcommittee has reviewed the plan in its current form, believes the plan to be comprehensive, but that it is necessary to establish priorities for action. (For background material, see Appendix VI.)

Mr. Etherington said that since many of the recommendations to which the action plan responds originate from sources other than the President's Commission report, the title of the Action Plan should be changed, and that an appendix should be provided to list all of the source documents. He said also that there is an absense of a quantitative basis for choosing the items to be addressed. He said that the report indicates, falsely, that only TMI-2 items are addressed, while in fact the report also addresses some pre-TMI-2 items.

B. NRC Staff Discussion

R. Mattson, NRC Staff, noted the existence of a memorandum to the Commissioners from the Executive Director for Operations that defines the prerequisites for the resumption of licensing (see Appendix VII).

R. Mattson said that the items listed in the Action Plan would have to have priorities assigned to them in order to complete the plan. Once the priorities have been assigned, each office will start at the bottom of the list of priority objects and identify those to be either deleted or deferred, until a total of 150 man years have been identified for the plan. This is the total of unbudgeted time available to the NRC Staff to work on TMI-2 issues. He suggested that either budgeted funds may have to be reassigned to get additional work done, or additional supplemental funds may have to be obtained from Congress. This latter is not likely. In addition, the necessary manpower may not be available to be recruited. In that case, work would have to be contracted out. He said that only items that need to be done are currently listed in the Action Plan. Priorities will be assigned on a basis of potential for risk or to reduce consequences.

It was the consensus of the Committee, that until the priorities could be assigned and evaluated, the Committee could not provide its final comments regarding the Action Plan.

Members of the NRC Staff recognized that in order for the Committee to review an action plan in detail, the current plan would have to be developed further. They noted that the following items are under consideration:

- The overall research level will not be significant when compared to the total program.
- The NRC is requesting its contractors to develop independent analytical methods so that vendor design errors can be identified.
- All licensees will be required to perform positive task analyses, evaluate their training requirements, and develop criteria for personnel training and licensing.

- The NRC is considering the licensing of suppliers to vendors, as well as architect-engineers and may become more deeply involved in quality assurance and quality control.

Mr. Okrent recommended that the NRC Staff should obtain information on the requirements in foreign plants, compare them with U.S. requirements, and perhaps adopt those foreign requirements that would improve safety in the U.S.

Mr. Moeller suggested that the NRC Staff develop data on the operating utilities to develop criteria that can assure that the best operators of the utilities be used as guides for evaluation of performance.

R. Scroggins, NRC Staff, said that siting issues are being addressed in the TMI-2 Action Plan by taking into consideration potential accidents greater than Class-8.

R. Bernero said that the NRC Staff has considered a problem of a degraded core without melting, and compared the potential consequences with those from a partial core melt. Mr. Lewis suggested that attention may be focusing on the wrong event and moving away from safety. He noted that the serious events at Three Mile Island were caused by purely human factors that could be reversed at any time.

R. Scroggins suggested that much information is being obtained through the Integrated Reliability Evaluation Program (IREP) that is being applied to six plants. The NRC Staff is studying a proposed requirement for a licensee to perform a mini IREP for a mid-term operating license.

J. A. Norberg, NRC Staff, said that in further efforts to assure better reliability in operating plants, the NRC Staff has sent letters to licensees requesting that they identify auxiliary feedwater system problems.

R. Mattson said that the current draft of the Action Plan does not adequately reflect the need for greater reliability in the auxiliary feedwater system, and that this matter will be addressed in future drafts.

W. Lipinski, ACRS Consultant, noted that a loss of feedwater can cause, in the long term, core melt in one-half of all current Westinghouse plants, and all Combustion Engineering plants, because the primary systems cannot cool the plants while under high pressure. Babcock and Wilcox plants can cool at high pressure by using feed and bleed methods along with the ECCS.

In answer to a question, V. Benaroya said that the NRC Staff is developing a plan to require environmental qualification of pressurizer heaters.

Mr. Kerr noted his opinion that the two main points of the President's Commission Report on the TMI-2 accident are that

- The accident at TMI-2 was too serious an accident to be tolerated again, and
- The NRC must be prepared to handle the same type or worse accident should it occur.

He asked members of the NRC Staff whether, even with its limited resources, they have made an effort to place a higher priority on accident prevention over accident mitigation.

R. Purple, NRC Staff, said that this concept is similar to the current NRC Staff thinking. However, there is some counter argument that prevention reaches a point of diminishing returns, and that more can be obtained on a cost benefit basis from litigation.

R. Purple said that the section in the current draft dealing with emergency plans will be rewritten since the President has directed that the Federal Emergency Management Administration (FEMA) will have the full authority to coordinate emergency plans with state and local agencies. In the meantime, the NRC will have to show how the public is being protected until the time arrives when FEMA can take over.

Mr. Okrent suggested that the NRC Staff consider upgrading requirements regarding radiation protection of the control room and the emergency center for operator habitability during accidents of such severity as are now being postulated.

R. Mattson said that the NRC Staff plans to determine whether DOE plans regarding worker protection and public protection are adequate, and if they are not, these matters will be covered in future drafts of the Action Plan.

J. Sinto, NRC Staff, said that Chapter 4 of the Action Plan deals with internal NRC organization, especially that related to the Commissioners and to upper management levels, and that the current draft was derived from the letter from the Commissioners to Dr. Frank Press. However, the implementation of any internal organization or reorganization is under the jurisdiction of the Commissioners themselves. Chapter 4, if necessary, will be revised to reflect the Commissioners' implementation.

C. Summary

Mr. Etherington noted that it appeared to be the consensus of the Committee that a letter, not containing detail, and without comment on priorities, could be written during this meeting. He noted that the current draft of the Action Plan is only a plan from which an action plan can be developed.

Mr. Ebersole suggested that in the Action Plan, the NRC should prohibit the "pass-through" of fines and penalties from the utilities to the customers.

Mr. Lewis suggested that

I'd support a compulsory ban
On pretending a list is a plan -
Without casting aspersion,
Decision aversion
Is a defect of fallible man.

IV. Meeting With the NRC Staff on Implementation of NRC Bulletins and Orders Resulting from the TMI-2 Accident and Small-Break LOCA Analysis (Open to Public)

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

A. Subcommittee Report

Mr. Mathis, Chairman of the TMI-2 Accident Bulletins and Orders Subcommittee, noted that the Subcommittee had met for a two day meeting in Los Angeles on January 3 and 4 to review the Bulletins and Orders Task Force reports, NUREG-0623 and NUREG-0645. Covered in these reports were mainly the analyses of loss of feedwater and of small-break LOCA events, viewed primarily with respect to systems reliability, analysis of transients, operator guidelines, plant procedures, and operator training. The Subcommittee also heard a report of the audit that the Bulletins and Orders Task Force had conducted at selected plants. In addition, the Subcommittee heard reports from owners groups representing the four light-water reactor vendors. Mr. Mathis also identified those ACRS members and ACRS consultants who attended this meeting.

The Chairman questioned whether the Committee could write a report on this subject at this meeting, since the Committee has not received and reviewed all of the pertinent reports yet.

D. Ross, NRC Staff, identified the reports yet to be issued as follows:

- NUREG-0611, on approximately January 18, 1980, dealing with Westinghouse plants,
- NUREG-0625, on approximately January 25, 1980, dealing with Combustion Engineering plants,
- NUREG-0565, due approximately January 25, 1980, dealing with Babcock and Wilcox plants, and
- NUREG-0626, due approximately February 1, 1980, dealing with General Electric Plants.

It was the consensus of the Committee that it would not be able to write a report on the implementation of NRC Bulletins and Orders until all of the above documents have been received and reviewed.

B. <u>NRC Staff Presentations</u>

W. Hodges, NRC Staff, noted his concern that vendors change their ECCS evaluation models over a short period, and that if these models calculate no fuel temperatures greater than 2200° F, they meet Appendix K requirements. He said he believes that this practice is counterproductive to safety.

D. Ross said that he does not believe that the NRC Staff has a choice in this matter. He said, however, that he believes that the analytical work today provides less margin to safety than the original compliance with Appendix K by the NSSS vendors. He said that he believes that it is important to adhere to Appendix K rules.

Mr. Kerr voiced his concern that many things appear to be done rapidly, and that there appears to be no effort to assess whether the changes do reduce risk rather than increase it.

Z. Rosztoczy, NRC Staff, noted his concern regarding the frequency of opening of the power operated relief valves (PORV), and believes something should be done to reduce this frequency.

Mr. Kerr noted his concern that if scram is induced to prevent PORV opening, the NRC should be sure that risk is not increased. If a scram is considered a challenge to safety, then any device that reduces the need for scrams must be considered a safety system, therefore PORVs must be considered to be safety-related, even though they are not required by the NRC.

In answer to a question, D. Ross said that the NRC Staff has not yet established what is adequate reliability for the feedwater system.

Mr. Okrent requested copies of memoranda relating to "identified research needs from the Bulletins and Orders Task Force". He questioned whether anyone in NRR has judged the requested R&D items in the above memoranda to be in balance with other NRC needs. He suggested that NRR could develop a broad spectrum of research requests related to current issues for which the return to public health and safety may not be as great as in other areas for which NRR could request research.

While discussing small-break LOCA analyses, Mr. Okrent chided the NRC Staff in that they had listed the conservatisms used in the calculations, but had not mentioned the unconservatisms. He maintained that a balanced presentation was not being given.

Z. Rosztoczy said that in the small-break LOCA analyses, the unconservatisms used were based on current practices, and are probably larger than the conservatisms required by Appendix K.

Mr. Okrent requested that the unconservatisms identified in the calculations, as well as the conservatisms and uncertainties, be made available to the Committee.

D. Ross offered to have the appropriate Branch Chiefs provide this information to the Committee in writing.

Z. Rosztoczy indicated his following concerns:

- the frequency of small-break LOCA-type events,
- adequacy of the ECCS criteria for small-break LOCA as these criteria exist today,
- calculations of small-break LOCA events give indications of inadequacy of the current criteria,
- a lack of consistency between Appendix K and best estimate calculations, indicating that Appendix K does not provide sufficient margin to cover the uncertainties in the calculations.

He suggested that while the industry developed its new models, the NRC should study the criteria to determine what is needed for small-break LOCA analysis.

D. Ross said that this matter can be discussed with the ECCS Subcommittee when the Subcommittee reviews the new ECCS models.

Z. Rosztoczy said that the NRC Staff expects to reach decisions regarding the small-break LOCA analysis deficiencies within the next four to six months.

Mr. Okrent suggested that the NRC should also calculate probabilities of the postulated events.

D. Ross discussed the work products of the Bulletins and Orders Task Force, a time table for resolving the issues, a schedule for implementation of the Task Force recommendations, and described recent audits of ten light-water reactor plants (see Appendix X).

In response to questions raised by Mr. Okrent in the Subcommittee meeting, D. Ross discussed the research needs identified from the B&O Task Force work (see Appendix XI).

D. Ross discussed additional recommendations not included in the Bulletins and Orders Task Force generic reports (see Appendix XII).

B. Sheron, NRC Staff, discussed ECCS rule status summaries and their applicability to small-break LOCAs (see Appendix XIII).

Z. Rosztoczy presented a summary of small-break LOCA and loss of feedwater accident evaluations (see Appendix XIV).

V. Meeting with NRC Staff on Proposed Revision of NRC Criteria for Siting Nuclear Power Plants (NUREG-0625) (Open to Public)

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

A. Subcommittee Report

Mr. Moeller, Site Evaluation Subcommittee Chairman, noted that the NRC Siting Group Task Force has concluded that 10 CFR 100 has allowed the use of engineering safeguards to compensate for site deficiencies, which has allowed, in some cases, the siting of nuclear power plants in more populated areas. In a sense, this allowed an erosion of the limits on distance and population density in terms of siting. This task force has made nine separate recommendations in their report, NUREG-0625.

The task force proposes that 10 CFR 100 be rewritten to allow for larger accidents, including Class-9. They are proposing to set up new means for assessing the efficacy of engineered safety features.

In addition, they propose to allow for consideration of post-licensing changes in offsite activities in the vicinity of the site.

The task force plan includes the specification of minimum distances for exclusion areas, minimum distances for emergency planning zones, and minimum distances to offsite hazards.

Mr. Moeller cautioned that too stringent site requirements could eliminate the option of nuclear power, thus imposing a greater risk on the public if the proposed nuclear plants were replaced with coal plants or other energy sources that provide greater risks than nuclear plants. He noted, that in practice, the use of more remote sites has in fact been dictated, and that large utilities, such as TVA and Commonwealth Edison, have all but adapted their best existing sites to be multi-unit facilities.

B. NRC Staff Presentation

D. Muller, NRC Staff, discussed the background of NUREG-0625, and the nine recommendations made in the report (see Appendix XV).

In answer to a question regarding the effect on the proposed siting revisions on underground sites for nuclear plants, D. Muller said that the task force considered only light water reactors of current design placed on conventional sites. He noted, however, that there is nothing in the concepts that are proposed that precludes underground siting.

V. Moore, NRC Staff, reinforcing the previous statement by D. Muller, said that the proposals do not preclude major breakthroughs in safety. The proposals are intended to deal with the current state of the art.

D. Muller noted that NUREG-0625 was written for the Commissioners, and requires that the divergent Staff views be included in the report. He said, however, that the recommendations are the operative part of the report.

Mr. Okrent noted that "safety" and "environmental" matters cannot be separated, since both have safety implications. As an example, he cited the matter of chemicals disposed of in the past now finding their way to human uptake.

In answer to a question, D. Muller said that he anticipates that it will take up to three years until the new rule is promulgated. (For NRC Staff responses to questions raised by Mr. Okrent at the Subcommittee meeting, see Appendix XVI).

VI. Executive Sessions (Open to Public)

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

A. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 238th ACRS Meeting (February) and several items for future meeting (see Appendix II).

2. Schedule for ACRS Meetings and Tours

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

B. Subcommittee Reports

1. Procedures Subcommittee

The Chairman noted that the Procedures Subcommittee has been informed that, because of work assignments at ORNL, Mr. Bender would be unable to devote his usual amount of time to Committee activities. For the remainder of Calendar Year 1980, Mr. Bender will be able to participate in approximately one out of three ACRS meetings. His subcommittee activities will also be curtailed. The Committee agreed that this proposed abreviated work schedule is acceptable.

2. Class-9 Accidents

Mr. Kerr, Subcommittee Chairman, requested that the Committee provide the subcommittee with guidance regarding the areas that the subcommittee should investigate. Members provided the following suggestions:

- The subcommittee should consider what can be done to both prevent and contain fuel melt-through accidents. The NRC Staff should be urged to broaden its approach in this matter, and to consider mitigating devices inside the reactor vessel as well as outside.

- Identify those reactor safety research programs that have arisen out of Class-9 considerations, to determine whether the current research program is not fragmented, and that this area is covered as well as needed.

- Define those studies that the NRC Staff should pursue.
- Consider the implications of Class-9 accidents with respect to licensing.
- Try to define reasonable evacuation distances from an affected plant.
- Evaluate the potentials and consequences of steam explosions.

(For suggested core-melt scenarios, see Appendix XVII.)

C. Reevaluation and Resolution of Generic Items

The Committee agreed to defer full committee action regarding reevaluation and resolution of generic items applicable to LWRs (per assignments made during the 235th ACRS Meeting) until the 242nd ACRS Meeting (June).

D. Proposed Schedule for ACRS Review of NRC Staff Documents

The Committee was informed by the NRC Executive Director, that the Office of Nuclear Reactor Regulation plans to provide copies of documents regarding proposed changes in technical, policy/positions, rules and regulations, resolution of generic items, etc. to the Committee for comment during the same period of time they are available for public comment. The Committee agreed that it would decide on an appropriate time for ACRS participation in such proceedings on a case-by-case basis.

E. Change in DNBR for Combustion Engineering Reactors

With the Committee's concurrence, the Chairman appointed an ad hoc subcommittee, with Mr. Shewmon, Chairman, and also consisting of Messrs. Carbon, Etherington, Okrent, and Plesset, to review the changes in departure from nucleate boiling ratio for Combustion Engineering reactors identified with respect to the increase in operating power granted to Millstone 2. The Committee also agreed that the Operating Reactors Subcommittee would continue to review other requests for increases in power in accordance with the memo from R. F. Fraley to L. V. Gossick dated May 12, 1978 (see Appendix XVIII).

F. ACRS Reports and Letters

1. ACRS Participation in NRC Rulemaking on Radioactive Waste Storage and Disposal

The Committee approved a memorandum to the Commissioners accepting their request for ACRS participation in the NRC rulemaking on storage and disposal of radioactive waste from nuclear facilities, and requesting an extension for the development of ACRS' comments (see Appendix XIX).

2. Recommendations of President's Commission on ACRS' Role

The Committee provided the Commissioners with its comments on the recommendations of the President's Commission on the Three Mile Island Accident related to strengthening the ACRS' role (see Appendix XX).

3. Comments on Draft NUREG-0660

The Committee provided the Commissioners with its comments regarding the draft NUREG-0660 dated 12-10-79, Action Plan for Implementing Recommendations of the President's Commission and other studies of the TMI-2 accident (see Appendix XXI).

4. Request for User Requests and Other Memoranda

The Committee approved a memorandum to the NRC Executive Director for Operations requesting that user requests and other memoranda which identify safety research needs be provided to the Committee automatically. Further, the Committee recommended that new research requests emanating from the current activities of the Bulletins and Orders Task Force and other ongoing activities be reviewed and evaluated within a broad perspective of the overall needs and responsibilities of the NRC (see Appendix XXII).

5. Review of Siting Policies

The Committee considered a draft of a report, Review of Siting Policies (NUREG-0625), but did not complete the report at this meeting. Further consideration of this matter is scheduled for the 238th ACRS Meeting (February).

DELETION 4

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

January 2, 1980

DETAILED SCHEDULE AND OUTLINE
FOR DISCUSSION
237TH ACRS MEETING
JANUARY 10-12, 1980
WASHINGTON, DC

Thursday, January 10, 1980, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 12:30 P.M. — Executive Session (Open)
   1.1) 8:30 A.M.-8:45 A.M.: Chairman's Report (MP/RFF)
      1.1-1) Proposed low power operation of Sequoyah Nuclear Power Plant
   1.2) 8:45 A.M. - 12:30 P.M.: Proposed ACRS Annual Report on the NRC Safety Research Program (CPS et al./TGM/DZ et al.)

   (Portions of this session will be closed as necessary to discuss information the premature release of which would frustrate the ACRS ability to perform its statutory function).

12:30 P.M. - 1:30 P.M. — LUNCH

2) 1:30 P.M. - 5:30 P.M. — Meeting with NRC Staff (Open)
   2.1) Discuss proposed NRC action plan to implement the recommendations of the President's Commission and other studies on the Three Mile Island, Unit 2 accident

   (Portions of this session will be closed as necessary to discuss Proprietary information applicable to these items.)

3) 5:30 P.M. - 6:30 P.M. — Executive Session (Open)
   3.1) Discuss proposed methods to strengthen ACRS function (MWC/RFF)

Friday, January 11, 1980, Room 1046, 1717 H Street, NW, Washington, DC

4) 8:30 A.M. - 12:30 P.M. — Meeting with NRC Staff (Open)

- 4.1) 8:30 A.M.-10:30 A.M.: Discuss implementation of NRC Bulletins and Orders resulting from the TMI-2 accident and small LOCA analysis (WJM/MP/PB/ALB)
- 4.2) 10:30 A.M.-12:30 P.M.: Discuss proposed revision of NRC Criteria for Siting Nuclear Power Plants (NUREG-0625) (DWM/RM)

12:30 P.M. - 1:30 P.M. — LUNCH

5) 1:30 P.M. - 2:00 P.M. — Meeting with NRC Staff (Open)

- 5.1) Discuss anticipated ACRS Subcommittee activities
  - 5.1-1) Activities of ACRS Subcommittee on Class-9 Accidents (WK/GRQ)
  - 5.1-2) Proposed activities for ACRS review/resolution of generic matters applicable to light-water reactors (HE/JCM)
  - 5.1-3) ACRS review of proposed power level increases (HE/RFF)
- 5.2) Discuss anticipated activities
  - 5.2-1) Response to Commissioner Gilinsky's inquiry regarding ACRS report on the Pause in Reactor Licensing (DO)
  - 5.2-2) Response to NRC request for ACRS participation/comments regarding rulemaking on waste storage and disposal (SL/DWM)

6) 2:00 P.M. - 6:30 P.M.

Executive Session (Open)

6.1) Discuss proposed ACRS reports on:
- . NRC Safety Research Program
- . NRC Action Plan

(Portions of this session will be closed as necessary to discuss Proprietary Information and information the premature release of which would frustrate the ability of the Committee to accomplish its statutory function.)

Saturday, January 12, 1980, Room 1046, 1717 H Street, NW, Washington, DC

7) 8:30 A.M. - 12:30 P.M.

Executive Session (Open)

7.1) Discuss proposed ACRS reports/comments on:
- . NRC Safety Research Program
- . NRC Action Plan
- . Proposed methods to strengthen ACRS function
- . NRC Siting Criteria

(Portions of this session will be closed as necessary to discuss Proprietary Information and information the premature release of which would frustrate the ability of the Committee to accomplish its statutory function.)

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

8) 1:30 P.M. - 4:00 P.M.

Executive Session (Open)

8.1) 1:30 P.M.-2:30 P.M.: Reports of ACRS Subcommittee on:

- 8.1-1) Babcock and Wilcox Water Reactors - Dynamic Performance of B&W Plants with once-through steam generators (HE/RM)
- 8.1-2) BWR Reactors with Mk I containment - proposed acceptance criteria (MP/ALB)
- 8.1-3) ACRS Procedures (MP/RFF)

8.2) 2:30 P.M.-4:00 P.M.: Miscellaneous

- 8.2-1) Activities of ACRS members
- 8.2-2) Complete discussion of items considered during this meeting

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meeting on the Surry Station) until conclusion of business. Thursday, January 24, 8:30 a.m. until the conclusion of business.

The Subcommittee will review the status of unresolved generic safety items involving pressure vessels, steam generators, and other pressure boundary components in its cognizant area of review.

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 cognizant Designated Federal Employee, Mr. Elpidio G. Igne (telephone 202/634-3314) between 8:15 a.m. and 5:00 p.m., EST.

Dated: December 20, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-30694 Filed 12-27-79; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Subcommittee on the Surry Nuclear Station; Meeting

The ACRS Subcommittee on the Surry Nuclear Station will hold a meeting on January 23, 1980 in Room 1046, 1717 H St., NW, Washington, DC 20555 to continue its review of the Surry Station steam generator replacement program. Notice of this meeting was published December 20, 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, January 23, 1980; 8:30 a.m. Until the Conclusion of Business

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, the Virginia Power and Electric Company, and their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

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 cognizant Designated Federal Employee, Mr. Gary Quittschreiber (telphone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be discussed at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555 and at the Swem Library, College of William and Mary, Williamsburg, VA 23185.

Dated: December 20, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-30695 Filed 12-27-79; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Subcommittee on Licensee Event Reports (LERs); Meeting

The ACRS Subcommittee on Licensee Event Reports (LERs) will hold an open meeting on January 23, 1980, in Room 1167, 1717 H St., NW., Washington, DC 20555. Notice of this meeting was published December 20, 1979.

The agenda for subject meeting shall be as follows:

Wednesday, January 23, 1980; 11:30 a.m. Until Conclusion of Business

The Subcommittee will discuss the evaluation of LER information with representatives of NRC's newly formed Office of Analysis and Evaluation of Operational Data.

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 cognizant Designated Federal Employee, Dr. Andrew L. Bates (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m. EST.

Dated: December [illegible], 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-30696 Filed 12-27-79; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards Nuclear Regulatory Commission; Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b.), the Advisory Committee on Reactor Safeguards will hold a meeting on January 10-12, 1980, in Room 1046, 1717 H Street, NW., Washington, DC. Notice of this meeting was published in the **Federal Register** on December 20, 1979.

The agenda for the subject meeting will be as follows:

Thursday, January 10, 1980

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 to the U.S. Congress regarding the NRC Safety Research Program.

Portions of this session will be closed as necessary to discuss information the premature disclosure of which would frustrate the ACRS ability to perform its statutory function.

1:30 P.M.-5:30 P.M.: Meeting with NRC Staff (Open)—The Committee will hear and discuss reports from representatives of the NRC Staff regarding proposed NRC action plans to implement recommendations of the President's Commission and other studies of the Three Mile Island, Unit 2 accident.

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

5:30 P.M.-6:30 P.M.: Executive Session (Open)—The Committee will discuss proposed methods to strengthen the role of the ACRS in accordance with the recommendations of the President's Commission on the accident at Three Mile Island.

Friday, January 11, 1980

8:30 A.M.-10:30 A.M.: Meeting with NRC Staff (Open)—The Committee will hear reports and will discuss proposed plans for NRC Implementation of the

Bulletins and Orders resulting from the accident at Three Mile Island.

10:30 A.M.–12:30 P.M.: Meeting with NRC Staff (Open)—The Committee will hear presentations and discuss proposed changes in NRC criteria for siting of nuclear powerplants (NUREG–0625).

1:30 P.M.–6:30 P.M.: Executive Session (Open)—The Committee will discuss proposed comments and recommendations regarding the NRC Safety Research Program and the NRC plans to implement the recommendations of the President's Commission and others on TMI-2.

Portions of this session will be closed as necessary to discuss Proprietary Information and information the premature disclosure of which would frustrate the ACRS ability to perform its statutory function.

Saturday, January 12, 1980

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 safety research program; NRC plans to implement recommendations of the President's Commission and others on TMI-2; implementation of NRC Bulletins and Orders resulting from the TMI-2 accident; proposed changes in NRC criteria for siting nuclear facilities; and proposed changes to strengthen the ACRS role.

The Committee will hear reports from its Subcommittees on Babcock and Wilcox Light Water Reactors and on ACRS Procedures.

