APPENDIXES
TO
MINUTES OF THE 236TH ACRS MEETING
DECEMBER 6-8, 1979

OCRS-1701

CERTIFIED

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SUMMARY: The National Science Foundation (NSF) is required to publish notice of permit applications received to conduct activities regulated under the Antarctic Conservation Act of 1978. NSF has published regulations under the Antarctic Conservation Act of 1978 at Title 45 Part 670 of the Code of Federal Regulations. This is the required notice of permit applications received.

DATE: Interested parties are invited to submit written data, comments, or views with respect to these permit applications by January 4, 1980. Permit applications may be inspected by interested parties at the Permit Office, address below.

ADDRESS: Comments should be addressed to Permit Office, Room 627, Division of Polar Programs, National Science Foundation, Washington, D.C. 20550.

FOR FURTHER INFORMATION CONTACT: Charles E. Myers at the above address or (202) 632-4238.

SUPPLEMENTARY INFORMATION: The National Science Foundation, as directed by the Antarctic Conservation Act of 1978 (Public Law 95-541), has developed regulations that implement the "Agreed Measures for the Conservation of Antarctic Fauna and Flora" for all United States citizens. The Agreed Measures, developed in 1964 by the Antarctic Treaty Consultative Parties, recommended establishment of a permit system for various activities in Antarctica and designation of certain mammals and certain geographic areas as requiring special protection. The Regulations were presented for public comment in draft form in the 6 March 1979 Federal Register. They appeared in final draft form in the 7 June 1979 Federal Register. Additional information was published in the 11 October 1979 Federal Register, page 58818.

The application received is:

1. *Applicant:* Charlene J. Denys, DePaul University, Department of Biological Sciences, Chicago, Illinois 60614.

Activity for which Permit Requested: Take Penguins. Twenty Adelie penguin adolescents will be sacrificed to obtain retinal and liver samples for use in research studies of visual pigments.

Import into USA—Some retinal and liver samples will be returned to the U.S. for further study.

Location: Torgersen Island, Antarctica.

Dates: February 1, 1980 to March 31, 1980.

Authority to take this action has been delegated by the Director, NSF to the Director, Division of Polar Programs and the Deputy Division Director DPP under National Science Foundation Staff Memorandum O/D 79-18, of May 28, 1979.

A. N. Fowler,
Deputy Division Director, Office of Polar Programs.

[FR Doc. 79-37000 Filed 11-30-79; 8:45 am]
BILLING CODE 7555-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on December 6-8, 1979, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published on November 21, 1979 (44 FR 67000).

The agenda for the subject meeting will be as follows:

Thursday, December 6, 1979

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

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

1:30 p.m.—6:30 p.m.: Executive Session (Open)—The Committee will hear and discuss reports from ACRS Subcommittee chairmen and designated ACRS members related to the safety and procedural aspects of the report of the President's Commission on the Accident at Three Mile Island; ACRS comments and recommendations regarding the NRC regulatory process; the proposed pause in licensing of nuclear facilities; the conceptual design of the Floating Nuclear Plant core ladle and its application to land-based nuclear plants.

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

Friday, December 7, 1979

8:30 a.m.—12:00 Noon: Meeting with NRC Staff (Open)—The Committee will hear reports and will discuss proposed plans for NRC implementation of the recommendations of the President's Commission on the Accident at Three Mile Island and the TMI-2 Lessons Learned Task Force Final Report (NUREG-0585).

1:00 p.m.—2:30 p.m.: Meeting with NRC Staff (Open)—The Committee will hear presentations and discuss the proposed pause in nuclear power plant licensing, and proposed alternatives, with representatives of the NRC Staff and the nuclear industry including an extended low-power start-up program for the Sequoyah Nuclear Plant.

2:30 p.m.—3:30 p.m.: Executive Session (Open)—The Committee will hear a report by one of its members and will discuss proposed plans by the President to implement recommendations of the President's Commission on Three Mile Island (tentative).

3:30 p.m.—4:30 p.m.: Meeting with Nuclear Regulatory Commission (Open)—The Committee will meet with the members of the Nuclear Regulatory Commission to discuss proposed plans for implementation of lessons learned from the accident at Three Mile Island and the related pause in licensing of nuclear power facilities. A portion of this session, if necessary, will be devoted to discussion of recent ACRS reports to the Commission on the NRC Safety Research Program (NUREG-0603) and Evaluation of Licensee Event Reports (NUREG-0572).

4:30 p.m.—5:45 p.m.: Meeting with NRC Staff (Open)—The Committee will hear a report and will discuss with members of the NRC Staff a proposed revision of NUREG-0606, Unresolved Safety Issues.

Saturday, December 8, 1979

8:30 a.m.—4:00 p.m.: Executive Session (Open)—The Committee will continue its discussion of proposed ACRS comments and recommendations regarding the NRC regulatory process and will also discuss proposed ACRS comments and recommendations regarding proposed action to implement the recommendations of the President's Commission on Three Mile Island; the TMI-2 Lessons Learned Task Force Final Report; and the proposed pause in nuclear power plant licensing and appropriate alternatives.

The Committee will discuss proposed replies to NRC Commissioners regarding follow-up and implementation of ACRS recommendations, and ACRS recommendations regarding NRC regulatory requirements which may warrant changes.

The Committee will discuss candidates nominated and will elect its officers for Calendar Year 1980.

The Committee will hear a report from its Subcommittee on the Mark I Pressure Suppression Containment Long Term Program.

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 release of which would represent an unwarranted invasion of personal privacy.

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

I have determined in accordance with Subsection 10(d) Pub. L. 92-463 that it is necessary to close portions of