The future schedule for Committee activities will also be discussed.

The Committee will complete discussion of items considered during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information related to matters being considered, and to protect information the premature release of which would frustrate the ACRS ability to perform its statutory function.

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on October 1, 1979 (44 FR 56408). In accordance with these procedures, oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Committee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a telephone call to the ACRS Executive Director (R. F. Fraley) prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

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

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

Dated: December 20, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-39597 Filed 12-27-79; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-318]

Baltimore Gas & Electric Co.; Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. DPR-69 issued to Baltimore Gas & Electric Company, which revised Technical Specifications for operation of the Calvert Cliffs Nuclear Power Plant, Unit No. 2 (the facility) located in Calvert County, Maryland. The amendment is effective as of its date of issuance.

The amendment revises the Appendix A technical Specifications of the facility to increase the measurement/calculational uncertainties for peaking factors F_r and F_q from 5.1 and 5.8 percent to 6.9 and 7.0 percent, respectively.

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

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

For further details with respect to this action, see (1) the application for amendment dated August 27, 1979, as supplemented October 1, 1979, (2) Amendment No. 24 to License No. DPR-69, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Calvert County Library, Prince Frederick, Maryland. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Md., this 11th day of December 1979.

For the Nuclear Regulatory Commission.

Robert W. Reid,

Chief, Operating Reactors Branch #4, Division of Operating Reactors.

[FR Doc. 79-39598 Filed 12-27-79; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-251]

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

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of the Turkey Point Nuclear Generating, Unit No. 4 (the facility) located in Dade County, Florida. The amendment is effective as of the date of issuance.

ACRS-1710

CERTIFIED

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JANUARY 10-12, 1980

ATTENDEES
237th ACRS Meeting
January 10-12, 1980

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Milton S. Plesset, Chairman
J. Carson Mark, Vice-Chairman
Myer Bender
Max W. Carbon
Jesse C. Ebersole
Harold Etherington
William Kerr
Stephen Lawroski
Harold W. Lewis
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
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

CONSULTANTS

C. Michelson
W. Lipinski

NRC ATTENDEES

237th ACRS Meeting
Jan. 10-12, 1980

Charles Kelber, RES
Peter Riehm, NRR
Guy Cunningham, ELD
R. H. Vollmer, NRR
R. P. Denise, NRR
Ronald Scroggins, RES
James A. Norberg, EMSB
Robert Purple, RHSS
Joseph Scinto, ELD

Div. of Project Management

W. Kane
B. Wilson
D. Ross

Div. of Systems Safety

W. Hodges
Brian Sheron
Philip R. Matthews
Roger Mattson
Z. Rosztoczy
V. Benaroya

PUBLIC ATTENDEES

237th ACRS MEETING

Jan. 10-12, 1980

January 10, 1980

Bill Horin, Debevoise & Liberman, Wash., DC
Mr. Leyse, EPRI
David Chaffee, Bus. Publishers, Inc.
Roger W. Huston, Consumers Power Co.
M. Banergi, Ebasco, NY
S. R. Phelps, EEI
L. S. Gifford, GE
R. Ross, O&M
H. Hamada, TIPCO

January 11, 1980

Rick Muench, Westinghouse
C. B. Brinkman, Combustion Engineering
L. S. Gifford, General Electric
M. Banerjei, Ebasco
R. Borsum, Babcock and Wilcox
Mr. Leyse, Electric Power Research Inst.
K. L. Huber, Westinghouse
T. Rogers, Pacific Gas and Electric
Joanne Dann, McGraw-Hill
T. Martin, NUTECH
R. Ross, D&M
C. Guchmel, Stone and Webster
S. R. Phyn, EEI
J. H.Baroff, Self
P. Higgins, Atomic Industrial Forum
B. Horin, D&L
Lynn Connor, Doe-Search Associates

APPENDIX II

FUTURE AGENDA

FEBRUARY

Bulletins and Orders	4 hours
RSR Report	5 hours
MK I Containment Acceptance Criteria	3 hours
TMI-1 Restart Review	4 hours
Revised Siting Rules	2 hours
Subcommittee Report re Response to Rep. Udall on Equipment Failure Rates and Davis Besse/ Rancho Seco incidents	1 hour
Subcommittee Reports	1 1/4 hour

ATWS
Surry Steam Generator Replacement
Fire Protection
La Cross Fuel Racks
Wolf Creek Seismic Design

Meeting with NRC on Recent Operating Problems:

Safe Shutdown Boron Capability (Midland Plant)

Inadequate Separation of Electrical Equipment and Systems at nuclear plants (e.g. WPPSS No. 2)

Loss of 480 volt bus and related plant equipment (San Onofre)

Contamination of Instrument Air with Service Air (Turkey Point)

Point Beach Steam Generator Tube Degradation

North Anna 1 Steam Generator Tube Degradation

Miscellanous

Report on Low Power Testing of Sequoyah et.al. nuclear plants

MARCH

Transient Stability of B&W Plants with Once-Through Steam Generators

Clarification of ACRS Report on the Pause in Licensing

FNP Core Ladle (conceptual design)

GETR Restart (seismic character of site)

ACRS comments re proposed revision of 10 CFR Part 50, Appendix K re clad ballooning

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

APPENDIX III

January 12, 1980

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS, AND TOURS

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

Most hotels currently being used by ACRS Members in the downtown Washington and Bethesda areas require a guaranteed reservation if arrival is scheduled after 6:00 p.m. Failure to use a room under these conditions involves 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
J. Jacobs

JANUARY

23 Surry Nuclear Station (GRQ)- HE, DWM. PS, MB

23-24 Metal Components (EI) - PS, MB, HE

25 ATWS (PB) - WK, JE, CM, JR

1/31-2/1 TMI, Unit 1 (Harrisburg, PA) (RM) - HE, JE, SL, HL, DWM, WM

FEBRUARY

6 Reliability and Probabilistic Ass. (GRQ) - DO, MB, JE, WK, JCM, HL, CPS

7-9 238th ACRS Mtg.

14 ECCS/Fuels (Clad ballooning models) (AB) - MP, PS, HE

20-21 Plant Arrangements (RKM) - MB, JE, SL, CM, JR

22 GETR (San Francisco, CA) (EI) - WK, CM, DO

6 PROCEDURES (RFF) 1:00 PM MP, MJC, MB, WK, SL, DWM, DO, CPS

MARCH

4 B&W Water Reactors (RM) - HE, JCE, JR, WM

5 Reg. Activities (SD) - WK, MB, HE, JR

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

CHAIRMAN

December 26, 1979

APPENDIX IV
PROPOSED LICENSING OF SEQUOYAH FOR LOW POWER TESTING

Mr. S. David Freeman
Chairman of the Board
Tennessee Valley Authority
Knoxville, Tennessee 37902

Dear Chairman Freeman:

Your December 3, 1979 letter to Dr. Hendrie requested that the Nuclear Regulatory Commission consider permitting TVA to conduct certain activities including fuel loading, zero power physics testing, special testing and operator training at the Sequoyah Nuclear Plant Unit 1 at no greater than five percent power.

Your proposal is an interesting one. While a distinction can be made between the risk to public health and safety from a special testing program at low power and operation at full power, further discussions between our respective staffs will be required to explore the details of your proposed program. However, until the Commission has completed the reviews necessary to ensure that operating reactors are adequately responding to the lessons learned from the TMI accident, only limited resources will be available for reviews associated with issuing new operating licenses.

Subject to this resource constraint, I have asked the staff to review your proposal and to make a recommendation to the Commission in this regard. The final decision on this matter will, of course, reside with the Commission.

I would also like to note that Commissioners Kennedy and Hendrie prefer that the NRC staff proceed promptly in this matter, particularly in light of the ACRS's strong endorsement of your proposal. They believe that the necessary resources can and should be made available under these circumstances.

Sincerely,

J.F. Ahearne

John F. Ahearne

TENNESSEE VALLEY AUTHORITY

KNOXVILLE, TENNESSEE 37902

OFFICE OF THE BOARD OF DIRECTORS

December 3, 1979

Joseph M. Hendrie, Chairman
U.S. Nuclear Regulatory Commission
1717 H Street, NW.
Washington, DC

Dear Chairman Hendrie:

We believe that there are advantages to be gained by pursuing certain limited activities in the case of those power plants where construction has been completed during the Commission's "pause" in issuing new construction permits and operating licenses, particularly where it can be demonstrated that the owner utility has taken the initiative in improving and promoting safety. We believe that the TVA program meets or exceeds the recommendations of the President's Commission and the NRC staff's short term lessons learned requirements. You will recall that TVA completed a detailed review of our nuclear program in May. TVA has implemented a series of major improvements as a result of that review. More recently, a special TVA nuclear safety task force has completed a review of the report by the President's Commission. This task force concluded, and we agree, that TVA meets all of the recommendations of the Kemeny commission report.

We are therefore asking that the NRC permit certain activities including fuel loading, zero power physics testing, "special" testing and operator training to be conducted at the Sequoyah Nuclear Plant unit 1.

We believe that using the Sequoyah unit to conduct tests of the natural circulation cooling phenomena is particularly advantageous at this time. There are questions about this mode of cooling under normal and degraded conditions which can be resolved by full scale demonstration testing. Since the fuel in the reactor at Sequoyah would not have been operated at significant power, the inventory of fission products present would be minimal.

We believe that significant testing and operator training can be performed which would permit operation of the reactor at no greater than five percent power. A summary description of the type of tests which TVA could perform is included as Enclosure 1.

Construction necessary for fuel loading was completed at Sequoyah unit 1 on November 15, 1979. The NRC staff has completed the review of the operating license application with the exception of items related to Three Mile Island. The TVA response to the NRC Staff Short Term Lessons

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Learned was submitted September 7, 1979, and your staff has been working with TVA to resolve these issues. Enclosed for your information are the TVA responses to the President's Commission on the Accident at Three Mile Island recommendations.

Our fuel loading and zero power testing would take approximately six weeks. We would then be able to begin special testing in mid-February. Should events in the interim dictate that modifications to the plant are required, the nuclear fuel could be removed from the reactor vessel and stored in the spent fuel pool with no hazard to the public health and safety.

Additionally, we know you will be interested to know that TVA has initiated a comparative risk analysis of the Sequoyah plant auxiliary feedwater system. This analysis will be complete by the time the proposed low power tests are finished. In addition, we are evaluating other areas of the Sequoyah plant where meaningful risks assessments could be completed before full power operation.

Very truly yours,

Dave Freeman

S. David Freeman
Chairman of the Board

Enclosures

Enclosure 1

SEQUOYAH NUCLEAR PLANT UNIT 1

SUMMARY OF SPECIAL TESTS

Prior to core loading, the plant nuclear instrumentation and temporary nuclear instrumentation will be checked out. Plant systems requiring boration will be borated to the specified concentration.

Following core loading and prior to initial criticality, baseline testing will be performed with the core completely assembled. Major items to be performed are moveable detector system checkout, rod drive mechanism and rod cluster control assembly operation tests, reactor internal vibration measurements, pressurizer system optimization and reactor coolant loop flow coastdown measurements.

After the reactor is brought critical, low power physics testing will begin. Plant baseline parameter measurements will be taken, reactivity measurements conducted, temperature coefficients determined, and boron endpoint measurements made. Reactivity measurements include integral and differential bank worth tests, minimum shutdown margins verification, and determination of the affect of a rod ejection.

These tests are the normal tests performed to verify that integrated system response meets design assumptions, verify the core design basis, and verify that adequate shutdown margin exists throughout cycle 1.

They are described in more detail in the Sequoyah Nuclear Plant Final Safety Analysis Report.

The following special tests conducted prior to exceeding 5 percent power are intended to provide a significant demonstration of reactor operation in the natural circulation mode under both normal and certain degraded conditions. These tests will also provide significant operator training and experience under these conditions. The tests will be repeated such that each operating shift participates in each test.

To simulate decay heat, the reactor will be operated at less than 5 percent power with the reactor coolant pumps tripped. This mode of operation will closely approximate natural circulation conditions (with subcooling) following a reactor trip from full power after several months of power operation.

Since detailed test procedures and safety evaluations for these tests have not been completed, some modifications in test scope or detail may be required. Test durations and methods of power level control will be provided in the detailed test procedures and evaluation. Once test procedures have been written and corresponding safety evaluations developed for the special tests, they will be submitted to NRC along with appropriate license amendments. We intend to have Westinghouse Electric Corporation review these special test procedures as they are doing with other selected emergency procedures.

I. Natural Circulation Verification

Purpose

Verify establishment of natural circulation in the primary system

Initial Conditions

Reactor Coolant Pumps operating

Steam Generators being fed by normal feedwater supply

Pressurizer Heater controlling pressure

Reactor Power Z 3%

Normal primary system temperature and pressure

Test Description

Test will be initiated by tripping of all reactor coolant pumps. Operator will verify establishment of natural circulation by observing response of the hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess core flow distribution.

II. Natural Circulation with Simulated Loss of Offsite Power

Purpose

Verify that natural circulation cooling can be established and maintained following loss of offsite power.

Initial Conditions

Reactor Power 1%.

Reactor Coolant Pumps operating.

Auxiliary Feed System operating on offsite power.

Pressurizer Heaters controlling pressure.

Normal primary system temperature and pressure.

Test Description

Test will be initiated by a simulated loss of offsite power. Reactor coolant pumps will be tripped, auxiliary feed pump and pressurizer heater loads will be transferred to diesel power. Operator will verify establishment of natural circulation by observing response of hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess the core flow distribution.

III. Natural Circulation with Loss of Pressurizer Heaters

Purpose

Verify establishment of natural circulation and determine the rate of decrease of margin to saturation while in this mode and the ability to reestablish margin through cooldown and makeup.

Initial Conditions

Reactor Power Z 3%

Reactor Coolant Pumps operating

Secondary system steam flow adjusted to maintain constant primary coolant temperature

Steam generators being feed by normal feedwater supply

Pressurizer heaters controlling pressure

Test Description

Test will be initiated by tripping pressurizer heaters and reactor coolant pumps. Establishment of natural circulation will be verified by observing response of hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess the core flow distribution. The operator will observe the saturation meter to verify margin. Prior to reaching saturation, secondary side steam flow will be increased to affect cooldown and reestablishment of saturation margin will be verified. In conjunction with cooldown, the operator feeds the primary system to compensate for shrinkage.

IV. Effect of Steam Generator Isolation (Secondary Side) on Natural Circulation

Purpose

Verify the effects of steam generator isolation (secondary side) on natural circulation.

Initial Conditions

Reactor Power 3%

All steam generators fed by normal feedwater supply

Reactor coolant pumps on

Secondary system steam flow adjusted to maintain constant temperature

Test Description

Trip reactor coolant pumps and verify establishment of natural circulation. Cooldown using steam dumps to provide sufficient margin to steam generator safeties. Isolate steam generators one at a time until three are isolated or primary system temperature starts to increase. Hot and cold leg temperatures will be monitored to ensure that sufficient heat is being removed by the natural circulation process. The steam generators will be returned to service one at a time and the reestablishment of natural circulation will be verified in each loop. Core exit thermocouples will be monitored to assess core flow distribution.

V. Natural Circulation at Reduced Pressure

Purpose

1) Verify operation and test accuracy of primary system saturation meter.
2) Provide operations personnel with online experience in using saturation meter to monitor and control margin to saturation.
3) Provide operational verification so that changes in saturation margin will not affect natural circulation provided adequate margin to saturation exists.

Initial Conditions

Reactor Power Z 3%

Reactor coolant pumps operating

Steam generators being fed by normal feedwater supply

Pressurizer heaters controlling pressure

Reactor coolant system pressure normal

Secondary system steam flow adjusted to maintain constant temperature

Test Description

Test is initiated by tripping of reactor coolant pumps and verifying establishment of natural circulation. Primary system pressure will be reduced as primary system temperature is held constant. Accuracy of saturation meter will be verified during pressure reductions. The effect of each pressure reduction on natural circulation will be observed. Core exit thermocouples will be monitored to assess core flow distribution.

VI. Determine the cooldown capability of the charging and letdown system

Purpose

Determine the cooldown capability of the charging and letdown system with the secondary plant isolated.

Initial Conditions

Reactor shutdown

Pressurizer heaters controlling pressure

Reactor coolant pumps running

All steam generators fed by normal feedwater flow

Test Description

Trip three reactor coolant pumps. Cooldown using steam dumps to provide margin to steam generator safeties. Isolate all steam generators. Establish charging and letdown for maximum cooling capability. Verify the cooldown capability of the charging and letdown system from the hot and cold leg temperatures in the active loop. This will be accomplished by periodically interrupting feed and bleed to permit heatup. Core exit thermocouples will be monitored to assess core flow distribution.

VII. Simulated Loss of All Onsite and Offsite AC Power

Purpose

To verify:

1. Hot standby conditions can be maintained,
2. Auxiliary feedwater can be controlled by manual means; i.e., with loss of AC power and control air,
3. Critical plant operations can be performed using emergency lighting,
4. Ability of 125-volt battery to supply 125-volt vital AC, and
5. Selected equipment areas do not exceed maximum design temperature.

Initial Conditions

Reactor critical at N1 percent power.

Reactor Coolant Pumps operating.

Pressurizer heaters controlling primary system pressure.

Test Description

Test will be initiated by:

1. Tripping RCP's and pressurizer heaters,
2. Tripping auxiliary building and control building lighting boards,
3. Removing AC power from auxiliary feedwater components and main steam power reliefs,
4. Tripping selected space and equipment coolers,
5. Tripping vital battery chargers and AC power to inverter,
6. Isolating main feedwater and main steam lines,
7. Establishing manual control of auxiliary feedwater,
8. After two hours, terminating the test by restoring AC power and returning equipment to normal service,
9. Shutdown reactor, and
10. Cooling down primary system and placing RHR system in service.

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 5, 1979

Mr. Steven A. Varga, Assistant Director (Acting)
Light Water Reactors Branch No. 3
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 1002
LQA/EAB:pwc

Docket No. 50-339

Dear Mr. Varga:

Messrs. E. A. Baum and B. R. Sylvia have reported to us on the results of recent licensing review activity by the Staff for North Anna Unit 2. It was encouraging to learn from them that a Task Force under the direction of Mr. P. J. Williams, Jr. has been assigned to concentrate on the review of North Anna 2, and further that positive direction has been given to move ahead with the review of our application.

They also were in attendance yesterday at an ACRS Subcommittee meeting which you, along with Mr. Denton, Mr. Vassallo, Mr. Williams and others of the Staff attended concerning a briefing on a proposed special test program which the Tennessee Valley Authority has agreed to carry out at a 3-5% reactor power level at its Sequoyah plant.

We have studied the summary of these special tests which are designed to verify establishment of natural circulation in the primary system, and to verify that it can be maintained under various operating conditions while their effects are observed. We feel, like you, that this test program has considerable merit and not only will provide good information to the operators, but also, will enable the plant's staff to verify procedures and design predictions. Further it will upgrade reactor operator training since these tests will emphasize key aspects of operation which were observed in the minutes and hours which followed the TMI-2 incident. Certainly the Tennessee Valley Authority people are to be commended for this imaginative and creative test program for operators.

We feel that these same tests will be helpful if conducted on North Anna 2. We therefore have followed the very same pattern, (enclosure) for North Anna and we commit to perform them on this unit. It is our intent to develop the necessary test procedures for North Anna Unit 2 this month and we will submit them to the NRC for review, comment and approval prior to implementation.

While North Anna 2 has already received a favorable letter from the Advisory Committee on Reactor Safeguards, we are hopeful that the staff would be in a position to also issue a 5% reactor power operating license in the very near future, once they have verified that the plant is completed in accordance with the operating license application.

Very truly yours,

B001
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1/1

C. M. Stallings
Vice President-Power Supply
and Production Operations

cc: Mr. J. P. O'Reilly

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NORTH ANNA POWER STATION UNIT 2

SUMMARY OF SPECIAL TESTS

Prior to core loading, the plant nuclear instrumentation and temporary nuclear instrumentation will be checked out. Plant systems requiring boration will be borated to the specified concentration.

Following core loading and prior to initial criticality, baseline testing will be performed with the core completely assembled. Major items to be performed are moveable detector system checkout, rod drive mechanism and rod cluster control assembly operation tests, reactor internal vibration measurements, pressurizer system optimization and reactor coolant loop flow coastdown measurements.

After the reactor is brought critical, low power physics testing will begin. Plant baseline parameter measurements will be taken, reactivity measurements conducted, temperature coefficients determined, and boron endpoint measurements made. Reactivity measurements include integral and differential bank worth tests, minimum shutdown margins verification, and determination of the affect of a rod ejection.

These tests are the normal tests performed to verify that integrated system response meets design assumptions, verify the core design basis, and verify that adequate shutdown margin exists throughout cycle 1.

They are described in more detail in the North Anna Final Safety Analysis Report.

The following special tests conducted prior to exceeding 5 percent power are intended to provide a significant demonstration of reactor operation in the natural circulation mode under both normal and certain degraded conditions. These tests will also provide significant operator training and experience under these conditions. The tests will be repeated such that each operating shift participates in each test.

To simulate decay heat, the reactor will be operated at less than 5 percent power with the reactor coolant pumps tripped. This mode of operation will closely approximate natural circulation conditions (with subcooling) following a reactor trip from full power after several months of power operation.

Since detailed test procedures and safety evaluations for these tests have not been completed, some modifications in test scope or detail may be required. Test durations and methods of power level control will be provided in the detailed test procedures and evaluation. Once test procedures have been written, they will be submitted to the NRC for their review and approval. We intend to have Westinghouse Electric Corporation review these special test procedures.

I. Natural Circulation Verification

Purpose

Verify establishment of natural circulation in the primary system

Initial Conditions

Reactor Coolant Pumps operating

Steam Generators being fed by normal feedwater supply

Pressurizer Heater controlling pressure

Reactor Power = 3%

Normal primary system temperature and pressure

Test Description

Test will be initiated by tripping of all reactor coolant pumps. The ~~Operator will~~ verify establishment of natural circulation by observing response of the hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess core flow distribution.

II. Natural Circulation with Simulated Loss of Offsite Power

Purpose

Verify that natural circulation cooling can be established and maintained following loss of offsite power.

Initial Conditions

Reactor Power 1%.

Reactor Coolant Pumps operating.

Auxiliary Feed System operating on offsite power.

Pressurizer Heaters controlling pressure.

Normal primary system temperature and pressure.

Test Description

Test will be initiated by a simulated loss of offsite power. Reactor coolant pumps will be tripped, auxiliary feed pump and pressurizer heater loads will be transferred to diesel power. The operator will verify establishment of natural circulation by observing response of hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess the core flow distribution.

III. Natural Circulation with Loss of Pressurizer Heaters

Purpose

Verify establishment of natural circulation and determine the rate of decrease of margin to saturation while in this mode and the ability to reestablish margin through cooldown and makeup.

Initial Conditions

Reactor Power = 3%

Reactor Coolant Pumps operating

Secondary system steam flow adjusted to maintain constant primary coolant temperature

Steam generators being fed by normal feedwater supply

Pressurizer heaters controlling pressure

Test Description

Test will be initiated by tripping pressurizer heaters and reactor coolant pumps. Establishment of natural circulation will be verified by observing response of hot leg and cold leg temperature instrumentation in each loop. Core exit thermocouples will be monitored to assess the core flow distribution. The operator will observe the saturation meter to verify margin. Prior to reaching saturation, secondary side steam flow will be increased to affect cooldown and reestablishment of saturation margin will verified. In conjunction with cooldown, the operator feeds the primary system to compensate for shrinkage.

IV. Effect of Steam Generator Isolation (Secondary Side) on Natural Circulation

Purpose

Verify the effects of steam generator isolation (secondary side) on natural circulation.

Initial Conditions

Reactor Power 3%

All steam generators fed by normal feedwater supply

Reactor coolant pumps on

Secondary system steam flow adjusted to maintain constant temperature

Test Description

Trip reactor coolant pumps and verify establishment of natural circulation. Cooldown using steam dumps to provide sufficient margin to steam generator safeties. Isolate steam generators one at a time until two are isolated or primary system temperature starts to increase. Hot and cold leg temperatures will be monitored to ensure that sufficient heat is being removed by the natural circulation process. The steam generators will be returned to service one at a time and the reestablishment of natural circulation will be verified in each loop. Core exit thermocouples will be monitored to assess core flow distribution.

V. Natural Circulation at Reduced Pressure

Purpose

Verify operation and test accuracy of primary system saturation meter.

Provide operations personnel with online experience in using saturation meter to monitor and control margin to saturation.

Provide operational verification so that changes in saturation margin will not affect natural circulation provided adequate margin to saturation exists.

Initial Conditions

Reactor Power = 3%

Reactor coolant pumps operating

Steam generators being fed by normal feedwater supply

Pressurizer heaters controlling pressure

Reactor coolant system pressure normal

Secondary system steam flow adjusted to maintain constant temperature

Test Description

Test is initiated by tripping of reactor coolant pumps and verifying establishment of natural circulation. Primary system pressure will be reduced as primary system temperature is held constant. Accuracy of saturation meter will be verified during pressure reductions. The effect of each pressure reduction on natural circulation will be observed. Core exit thermocouples will be monitored to assess core flow distribution.

VI. Determine the cooldown capability of the charging and letdown system

Purpose

Determine the cooldown capability of the charging and letdown system with the secondary plant isolated.

Initial Conditions

Reactor shutdown

Pressurizer heaters controlling pressure

Reactor coolant pumps running

All steam generators fed by normal feedwater flow

Test Description

Trip [illegible] reactor coolant pumps. Cooldown using steam dumps to provide margin to steam generator safeties. Isolate all steam generators. Establish charging and letdown for maximum cooling capability. Verify the cooldown capability of the charging and letdown system from the hot and cold leg temperatures in the active loop. This will be accomplished by periodically interrupting feed and bleed to permit heatup. Core exit thermocouples will be monitored to assess core flow distribution.

VII. Simulated Loss of All Onsite and Offsite AC Power

Purpose

To verify:

1. Hot standby conditions can be maintained,
2. Auxiliary feedwater can be controlled by manual means; i.e., with loss of AC power and control air,
3. Critical plant operations can be performed using emergency lighting,
4. Ability of 125-volt battery to supply 125-volt vital AC, and
5. Selected equipment areas do not exceed maximum design temperature.

Initial Conditions

Reactor critical at ~1 percent power.

Reactor Coolant Pumps operating.

Pressurizer heaters controlling primary system pressure.

Test Description

Test will be initiated by:

1. Tripping RCP's and pressurizer heaters,
2. Tripping auxiliary building and control building lighting boards,
3. Removing AC power from auxiliary feedwater components and main steam power reliefs,
4. Tripping selected space and equipment coolers,
5. Tripping vital battery chargers and AC power to inverter,
6. Isolating main feedwater and main steam lines,
7. Establishing manual control of auxiliary feedwater,

8. After two hours, terminating the test by restoring AC power and returning equipment to normal service,

9. Shutdown reactor, and

10. Cooling down primary system and placing RHR system in service.

PROJECT STATUS REPORT
ACRS Subcommittee on Reactor Safety Research
Meeting of January 9, 1980

Purpose:

To continue the discussion regarding the preparation of the Annual ACRS Report to Congress on NRC Reactor Safety Research.

Presentations:

Drs. Budnitz and Murley are expected to be present to discuss OMB changes to the FY-80 budget supplement and to FY-81 budget. Tom Murley has promised to provide the ACRS with copies of the President's NRC research budget at that time.

Status of Activities:

- The President's State-of-the-Union address is planned for January 23, 1980. Budget information can be openly discussed after that.
- Congress is scheduled to reconvene on January 22, 1980. Initial conference of House and Senate committees on NRC FY-80 budget has taken place on the Authorization Act. The next meeting is expected after January 22. Congressional Affairs expects no major problems between House and Senate Committees with the level of research fundings, and with the FY 80 budget supplement.

Draft Report Status:

- A technical chapter for each of the line items has been prepared by Dot, Project Engineer or me in the format suggested in Dr. Siess's memo of November 9, 1979 from information provided by ACRS authors. We have tried to preserve the comments and recommendations.
- Dot has prepared the attached format based on Dr. Siess's memo.
- Dot is compiling a list of research recommendations included in ACRS letters since the Committee's last report to Congress. She hopes to have the compilation available when you need it.

Things to be Accomplished:

(My views)

- Agree to overall format
- Decide on how much will be included on research priorities
- Complete initial reading of each technical chapter
- Agree on general conclusions of report.

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FY 8[illegible] SUMMARY
(Budget in Millions)

	SUPPLEMENT					TOTAL BUDGET			
	RES (6/15/79)	BRG ([illegible] 0603)	ACRS (on RES)	EDO (7/23/79)	Commission Allowance (8/14/79)	Budget	Cong. Approp.	Cong. & Comm. Allow	Congress (1/2/80)
1. Systems Engineering	$ 8.1	$ 6.5	----	$ 6.5	$ 6.5	$ 34.8	$ 34.8	$ 41.3	41.3
2. LOFT	2.0	2.0	----	2.0	$ 2.0	42.9	42.1	44.1	42.3
3. Code Development	3.5	3.1	----	3.1	$ 3.1	8.9	8.9	12.0	12.0
4. Fuel Behavior	5.6	5.6	----	5.6	$ 5.6	23.1	22.1	27.7	27.6
5. Primary System Integrity	1.0	1.0	----	1.0	$ 1.0	8.6	9.0	10.0	8.6
6. Seismic Engineering Safety	2.0	2.0	----	2.0	2.0	10.0	8.4	10.4	10.5
7. Reactor Environmental Effects	0.7	0.7	----	0.7	0.7	3.8	3.2	3.9	7.0*
8. Waste Management	0.0	3.0	----	3.0	3.0	6.7	5.6	8.6	8.6
9. Safeguards	0.2	0.0	----	0.0	0.0	5.0	4.0	4.0	4.0
10. Risk Assessment	3.3	3.3	----	3.3	3.3	5.7	5.2	8.5	8.5
11. Improved Reactor Safety	3.4	($ 3.4)	Restore funds	($ 3.4)	0.0	1.0	1.0	1.0	1.0
TOTAL (1-11)	$29.8	$27.2	$27.2	$27.2	$27.2	$150.5	$144.3	$171.5	$171.4
12. Fast Breeder	----	----	----	----	-	$ 13.7	$ 13.7	$ 13.7	13.7
13. Adv. Converter	----	----	----	----	-	0.0	1.7	1.7	1.7
14. Fuel Cycle	----	----	----	----	----	3.8	3.1	3.1	-
TOTAL (1-14)	$29.8	$27.2	$27.2	$27.2	$27.2	$168.0	$162.8	$190.0	$186.8

*Reactor Environmental Effec
include Fuel Cycle funds.

( ) = funds set aside

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FY 81 BUDGET SUMMARY

(Budget In Millions)

	RES (6/15/79)	BRG	ACRS (on RES)	RES (7/23/79)	EDO		RES REQ (8/14/79)	COMM ALLOW (8/14/79)	Congress (1/2/80)
1. Systems Engineering	$ 45.3	$ 32.8	----	$ 38.0	$ 35.6		$ 38.0	$ 38.0	38.0
2. LOFT	49.3	48.0	$ 49.3	48.3	48.0		48.0	48.0	43.0
3. Code Development	15.2	13.2	----	15.2	13.2		15.2	14.2	14.2
4. Fuel Behavior	28.5	27.9	----	27.9	27.9		27.9	27.9	27.9
5. Primary System Integirty	15.1	15.1	----	15.1	15.1		15.1	14.3	14.3
6. Seismic Engineering Study	19.9	13.9	17.0	19.9	13.9		19.9	16.9	16.9
7. Fast Breeder Reactor	22.1	(22.1)	22.1	22.1	(15.0)		22.1	18.0	5.0
8. Advanced Converters	3.9	(3.9)	3.9	3.9	(3.9)		3.9	2.5	0
9. Reactor Environmental Effects	9.8	6.2	6.2	9.8	6.2		9.8	7.8	12.2*
10. Fuel Cycle	5.9	5.0	----	5.0	5.0		5.0	4.4	-
11. Waste Management	15.9	12.9	----	14.8	12.9		14.8	13.6	13.6
12. Safeguards	6.7	4.9	4.9	5.7	5.3	($0.4)	5.7	4.9	4.9
13. Risk Assessment	12.6	7.3	12.6	12.6	9.0		12.6	12.6	12.6
14. Improved Reactor Safety	6.6	(6.6)	6.6	6.6	(6.6)		6.6	4.5	4.5
TOTAL	$256.8	$187.2		$244.9	$192.1		$244.6	$227.6	$207.1

( ) = funds set aside - not included in total sums

* Reactor Environmental Effects includes fuel cycle funds.

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Staff Chapter Assignments

Systems Engineering ....Andy Bates

LOFT......................................Andy Bates

Code Development..........................Andy Bates

Fuel Behavior.............................Paul Boehnert

Primary System Integrity..................Al Igne

Reactor Environmental Effects.............Rags Muller

Fuel Cycle................................Peter Tam

Waste Management..........................Peter Tam

Seismic Engineering.......................Dick Savio

Advanced Reactors.........................Dick Savio

Safeguards................................Dick Major

Risk Assessment...........................Gary Quittschreiber

Improved Reactor Safety...................Sam Duraiswamy

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

January 4, 1980

ACRS Members

OMB RECOMMENDATION ON THE FY-81 BUDGET IN FAST BREEDER REACTORS

The Commission's recommendation for the FY-81 budget ($18.0M) Fast Breeder Reactor line item has been rejected by OMB. They have, instead, recommended that the budget for FY-81 be $5.0M and that these funds be used to terminate the NRC program in this area. OMB has also recommended a severe cut in DOE's requested funding for breeder systems. Their requested funding was $550M and the OMB recommendation is $320M. I will keep you informed as to future developments.

for Mary Ellen Almond

R. Savio
Staff Engineer

cc: T.G. McCreless
M.W. Libarkin
R.F. Fraley

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GISA

ackground Info

TABLE 3.1
FY 81 BUDGET

	BUDGET (In millions) RES	BRG	Commission Mark
1. SYSTEMS ENGINEERING			
a. Semiscale	$ 8.1	$ 8.1	8.1
b. Blowdown & Reflood Heat Transfer	8.4	8.4	8.4
c. 3-D Flow Distribution	12.0	10.0/2.0(1)	10.0
d. ECC Bypass Research	0.9	0.9	0
e. Model Development Experiments	3.5	3.5	3.5
f. Operational Safety	9.8	0.0/9.8(2)	6.1
g. Technical Support	2.6	1.9	1.9
	$45.3	$32.8	$38.0

(1) BRG set aside for NRC consideration due to change in scope of the effort and also because $1 million is for contingencies not included in NRC directed ceiling of $59 million for this program.

(2) BRG set aside for NRC consideration because this funding is generally for new efforts proposed by RES and out-year impacts reflect significant growth.

	BUDGET (In millions) RES	BRG	
2. LOFT			
a. Program Planning and Analysis	$ 5.0	$ 5.0	Reallocation being planned
b. Fuel	8.3	8.3	
c. Operations	8.9	8.9	
d. Instrumentation	9.0	9.0	
e. Facility Support	11.3	10.0(3)	
f. Engineering and Physics	6.5	6.5	
g. Advanced Fuel Instrumentation	0.3	0.3	
	$49.3	$48.0	45.0

(3) BRG said that accuracy and timing of scheduling and testing not precise or exact enough that full request is required by FY 81.

TABLE 3.1 (Cont)

	BUDGET (In millions) RES	BRG	
3. CODE DEVELOPMENT			
a. Systems Codes	$ 6.3	$ 6.3	6.3
b. Component Codes	1.6	1.6	1.6
c. TRAC Assessment and Applications	7.3	5.3 (4)	6.3
	$15.2	$13.2	14.2

not firm

(4) BRG said that RES has not adequately demonstrated that $2.0 million TRAC application is not duplicative with the NRR program.

	BUDGET (In millions) RES	BRG
4. FUEL BEHAVIOR		
a. Clad and Fuel	$ 2.6	$ 2.6
b. Fuel Codes	1.5	1.5
c. In-Pile Testing (PBF)	16.1	16.1
d. In-Pile Testing (Other)	4.2	4.2
e. Fuel Melt	4.1	3.5 (5)
	$28.5	$27.9

no change. ✓

(5) BRG deleted low priority fuel melt effort for FNP.

	BUDGET (In millions) RES	BRG	
5. PRIMARY SYSTEM INTEGRITY			
a. Fracture Mechanics	$ 5.9	$ 5.9	5.1
b. Operating Effects	6.3	6.3	6.3
c. Nondestructive examination	2.9	2.9	2.9
	$15.1	$15.1	14.3

TABLE 3.1 (Cont)

	BUDGET (In millions) RES	BRG	
6. SEISMIC ENGINEERING SAFETY			
a. Structural Engineering	$6.0	$ 3.9	5.0
b. Mechanical Engineering	7.4	3.8	5.7
c. Site Safety	6.5	6.2	6.2
	$19.9	$13.9 (6)	16.9

(6) BRG reduction was based on low priority of this research (as assigned by RES). BRG level was said to be sufficient for RES to pursue a logical progression of effort started with FY 80 supplement.

	BUDGET (In millions) RES	BRG	
7. FAST BREEDER REACTOR			← to close out program
a. Analysis	$ 7.8	-	
b. Safety Test Facility Studies	.7	-	
c. Aerosol Release and Transport	3.0	-	
d. Materials Interactions	4.6	-	
e. Systems Integrity	6.0	-	
	$22.1	0.0/22.1 (7)	5.0

(7) BRG set aside entire amount for Commission consideration. BRG recommended that NRC priorities should be on LWR programs.

	BUDGET (In millions) RES	BRG	
8. ADVANCED CONVERTERS			
a. GCR Program	$ 3.9	0.0/3.9 (8)	
	$ 3.9	$0.0/3.9 (8)	0

(8) BRG set aside entire program based on the Administration's decision to terminate domestic program in FY 79.

TABLE 3.1 (Cont)

9. REACTOR ENVIRONMENTAL EFFECTS	BUDGET (In millions) RES	BRC	
a. Physical Transport and Effluent Characteristics	$2.1	$1.3	1.7
b. Ecological Processes	0.6	0.4	0.6
c. Radiation Dosimetry and Health Effects	1.2	1.1	1.0
d. Ecological Impacts	1.7	0.3	1.0
e. Socioeconomics and Regional	1.2	0.5	0.8
f. Occupational Radiation Exposure	1.1	0.9	0.9
g. Effluent Control	1.0	0.8	0.9
h. Decommissioning	0.9	0.9	0.9
	$9.8	$6.2 (9)	7.8

(9) BRC provided minimum level on many areas (except those dealing with problem of low level radiation exposure) because of low priority of programs.

10. FUEL CYCLE	BUDGET (In millions) RES	BRC	
a. Effluent Control	$0.7	$0.3	0.3
b. Safety	1.4	1.3	1.0
c. Occupational/Health	1.6	1.2	1.3
d. Environmental Impacts	0.1	0.1	0.3
e. Transportation	1.5	1.5	1.2
f. Decommissioning	0.6	0.6	0.3
	$5.9	$5.0 (10)	4.4

(10) BRC reduced funding as some parts of program are not clearly supported by user needs or are supported by outdated requests.

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

January 9, 1979

APPENDIX VI
BACKGROUND MATERIAL FOR DISCUSSIONS OF
TMI-2 ACCIDENT ACTION PLAN

TO: E. Etherington, Chairman of the Ad Hoc Subcommittee on Three Mile Island 2 Accident Action Plan

FROM: R. Major, Reactor Engineer RWM

SUBJECT: AD HOC SUBCOMMITTEE ON THREE MILE ISLAND ACCIDENT PLAN MEETING OF JANUARY 7, 1980

I have prepared the attached proposed meeting summary for your review. Copies are being distributed to the other ACRS members and Subcommittee consultants for their information and comment. Corrections and additions will be included in the minutes of the meeting.

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
ACRS Participating Consultants
E. Case, NRR
R. Purple, OSD
J. O'Reilly I&E
R. Scroggin , RES
J. Scinto, LD
W. Minners, NRR

PROPOSED SUMMARY OF THE JANUARY 7, 1980 MEETING
OF THE AD HOC SUBCOMMITTEE ON THREE MILE ISLAND 2 ACCIDENT ACTION PLAN

PURPOSE:

The purpose of this meeting was to discuss the NRC Staff's "Draft Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the Three Mile Island, Unit 2 Accident," NUREG-0660.

ATTENDEES:

H. Etherington, ACRS
H. Lewis, ACRS
W. Mathis, ACRS
W. Lipinski, ACRS Consultant
C. Michelson, ACRS Consultant
T. Theofanous, ACRS Consultant
R. Purple, NRC Staff
J. O'Reilly, NRC Staff
R. Scroggins, NRC Staff
J. Scinto, NRC Staff
W. Minners, NRC Staff
S. Hanauer, NRC Staff

MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS:

1. The Staff was asked to described the purpose of a separate plan for TMI-related work as opposed to other ongoing agency efforts such as work to resolve generic items. The Staff was asked to describe how priorities would be established between other programs and the NUREG-0660 items and how the Staff would insure that NUREG-0660 really includes all the recommendations resulting from the TMI-2 accident from the various sources?

   The Staff noted that the development of the Action Plan was neither political expediency nor was it to provide a document to support a supplementary budget request. The report is better characterized as being a document that represents the best judgment of the Staff as to the items that need to be accomplished as a result of the TMI accident.

2. The Action Plan includes consideration of the Kemeny Commission Report, the President's statement that followed the Kemeny Commission Report, ACRS recommendations, the results of the Lessons Learned reports, commitments made by the Commissioners in Congressional testimony since the TMI-2 accident, and other ideas for improvements generated within the Staff itself. It will include consideration of all of the recommendations that may come out of the NRC Special Inquiry (Rogovin Report).

3. One of the aims of the Action Plan when it is approved by the Commission would be to define the end of the licensing pause.

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4. A scoring system for judging priorities like that used on the generic safety issues will be used to help set priorities within the Action Plan. The scoring system judges items on criteria such as safety significance, whether the item effects the human element, or whether it improves hardware. It gets a score for whether it's a cheap thing to do or expensive. The scoring system is not intended to eliminate any items from the plan. The scoring system will be a tool for management at the office-director-level to make judgment calls as far as what items can be delayed. (See attachment)

5. Scheduling estimates in the Action Plan are based on the judgment of a single task manager; they do not take into consideration resource limitations. All tasks are assumed to begin at once as if there were no resource limitations.

6. The Staff noted that the Commission has asked that those items applicable to near-term OL plants be pulled from the Action Plan. The Staff will discuss this item with the Commission at a future meeting. (See attached Jan. 5, 1980 memo to Commissioners)

7. During a brief executive session, the Subcommittee recognized the fact that NUREG-0660 is a draft document that will undergo change. Some of the material could use additional description to clarify the aims of certain tasks. It was also recognized that there is a need to set priorities among the items in the Action Plan and between Action Plan items and the balance of the agency's work. Overall, the document was described as impressive and useful.

FUTURE MEETINGS:

The full ACRS will hear a presentation on the Action Plan from the NRC Staff on January 10, 1980 from 1:30 p.m. to 5:30 p.m.

DRAFT

Priority Ranking System

Purpose: This ranking system is for use in prioritizing both the necessary and the desirable elements of the Action Plan. It is not intended to be used to eliminate elements from the plan. The only basis for removal of elements will be a finding that they are either not necessary for safety or not related to TMI.

I. Safety Significance

(see Attachment for judgment factors)

High. 100

Medium. 50

Low 0

II. Type of Improvement

Improves the human element 20

Improves the hardware. 10

III. Utilization of Resources

A. Waste: Project is ongoing, and significant resources would be wasted if stopped. 20

Project has not yet been initiated or small resources now assigned 10

B. Staff resource requirement

(Score only if staff is involved in the action item)

Small (<2MY). 20

Medium (>2<10MY). 10

Large (>10MY) 0

C. Industry Resource Requirement

(score only if industry is involved in the action item)

Small (<$1.0M) 20

Large 0

IV. Timing of Improvement

(i.e., how soon will the expected benefit be realized?)

Within one year 30

Within two years 20

Within three years 10

Beyond three years 0

Judgment Factors

for

Safety Significance

A. Accident probability

Judge whether the action item has the potential for a large, moderate, or small reduction in accident probability. Where numbers can be estimated, a factor of 10 is large, a factor of 2 is small. Otherwise, use judgment to assess degree of reduction and consider the directness of the item's relationship to accident initiators.

B. Dose consequence

Consider whether the quantity of radioactive material that could be released if the action item was not done would be large or small. Also, consider he degree of dose reduction that the item could provide.

C. Number of levels of defense in depth affected.

D. People Affected

An item that provides added protection for the general public should be given more weight than one limited to worker protection.

E. Organization Level of Action

Actions that improve the licensee's or the local authority's capability to mitigate the consequences of an accident are more important than items designed to improve the State or Federal capability. Things that can be done at the site, right away, should be given more weight than things that require long distance response by State or Federal authorities.

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

January 5, 1980

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

FROM: Lee V. Gossick
Executive Director for Operations

SUBJECT: TMI ACTION PLAN -- PREREQUISITES FOR RESUMPTION OF LICENSING

In response to the Secretary's memorandum of December 28 and the guidance of the Commission at the meeting on December 21, the TMI Action Plan Steering Group has developed a proposed definition of the actions that would be required to be taken before reactor licensing could be resumed. This memorandum provides that proposed definition and reports on an important initial step in that effort -- the identification of the specific licensing requirements for the near-term operating license applications. This memorandum and its attached list of near-term OL requirements were presented to and discussed with the Directors of NRR, IE, RES, and SD on January 4, and they have concurred. The Executive Legal Director has no legal objections.

The licensing pause has been described previously, but not in the detail now needed. It was broadly defined by the Commission in its November 9, 1979 letter to Dr. Press in the Executive Office of the President. In providing its analysis and views of the recommendations of the President's Commission, the NRC said in that letter, in part,

> "NRC has decided that new plants will not be licensed until the required criteria have been developed. This approach assures that the NRC staff can give the necessary attention to implementation of the changes on operating plants.
>
> NRC plans to proceed systematically in the following manner: (1) review and correlate the recommendations of the President's Commission, those of internal lessons learned groups, those of the Advisory Committee on Reactor Safeguards, the findings of NRC Special Inquiry (when available), the findings of ongoing Congressional investigations (when available), and other inputs; (2) transform the recommendations in each subject area into a statement of goals (i.e., define the new or improved safety objectives to be accomplished in each area); (3) develop task action plans to transform the goals into organizational or procedural changes as they apply to NRC,

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or into regulatory requirements as they apply to licensees; (4) initiate implementation of the new regulatory requirements on operating plants; and (5) initiate implementation of the new regulatory requirements on plants under construction."

The "action plans" called for in the November 9 letter have now come to be known as the draft TMI Action Plan (NUREG-0660). The desired format and content of the action plan in the context of the Commission's licensing pause were described in Commissioner Hendrie's memorandum of November 16, as "...essentially a matrix formed by listing the points in the November 9th paper, plus any other actions we think necessary, along one axis and the various classes of cases along the other axis." Table 1 of NUREG-0660 is the matrix of licensing and other actions developed by the staff in response to this guidance from the Commission.

The Action Plan contains what the staff presently believes constitutes the complete set of additional requirements and programs for NRC, for operating reactors, for operating license applicants, for reactors under construction, and for construction permit applicants. In its totality, the Action Plan will identify all actions considered to be necessary as a result of the accident at TMI. Some will be required to be finished before the resumption of licensing. Others may be required to be undertaken before resumption of licensing. Still other, longer term actions may not be undertaken until well after licensing has been resumed. Adoption of the Plan describing all of these actions by the NRC would constitute "getting its house in order." We do not believe that the isolated approval of any particular subset of action items -- for example the licensing requirements that are applicable to near-term operating licenses -- is a sufficient condition to justify the resumption of licensing.

We believe that Commission consideration and approval of the Action Plan in its entirety is a necessary action. Approval of the plan would mean Commission endorsement that the total program defined in the Plan constitutes the sufficient measures to be undertaken to permit resumption of licensing. This is important and necessary guidance for licensees, license applicants, the staff, and the hearing boards. In this connection, the form of the Commission approval of the Plan is an important subject that needs further consideration. Some preliminary thoughts by ELD on this subject are attached.

There are several deficiencies in the present draft that render it inadequate for approval at this time. First, it is incomplete. Recognizing that the NRC Special Inquiry report may contain additional requirements not presently identified in the draft Action Plan and that there is staff review of the plan still ongoing, we are not recommending approval of the existing draft Action Plan. Second, the plan as presently drafted is a mixture of policy objectives, program descriptions, and specific licensing criteria. Some of this material is at a level of detail that is too specific for Commission approval (i.e., it is at a level of detail more appropriate for staff action and interpretation). We anticipate furnishing to the Commission another draft of the plan within about a month of issuance of the NRC Special Inquiry Report. It is our intent that it will correct these sorts of deficiencies. In addition, at that time, we expect to furnish an analysis of the resource and programmatic implications of the Plan, including the identification of necessary reprogramming, future budget requirements, and effect on present programs.

We recognize that there are many action items in the present draft of the Plan that require clearer description, fuller explanation of need, development of detailed criteria, consideration of alternative approaches, and the like, before final decisions on them could be expected. We plan, for the next draft, to identify each of those actions and a proposed schedule and method for obtaining Commission approval. We propose that those approvals can be granted external to or subsequent to Commission approval of the Action Plan itself. Approval of the Plan will simply mean, in these areas, that the Commission agrees in principal with the indicated action but intends to treat them separately and on specific schedules and according to methods or procedures outlined in the Plan. The balance of the action items in the Plan will be sufficiently well-described that Commission approval of the overall Plan will constitute specific approval of those items. Examples of the sort of detailed requirements that can be decided by Commission approval of the overall Plan are the specific near-term operating license requirements described below.

There are several subsets of requirements that could be extracted from the Plan for separate consideration and decision by the Commission. Consistent with our understanding of the Commission's request at the December 21 meeting, we have extracted those actions that are uniquely applicable to near-term operating licenses. We have defined "near-term operating licenses" as those that would be issued before July 1980. A longer time period would add, subtract, or modify requirements. It is necessary to establish such a temporal definition because the subset of actions required to be accomplished by applicants before obtaining an OL differs depending on that definition. The set of requirements for near-term OL applicants according to a July 1980 definition is attached as Enclosure 1.

A similar listing of requirements could be extracted for other classes of activities, such as the set of short-term lessons learned already applied to operating reactors, the additional requirements for operating reactors beyond the short-term lessons learned, the actions required to be taken by holders of construction permits, and the internal actions required to be taken by the NRC that would define "putting our house in order." It is our intent that an improved Table 1 in the next draft of NUREG-0660 will more clearly identify such subgroupings of all the actions contained in the Plan.

Besides the information discussed above, the Steering Group will be prepared at its meeting with the Commission on January 9 to discuss the status of ongoing work to revise the action plan generally, to identify the method being used to identify resource reprogramming candidates in the current NRC operating plan and budget submissions, and to propose a method for obtaining feedback and ideas from reactor operators and others involved in the implementation of the TMI-related requirements.

Lee V. Gossick
Executive Director for Operations

Enclosures:
1. Near-Term Operating License Requirements
2. ELD Comments on Form of Commission Approval

cc: Office Directors
Steering Group Members
Task Managers

TMI ACTION PLAN

NEAR-TERM OPERATING LICENSE REQUIREMENTS

Requirement	Already Approved	When Applicable*
I.A.1.1 Shift Technical Advisor		
Provide technical advisors with engineering expertise on each shift.	Yes	FL
I.A.1.2 Shift Supervisor Duties		
Minimize administrative duties.	Yes	FL
I.A.1.3 Shift Manning		
(1) SRO and RO in control room.	No	FL
(2) Administrative aide to shift supervisor on each shift.	No	FL
(3) Restrictions on use of overtime.	No	FL
I.B.1.1 Organization and Management Criteria		
Interoffice NRC review of licensee management to determine organizational and managerial capabilities, pending development of criteria.	No	FL

*FL = before fuel load
FP = before full power

Requirement	Already Approved	When Applicable
I.B.3.1 Safety Engineering Group		
Licensee provide onsite safety engineering group to provide supplemental engineering review and support. Interoffice NRC review of the adequacy of this group, pending development of formal criteria.	No	FL
I.B.3.4 Resident Inspector		
NRC resident inspector at each site for new OL.	No	FL
I.C.1.1 Analysis and Procedure Modifications		
(1) Phase I - small break LOCA's.	Yes	FL
(2) Phase II - inadequate core cooling.	Yes	FL
I.C.1.2 Shift Relief and Turnover Procedures		
Plant procedures for shift and relief turnover.	Yes	FL
I.C.1.3 Shift Personnel Responsibilities		
Plant procedures specifying responsibilities of shift personnel for safe operation of the plant.	Yes	FL

Requirement	Already Approved	When Applicable
I.C.1.4 Control Room Access Plant procedures for limiting access to the control room.	Yes	FL
I.C.2 Vendor Review of Procedures NSSS vendor review of licensee emergency procedures, low power test procedures, and power ascension procedures.	No	FP
I.C.3 Pilot Program for Review of Selected Emergency Procedures NRC conduct in-depth review of development and use of selected emergency procedures on NTOL plants.	No	FP
I.E.1 Licensee Operating Experience Evaluation Capability Onsite and offsite capability for evaluation of operating experiences at nuclear power plants.	Partial	FL
I.E.2 Licensee Dissemination of Operating Experiences Procedures that assure feedback of operating experiences to operators and other personnel.	No	FL

Requirement	Already Approved	When Applicable
I.G Training During Low Power Testing Conduct "hands on" training in selected plant evolutions and off-normal events for shift personnel.	No	FP
II.B.1 Degraded Core - Primary System Vent Provide design of remotely operable high-point reactor coolant system vents.	Yes	FP
II.B.2 Degraded Core - Shielding Provide design of additional shielding required to provide access to vital areas and protect safety equipment.	Yes	FP
II.B.3 Degraded Core - Sampling Provide interim procedures and final system design for sampling and analyzing reactor coolant and containment atmosphere.	Yes	FP
II.B.4 Degraded Core - Training (1) Establish training program for all operating personnel in the mitigation of severe core damage using existing equipment.	No	FL
(2) Complete initial training.	No	FP

Requirement	Already Approved	When Applicable
II.B.8 Degraded Core - Rulemaking Issue notice of intent to conduct rulemaking on requirements for design features for accidents involving severely damaged cores.	No	FP
II.B.9 Interim Hydrogen Control Requirements for Small Containments Under development.	No	FP
II.C.1.1 Mini-IREP	No	FP
II.C.1.8 Reliability Assurance Establish a reliability assurance program for engineered safety features systems.	No	FP
II.D.1.1 Relief and Safety Valve Test Commit to performance testing of RCS relief and safety valves under the full range of normal and accident conditions by July 1981.	Yes	FL
II.D.1.5 Relief and Safety Valve Position Install direct indication of relief and safety valve position.	Yes	FL

Requirement	Already Approved	When Applicable
II.E.1.1 Auxiliary Feedwater System Reliability		
Perform simplified reliability analysis of AFW system and modify as necessary.	No	FP
II.E.1.3 Auxiliary Feedwater Initiation		
Install safety grade automatic start of AFW and safety grade flow indicators.	Yes	FP
II.E.3 Emergency Power for Decay Heat Removal		
Install capability to supply some pressurizer heaters and controls from emergency power supply and implement necessary training and procedures.	Yes	FP
II.E.4.1 Containment Penetrations		
Provide design of redundant dedicated containment penetrations for external hydrogen recombiner, if applicable.	Yes	FL
II.E.4.3 Containment Isolation		
Install diverse containment isolation signal.	Yes	FP
II.E.4.5 Containment Purge		
Restrict containment purge operation and demonstrate purge valve operability.	Yes	FP

Requirement	Already Approved	When Applicable
II.F.2 Inadequate Core Cooling Instruments		
(1) Install subcooling meter.	Yes	FL
(2) Submit design of vessel level indicator.	Yes	FL
II.G Emergency Power for Pressurizer Equipment		
Modify power supplies for the pressurizer relief valves, block valves, and level indicators to be from emergency power sources.	Yes	FL
III.A.1.1 Role of NRC		
More detailed definition of role of NRC in emergencies than presently contained in Action Plan.	No	FP
III.A.1.5 Communications		
Install two direct dedicated telephone lines between plant and NRC.	Yes	FL
III.A.2.1 Technical Support Center		
Establish initial onsite TSC and provide plans, procedures, staffing, communications, and radiation monitoring equipment. (Upgrade on same schedule as present OR's.)	Yes	FL

Requirement	Already Approved	When Applicable
III.A.2.2 Onsite Operational Support Center Establish an OCS as described in the 10/30/79 letter to licensees. (Upgrade on same schedule as present OR's.)	Yes	FL
III.A.2.3 Near-Site Emergency Operations Center Establish an EOC as a base for coordinating onsite and offsite activities and interface with State, local, and Federal agencies. (Upgrade on same schedule as present OR's.)	Yes	FL
III.A.3 Upgrade Licensee Emergency Preparedness Upgrade emergency plans in accordance with Regulatory Guide 1.101 and NUREG-0610.	Yes	FL
III.B.3.2 FEMA-NRC Concurrence in State and Local RERP Concurrence must be obtained.	Yes	FL
III.D.1.3.a Area Radiation Monitors (Partial) Provide instrumentation to determine in-plant airborne radioiodine concentrations.	Yes	FL

Requirement	Already Approved	When Applicable
III.D.2.1 Control Room Habitability Confirm compliance with existing regulatory requirements or establish schedule for necessary modifications to achieve compliance.	No	FP
III.D.2.2.b Evaluation of Secondary Side Hazards Evaluate secondary side leakage and radiological hazards which could result from major accident, and make modifications to reduce hazards.	Yes	FP
III.D.2.2.c Improve Auxiliary Building Identify improvements to control radioactive leakage from auxiliary buildings, including requirements for building exhaust filtration where it doesn't already exist, and provide schedule for modifications.	No	FP
III.E.1.1 Improved Vent Gas Systems Review vent gas and leak detection systems against new design criteria and provide schedule for modifications.	No	FP

Requirement	Already Approved	When Applicable
III.E.1.2.a Surveillance Testing (Filtration Systems) (Partial) Implement surveillance testing program for non-ESF filtration systems.	No	FP
III.E.2.1.b NRC Monitoring NRC establish TLD surveillance network around site.	Yes	FL

January 5, 1980

MEMORANDUM FOR: Roger Mattson

FROM: Guy Cunningham

SUBJECT: TMI ACTION PLAN -- PREREQUISITES FOR RESUMPTION OF LICENSING

At their meeting on January 4, the office directors were unanimously agreed that Commission approval of the recommendations of this paper should be obtained before their full implementation. There was disagreement, however, as to whether that approval should be in the form of a general statement of policy or one or more rules (made immediately effective as appropriate). OELD believes that the difference between the approaches should be highlighted and the consequences of the choice made clear. A good discussion of this subject is presented in Pacific Gas and Electric Co. v. FPC (D.C. Cir. 1974) 506 F.2d 33. In part, the Court said:

> The critical distinction between a substantive rule and a general statement of policy is the different practical effect that these two types of pronouncements have in subsequent administrative proceedings. A properly adopted substantive rule establishes a standard of conduct which has the force of law. In subsequent administrative proceedings involving a substantive rule, the issues are whether the adjudicated facts conform to the rule and whether the rule should be waived or applied in that particular instance. The underlying policy embodied in the rule is not generally subject to challenge before the agency.
>
> A general statement of policy, on the other hand, does not establish a "binding norm." It is not finally determinative of the issues or rights to which it is addressed. The agency cannot apply or rely upon a general statement of policy as law because a general statement of policy only announces what the agency seeks to establish as policy. A policy statement announces the agency's tentative intentions for the future. When the agency applies the policy in a particular situation, it must be prepared to support the policy just as if the policy statement had never been issued. An agency cannot escape its responsibility to present evidence and reasoning supporting its substantive rules by announcing binding precedent in the form of a general statement of policy.
>
> (Citations and footnotes omitted.)

In the present situation, utilization of a policy statement to announce the agency's intention to require implementation of the recommendations of this paper will mean that the proposed requirement will be a proper subject for litigation in every contested case before the issuance of any permit or license. OELD believes that immediately effective rules can be promulgated in the same

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period of time as a general statement of policy. Moreover, to the extent that the rules are merely "interpretive" of present regulations, they may be promulgated, as may a general statement of policy, without following APA rulemaking procedures.

III

Guy H. Cunningham, ~~III~~
Chief Regulations Counsel, OELD

Attachment 2

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

January 5, 1980

APPENDIX VII
TMI ACTION PLAN--PREREQUISITES FOR RESUMPTION OF LICENSING

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

FROM: Lee V. Gossick
Executive Director for Operations

SUBJECT: TMI ACTION PLAN -- PREREQUISITES FOR RESUMPTION OF LICENSING

In response to the Secretary's memorandum of December 28 and the guidance of the Commission at the meeting on December 21, the TMI Action Plan Steering Group has developed a proposed definition of the actions that would be required to be taken before reactor licensing could be resumed. This memorandum provides that proposed definition and reports on an important initial step in that effort -- the identification of the specific licensing requirements for the near-term operating license applications. This memorandum and its attached list of near-term OL requirements were presented to and discussed with the Directors of NRR, IE, RES, and SD on January 4, and they have concurred. The Executive Legal Director has no legal objections.

The licensing pause has been described previously, but not in the detail now needed. It was broadly defined by the Commission in its November 9, 1979 letter to Dr. Press in the Executive Office of the President. In providing its analysis and views of the recommendations of the President's Commission, the NRC said in that letter, in part,

"NRC has decided that new plants will not be licensed until the required criteria have been developed. This approach assures that the NRC staff can give the necessary attention to implementation of the changes on operating plants.

NRC plans to proceed systematically in the following manner: (1) review and correlate the recommendations of the President's Commission, those of internal lessons learned groups, those of the Advisory Committee on Reactor Safeguards, the findings of NRC Special Inquiry (when available), the findings of ongoing Congressional investigations (when available), and other inputs; (2) transform the recommendations in each subject area into a statement of goals (i.e., define the new or improved safety objectives to be accomplished in each area); (3) develop task action plans to transform the goals into organizational or procedural changes as they apply to NRC,

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or into regulatory requirements as they apply to licensees; (4) initiate implementation of the new regulatory requirements on operating plants; and (5) initiate implementation of the new regulatory requirements on plants under construction."

The "action plans" called for in the November 9 letter have now come to be known as the draft TMI Action Plan (NUREG-0660). The desired format and content of the action plan in the context of the Commission's licensing pause were described in Commissioner Hendrie's memorandum of November 16, as "...essentially a matrix formed by listing the points in the November 9th paper, plus any other actions we think necessary, along one axis and the various classes of cases along the other axis." Table 1 of NUREG-0660 is the matrix of licensing and other actions developed by the staff in response to this guidance from the Commission.

The Action Plan contains what the staff presently beli... s constitutes the complete set of additional requirements and programs ... NRC, for operating reactors, for operating license applicants, for reactors under construction, and for construction permit applicants. In its totality, the Action Plan will identify all actions considered to be necessary as a result of the accident at TMI. Some will be required to be finished before the resumption of licensing. Others may be required to be undertaken before resumption of licensing. Still other, longer term actions may not be undertaken until well after licensing has been resumed. Adoption of the Plan describing all of these actions by the NRC would constitute "getting its house in order." We do not believe that the isolated approval of any particular subset of action items -- for example, the licensing requirements that are applicable to near-term ...erating licenses -- is a sufficient condition to justify the resumption of licensing.

We believe that Commission consideration and approval of the Action Plan in its entirety is a necessary action. Approval of the plan would mean Commission endorsement that the total program defined in the Plan constitutes the sufficient measures to be undertaken to permit resumption of licensing. This is important and necessary guidance for licensees, license applicants, the staff, and the hearing boards. In this connection, the form of the Commission approval of the Plan is an important subject that needs further consideration. Some preliminary thoughts by ELD on this subject are attached.

There are several deficiencies in the present draft that render it inadequate for approval at this time. First, it is incomplete. Recognizing that the NRC Special Inquiry report may contain additional requirements not presently identified in the draft Action Plan and that there is staff review of the plan still ongoing, we are not recommending approval of the existing draft Action Plan. Second, the plan as presently drafted is a mixture of policy objectives, program descriptions, and specific licensing criteria. Some of this material is at a level of detail that is too specific for Commission approval (i.e., it is at a level of detail more appropriate for staff action and interpretation). We anticipate furnishing to the Commission another draft of the plan within about a month of issuance of the NRC Special Inquiry Report. It is our intent that

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it will correct these sorts of deficiencies. In addition, at that time, we expect to furnish an analysis of the resource and programmatic implications of the Plan, including the identification of necessary reprogramming, future budget requirements, and effect on present programs.

We recognize that there are many action items in the present draft of the Plan that require clearer description, fuller explanation of need, development of detailed criteria, consideration of alternative approaches, and the like, before final decisions on them could be expected. We plan, for the next draft, to identify each of those actions and a proposed schedule and method for obtaining Commission approval. We propose that those approvals can be granted external to or subsequent to Commission approval of the Action Plan itself. Approval of the Plan will simply mean, in these areas, that the Commission agrees in principal with the indicated action but intends to treat them separately and on specific schedules and according to methods or procedures outlined in the Plan. The balance of the action items in the Plan will be sufficiently well-described that Commission approval of the overall Plan will constitute specific approval of those items. Examples of the sort of detailed requirements that can be decided by Commission approval of the overall Plan are the specific near-term operating license requirements described below.

There are several subsets of requirements that could be extracted from the Plan for separate consideration and decision by the Commission. Consistent with our understanding of the Commission's request at the December 21 meeting, we have extracted those actions that are uniquely applicable to near-term operating licenses. We have defined "near-term operating licenses" as those that would be issued before July 1980. A longer time period would add, subtract, or modify requirements. It is necessary to establish such a temporal definition because the subset of actions required to be accomplished by applicants before obtaining an OL differs depending on that definition. The set of requirements for near-term OL applicants according to a July 1980 definition is attached as Enclosure 1.

A similar listing of requirements could be extracted for other classes of activities, such as the set of short-term lessons learned already applied to operating reactors, the additional requirements for operating reactors beyond the short-term lessons learned, the actions required to be taken by holders of construction permits, and the internal actions required to be taken by the NRC that would define "putting our house in order." It is our intent that an improved Table 1 in the next draft of NUREG-0660 will more clearly identify such subgroupings of all the actions contained in the Plan.

Besides the information discussed above, the Steering Group will be prepared at its meeting with the Commission on January 9 to discuss the status of ongoing work to revise the action plan generally, to identify the method being used to identify resource reprogramming candidates in the current NRC operating plan and budget submissions, and to propose a method for obtaining feedback and ideas from reactor operators and others involved in the implementation of the TMI-related requirements.

Lee V. Gossick
Executive Director for Operations

Enclosures:
1. Near-Term Operating License Requirements
2. ELD Comments on Form of Commission Approval

cc: Office Directors
Steering Group Members
Task Managers

TMI ACTION PLAN

NEAR-TERM OPERATING LICENSE REQUIREMENTS

Requirement	Already Approved	When Applicable*
I.A.1.1 Shift Technical Advisor		
Provide technical advisors with engineering expertise on each shift.	Yes	FL
I.A.1.2 Shift Supervisor Duties		
Minimize administrative duties.	Yes	FL
I.A.1.3 Shift Manning		
(1) SRO and RO in control room.	No	FL
(2) Administrative aide to shift supervisor on each shift.	No	FL
(3) Restrictions on use of overtime.	No	FL
I.B.1.1 Organization and Management Criteria		
Interoffice NRC review of licensee management to determine organizational and managerial capabilities, pending development of criteria.	No	FL

*FL = before fuel load
FP = before full power

Requirement	Already Approved	When Applicable
I.B.3.1 Safety Engineering Group		
Licensee provide onsite safety engineering group to provide supplemental engineering review and support. Interoffice NRC review of the adequacy of this group, pending development of formal criteria.	No	FL
I.B.3.4 Resident Inspector		
NRC resident inspector at each site for new OL.	No	FL
I.C.1.1 Analysis and Procedure Modifications		
(1) Phase I - small break LOCA's.	Yes	FL
(2) Phase II - inadequate core cooling.	Yes	FL
I.C.1.2 Shift Relief and Turnover Procedures		
Plant procedures for shift and relief turnover.	Yes	FL
I.C.1.3 Shift Personnel Responsibilities		
Plant procedures specifying responsibilities of shift personnel for safe operation of the plant.	Yes	FL

Requirement	Already Approved	When Applicable
I.C.1.4 Control Room Access Plant procedures for limiting access to the control room.	Yes	FL
I.C.2 Vendor Review of Procedures NSSS vendor review of licensee emergency procedures, low power test procedures, and power ascension procedures.	No	FP
I.C.3 Pilot Program for Review of Selected Emergency Procedures [illegible]C conduct in-depth review of development and use of selected emergency procedures on NTOL plants.	No	FP
I.E.1 Licensee Operating Experience Evaluation Capability Onsite and offsite capability for evaluation of operating experiences at nuclear power plants.	Partial	FL
I.E.2 Licensee Dissemination of Operating Experiences Procedures that assure feedback of operating experiences to operators and other personnel.	No	FL

Requirement	Already Approved	When Applicable
I.G Training During Low Power Testing		
Conduct "hands on" training in selected plant evolutions and off-normal events for shift personnel.	No	FP
II.B.1 Degraded Core - Primary System Vent		
Provide design of remotely operable high-point reactor coolant system vents.	Yes	FP
II.B.2 Degraded Core - Shielding		
Provide design of additional shielding required to provide access to vital areas and protect safety equipment.	Yes	FP
II.B.3 Degraded Core - Sampling		
Provide interim procedures and final system design for sampling and analyzing reactor coolant and containment atmosphere.	Yes	FP
II.B.4 Degraded Core - Training		
(1) Establish training program for all operating personnel in the mitigation of severe core damage using existing equipment.	No	FL
(2) Complete initial training.	No	FP

Requirement	Already Approved	When Applicable
II.B.8 Degraded Core - Rulemaking Issue notice of intent to conduct rulemaking on requirements for design features for accidents involving severely damaged cores.	No	FP
II.B.9 Interim Hydrogen Control Requirements for Small Containments Under development.	No	FP
II.C.1.1 Mini-IREP	No	FP
II.C.1.8 Reliability Assurance Establish a reliability assurance program for engineered safety features systems.	No	FP
II.D.1.1 Relief and Safety Valve Test Commit to performance testing of RCS relief and safety valves under the full range of normal and accident conditions by July 1981.	Yes	FL
II.D.1.5 Relief and Safety Valve Position Install direct indication of relief and safety valve position.	Yes	FL

Requirement	Already Approved	When Applicable
II.E.1 Auxiliary Feedwater System Reliability Perform simplified reliability analysis of AFW system and modify as necessary.	No	FP
II.E.1.3 Auxiliary Feedwater Initiation Install safety grade automatic start of AFW and safety grade flow indicators.	Yes	FP
II.E.3 Emergency Power for Decay Heat Removal Install capability to supply some pressurizer heaters and controls from emergency power supply and implement necessary training and procedures.	Yes	FP
II.E.4.1 Containment Penetrations Provide design of redundant dedicated containment penetrations for external hydrogen recombiner, if applicable.	Yes	FL
II.E.4.3 Containment Isolation Install diverse containment isolation signal.	Yes	FP
II.E.4.5 Containment Purge Restrict containment purge operation and demonstrate purge valve operability.	Yes	FP

Requirement	Already Approved	When Applicable
II.F.2 Inadequate Core Cooling Instruments		
(1) Install subcooling meter.	Yes	FL
(2) Submit design of vessel level indicator.	Yes	FL
II.G Emergency Power for Pressurizer Equipment		
Modify power supplies for the pressurizer relief valves, block valves, and level indicators to be from emergency power sources.	Yes	FL
III.A.1.1 Role of NRC		
More detailed definition of role of NRC in emergencies than presently contained in Action Plan.	No	FP
III.A.1.5 Communications		
Install two direct dedicated telephone lines between plant and NRC.	Yes	FL
III.A.2.1 Technical Support Center		
Establish initial onsite TSC and provide plans, procedures, staffing, communications, and radiation monitoring equipment. (Upgrade on same schedule as present OR's.)	Yes	FL

Requirement	Already Approved	When Applicable
III.A.2.2 Onsite Operational Support Center Establish an OCS as described in the 10/30/79 letter to licensees. (Upgrade on same schedule as present OR's.)	Yes	FL
III.A.2.3 Near-Site Emergency Operations Center Establish an EOC as a base for coordinating onsite and offsite activities and interface with State, local, and Federal agencies. (Upgrade on same schedule as present OR's.)	Yes	FL
III.A.3 Upgrade Licensee Emergency Preparedness Upgrade emergency plans in accordance with Regulatory Guide 1.101 and NUREG-0610.	Yes	FL
III.B.3.2 FEMA-NRC Concurrence in State and Local RERP Concurrence must be obtained.	Yes	FL
III.D.1.3.a Area Radiation Monitors (Partial) Provide instrumentation to determine in-plant airborne radioiodine concentrations.	Yes	FL

Requirement	Already Approved	When Applicable
III.D.2.1 Control Room Habitability Confirm compliance with existing regulatory requirements or establish schedule for necessary modifications to achieve compliance.	No	FP
III.D.2.2.b Evaluation of Secondary Side Haz[illegible] Evaluate secondary side leakage and radiological hazards which could result from major accident, and make modifications to reduce hazards.	Yes	FP
III.D.2.2.c Improve Auxiliary Building Identify improvements to control radioactive leakage from auxiliary buildings, including requirements for building exhaust filtration where it doesn't already exist, and provide schedule for modifications.	No	FP
III.E.1.1 Imp[illegible]ed Vent Gas Systems Review vent gas and leak detection systems against new design criteria and provide schedule for modifications.	No	FP

Requirement	Already Approved	When Applicable
III.E.1.2.a Surveillance Testing (Filtration Systems) (Partial)		
Implement surveillance testing program for non-ESF filtration systems.	No	FP
III.E.2.1.b NRC Monitoring		
NRC establish TLD surveillance network around site.	Yes	FL

January 5, 1980

MEMORANDUM FOR: Roger Mattson

FROM: Guy Cunningham

SUBJECT: TMI ACTION PLAN -- PREREQUISITES FOR RESUMPTION OF LICENSING

At their meeting on January 4, the office directors were unanimously agreed that Commission approval of the recommendations of this paper should be obtained before their full implementation. There was disagreement, however, as to whether that approval should be in the form of a general statement of policy or one or more rules (made immediately effective as appropriate). OELD believes that the difference between the approaches should be highlighted and the consequences of the choice made clear. A good discussion of this subject is presented in Pacific Gas and Electric Co. v. FPC (D.C. Cir. 1974) 506 F.2d 33. In part, the Court said:

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> A general statement of policy, on the other hand, does not establish a "binding norm." It is not finally determinative of the issues or rights to which it is addressed. The agency cannot apply or rely upon a general statement of policy as law because a general statement of policy only announces what the agency seeks to establish as policy. A policy statement announces the agency's tentative intentions for the future. When the agency applies the policy in a particular situation, it must be prepared to support the policy just as if the policy statement had never been issued. An agency cannot escape its responsibility to present evidence and reasoning supporting its substantive rules by announcing binding precedent in the form of a general statement of policy
>
> (Citations and footnotes omitted.)

In the present situation, utilization of a policy statement to announce the agency's intention to require implementation of the recommendations of his paper will mean that the proposed requirement will be a proper subject for litigation in every contested case before the issuance of any permit or license. OELD believes that immediately effective rules can be promulgated in the same

Attachment 2

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UNITED STATE
NUCLEAR REGULATORY
ADVISORY COMMITTEE ON REAC
WASHINGTON, D. C. 2055

APPENDIX VIII
BACKGROUND MATERIAL FOR DISCUSSIONS ON
IMPLEMENTATION OF NRC TMI-2 ACCIDENT-
RELATED BULLETINS AND ORDERS

January 9, 1980

TO: W. Mathis, Chairman B&O Subcommittee
M. Plesset, Chairman ECCS Subcommittee

FROM: P. Boehnert PB

SUBJECT: COMBINED SUBCOMMITTEES ON B&O/ECCS MEETING OF JANUARY 3-4, 1980

I have prepared the attached proposed meeting summary for your review. Copies are being distributed to the other ACRS members for their information and comment. Corrections and additions will be included in the minutes of the meeting.

Attachment As Stated

cc: ACRS Members
ACRS Technical Staff

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January 9, 1980

PROPOSED SUMMARY OF THE COMBINED BULLETINS AND ORDERS/ECCS SUBCOMMITTEE MEETING, JANUARY 3-4, 1980, LOS ANGELES, CALIFORNIA

The ACRS Bulletins & Orders and ECCS Subcommittees held a joint meeting in Los Angeles on January 3-4, 1980 to continue discussion of the NRC and industry response to NRC Bulletins and Orders. ACRS Members in attendance included W. Mathis, M. Plesset, H. Etherington and D. Okrent (January 3 - PM only). Consultants in attendance included A. Acosta, I. Catton, W. Lipinski, C. Michelson, V. Schrock, T. Wu and Z. Zudans.

MEETING HIGHLIGHTS

1. During a brief open executive session, ACRS Members and consultants made the following observations: (1) Mr. Mathis felt that some of the NRC recommendations evidenced a lack of consistency - and cited as an example the requirement for automatic initiation of auxiliary feedwater but with a provision for overriding the auto initiation; (2) Dr. Plesset expressed concern over the limitations of Semiscale and the Two Loop Test Apparatus vis-a-vis scaling test results to apply to full size reactors. He said careful consideration of the use of these facilities for full-scale interpetation is necessary. (3) Mr. Michelson observed that there was little discussion in the NRC NUREG reports of single failure considerations for small-break analyses. He cited the example of a secondary side blowdown causing a single failure in the primary system leading to combined primary/secondary blowdown; (4) Dr. Acosta suggested that the idea be explored of running the reactor coolant pumps at a reduced speed (10%) to allow pumping for a non-break accident, when the specific accident is unknown to the operator.

2. Dr. Ross, B&O Task Force leader, gave an overview of the B&O Task Force activities. The Task Forces' charge is to review generic implications of the TMI-2 accident for all operating plants to confirm bases for their continued safe operation. The scope of review encompassed the loss of feedwater and small-break LOCA events in the areas of systems reliability, analyses and operator guideliens, plant procedures, and operator training. The principal work products of the Task Force are

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4 generic reports, - (for each vendors plants), plus a summary report (NUREG-0645). (Note - copies of 0645 will be distributed to the Committee during the January meeting.) The majority of the B&O Task Force has been disbanded. A small group remains to supervise the issuance of the above NUREG reports; however as of this time, it has not been decided what NRC group will review the industry responses to the requirements specified in the NUREGs.

3. W. Kane (NRC) reviewed the status of the long-term requirements of the NRC Orders issued to all B&W plants in May 1979. There are four items generic to all B&W plants: (1) continued upgrade of the auxiliary feedwater system; (2) perform a FMEA on the ICS, (3) upgrade the anticipatory reactor trip to safety grade; and (4) continued operator training and drilling. Figure 1 lists the plant-specific requirements of the orders. The bulk of the work on these items has been submitted for NRC review and approval.

4. Mr. Mathews (NRC) discussed the status of the auxiliary feedwater system review program. The main B&O requirements for this topic were incorporated into Lessons Learned Items - 2.1.7.A (automatic AFW Initiation), and 2.1.7.B (AFW flow indication). For 2.1.7.A, all but one plant (Yankee Rowe) has complied or committed to comply; all plants have complied or committed to comply with the provisions of 2.1.7.B. Yankee Rowe will install two motor driven auxiliary feedwater pumps as a long-term fix. The Staff is also awaiting response to requests for a AFW reliability review and AFW flow design basis information.

5. Z. Rosztoczy described the B&O Task Force efforts in the area of analysis of design and off-normal transients and accidents. This category embrased work in the following areas: (1) small-break LOCA analysis methods, (2) inadequate core cooling, (3) evaluation of off-normal transients and accidents, (4) reactor coolant pump trip and HPI termination criteria, and (5) experimental programs for small-break LOCA.

Among the significant conclusions/recommendations noted by Dr. Rosztoczy include the following:

- (a) analysis methods used for small-break (SB) LOCA analysis should be revised, documented, and submitted for NRC review by June 30, 1980, (b) plant specific calculations using NRC approved methods should be provided by all licensees by December 31, 1980 and (c) an NRC position on required conservatism in SB LOCA analyses should be issued by June 30, 1980. It was noted that there is a difference of opinion between Dr. Rosztoczy and D. Ross over recommandation (c) above. Dr. Ross has recommended item (c) be postponed and that items (a) and (b) be forwarded for action. Dr. Rosztoczy believes (a) and (c) can be performed together, then one can procede to item (b). There was considerable Subcommittee discussion on this point and it will be addressed at the Full Committee on Friday, January 11, 1980.

- Unless the capability of the relief valves to provide sufficient depressurization in case of loss of heat removal through the steam geneators can be demonstrated with due account of input and calculational uncertainties, continued operation of PWRs with low-head HPI systems should be conditioned on a timely design change. Replacement of HPI pumps with high-head pumps, increased relief valve capacity, or installation of a high pressure RHR system are possible acceptable design changes. (Note: this item effects all CE plants, one-half of W plants, and Davis-Besse (B&W)).

- Technical Specifications for ECC Systems should include an appropriate limit on the accumulated outage time per year.

- The frequency of relief valve challenges should be reduced substantially (by an order of magnitude) in CE, W and GE plants. This can be accomplished by careful selection of relief valve and overpressure reactor trip setpoints together with anticipatory reactor trips on turbine trip and loss of feedwater.

• For the analysis of transients and accidents topic, an event tree analysis of selected transients and accidents will be performed by the vendors through the Owner's Group. The purpose of this study is to upgrade operator training programs and emergency procedures, as well as identify essential instrumentation needed to follow these procedures. Single active failures will be considered for each system called upon to function for a particular event; multiple failures and passive failures will only be considerd if extenuating circumstances warrant, e.g., use of a non-safety grade AFW system. Dr. Okrent asked the Staff for their position on consideration of multiple failures. Dr. Ross replied that this topic is being addressed in the NRC Action Plan. A three-pronged approach is being used to provide more protection for DBAs. These three aspects are: (1) upgraded operator training, (2) better control room information, and (3) better operating procedures.

• Concerning the question of RCP trip during SB LOCAs, the NRC is requiring automatic RCP trip for a SB LOCA accident. NRC believes insufficient data exists to verify models with pumps running. NRC is encouraging the industry the seek means to leave pumps running during SB LOCAs. CE has proposed increasing HPI capacity and leaving two of the four RCPs running.

• The preliminary conclusions of the LOFT L3-1 SB LOCA test, run on November 20, 1979, were discussed. B&W, CE, and W along with INEL and LASL submitted pretest, or "blind" analyses of the test. The test data have been compared with the analyses but the value of this comparison is limited because the various flow bypass paths seen in the test were not modeled in the analyses.

• The Semiscale SB LOCA test (S-07-10B) was run on January 19, 1980. Pretest analyses were submitted by B&W, CE, W, INEL, and LASL. A "quick-look" report summarizing the preliminary data analyses is due out in late January.

- A small break LOCA test on the two-loop test apparatus (BWR facility) was run on December 18, 1979. GE submitted a pre-test prediction on the same date. Test data are locked up, pending receipt of predictions from INEL.

6. Mr. Bruce Wilson (NRC) discussed the approved HPI termination criteria Figures 2-5 attached describe these criteria for each vendor's plants.

7. The results of an NRC audit of selected plants was reviewed. The B&O Task Force performed these audits to assess four areas: (1) comparison of the plant's SB LOCA procedures to the approved vendor's guidelines; (2) review of the SB LOCA procedures training given reactor operators; (3) quizzing of the operators to determine their proficiency vis-a-vis the SB LOCA procedures; and (4) review of the system-related aspects of the procedures to assure the operator actions can be performed. Problem areas noted during the audits included: (1) misinterpretation of the guidelines, (2) refusal to follow guidelines believed to be unsafe, (3) plant operators lack of depth of knowledge in fluid dynamics, thermo-hydraulics, and heat transfer, and (4) control room design problems.

8. W. Kane (NRC) detailed the schedule for implementation of the B&O Task Force recommendations (Figure 6). (Note: It is my understanding that H. Denton has recently extended the 1/1/80 deadline dates to 1/31/80.)

9. A representative from each of the four utility Owner's Groups made a presentation before the Subcommittee. The tone and tennor of these presentations evidenced a mood of hostility towards the NRC Staff. In general, the Owner's Groups complained of unrealistic schedule demands, lack of clear and concise responses from the Staff, and a general lack of operating experience among Staff members. Both Mr. Mathis and Dr. Plesset expressed dismay at the lack of cooperation shown by the Owner's Groups, particularly in light of the TMI-2 experience.

10. The B&O Task Force will make a presentation before the ACRS on Friday, January 11, 1980, at 8:30 a.m., to present an overview of the Task Force Activities. The Task Force would like Committee comment on their efforts.

REQUESTS FOR INFORMATION

1. Mr. Etherington asked if the NRC has studied the possibility of water hammer in RHR system condensors. NRC will respond to this question at the full Committee meeting.

2. Dr. Plesset requested a brief report on the status of UHI plants vis-a-vis NRC post-TMI review. Dr. Ross said he would respond at the ACRS meeting.

3. Mr. Etherington asked who in NRC will be responsible for tracking the B&O recommendations after the Task Force has disbanded. NRC has been asked to respond to this question at the full Committee meeting.

FUTURE MEETINGS

No future meetings have been scheduled at this time.

STATUS OF PLANT SPECIFIC REQUIREMENTS OF THE ORDERS (LONG-TERM)

OCONEE 1, 2 & 3

1. INSTALL TWO MOTOR DRIVEN EMERGENCY FEEDWATER PUMPS PER UNIT - COMPLETE

ANO-1

1. CONNECT MOTOR-DRIVEN EFW PUMP TO VITAL BUS - COMPLETE
2. INSTALL EFW CONTROL SYSTEM (BEING DEVELOPED BY B&W) - AP&L READY TO INSTALL SYSTEM DURING JANUARY 1980 S/D. REQUIRES NRC APPROVAL. DUE TO OTHER PRIORITIES, WE COULD NOT REVIEW IN TIME TO INSTALL THIS OUTAGE. MUST WAIT UNTIL FALL REFUELING.
3. MODIFY EFW SUCTION PIPING TO IMPROVE SEPARATION - WILL BE COMPLETED DURING PRESENT OUTAGE (COMPLETE BY JANUARY 23, 1980)
4. PROVIDE CONTROL ROOM ANNUNCIATION FOR ALL AUTO START CONDITIONS OF EFW - COMPLETE
5. ADD REDUNDANT PRESSURE SWITCH TO EFW PUMP SUCTION AND REDUNDANT LOW PRESSURE ANNUNCIATION IN THE CONTROL ROOM - WILL BE COMPLETED DURING DECEMBER 1980 REFUELING OUTAGE

CRYSTAL RIVER 3

1. PROVIDE AFW FLOW VERIFICATION IN THE CONTROL ROOM - COMPLETE

DAVIS-BESSE 1

CONTINUED ATTENTION TO TRANSIENT ANALYSIS AND PROCEDURES FOR THE MANAGEMENT OF SMALL BREAKS - *

* THIS SPECIFIC REQUIREMENT HAS THE SAME INTENT OF SECTION 2.1.9 OF NUREG-0578 TECO HAS BEEN PARTICIPATING WITH THE B&W OWNERS' GROUP TO COMPLETE THE THREE SPECIFIC AREAS COVERED BY THIS SECTION: SMALL BREAK ANALYSIS AND PROCEDURES INADEQUATE CORE COOLING AND ANTICIPATED TRANSIENTS ANALYSI PROCEDURES FIGURE 1

B. WILSON - MKK, IN

B&W HPI TERMINATION CRITERIA

1. THE LPI SYSTEM IS IN OPERATION AND FLOWING AT A RATE IN EXCESS OF 1000 GPM AND THE SITUATION HAS BEEN STABLE FOR 20 MINUTES.

OR

2. ALL HOT AND COLD LEG TEMPERATURES ARE AT LEAST 50° BELOW THE SATURATION TEMPERATURE FOR THE EXISTING RCS PRESSURE, THE HOT LEG TEMPERATURES ARE NOT MORE THAN 50° HOTTER THAN THE SECONDARY SIDE SATURATION TEMPERATURE, AND THE ACTION IS NECESSARY TO PREVENT THE INDICATED PRESSURIZER LEVEL FROM GOING OFF-SCALE HIGH.

FIGURE 2

WESTINGHOUSE HPI TERMINATION CRITERIA

1. REACTOR COOLANT PRESSURE IS GREATER THAN 2000 PSIG AND INCREASING

AND

2. PRESSURIZER WATER LEVEL IS GREATER THAN NO LOAD WATER LEVEL

AND

3. THE REACTOR COOLANT INDICATES SUBCOOLING IS GREATER THAN (PLANT SPECIFIC VALUE)

AND

4. WATER LEVEL IN AT LEAST ONE STEAM GENERATOR IS IN THE NARROW RANGE SPAN OR IN THE WIDE RANGE SPAN AT A LEVEL SUFFICIENT TO ASSURE THAT THE U-TUBES ARE COVERED

FIGURE 3

CE HPI TERMINATION CRITERIA

AFTER ANY SIAS, OPERATE THE SIS UNTIL RCS HOT AND COLD LEG TEMPERATURES ARE AT LEAST 50°F BELOW SATURATION TEMPERATURE FOR THE RCS PRESSURE AND A PRESSURIZER LEVEL IS INDICATED, UNLESS THE CAUSE OF THE SIAS HAS BEEN VERIFIED TO BE AN INADVERTANT ACTUATION.

FIGURE 4

GE ECCS TERMINATION CRITERIA

ANY EMERGENCY CORE COOLING SYSTEM SHOULD NOT BE SHUT OFF UNLESS THERE ARE MULTIPLE CONFIRMING PROCESS PARAMETER INDICATIONS (SUCH AS LEVEL INDICATIONS FROM SEVERAL INSTRUMENTS) THAT THE CORE AND CONTAINMENT ARE IN A SAFE, STABLE CONDITION.

FIGURE 5

B Lane-Watching

TABLE 3-1

SCHEDULE FOR IMPLEMENTATION OF
BULLETINS & ORDERS TASK FORCE RECOMMENDATIONS

Recommendation		Schedule
GS-1	AFW System Train Outage Time Limit	1/1/80
GS-2	Lock & Verify Position of Manual AFW System Valves	1/1/80
GS-3	Throttling AFW System Flow	1/1/80
GS-4	Initiating Backup Water Supplies	1/1/80
GS-5	AFW Flow Following Loss of All Ac Power	1/1/80
GS-6	AFW System Flow Path Verification	1/1/80
GS-7	Non-safety Grade Non-redundant AFW System Auto Initiation Signals	1/1/80
GS-8	Auto Initiation of AFW Systems	1/1/80
	Primary AFW Source Low Level Alarm	1/1/80
	AFW Pump Endurance Test	1/1/80
	AFW Flow Indication	1/1/80
	AFW System Availability during Periodic Surveillance Testing	1/1/80
GL-1	Auto Initiation of AFW Systems	1/1/81

FIGURE 6

SCHEDULE FOR IMPLEMENTATION OF
BULLETINS & ORDERS TASK FORCE RECOMMENDATIONS

Recommendation		Schedule
GL-2	Single Valve in AFW System Flow Path	1/1/81
GL-3	Elimination of ac Power Dependency	1/1/81
GL-4	Loss of Pump Suction Due to Natural Phenomena	1/1/81
GL-5	Non-Safety Grade Non-redundant Auto Initiation Signals	1/1/81
3.2.1(a)	Analysis Methods Appendix K	7/1/80
3.2.1(b)	Plant-Specific Appendix K Calculations	1/1/81
3.2.2(a)	RCP Pump Trip	See Section 7.3.1 of NUREG-0623
3.2.2(b)	Reliability & Redundancy of Equipment	NRC Action Schedule in TMI-2 Action Plan
3.2.3(a)	Two-phase natural circulation experiments	1/1/81
3.2.3(b)	Instrumentation to verify natural circulation	4/1/80
3.3.1	Confirmation of Anticipatory trip	4/1/80
3.3.2	Interrelationship between Safety and Relief Valves	Schedule for NRC Action in TMI-2 Action Plan
3.3.3	PID Controller Modification	4/1/80
3.3.4	Proposed Anticipatory Trip Modification	Plant-specific

SCHEDULE FOR IMPLEMENTATION OF

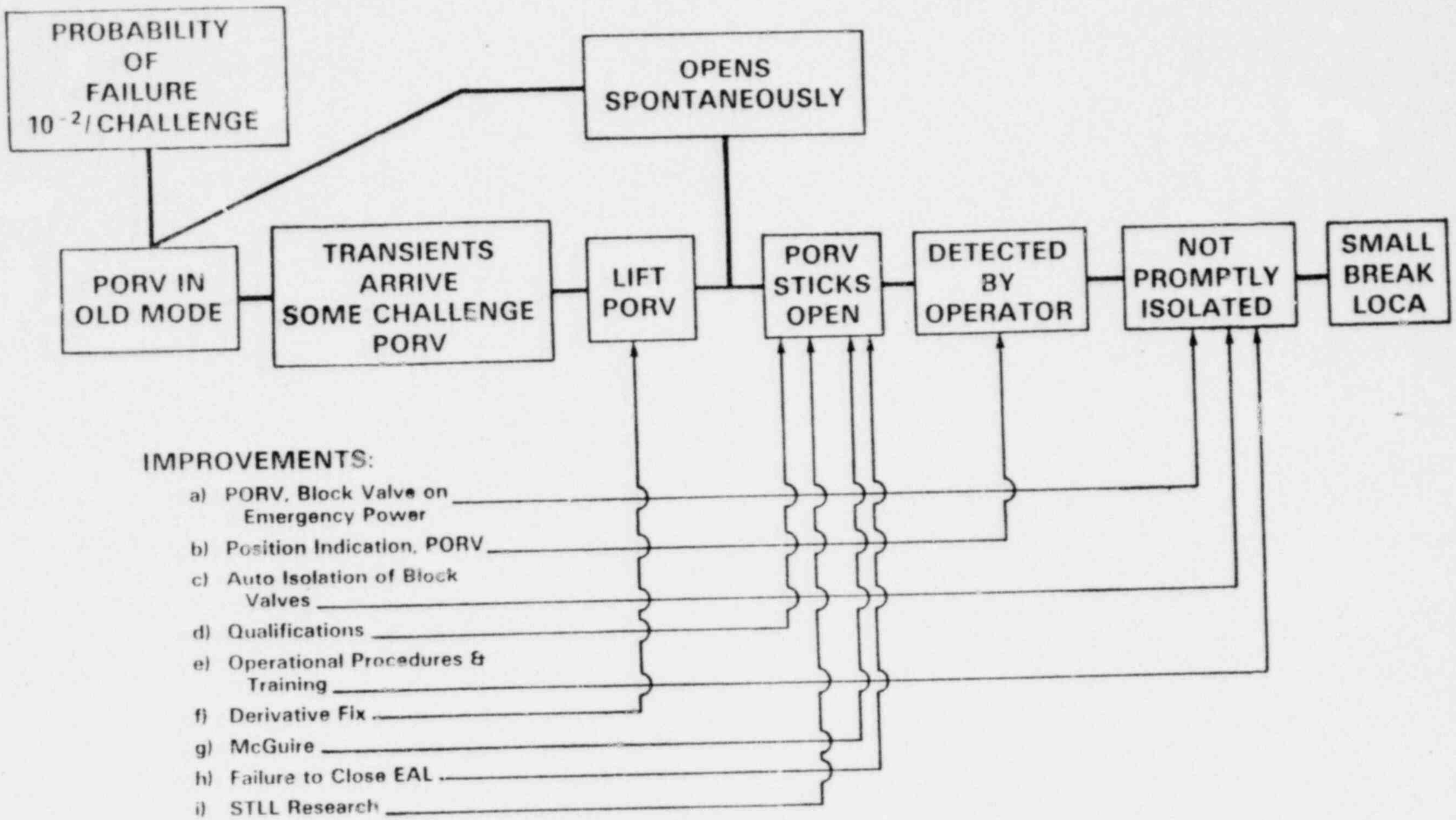
BULLETINS & ORDERS TASK FORCE RECOMMENDATIONS

Recommendation		Schedule
3.3.5	CCI-supplied PORV	Plant-specific
3.3.6	Auto Isolation of PORV	
	(a) Installation	7/1/80
	(b) Test	During first refueling outage following installation.
	(c) Westinghouse Report on PORV Modifications	10/1/80
	(d) Reporting Failures and Challenges to PORVs and SVs	
	- Challenges	Document in Annual Report
	- Failures	Promptly in Conformance with NUREG-0610
3.4.1	(a) Modifications to RELAP4 Heatup Calculation	NRC Action
3.4.1	(b) Effects of accumulator injection on RELAP4 Calculations	NRC Action
3.5.1	(a) Operator Training at Simulator	Commitment by 4/1/80
	(b) Simulator Modifications	7/1/80
3.6.1	NRC Review of Procedures	To be developed in NRC TMI-2 Action Plan

SCHEDULE FOR IMPLEMENTATION OF

BULLETINS & ORDERS TASK FORCE RECOMMENDATIONS

Recommendation		Schedule
3.6.2	NSSS Vendor Review of Procedures	To be developed in NRC TMI-2 Action Plan
3.6.3	Symptom-Based Procedures	NRC Action
3.7.1	Monitoring Control Board	1/1/80


PROBABILITY OF FAILURE 10^{-2}/CHALLENGE
OPENS SPONTANEOUSLY
PORV IN OLD MODE
TRANSIENTS ARRIVE SOME CHALLENGE PORV
LIFT PORV
PORV STICKS OPEN
DETECTED BY OPERATOR
NOT PROMPTLY ISOLATED
SMALL BREAK LOCA
IMPROVEMENTS:
a) PORV, Block Valve on Emergency Power
b) Position Indication, PORV
c) Auto Isolation of Block Valves
d) Qualifications
e) Operational Procedures & Training
f) Derivative Fix
g) McGuire
h) Failure to Close EAL
i) STLL Research

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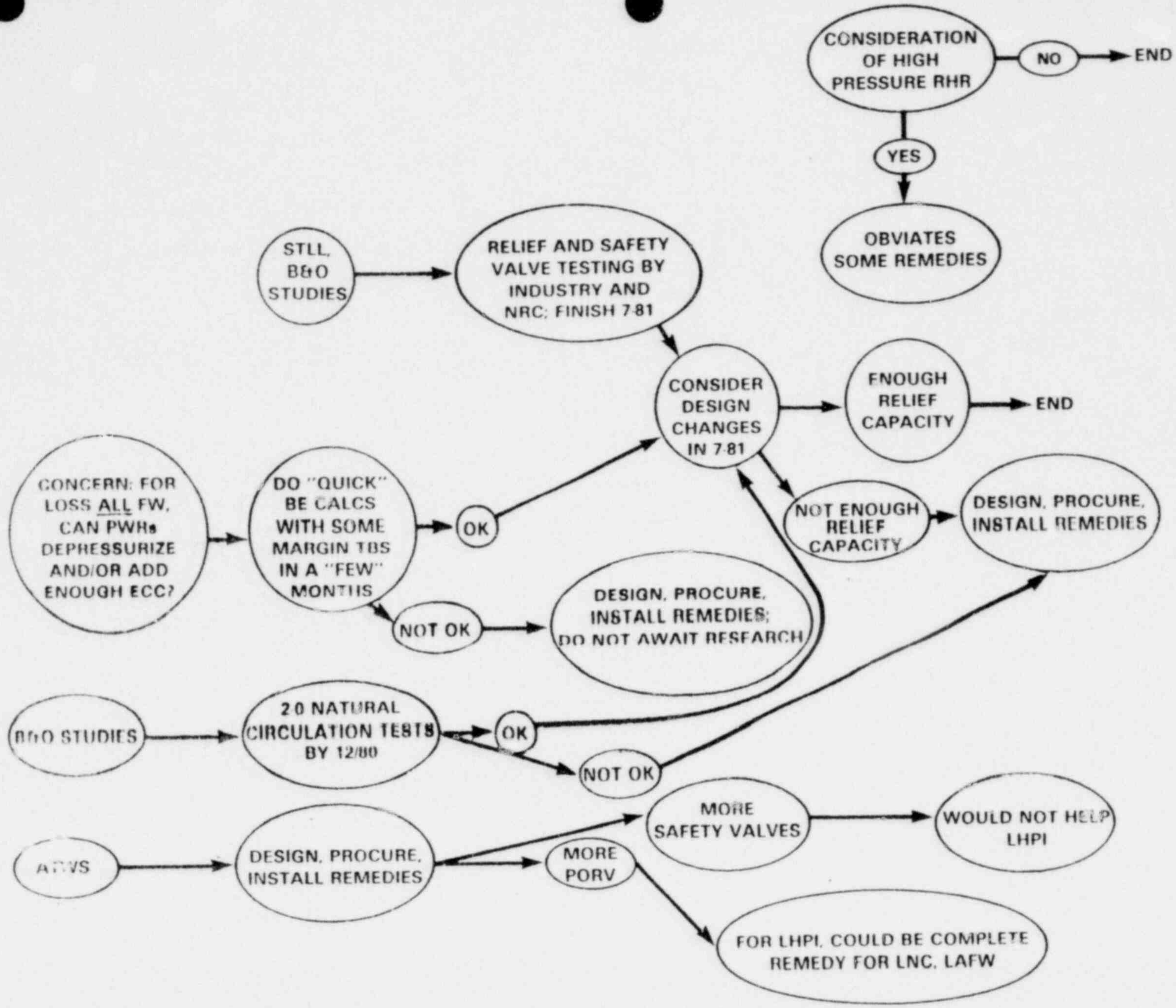
CONSIDERATION OF HIGH PRESSURE RHR
NO
END
YES
OBVIATES SOME REMEDIES
STLL, B&O STUDIES
RELIEF AND SAFETY VALVE TESTING BY INDUSTRY AND NRC; FINISH 7-81
CONSIDER DESIGN CHANGES IN 7-81
ENOUGH RELIEF CAPACITY
END
NOT ENOUGH RELIEF CAPACITY
DESIGN, PROCURE, INSTALL REMEDIES
CONCERN: FOR LOSS ALL FW, CAN PWRs DEPRESSURIZE AND/OR ADD ENOUGH ECC?
DO "QUICK" BE CALCS WITH SOME MARGIN TBS IN A "FEW" MONTHS
OK
NOT OK
DESIGN, PROCURE, INSTALL REMEDIES; DO NOT AWAIT RESEARCH
B&O STUDIES
2.0 NATURAL CIRCULATION TESTS BY 12/80
OK
NOT OK
MORE SAFETY VALVES
WOULD NOT HELP LHPI
DESIGN, PROCURE, INSTALL REMEDIES
MORE PORV
FOR LHPI, COULD BE COMPLETE REMEDY FOR LNC, LAFW

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING

JANUARY 11, 1980

APPENDIX X
WORK PRODUCTS OF BULLETINS AND ORDERS
TASK FORCE

WORK PRODUCTS OF BOTF

- BULLETINS TO ACHIEVE SHORT-TERM CHANGES IN PROCEDURES AND SYSTEMS; FOLLOWED UP BY LICENSEE SUBMITTALS AND STAFF SERs.

- REVIEW OF CONFIRMATORY ORDER SUBMITTALS (FOR B&W PLANTS) AND PREPARATION OF STAFF SERs.

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- INQUIRY INTO SBLOCA AND RELATED ANALYSES; REVIEW OF GENERIC SUBMITTALS; PREPARATION OF 6 GENERIC REPORTS.

- REVIEW AND APPROVAL OF NEW OR REVISED GUIDELINES FOR OPERATORS FOR EVENT DIAGNOSTICS AND SBLOCA.

- REVIEW OF AFW AND ISSUANCE OF PLANT-SPECIFIC UPGRADE REQUIREMENTS

- AUDIT OF PLANT OPERATORS FOR UNDERSTANDING OF TMI-2 EVENTS AND AWARENESS OF NEW PROCEDURES.

Tasks and Responsibilities Timetable for B&W-Designed Operating Plants

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Task Description	Completion Date
Documentation of Small Break Analytical Methods Including an Evaluation of Noding	July 1, 1980
Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	January 1, 1981
Evaluation of Two-Phase Natural Circulation Test Data Including the Effect of Noncondensable Gases	January 1, 1981
Evaluation of Pressurizer Spray Line and Isolation Valve Failure	May 1, 1980
Evaluation of a Small Break that is Isolated and Repressurizes the System to the PORV Setpoint	June 1, 1980
Evaluation of Water Slugs in Piping, Including Inertial Motions, Impact and Pressure Oscillations	March 31, 1980
Evaluation of Reactor Coolant Pump Seal Damage and Leakage During a Small Break LOCA	March 1, 1980

Tasks and Responsibilities Timetable for B&W-Designed Operating Plants (Cont.)

Task Description	Completion Date
Pump Running Evaluation of LOFT L3-6 (Presently Scheduled for Late March 1980)	Pre-Test
Effect of Noncondensible Gases: (a) From Radiolytic Decomposition; (b) On Steam Generator Heat Transfer; and, (c) Long-Term Cooling if Core Flood Tank Gas is Inadvertently Discharged.	March 31, 1980
Operator Guidelines for the Presence of Noncondensible Gases in the Reactor Coolant System	March 31, 1980
Evaluation of Slug Flow in Steam Generator Tubes, Condensation Loads	March 31, 1980
Conceptual Design and Justification for an Automatic Block Valve Closure System	March 1, 1980
Evaluation of PORV Opening Probability During Over-Pressure Transients	March 1, 1980
Safety Valve Reliability Evaluation	June 1, 1980

Schedule for Implementation of Bulletins & Orders Task Force Recommendations for W-Designed Operating Plants

Section No.	Title of Recommendation	Schedule
3.1.3.1	Technical Specification (TS) Time Limit on AFW System Train Outage (GS-1)	1/1/80
3.1.3.2	TS Administrative Control on Manual AFW System Valves Lock and Verify Position Valves (GS-2)	1/1/80
3.1.3.3	AFW System Flow Throttling-Water Hammer (GS-3)	1/1/80
3.1.3.4	Emergency Procedures for Initiating Backup Water Supplies (GS-4)	1/1/80
3.1.3.5	Emergency Procedures for Initiating AFW Flow Following Loss of All AC Power (GS-5)	1/1/80
3.1.3.6	AFW System Flow Path Verification (GS-6)	1/1/80
3.1.3.7	Non-Safety Grade Non-Redundant AFW System Automatic Initiation Signals (GS-7)	1/1/80
3.1.3.8	Automatic Initiation of AFW Systems (GS-8)	1/1/80
3.1.4.1	Primary AFW Source Low Level Alarm	1/1/80
3.1.4.2	AFW Pump Endurance Test	1/1/80
3.1.4.3	Indication of AFW Flow to the Steam Generators	1/1/80
3.1.4.4	AFW System Availability During Periodic Surveillance Testing	1/1/80
3.1.5.1	Automatic Initiation of AFW Systems (GL-1)	1/1/81
3.1.5.2	Single Valves in the AFW System Flow Path (GL-2)	1/1/81

Schedule for Implementation of Bulletins & Orders Task Force Recommendations for W-Designed Operating Plants (Cont.)

Section No.	Title of Recommendation	Schedule
3.1.5.3	Elimination of AFW System Dependency on AC Power Following a Complete Loss of AC Power (GL-3)	1/1/81
3.1.5.4	Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena (GL-4)	1/1/81
3.1.5.5	Non-Safety Grade, Non-Redundant AFW System Automatic Initiation Signals (GL-5)	1/1/81
3.2.1	Small Break LOCA Analyses	
(a)	Analysis Methods Appendix K	7/1/80
(b)	Plant-Specific Appendix K Calculations	1/1/81
3.2.2	Role of Non-Safety Equipment in Mitigating Small Break LOCAs	
(a)	RCP Pump Trip	1/1/81
(b)	Interaction of Safety and Non-Safety Systems	TMI-2 Action Plan
3.2.3	Michelson's Concerns	
(a)	Two-Phase Natural Circulation Experiments	1/1/81
(b)	Instrumentation to Verify Natural Circulation	4/1/80
3.2.4	PORV Failures in W-Designed Plants	
(a)	Confirmation of Anticipatory Trip	4/1/80
(b)	Evaluate Elimination of PORV Function	TMI-2 Action Plan

Schedule for Implementation of Bulletins & Orders Task Force Recommendations for W-Designed Operating Plants (Cont.)

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Section No.	Title of Recommendation	Schedule
(c)	PID Controller Modification	4/1/80
(d)	Proposed Anticipatory Trip Modification	Plant-Specific
(e)	CCI-Supplied PORV	Plant-Specific
(f)	Installation of Auto Isolation of PORVs*	7/1/80
(g)	Testing of Auto Isolation of PORVs*	During First Refueling Outage Following Installation
(h)	Westinghouse Report on PORV Failure Reduction	10/1/80
(i)	Reporting Failures and Challenges of PORVs and Safety Valves	Failures: Promptly per NUREG-0610 Challenges: In Annual Report
3.4.1	Audit Calculations	
(a)	Modifications to RELAP4 Heatup Calculation	NRC Action
(b)	Effects of Accumulator Injection on RELAP4 Calculations	NRC Action
(c)	Modification of RELAP4 to Represent Steam Generator Realistically	NRC Action
3.5.1	Expanded Use of Simulators in Operator Training	
(a)	Simulator Training Program	7/1/80
(b)	Simulation of Small Break LOCA	7/1/81
3.6.1	Review of Procedures (NRC)	TMI-2 Action Plan
3.6.2	Review of Procedures (NSSS Vendors)	TMI-2 Action Plan
3.6.3	Symptom-Based Emergency Procedures	NRC Action
3.7.1	Monitoring Control Board	4/1/80

Schedule for Implementing The Bulletins & Orders Task Force Recommendations on C-E Designed Plants

Section of Recommendation	Title of Recommendation	Schedule Date
3.1.3	Short Term Generic Recommendations (AFW Systems):	
3.1.3.1	Technical Specification (TS) Time Limit on AFW System Train Outage (GS-1)	01/01/80
3.1.3.2	TS Administrative Control on Manual Valves — Lock and Verify Position (GS-2)	01/01/80
3.1.3.3	AFW System Flow Throttling — Water Hammer (GS-3)	01/01/80
3.1.3.4	Emergency Procedures for Initiating Backup Water Supplies (GS-4)	01/01/80
3.1.3.5	Emergency Procedures for Initiating AFW Flow Following Loss of All AC Power (GS-5)	01/01/80
3.1.3.6	AFW System Flow Path Verification (GS-6)	01/01/80
3.1.3.7	Automatic Initiation of AFW System (GS-8)	01/01/80
3.1.4	Additional Short-Term Recommendations (AFW Systems):	
3.1.4.1	Primary AFW Source Low Level Alarm	01/01/80
3.1.4.2	AFW Pump Endurance Test	01/01/80
3.1.4.3	Indication of AFW Flow to the Steam Generators	01/01/80

Schedule for Implementing the Bulletins & Orders Task Force Recommendations on C-E Designed Plants (Cont.)

Section of Recommendation	Title of Recommendation	Schedule Date
3.1.4.4	AFW System Availability During Periodic Surveillance Testing	01/01/80
3.1.5	Long-Term Generic Recommendations (AFW Systems):	
3.1.5.1	Automatic Initiation of AFW System (GL-1)	01/01/81
3.1.5.2	Single Valves in AFW System Flow Path (GL-2)	01/01/81
3.1.5.3	Elimination of AC Power Dependency (GL-3)	01/01/81
3.1.5.4	Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena (GL-4)	01/01/81
3.2	Analysis:	
3.2.1	Confirmation of Small-Break LOCA Analysis Methods:	
(a)	Analysis Methods Appendix K	07/01/80
(b)	Plant-Specific Appendix K Calculations	01/01/81
3.2.2	Role of Non-Safety Equipment in Mitigating S-B LOCAs:	
(a)	Automatic Trip of RCPs	01/01/81
(b)	Review of Reliability & Redundancy of Equipment	TMI-2 Action Plan
3.2.3	Michelson Concerns:	
(a)	Two-Phase Natural Circulation Experiments	01/01/81
(b)	Instrumentation to Verify Natural Circulation	04/01/80
3.2.4	PORV Failures in C-E Plants:	
(a)*	Installation of Automatic Isolation of PORVs	07/01/80
(b)*	Testing Automatic Isolation of PORVs	First Refueling Outage After Installation
(c)	C-E Report on PORVs Failure Reductions	10/01/80

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Schedule for Implementing the Bulletins & Orders Task Force Recommendations on C-E Designed Plants (Cont.)

Section of Recommendation	Title of Recommendation	Schedule Date
(d)	Reporting Future Failures and Challenges of PORVs and SVs	Failures: Promptly per NUREG-0610; Challenges: In Annual Report
(e)	Evaluate Elimination of PORV Function	TMI-2 Action Plan
3.2.5	Audit Calculations:	
(a)	Modification to RELAP and CEFLASH-4AS Due to Uncertainties in Heatup Calculations	RELAP: TMI-2 Action Plan CEFLASH: 07/01/80
(b)	Effects of Accumulator Injection on RELAP-4 Calculation	NRC Action
(c)	Modification of RELAP-4 to Represent SG Behavior Realistically	NRC Action
3.3	Operator Training:	
3.3.1	Expanded Use of Simulators in Operator Training:	
(a)	Simulator Training Program	07/01/80
(b)	Simulation of Small-Break LOCAs	01/01/81
3.4	Operating Procedures:	
3.4.1	Review of Procedures (NRC)	TMI-2 Action Plan
3.4.2	Review of Procedures (NSSS Vendors)	TMI-2 Action Plan
3.4.3	Symptom-Based Emergency Procedures	TMI-2 Action Plan
3.5	Human Factors:	
3.5.1	Monitoring Control Board	04/01/80

Implementation of Recommendations for Operating and Near-Term OL BWR's

Recom. Number	Abbreviated Title	Action Required	Implementation Category[a]
A.1	Separation of HPCI and RCIC Initiation Levels	1) Analysis	S
		2) Implementation	L
A.2	Isolation of Isolation Condensers on High Radiation	Modify Isolation Circuitry	S
A.3	Spurious Isolation of HPCI and RCIC	Modify Break Detection Circuitry	S
A.4	Reduction of Challenges and Failures of Relief Valves	1) Feasibility Study	S
		2) System Modification	L
A.5	Identify Water Source Prior to Manual ADS	Modify Guidelines and Procedures	S
A.6*	Report on Outage of ECC Systems	1) Report Submittal	SS
		2) Plant-Specific Tech Spec Changes	S
A.7	Modification of ADS Logic	1) Feasibility Study for Staff Review	S
		2) Modification to ADS Logic	L
A.8	Interlock on Recirculation Pump Loops	Install Interlocks for Non-Jet Pump Plants	S
A.9	Loss of Service Water for Big Rock Point	Verify Acceptability of Consequences	S
A.10	Restart of Core Spray and LPCI on Low Level	1) Preliminary Design	S
		2) Modification of Restart Logic	L

[a] Category S: Implement by June 30, 1980
Category SS: Implement Within 60 Days of this Report
Category SSS: Implement by January 31, 1980
Category L: Implementation by January 1, 1981
Category LL: Implementation by January 1, 1983

Implementation of Recommendations for Operating and Near-Term OL BWR's (Cont.)

Recom. Number	Abbreviated Title	Action Required	Implementation Category[a]
A.11	Revised Emergency Procedures	All Operators Must Have Read Prior to Going on Duty	SSS
A.12*	Revise Small Break LOCA Model for Compliance with Appendix K	1) Revise Model	S
		2) Compare with TLTA Data	S
A.13	Plant Specific Analysis with Revised Model	Submit Analyses with Revised Model	L
A.14	No Fuel Failure Requirement for Anticipated Transient with Single Failure	Verify Compliance with Requirement	S
A.15	Depressurization with Other than ADS	Analyses to Support Other Modes	S
A.16	Two Operators in Control Room	Minimum of Two Operators in Control Room	SSS
A.17	Michelson Concerns	GE Address Concerns	SSS
B.1	Automatic Switchover of RCIC Suction	1) Verify Procedures	SS
		2) Design Modification	L
B.2	Central Water Level Recording	Installation of Recorders	L
B.3	Space Cooling for HPCI and RCIC	Demonstrate Minimum of Two Hour Capability	L
B.4	Effect of Loss of AC Power on Pump Seals	Demonstrate Adequacy of Seal Design	L
B.5	Use of RHR for Fuel Pool Cooling	Risk Assessment	L
B.6	Common Reference for Level Instruments	Modify Scale to Obtain Common Reference	S
B.7	Qualification of Accumulators on ADS Valves	Show Acceptability	L

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Implementation of Recommendations for Operating and Near-Term OL BWR's (Cont.)

Recom. Number	Abbreviated Title	Action Required	Implementation Category[a]
B.8	Guidelines for Symptom-Based Emergency Procedures	Develop New Guidelines	LL
B.9	Test Program for SBLOCA Model Verification	1) Pre-Test Pred. of 1st Two Tests	SSS
		2) Develop Test Program	S
		3) Model Verification	LL
B.10	Diverse Initiation Signal for RCIC	Upgrade if Required	L
B.11	Small Break LOCA on Simulators	Upgrade Simulator	L
B.12	Use of Non-ECC Systems in Analysis	1) Review System Capability	L
		2) Upgrade if Needed	
B.13	Performance of Isolation Condensers with Non-Condensables	Demonstrate Adequacy	L
B.14	Reporting of Failures and Challenges to SRVs	Prompt Reporting of Failures and Annual Report of Challenges	N/A
B.15	Impact of B&O Recommendations	Assess Impact on Safety and Reliability	L

Oless

PLANT AUDITS

OBJECTIVE

ENSURE SMALL BREAK LOCA EMERGENCY PROCEDURE WAS CONFORMED TO GUIDELINES.

EVALUATE TRAINING OF LICENSED OPERATORS AND SENIOR OPERATORS CONCERNING:

TMI-2 ACCIDENT

SMALL BREAK PHENOMENON

REVISED LOCA PROCEDURES

CONCLUSIONS

1. SMALL BREAK LOCA PROCEDURES CAN BE IMPLEMENTED BY DECEMBER 31, 1979

2. RETRAINING ASSOCIATED WITH SMALL BREAK LOCA's CAN BE ACCOMPLISHED BY DECEMBER 31, 1979.

3. IMPROVEMENTS NEEDED IN:

   A. PWR'S -- OPERATOR'S DEPTH OF KNOWLEDGE IN THERMODYNAMICS, HEAT TRANSFER, AND FLUID FLOW.

   B. BWR'S -- OPERATOR'S UNDERSTANDING OF REACTOR VESSEL LEVEL INSTRUMENTATION.

PLANTS AUDITED

	PLANT	DATE	NO. OF LICENSED PEOPLE
1.	OCONEE	MAY 11-15	14
2.	ARKANSAS I	MAY 20-23	9
3.	RANCHO SECO	JUNE 1 & 2	7
4.	DAVIS-BESSE	JUNE 6 - 8	9
5.	CRYSTAL RIVER	JUNE 19	8
6.	SALEM	DEC 10	4
7.	NINE MILE POINT	DEC 10	3
8.	FITZPATRICK	DEC 11	3
9.	DRESDEN 2/3	DEC 12	3
10.	MILLSTONE II	DEC 19	4

IDENTIFIED RESEARCH NEEDS FROM B&OTF

1. DRAFT MEMO DENTON/LEVINE	REQUESTS RES TO FOLLOW EPRI PROGRAM FOR PERFORMANCE TESTING OF PORV AND SAFETY VALVES 1. MONITOR PROGRAM - COLLECT DATA - EVALUATE DATA 2. DEVELOP FLOW MODEL 3. SPECIFIC INFORMATION REQUESTED: FOR CONDITIONS OF SUBCOOLED LIQUID/TWO-PHASE FLOW AND SATURATED STEAM + CALCULATE FLOW THROUGH VALVES + DETERMINE VALVE OPERATING CHARACTERISTICS + DETERMINE EFFECT OF BACK PRESSURE ON CAPACITY + CALCULATE FORCES ON VALVE
2. MEMO DENTON/LEVINE	IDENTIFIED EXPERIMENTAL NEEDS FOR TWO-PHASE NATURAL CIRCULATION AND RCP PERFORMANCE 1. NEED LOFT TESTS WHICH DEMONSTRATE VARIOUS MODES OF TWO-PHASE NATURAL CIRCULATION 2. IDENTIFY CONDITIONS WHICH COULD STOP NATURAL CIRCULATION 3. DETERMINE ABILITY TO HAVE NATURAL CIRCULATION IN THE PRESENSE OF NONCONDENSIBLE GASES 4. LOFT TEST - SBLOCA WITH RCPs RUNNING WITH NUCLEAR HEAT 5. PERFORMANCE OF RCPs IN TWO-PHASE FLUID (FULL OR PART SCALE)

3. MEMO DENTON/LEVINE 12/03/79

NRR ENDORSEMENT OF THE FY 80 WORKSCOPE FOR THE PWR BDHT PROGRAM AND THE REDIRECTED SEMISCALE PROGRAM AND FACILITY UPGRADE

1. SEMISCALE TESTING NEEDS AND PRIORITIES
   - 1A - CL-SBLOCA-RCPs RUNNING THROUGHOUT ENTIRE TEST
   - 1B - CL-SBLOCA-RCPs TRIPPED AT HIGH VOID FRACTION
   - 1C - CL-SBLOCA-RCPs TRIPPED UPON REACTOR TRIP
   - 2A - CL-SBLOCA-SYSTEM DEPRESSURIZATION
   - 2B - CL-SBLOCA-SYSTEM PRESSURE STABILIZATION AT INTER. PRESS.
   - 2C - CL-SBLOCA-SYSTEM REPRESSURIZATION
   - 3A - SAME AS 1A BUT HOT LEG BREAK (HL)
   - 3B - SAME AS 1B BUT HL BREAK
   - 3C - SAME AS 1C BUT HL BREAK
   - 4A - TWO-PHASE NATURAL CIRCULATION-POOL BOILING IN THE CORE STEAM CONDENSATION IN THE STEAM GENERATORS
   - 4B - TWO-PHASE NATURAL CIRCULATION WITH VESSEL LEVEL AT TOP OF HOT LEG TO ASSESS SLUG FLOW
2. LONGER TERM TESTING
   - S-07-6 REPEAT
   - UHI SERIES INCLUDING UHI SBLOCA

4. MEMO DENTON/LEVINE 11/28/79

REQUESTED ARRANGEMENT OF AN ADDITIONAL TLTA TEST (WITH ECC AVAILABLE)

5.

VERBAL REQUEST TO PERFORM INDEPENDENT PRETEST PREDICTIONS FOR TLTA TESTS UNDER EXISTING INEL CONTRACT

6.	MEMO ROSS/TONG	11/08/79	REQUESTED INFORMATION ON CORE THERMAL-HYDRAULIC BEHAVIOR DURING SBLOCA 1. DATA ON CLADDING TO STEAM HEAT TRANSFER 2. DATA ON STEAM ENTHALPY RISE 3. TWO-PHASE MIXTURE LEVEL 4. QUENCH BEHAVIOR DURING SBLOCA CORE UNCOVERY
7.	VERBAL AGREEMENT DENTON/LEVINE	10/01/79	IREP - INITIAL PLANT STUDY - CRYSTAL RIVER 3 CONDUCT A LIMITED RISK ASSESSMENT OF A B&W REACTOR AIMED AT IDENTIFYING ANY UNIQUE RISK-IMPACTING SEQUENCES RELATIVE TO THE REACTOR SAFETY STUDY

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RECOMMENDATIONS NOT INCLUDED IN B&OTF GENERIC REPORTS

* 1. AN NRC POSITION ON REQUIRED CONSERVATISM IN SMALL BREAK ANALYSIS SHOULD BE BY 6/30/80.

* 2. UNLESS THE CAPABILITY OF THE RELIEF VALVES TO PROVIDE SUFFICIENT DEPRESSURIZATION IN CASE OF LOSS OF HEAT REMOVAL THROUGH THE STEAM GENERATORS CAN BE DEMONSTRATED WITH DUE ACCOUNT OF INPUT UNCERTAINTIES AND CALCULATIONAL UNCERTAINTIES. CONTINUED OPERATION OF PWR'S WITH LOW CUTOFF HEAD HPI SYSTEMS SHOULD BE CONDITIONED ON A TIMELY DESIGN CHANGE. REPLACEMENT OF THE HPI PUMPS WITH HIGH CUTOFF HEAD PUMPS, INCREASED RELIEF VALVE CAPACITY OR INSTALLATION OF A HIGH PRESSURE RHR SYSTEM ARE POSSIBLE ACCEPTABLE CHANGES.

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* 3. THE TECHNICAL SPECIFICATIONS FOR ECC SYSTEMS SHOULD BE MODIFIED TO INCLUDE AN APPROPRIATE LIMIT ON THE ACCUMULATED OUTAGE TIME PER YEAR.

** 4. THE FREQUENCY OF RELIEF VALVE CHALLENGES SHOULD BE REDUCED SUBSTANTIALLY (BY AN ORDER OF MAGNITUDE) IN CE, W, AND GE PLANTS.

* DISCUSSED IN B&OTF FINAL REPORT (NUREG-0645).

** ALTERNATIVE RECOMMENDATIONS APPEAR IN B&OTF GENERIC REPORTS.

OTHER MATTERS

(1) CONSERVATISMS IN LOCA ANALYSES

(2) LOSS OF ALL FEEDWATER IN PWRs

(3) DESIGN BASES FOR AFW SYSTEMS

(4) AUDITS OF LICENSED OPERATORS

(5) RECOMMENDATIONS RESULTING FROM AUDITS OF BWR PLANTS

(6) LOSS OF ALL AC POWER

(7) TECHNICAL SPECIFICATION LIMIT ON CUMULATIVE OUTAGE TIME FOR SAFETY EQUIPMENT

ECCS RULE

STATUS SUMMARY

&

APPLICABILITY TO SMALL BREAK

LOCAs

ECCS RULE CHANGE

- STAFF REQUESTED COMMISSION APPROVAL OF PROPOSED ACTION PLAN TO MODIFY ECCS RULE IN 10 CFR 50.46 AND APPENDIX K TO 10 CFR PART 50 ON JANUARY 18, 1978 (SECY 7826).
- COMMISSION APPROVED PROPOSED ACTION PLAN AT JUNE 12, 1978 AFFIRMATION SESSION.
- FEDERAL REGISTER NOTICE OF PROPOSED RULEMAKING ISSUED IN FR, VOL. 43, NO. 235 - WEDNESDAY, DECEMBER 6, 1978.
    - NOTICE REQUESTED ADVICE AND RECOMMENDATIONS ON PROPOSED AREAS OF REVISION
    - COMMENTS REQUESTED ON 5 SPECIFIC QUESTIONS RELATED TO PROPOSED REVISIONS

RULE CHANGE (PRIOR TO TMI-2)

- TWO PHASES OF RULE CHANGE
  - PROCEDURAL RULE CHANGE
    - NEGLIGIBLE EFFECT ON OVERALL CONSERVATISM
    - 6 MONTHS FOR PREPARATION IN NRR FOR RULE CHANGE REQUEST
    - 12-18 MONTHS TO COMPLETE RULEMAKING (NO PUBLIC HEARING)
  - TECHNICAL RULE CHANGE
    - BASED ON NEW INFORMATION FROM RESEARCH AND EXPERIENCE
    - WOULD AFFECT OVERALL CONSERVATISM - ASSESSMENT NEEDED
    - 18 MONTHS FOR CONSERVATISM IMPACT ASSESSMENT AND RULE CHANGE REQUEST
    - 18-24 MONTHS TO COMPLETE RULEMAKING PROCESS. (NO PUBLIC HEARING)

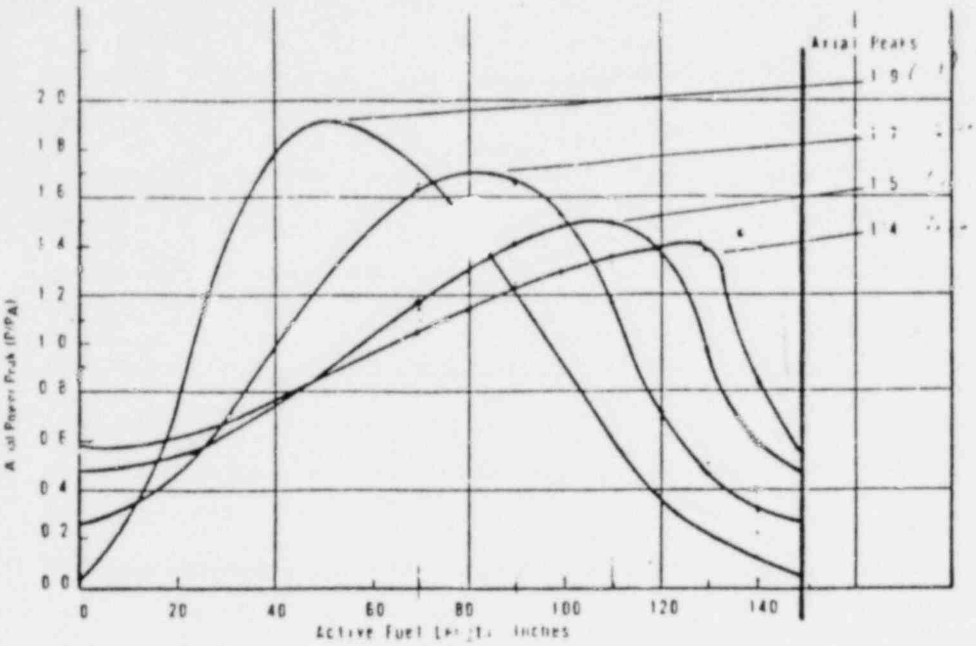
REQUIREMENTS OF APPENDIX K TO 10CFR50

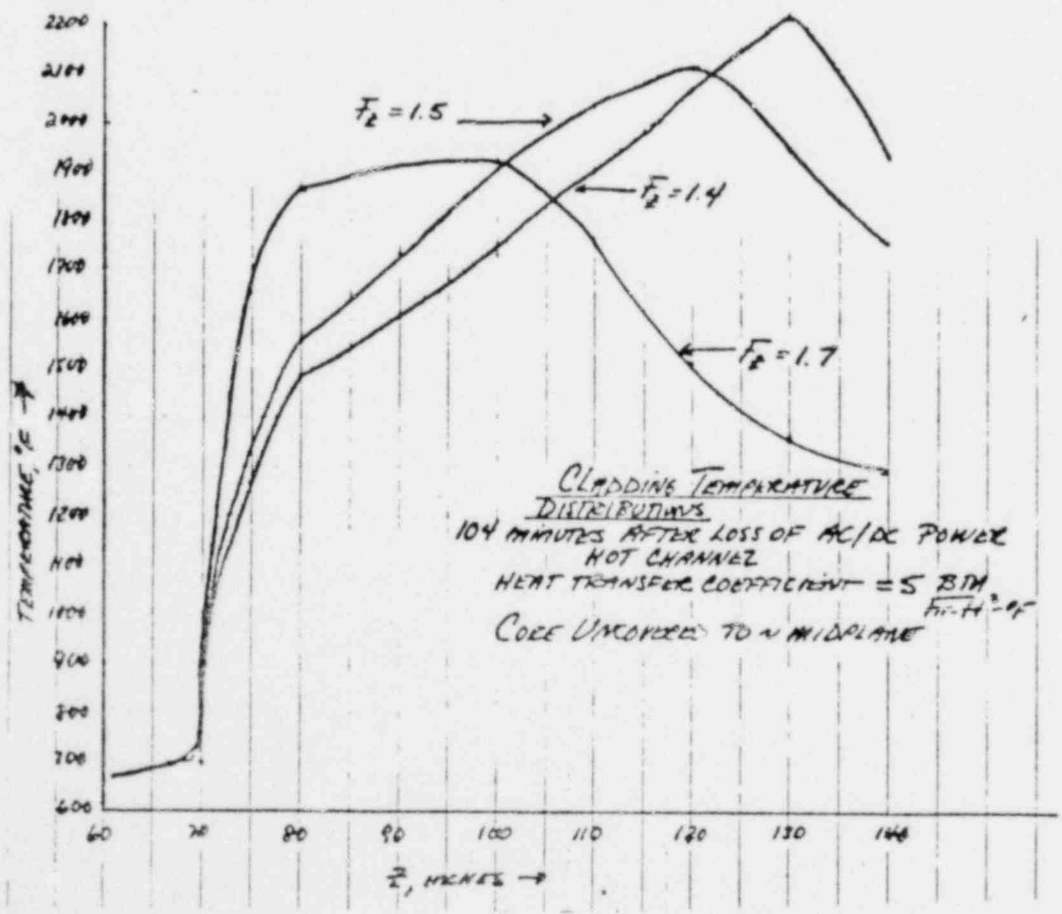
A.	Heat Sources Power Level	1.02 x Licensed Power Level
	Prior Operation	Continuous
	Peaking Factor	Maximum Technical Specification
	Power Distribution Shapes	Examine for most conservative shape/peaking factor combination
	Stored Energy in Fuel	Use Burnup or Stored Energy that Yields highest PCT
	Void and Temp. Coefficients of Reactivity	Minimum plausible values including uncertainties
	Fission Product Decay	1.2 x 1971 ANS Proposed Standard infinite irradiation at maximum peaking factor
	Metal-Water Reaction	Baker-Just/Not steam-limited/ Inside-Outside Reaction 1.5 inch minimum reaction distance in each direction from rupture
B.	Swelling and Rupture of the cladding and Fuel Rod Thermal Parameters	None Specified
C.	Blowdown Phenomena Break Time	Instantaneous
	Break Flow	Use Moody Model when flow is two-phase
	Break Area	Find worst side

	Return to Nucleate Boiling	No return to Nucleate Boiling until reflood once CHF predicted to occur
	End of Blowdown ECC Water Injected During Bypass (PWR Only)	Subtract from vessel inventory
D.	Post-Blowdown Phenomena; Heat Removal by the ECCS	
	Single Failure Criteria	Assume most damaging single failure of ECCS equipment
	Containment Pressure	Calculate conservatively and assume all pressure-reducing equipment operable
	Pump Condition duirng reflood	Locked Rotor if worst case

APPENDIX K CONSERVATISMS APPLICABLE TO SMALL BREAK LOCAS

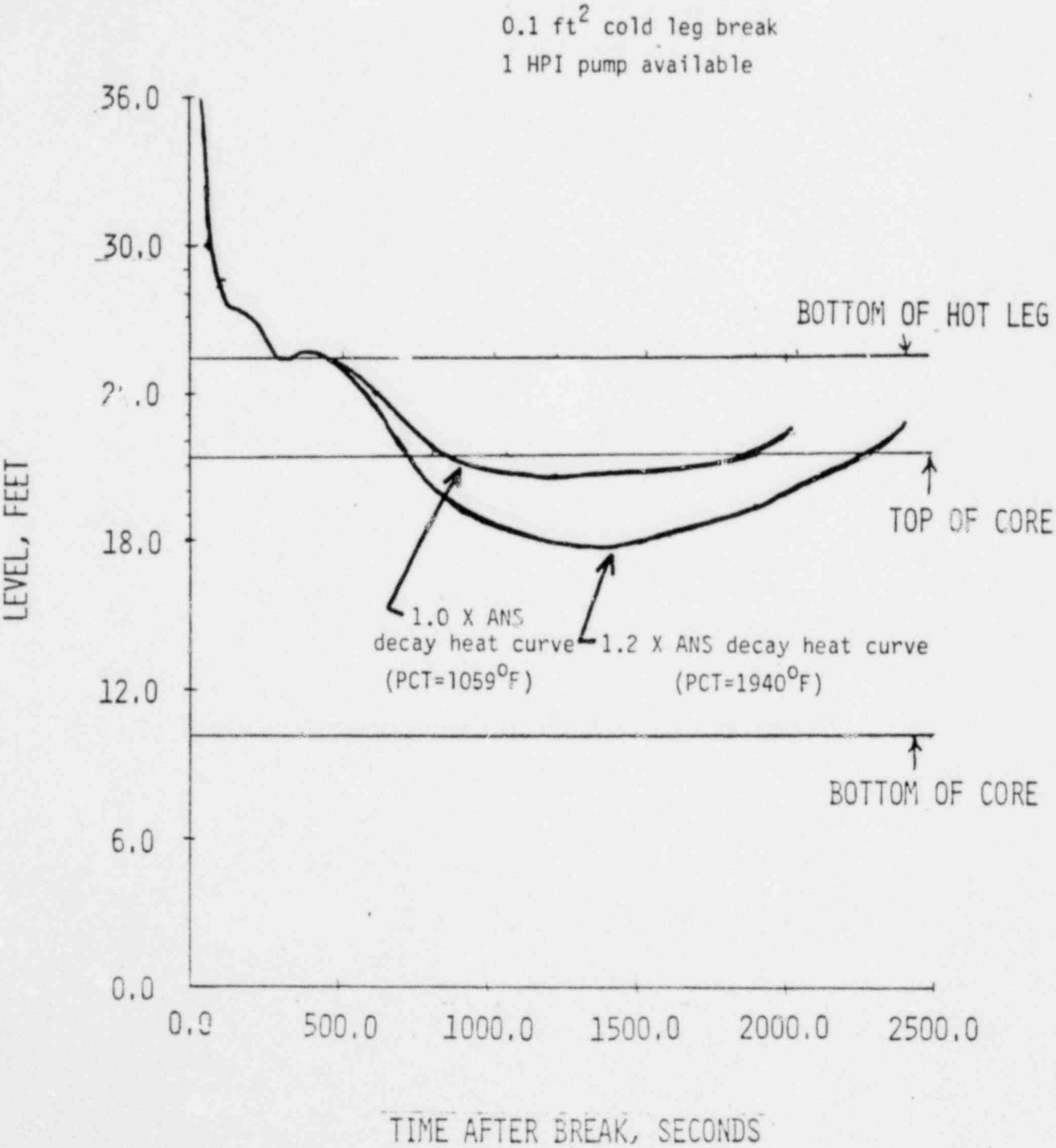
- 1.02 X LICENSED POWER LEVEL
- INFINITE IRRADIATION ASSUMPTION
- CONSERVATIVE AXIAL SHAPE/PEAKING FACTOR COMBINATION
- 1.2 X 1971 ANS DECAY HEAT CURVE
- SINGLE FAILURE CRITERIA





EFFECT OF AXIAL SHAPE/PEAKING FACTOR COMBINATION
ON CLADDING TEMPERATURE

EFFECT OF DECAY HEAT MULTIPLIER (1.2) ON TWO-PHASE MIXTURE LEVEL FOR SMALL BREAKS IN CE PLANTS



RESPONSE TO ADVANCED NOTICE ON ECCS RULEMAKING

Private	Utilities	Vendors	Government
3	15	5	2

Major Comments

1. Model Should Be Based on Realistic Analysis

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2. Rule Should Permit Greater Flexibility to Meet Criteria and Use Research Information

3. Phase 1 Scope Should Be Expanded to Include New Decay Heat and Zircaloy Oxide Data

4. ECCS Should Be Treated as Other DBA's

5. No Extensive Rulemaking - Just Reinterpretation

SUMMARY OF SMALL LOSS-OF-COOLANT AND LOSS-OF-FEEDWATER ACCIDENT EVALUATIONS

SLIDES PRESENTED BY ZOLTAN R. ROSZTOCZY AT THE JANUARY 11TH MEETING OF THE ACRS

JANUARY, 1980

	B&W		CE	WESTINGHOUSE		GE
	before 5/1/79	after 5/1/79		High head HPI	Low Head HPI	
CHALLENGE RATE OF RELIEF VALVES (NO. OF VALVES OPENED PER REACTOR YEAR)	3	0.2	1	2*		15
FAILURE RATE OF RELIEF VALVES (NO. OF FAILURES TO CLOSE PER OPENING)	1/20		1/20	1/20*		1/20
SIZE OF RELIEF VALVE (IN^2)	1.1		1.4	1.4		14
CONSEQUENCE OF A STUCK OPEN RELIEF VALVE WITH SINGLE FAILURE ASSUMPTIONS	NO CORE UNCOVERY		NO CORE UNCOVERY	NO CORE UNCOVERY		PARTIAL CORE UNCOVERY
LIKELIHOOD OF EXTENDED LOSS OF NATURAL CIRCULATION	UNLIKELY		HIGHLY UNLIKELY	HIGHLY UNLIKELY		N/A
CONSEQUENCE OF A STUCK OPEN RELIEF VALVE WITH EXTENDED LOSS OF NATURAL CIRCULATION	** 40 min. available to initiate HPI		Possible Core Melt	60 min available to initiate HPI	Possible Core Melt	N/A
CONSEQUENCE OF TEMPORARY LOSS OF ALL FEEDWATER (FEEDWATER AND RCIC FOR BWRs)	*** 20 min. available to initiate HPI or AFW		30 min. available to initiate AFW	30 min. available to initiate HPI or AFW	30 min. available to initiate AFW	NO CORE UNCOVERY
CONSEQUENCE OF STUCK OPEN RELIEF VALVE WITH TEMPORARY LOSS OF ALL FEEDWATER	*** 40 min. available to initiate HPI or AFW		60 min. available to initiate AFW	60 min. available to initiate HPI or AFW	60 min. available to initiate AFW	7 min. available to initiate ADS
CONSEQUENCE OF EXTENDED LOSS OF ALL FEEDWATER	** 20 min. available to initiate HPI		Possible Core Melt	30 min. available to initiate HPI	Possible Core Melt	20 min. available to initiate ADS
CONSEQUENCE OF STUCK OPEN RELIEF VALVE WITH EXTENDED LOSS OF ALL FEEDWATER	** 40 min. available to initiate HPI or AFW		Possible Core Melt	60 min. available to initiate HPI or AFW	Possible Core Melt	4 min. available to initiate ADS

1. PWR calculations assumed prompt tripping of the reactor coolant pumps
2. Results apply only if accumulative outage time of ECCS is very small
3. Time available for operator action is based on no core uncovery

*Information incomplete, actual values might be more favorable

**In case of the Davis Besse plant the consequence is "possible core melt"

***In case of the Davis Besse plant AFW has to be initiated

SUMMARY OF SMALL BREAK LOCA EVALUATION

- POSSIBLE FAILURE OF A RELIEF VALVE TO CLOSE ON A NUCLEAR PLANT, AS IT HAPPENED AT TMI, IS A GENERIC, INDUSTRY WIDE PROBLEM, APPLICABLE TO ALL U.S. DESIGNS INCLUDING PWR'S AND BWR'S.

- AT THE PRESENT MODE OF OPERATION OF THE PLANTS THE GE BWR'S ARE EXPECTED TO EXPERIENCE THE LARGEST NUMBER OF RELIEF VALVE FAILURES FOLLOWED BY THE WESTINGHOUSE PWR'S, THE CE PWR'S AND FINALLY THE B&W PWR'S.

- BWR RELIEF VALVES ARE TYPICALLY TEN TIMES LARGER THAN PWR RELIEF VALVES. CONSEQUENTLY FAILURE OF A BWR VALVE TO CLOSE COULD HAVE MORE SERIOUS CONSEQUENCES, IT COULD RESULT IN REACTOR CORE UNCOVERY, WHILE CORE UNCOVERY IS NOT EXPECTED FOR PWR'S.

- PWR'S WITH LOW CUT OFF HEAD HPI ARE POSSIBLY NOT PROTECTED FOR THE EXTENDED LOSS OF ALL FEEDWATER AND FOR THE EXTENDED LOSS OF NATURAL CIRCULATION EVENTS. ALL CE PLANTS, HALF OF THE WESTINGHOUSE DESIGNS AND DAVIS-BESSE AMONG THE B&W DESIGNS FALL INTO THIS CATEGORY.

- CALCULATIONAL METHODS USED FOR SMALL BREAK LOCA ANALYSIS HAVE NOT YET BEEN PROPERLY VERIFIED AND HAVE LARGE UNCERTAINTIES. THE UNCERTAINTIES OF THE CALCULATIONS POSSIBLY EXCEED THE MARGIN REQUIRED BY THE ECCS ACCEPTANCE CRITERIA.

- APPROPRIATE CORRECTIONS HAVE BEEN RECOMMENDED FOR THE EXISTING SHORTCOMINGS. THE RECOMMENDATIONS, WHEN IMPLEMENTED, WILL PROVIDE REASONABLE ASSURANCE THAT CONTINUED OPERATION OF THE PLANTS DOES NOT REPRESENT AN UNDUE RISK TO PUBLIC HEALTH AND SAFETY.

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REPORT OF THE SITING POLICY TASK FORCE

APPENDIX XV
HIGHLIGHTS OF THE SITING POLICY TASK FORCE REPORT, NUREG-0625

CHARACTERISTICS OF PRESENT POLICY AND PRACTICE

- SITING DECISION IS NOW CLOSELY COUPLED WITH PLANT DESIGN DECISION
  - HAS RESULTED IN IMPROVED DESIGN
  - SITING HAS BEEN DEEMPHASIZED AS A FACTOR IN DEFENSE IN DEPTH
- AMBIVALENT TREATMENT OF CLASS 9 ACCIDENTS
  - STATEMENT OF CONSIDERATIONS TO PART 100 INCLUDES LARGE ACCIDENTS
  - REGULATIONS EMPHASIZE DBA
- GENERAL SITING POLICY ALLOWING FLEXIBILITY BUT PROVIDING LITTLE DEFINITIVE GUIDANCE

SITING POLICY CHANGES

- GOALS

  - TO STRENGTHEN SITING AS A FACTOR IN DEFENSE IN DEPTH BY ESTABLISHING REQUIREMENTS FOR SITE APPROVAL THAT ARE INDEPENDENT OF PLANT DESIGN CONSIDERATION.

  - TO TAKE INTO CONSIDERATION IN SITING THE RISK ASSOCIATED WITH ACCIDENTS BEYOND THE DESIGN BASIS (CLASS 9) BY ESTABLISHING POPULATION DENSITY AND DISTRIBUTION CRITERIA.

  - TO REQUIRE THAT SITES SELECTED WILL TEND TO MINIMIZE THE OVERALL RISK FROM ENERGY GENERATION

- RECOMMENDATIONS

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RECOMMENDATION 1

REVISE PART 100 TO CHANGE THE WAY PROTECTION IS PROVIDED FOR ACCIDENTS BY INCORPORATING A FIXED EXCLUSION AND PROTECTION ACTION DISTANCE AND POPULATION DENSITY AND DISTRIBUTION CRITERIA.

1. SPECIFY A FIXED MINIMUM EXCLUSION DISTANCE BASED ON LIMITING THE INDIVIDUAL RISK FROM DESIGN BASIS ACCIDENTS. FURTHERMORE, THE REGULATIONS SHOULD CLARIFY THE REQUIRED CONTROL BY THE UTILITY OVER ACTIVITIES TAKING PLACE IN LAND AND WATER PORTIONS OF THE EXCLUSION AREA.

2. SPECIFY A FIXED MINIMUM EMERGENCY PLANNING DISTANCE OF 10 MILES. THE PHYSICAL CHARACTERISTICS OF THE EMERGENCY PLANNING ZONE SHOULD PROVIDE REASONABLE ASSURANCE THAT EVACUATION OF PERSONS, INCLUDING TRANSIENTS, WOULD BE FEASIBLE IF NEEDED TO MITIGATE THE CONSEQUENCES OF ACCIDENTS.

RECOMMENDATION 1
(CONT'D.)

3. INCORPORATE SPECIFIC POPULATION DENSITY AND DISTRIBUTION LIMITS OUTSIDE THE EXCLUSION AREA THAT ARE DEPENDENT ON THE AVERAGE POPULATION OF THE REGION.

4. REMOVE THE REQUIREMENT TO CALCULATE RADIATION DOSES AS A MEANS OF ESTABLISHING MINIMUM EXCLUSION DISTANCES AND LOW POPULATION ZONES.

RECOMMENDATION 2

REVISE PART 100 TO REQUIRE CONSIDERATION OF THE POTENTIAL HAZARDS POSED BY MAN-MADE ACTIVITIES AND NATURAL CHARACTERISTICS OF SITES BY ESTABLISHING MINIMUM STANDOFF DISTANCES FOR:

1. MAJOR OR COMMERCIAL AIRPORTS,
2. LNG TERMINALS,
3. LARGE PROPANE PIPELINES,
4. LARGE NATURAL GAS PIPELINES,
5. LARGE QUANTITIES OF EXPLOSIVE OR TOXIC MATERIALS,
6. MAJOR DAMS, AND
7. CAPABLE FAULTS

RECOMMENDATION 3

REVISE PART 100 BY REQUIRING A REASONABLE ASSURANCE THE INTERDICTIVE MEASURES ARE POSSIBLE TO LIMIT GROUNDWATER CONTAMINATION RESULTING FROM CLASS 9 ACCIDENTS WITHIN THE IMMEDIATE VICINITY OF THE SITE.

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RECOMMENDATION 4

REVISE APPENDIX A TO 10 CFR 100 TO BETTER REFLECT THE EVOLVING TECHNOLOGY IN ASSESSING SEISMIC HAZARDS.

RECOMMENDATION 5

REVISE PART 100 TO INCLUDE CONSIDERATION OF POST-LICENSING CHANGES IN OFFSITE ACTIVITIES:

1. THE NRC STAFF SHALL INFORM LOCAL AUTHORITIES (PLANNING COMMISSION, COUNTY COMMISSIONS, ETC.) THAT CONTROL ACTIVITIES WITHIN THE EMERGENCY PLANNING ZONE (EPZ) OF THE BASIS FOR DETERMINING THE ACCEPTABILITY OF A SITE.

2. THE NRC STAFF SHALL NOTIFY THOSE FEDERAL AGENCIES AS IN ITEM 1 ABOVE THAT MAY REASONABLY INITIATE A FUTURE FEDERAL ACTION THAT MAY INFLUENCE THE NUCLEAR POWER PLANT.

RECOMMENDATION 5
(CONT'D.)

3. THE NRC STAFF SHALL REQUIRE APPLICANTS TO MONITOR AND REPORT POTENTIALLY ADVERSE OFFSITE DEVELOPMENTS.

4. IF, IN SPITE OF THE ACTIONS DESCRIBED IN ITEMS 1 AND 3, THERE ARE DEVELOPMENTS OFF SITE THAT HAVE THE POTENTIAL FOR SIGNIFICANTLY INCREASING THE RISK TO THE PUBLIC, THE NRC STAFF WILL CONSIDER RESTRICTIONS ON A CASE-BY-CASE BASIS.

RECOMMENDATION 6

CONTINUE THE CURRENT APPROACH RELATIVE TO SITE SELECTION FROM A SAFETY VIEWPOINT, BUT SELECT SITES SO THAT THERE ARE NO UNFAVORABLE CHARACTERISTICS REQUIRING UNIQUE OR UNUSUAL DESIGN TO COMPENSATE FOR SITE INADEQUACIES.

RECOMMENDATION 7

REVISE PART 100 TO SPECIFY THAT SITE APPROVAL BE ESTABLISHED AT THE EARLIEST DECISION POINT IN THE REVIEW AND TO PROVIDE CRITERIA THAT WOULD HAVE TO BE SATISFIED FOR THIS ~~APPROACH~~ Decision TO BE SUBSEQUENTLY REOPENED IN THE LICENSING PROCESS.

RECOMMENDATION 8

REVISE PART 51 TO PROVIDE THAT A FINAL DECISION DISAPPROVING A PROPOSED SITE BY A STATE AGENCY WHOSE APPROVAL IS FUNDAMENTAL TO THE PROJECT WOULD BE A SUFFICIENT BASIS FOR NRC TO TERMINATE REVIEW. SUCH TERMINATION OF A REVIEW WOULD THEN BE REVIEWED BY THE COMMISSION

RECOMMENDATION 9

DEVELOP COMMON BASES FOR COMPARING THE RISKS FOR ALL EXTERNAL EVENTS

A-143

DIFFERING TASK FORCE OPINION

- POPULATION DENSITY AND DISTRIBUTION

DIFFERING WORKING GROUP OPINION

- USE AND DISCLOSURE OF BENEFITS AND RISKS IN SITING
- CONSIDER METEOROLOGIC CHARACTERISTICS OF SITES IN ESTABLISHING POPULATION CRITERIA

STAFF RESPONSES TO QUESTIONS SUBMITTED BY DR. OKRENT

ATTACHMENT M to Minutes of Site Eval Subcte Mtg Oct 16, 17

1. **Should the combination of siting criteria and reactor design consider Class 9 accident effects on neighboring reactors at the same site? If yes, how? If not, why not?**

The Siting Policy Task Force has concentrated on examining how siting might be strengthened as a factor in the defense-in-depth concept as applied to the general public. For this reason, the Task Force has not devoted sufficient study to examine the combination of siting criteria and/or design requirements to consider the effect of a Class 9 accident at one reactor upon a neighboring reactor at the same site. We note, however, that the radiological doses to the control room operators of a power reactor are required to meet the stringent limits (5 rem whole body, 30 rem thyroid) given in Criterion 19 of Appendix A to 10 CFR Part 50, in conjunction with a large fission product source term, not unlike that to be expected from a seriously degraded core. It appears, therefore, that there may be significant inherent capability for the control room operators of a nuclear power reactor to withstand the effects of highly degraded conditions at a neighboring plant. Non-essential plant personnel not located in the control room would be required to be evacuated, of course.

APPENDIX XVI
NRC STAFF ANSWERS TO QUESTIONS RAISED BY MR. OKRENT AT OCT 16-17, 1979 SITE EVALUATION SUBCOMMITTEE MEETING

2. The ACRS letters of October and December 1960 urged that even uncontrolled releases not result in a catastrophe. (Defined as lethal doses at a population center.) Is this a practical criterion to meet in terms of today's reactors and future sites?

One commonly accepted criterion of the population center distance around 1960 was that even in the event of an uncontrolled release, such as a core melt with containment failure, the population center distance was such that acute fatalities would be unlikely at the population center. In terms of today's reactors, based upon the consequence calculations taken from WASH-1400 and other similar studies, it appears that acute fatalities are unlikely beyond distances of about 10 to 15 miles, even for large uncontrolled releases. (See, for example, Figure VI 13-7 from WASH-1400, attached.) A recent staff survey (NUREG-0348, Demographic Statistics Pertaining to Nuclear Power Reactor Sites) has indicated that only for about 11 out of 106 sites is the nearest population center located less than 10 miles away. Consequently, it is the judgment of the Siting Policy Task Force that this criterion may not be impractical for future sites. The Task Force notes, however, that the concept of an isolated population center may also be outmoded, and for this reason proposed a set of population density limits coupled with limits in a given sector.

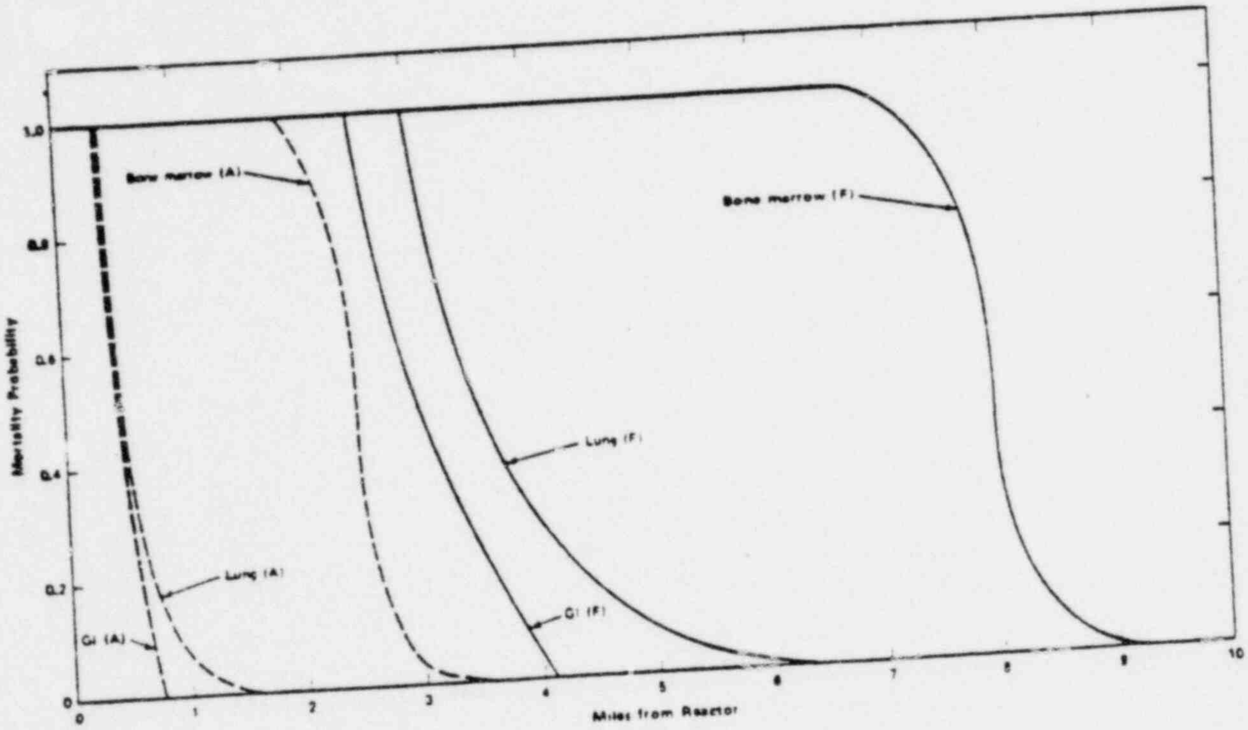

FIGURE VI 13-7 Mortality probability for an affected population versus distance from reactor for two hypothetical weathers: stability category A, wind speed = 0.5 m/sec; stability category F, wind speed = 2.0 m/sec.

3. How large a city would be allowed by the proposed criteria, at a radius of 10-15 miles? 5-10 miles?

The example given in NUREG-0625 regarding population density as a function of distance was keyed to the average population density of the region. The Siting Policy Task Force did not define what was meant by a region, however. Assuming that the region is defined to be the same geographic area as the State where the reactor is proposed to be located, and using data from the 1970 Census, the maximum population in any given sector can be computed as shown in the table that follows. For comparison, the values given by the present staff "trip level" of 500 persons per square mile (Regulatory Guide 4.7) are also shown. As stated in NUREG-0625, the Task Force has not completed a definitive study on the population densities, and the example given was merely to illustrate the concept.

MAXIMUM POPULATION IN ANY

22-½° SECTOR*

Distance, Miles	Reg. Guide** 4.7 (500/mi²)	NUREG-0625 Example NJ (950/mi²)	MA (730/mi²)	MD/NY (390/mi²)	PA/OH (2[illegible]/mi²)	OTHERS (less than 200)
0-5	39,270	18,700	14,270	7,775	5,140	3,930
5-10	117,800	84,200	64,240	34,990	23,150	17,670
10-20	471,240	898,180	685,180	373,220	246,930	94,250

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* No more than half of the total population in any annular region to be in a given 22-½° sector.

** No sector limit is given in Regulatory Guide 4.7; hence the value shown is for the entire radial interval.

. What is judged to be the risk to an individual living just outside the proposed exclusion distance of about 1/2 mile? What is the acceptable risk?

Response:

The Siting Policy Task Force has not established 1/2 mile as an acceptable minimum ER; rather it used 1/2 mile as an illustration of what the range of the final ER minimum distance would likely be to aid the Commission to visualize the various Task Force recommendations. Final minimum distances and limits on population densities and distributions will be developed only after a deliberate study of the risk from both Class 9 and other credible accidents. At this time the Task Force has not established a risk to an individual residing at 1/2 mile; nor has it established an acceptable level of risk. Such levels of risk will have to be developed in consultation with a variety of sources, including the ACRS, that would provide a spectrum of levels that society would perceive as acceptable.

6. Would a reactor site twenty-one miles from New York City be allowed if it met the criteria? Would a power park be allowed on such a site?

Response:

The intent of the proposed maximum population density and distribution criteria is to unequivocally exclude all areas having population greater than the stated maximum from siting consideration. This is not the same as saying that all sites meeting the criteria are unequivocally acceptable.

Based on staff experience the proposed criteria would eliminate 95% of the "problem" sites, but not all. A site, although meeting the population criteria, may possess such unusual topographic or meteorologic features which in combination with marginal population characteristics, may render it unacceptable from the safety standpoint.

In reality, the New York City urban population densities do not increase in a step function fashion, rather they increase monotonically with distance as one approaches outlying suburban areas until the core densities are reached. Thus, the closest areas meeting the proposed population criteria (the actual figures of distances and densities are yet to be determined), may well be in excess of 50 to 75 miles away from New York City.

In addition, the proposed rulemaking on alternative sites (SECY 79-481), which was endorsed by the Siting Policy Task Force, requires comparison of population density and distribution as part of determination of the "environmentally preferable site." It is rather unlikely that a site 21 miles from New York City would be found not having an "obviously superior alternative."

The answer then to both questions is not very likely.

Why was five miles downstream selected as minimum separation between a reactor site and a dam?

Response:

Staff practice in applying Regulatory Guides 1.59 (Design Basis Floods) for NPP and 1.102 (Flood Protection at NPP) has indicated that the design of major dams often reflects less severe criteria than is used for nuclear power plants and, as a result, failures must be postulated to assure nuclear health and safety. Furthermore, in analyzing the consequences of dam failures, the resulting hydrodynamic effects on nuclear power plant facilities within several miles of major dams were often very difficult to define, involved considerable staff and applicant evaluation efforts, and often required considerable flood protection provisions. Lastly, no need to locate nuclear power plants in floodplains downstream and close to major dams has been established.

To streamline the licensing process, and since no basic need to locate nuclear power plants downstream and close to major dams has been established, the Task Force recommended a nuclear power plant floodplain standoff distance of 5 miles downstream of a major dam. The intent is to minimize the need to design facilities for the potentially very large hydrodynamic forces that can result in such distances. The specific distance of 5 miles was suggested based upon the experience of one Task Force work group member for illustration purposes. The intent is to evaluate different siting situations to establish a comparable distance, or set of criteria, if the Task Force recommendation is implemented.

8. Why 12.5 miles from "a small capable fault"?

Response:

This question addresses section 3.2.2 and Recommendation 2, statement number 5, and asks why 12 1/2 miles should be used for all capable faults, even small ones. It should be noted that on page 51 of NUREG-0625 it is stated that:

> "For those hazards for which practicable standoff distances can be set, the Task Force recommends that specific distances be established. Although the Task Force has not conducted a comprehensive study, the objective would be that an accident at a facility hosting a hazardous activity would not endanger the nuclear plant. In the opinion of the Task Force, such distance could be approximately the following:"

The intent of this section was to suggest some examples of the type of standoff distances that should be considered as a concept by the Commission. A better statement of the intent of statement 5 would be:

> "sites should not be located within the range of strong nearfield ground motion from earthquakes on larger capable faults (a distance of 20 kilometers from such faults would usually be acceptable)."

The concept being proposed herein is that, although we feel confident that nuclear facilities can be sited safely in areas of strong nearfield ground motion, we recognize that difficult seismic issues must be addressed. Addressing these issues adequately has required extensive applicant and staff review time. The word "large" should be inserted in statement 5 and a clear reference to nearfield ground motion should be added.

9. How can one accept removal of guidance from 10 CFR 100 Appendix A without knowing what is substituted therefore? If current guidance leaves room for ambiguity, how would a less specific Appendix A provide better guidance?

The Task Force recommendation 4 indicates that:

"The Task Force established that Appendix A contains concepts based on the state-of-the-art existing at the time the appendix was prepared that are not clearly defined and lack a clear statement of the intent of the regulation.

The Task Force recommends that Appendix A to 10 CFR 100 be revised to better reflect evolving technology in assessing seismic hazards and to be more specific with respect to the definition of the terms and concepts it contains. In addition, the Task Force recommends that specific guidance material be removed from Appendix A and be placed in Regulatory Guides."

The intent of any future modifications to Appendix A to 10 CFR 100 would be to remove those aspects of the regulation which are changeable based on availability new data (i.e. tectonic provinces) and put them into pertinent Regulatory Guides. For example, the regulation might state that a reactor should not be built on a capable fault but the Regulatory Guide on Capable Faulting would indicate how to identify and evaluate such faults. The problem with the current Regulation is that guidance which should be modified and changed after proper evaluation of new data is law and cannot be modified except with great difficulty and after rulemaking.

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

October 31, 1979

D. Okrent, Chairman, Reactor Safety Research Subcommittee
P. Shewmon, Chairman, Reactor Fuel Subcommittee
W. Kerr, Chairman, Consideration of Class 9 Accident Subcommittee

LONG RANGE RESEARCH PLANS FOR CORE MELT PROGRAM

Attached is a copy of the NRC's preliminary long range research plan for the core melt program. This is a significant accomplishment of the NRC Staff. I have not previously seen or heard of anything other than fragmented planned tasks in this area with no apparent long range goal.

I suggest that an appropriate Subcommittee be assigned to review the overall core melt program while the final long range research plan for the core melt program is being developed. This would be an appropriate subject for discussion at the December ACRS meeting when Dr. Kerr will be scheduled for about 15 minutes to lead a discussion on the role of the Subcommittee on Consideration of Class 9 Accidents.

G. Quittschreiber
Senior Staff Engineer

Attachment:
Preliminary Core Melt Long
Range Research Plans

cc: ACRS Members
Tech Staff
D. Zukor

A-15 8

APPENDIX XVIII
REVIEWS OF PROPOSALS TO INCREASE POWER

Lee V. Gossick
Executive Director for Operations

REVIEWS OF PROPOSALS TO INCREASE POWER

This memo is to confirm the discussion during the May 4, 1978 meeting between the Commission and the ACRS concerning ACRS reviews of proposals to increase power levels at operating reactors. The Committee expressed its desire for the opportunity to review proposed power level increases at operating facilities including those that involve an increase from a reduced power level to the design power level. Such proposals will be routinely reviewed by the Committee's Subcommittee on Operating Reactors, with a case-by-case decision as to the need for full Committee review.

It is our understanding that proposals to extend operating power levels beyond that originally established as the design power level will normally involve a formal ACRS review and report.

/S/

R. F. Fraley
Executive Director

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

January 15, 1980

APPENDIX XIX
ACRS PARTICIPATION IN NRC RULEMAKING ON
RADIOACTIVE WASTE STORAGE AND DISPOSAL

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Ahearne:

The Committee has received your letter of January 9, 1980 regarding ACRS participation in the NRC rulemaking on storage and disposal of radioactive waste from nuclear facilities.

The ACRS welcomes the opportunity to participate in this proceeding as you have proposed. However, there is a problem with proposed timing with respect to providing ACRS comments on the statements and cross-statements filed by parties to the proceeding. The 30-45 day period you have suggested would be difficult to meet. If this could be extended to approximately 60 days, it would be helpful to the Committee.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

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

January 15, 1980

APPENDIX XX
RECOMMENDATIONS OF PRESIDENT'S COMMISSION
ON ACRS' ROLE

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: RECOMMENDATIONS OF PRESIDENT'S COMMISSION ON ACRS ROLE

Dear Dr. Ahearne:

The following comments are offered in response to Mr. Chilk's letter of November 9, 1979 requesting that the ACRS provide the Commission with its views and analysis of the role of the ACRS as contained in the recommendations of the report of the President's Commission (PC) on the Accident at Three Mile Island. Individual recommendations from the report are listed below with ACRS comments following.

1. "The Advisory Committee on Reactor Safeguards (ACRS) should be retained, in a strengthened role, to continue providing an independent check on safety matters." The ACRS agrees.

2. "The members of the Committee should continue to be part-time appointees;...." The ACRS agrees.

3. "The staff of ACRS should be strengthened to provide increased capacity for independent analysis." The ACRS agrees that current staff support is inadequate to provide suitable independent-analysis capability; to keep abreast of NRC Staff, industry, and foreign group activities on specific safety matters; to provide technical and background information to the members so the latter can make the best use of their limited time; and to provide proper support to the numerous ACRS subcommittees. The Committee therefore requests that ten additional, senior-staff positions be authorized for the ACRS staff in order to meet the sense of the PC's recommendations and to provide an adequate technical support base for improved operation of the Committee. These positions are intended to be in addition to those authorized in the Fellowship Program. However, if budgetary limitations prevent this level of support, the Committee would accept some conversion of Fellowship positions into permanent, senior positions.

   In connection with strengthening the staff, it is noted that the help of some outside organization could occasionally be very useful in the assembly of information and data or in carrying out some specific analysis. It is requested that means be explored whereby the ACRS could obtain such short-term studies as needed.

4. "Special consideration should be given to improving ACRS' capabilities in the field of public health." At the present time, the Committee has one member who is a specialist in the field of public health, and it can call upon an extensive list of highly qualified consultants. One of the initial group of ACRS Fellows was qualified in this area, and new Fellows, or possibly full-time staff members, knowledgeable in this field could be added to our staff as needed. Consequently, the Committee believes it has adequate competence in this area.

5. "The ACRS should not be required to review each license application." The ACRS concurs with this recommendation and suggests that legislation be passed such that, unless the Commission specifically requests a review and report on an application or portion thereof, the Committee may dispense with such review and report by notifying the Commission in writing that review by the Committee is not warranted. We would expect that such notification by the Committee would be made part of the public record.

6. "When ACRS chooses to review a license application, it should have the statutory right to intervene in hearings as a party. In particular, ACRS should be authorized to raise any safety issue in licensing proceedings, to give reasons and arguments for its views, and to require formal response by the Agency to any submission it makes." While the ACRS agrees that additional emphasis should be given to ACRS recommendations during the hearing process, it believes that a more desirable method of achieving this purpose would be to alter the statute to require that all recommendations made by the ACRS on given licensing proceedings be treated as substantive issues during the hearing. In order to protect the advisory role and collegiality of the ACRS, the statute should also specify that neither the Committee nor its members should be involved as a party nor be subject to subpoena in connection with the hearings.

7. "Any member of the ACRS should be authorized to appear and testify in hearings, ..." The Committee believes that one of its main strengths results from its collegial approach and that this would be jeopardized if members departed from the collegial forum. Although members can express disagreement with full Committee views by adding separate comments to our reports, we believe the collective aspect is overriding and we cannot support the recommendation. A member should be free, of course, to participate as an intervenor in his capacity as a private citizen.

8. "ACRS should have similar rights in rulemaking proceedings. In particular, it should have the power to initiate a rulemaking proceeding before the agency to resolve any generic issue it identifies." The Committee agrees with the thrust of this recommendation but believes that the Commission would, as a matter of course, initiate a rulemaking proceeding when recommended by the ACRS. However, as noted in our letter of December 13, 1979 to Commissioner Bradford, we believe that well-defined

procedures for ACRS input to the rulemaking process would be useful for clarifying the roles and responsibilities of the ACRS and the NRC Staff in this area. Such procedures should include enough flexibility to allow those departures which may be required by special circumstances.

We have also informally sought comment from the President's Office, the Commission, the ASLBP, the NRC Staff, Congressional Staff, and from the Committee members on ways to strengthen the role of the ACRS. Four major suggestions have surfaced, and these are addressed below.

1. It has been suggested that it would be of considerable value to the Commission if the ACRS could periodically assist in establishing priorities among the many safety matters needing attention. One approach to accomplish such an assignment, which we are prepared to undertake, would be for the Committee to comment on the priorities indicated in the report on unresolved safety issues which is submitted annually by the NRC to the Congress. Such a review should include consideration of other issues which are potential candidates for the list.

   A second, more time-consuming approach, somewhat experimental in nature, might be for the ACRS to evaluate and provide comments to the Commission on the general objectives, priorities, and resource allocations of the Office of Nuclear Reactor Regulation or other NRC Offices. We would be pleased to work with the Commissioners to determine whether this or some other approach might prove useful.

2. It has been suggested that the NRC needs a senior advisory group to assist in consideration of problems covering all aspects of the fuel cycle and that the PC seems to suggest that this role be filled by the ACRS. As you are aware, the ACRS, at the request of the Commissioners, either is or has been involved in safety-related aspects of reactor power plant design and operation, advanced reactor development, Department of Energy and Naval reactors, research, siting, chemical processing facilities, nuclear safeguards, transportation of radioactive materials, industrial sabotage, waste management, emergency planning, and spent fuel storage capacity. Thus, it already serves as an advisory body on subjects covering most of the breadth of the safety aspects of the fuel cycle. Although the Committee's time is limited, it could undertake additional work on the few remaining safety aspects of the full fuel cycle.

3. The Committee feels that some of its recommendations have not been followed up by the Commission and the NRC Staff in an adequate or timely fashion. We are pleased to see that you have initiated actions recently to resolve this matter, and we are prepared to work with you or your staff as needed. We believe that the Commission and Staff should develop a specific procedure for handling ACRS recommendations and for commenting on the reasons for the actions taken.

4. It has been suggested that the ACRS should devote a greater fraction of its time to some of the broader, as contrasted to detailed, aspects of reactor safety. The Committee is in agreement with this point and had begun moving farther in this direction prior to TMI-2.

We would welcome the opportunity to discuss any aspects of this letter on which you have questions.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

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

January 15, 1980

APPENDIX XXI
COMMENTS ON DRAFT NUREG-0660

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: DRAFT NUREG-0660, "ACTION PLANS FOR IMPLEMENTING RECOMMENDATIONS OF THE PRESIDENT'S COMMISSION AND OTHER STUDIES OF THE TMI-2 ACCIDENT"

Dear Dr. Ahearne:

During its 237th meeting, January 10-12, 1980, the Advisory Committee on Reactor Safeguards reviewed Draft NUREG-0660, dated December 10, 1979. The draft had previously been discussed at an ACRS Subcommittee meeting in Washington, D.C., on January 7, 1980. During its review, the Committee had the benefit of discussions with the NRC Staff.

The draft is a compilation of recommendations made by the several organizations and commissions that have investigated the TMI-2 accident. The Committee understands that a primary purpose of the document is to establish criteria for termination of the pause in licensing. Other purposes are to provide a complete action plan relating to all the unresolved issues and unimplemented recommendations from the lessons learned from the TMI-2 accident, and to establish priorities and requirements of funds and manpower. The draft gives preliminary target dates and estimates of the necessary resources, but does not yet recommend priorities.

The Committee believes the Plan is comprehensive, but not selective; this comprehensiveness serves to dilute the items important to safety, and therefore important to termination of the licensing pause. In the absence of priorities and identification of the items that the NRC Staff considers important, the ACRS finds it difficult to make objective comments on the Plan. The Committee understands that the Staff is proceeding to develop priorities and identification of items of primary importance, and the Committee will expect to review the important aspects of the Plan when this has been done.

The Committee is also concerned that preoccupation with the Plan may lead to neglect of pre-TMI-2 accident safety concerns, some of which are of long standing and of greater importance than some of the listed items. It is important to establish priorities on an overall consideration of both "old" and "new" items.

The Plan lists a large number of proposed changes in plant equipment, plant staffing, operating procedures, and licensing requirements. The ACRS believes that the scheduled time for establishing a complete plan setting detailed requirements for all items is too short to give reasonable assurance that all changes will be in the direction of greater safety. In illustration of this concern, the Committee points to the controversy that arose over the directive prohibiting tripping of the reactor coolant pumps following high pressure injection initiation.

The Committee believes that a two step process is more appropriate in developing the Action Plan. On an expedited basis, the Staff should develop those recommendations for safety improvement that it believes can and should be adopted as requirements for a termination in the pause in licensing. On a longer but defined time schedule, the Staff should develop a plan for dealing with other issues and implications of the TMI-2 accident.

Additional comments by member H. Lewis are presented below.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Additional Comments by Member H. Lewis

The letter of January 5, 1980 from L. V. Gossick, Executive Director for Operations, to the Commissioners describes the Action Plan as the complete list of all actions necessary as a result of the accident at TMI-2, and states that complete approval of the Plan, in its entirety, by the Commission, should be regarded as a prerequisite for the resumption of licensing. The Staff has further told us that, though they plan to assign priority scores to the items on the list (through a scoring system of dubious relevance), it is expected that all items on the list will be accomplished, in time.

It is my view that such an unselective approach to the lessons of TMI-2 is inappropriate, and that the Plan consists of an uncritical listing of anything anyone has suggested be done in the aftermath of (not necessarily as a result of) the accident at TMI-2. In particular, the Plan provides no guidance, and reflects no analysis, with respect to the safety relevance of the items, or even whether they would enhance safety. I believe adoption of the Plan would make no demonstrated contribution to a reordering of NRC priorities toward those safety weaknesses highlighted in the various reports on TMI-2.

It would be preferable to bite the bullet, and identify those twenty items that need attention, in terms of their impact on safety, as determined by any reasonable analysis. This has not been done, nor is it contemplated.

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

January 15, 1980

APPENDIX XXII
REQUEST FOR USER REQUESTS AND OTHER MEMORANDA

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

Dear Mr. Gossick:

The various offices of the NRC Staff issue user requests and other memoranda which identify safety research needs for the use of the Office of Nuclear Regulatory Research and other groups. The ACRS has not generally been a recipient of copies of such memoranda in the past. The Committee requests that in the future, it automatically receive such memoranda for its information.

In NUREG-0603 the ACRS recommended that the Research and Regulatory Staffs of the NRC should, in the reasonably near future, reevaluate the overall priorities, levels of expenditure, and focus of the safety research program. Many new research requests are emanating from the current activities of the Bulletins and Orders Task Force and other ongoing activities related to the TMI-2 accidents. The Committee believes it is important that these requests be reviewed and evaluated within a broad perspective of the overall needs and responsibilities of the NRC, keeping in mind all safety matters of primary importance to the protection of the public health and safety.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

cc: Chairman Ahearne

DELETION 1

pages A-167 - A-170

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, Chairman J. F. Ahearne to NRC Commissioners, Nuclear Advisory Board, dtd Dec 20, 1979

2. Minutes, Combined Meetings of ACRS Bulletins and Orders and ECCS Subcommittee, Jan 3-4, 1980, Los Angeles, CA

3. Letter, W. C. Counsil, Vice Pres, Connecticut Yankee Power Co to NRC Chairman J. M. Hendrie, Automatic Initiation of Auxiliary Feedwater, dtd Nov 3, 1979

4. Westinghouse Evaluation of North Anna Unit 1 Cooldown Incident of Sep 25, 1979

5. Letter, L. O. Mayer, Northern States Power Co to J. G. Keppler, Region III, IE, NRC, Prairie Island, Tube Rupture in No. 11 Steam Generator

6. Letter, D. A. Bixel, Consumers Power Co to J. G. Keppler, Region III, IE, NRC, Palisades, Response to IE Bulletins 79-05C and 79-06C July 26, 1979 Supplement, dtd Aug 2, 1979

7. Memorandum, D. Ross, Jr., Director, Bulletins and Orders Task Force to H. Denton, Director, NRR, B&O Report on W Plants, dtd Dec 4, 1979

8. Memorandum, Z. Rosztoczy, B&O Task Force to H. Denton, Director NRR and D. Ross, Jr, Director, Bulletins and Orders Task Force, Appendix VIII of NUREG-0611, Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants, dtd Dec 7, 1979

9. Draft, Program Plan NRC Fuel Melt Research, dtd Oct 1979

10. Memorandum, V. Gilinsky, NRC to Chairman, ACRS Request for Expansion of the ACRS Report Comments on the Pause in Licensing, dtd Dec 11, 1979

11. ACRS Consultants' Reports on the ACRS Review of the NRC Reactor Safety Research Program conducted in 1979, dated Jan 7, 1980, Dec 28, 1979, Aug 20, 1979, and Aug 18, 1979

12. Memorandum, T. E. Murley, RES to T. G. McCreless, ACRS, Severe Fuel Damage Test Program and PBF, dtd Jan 3, 1980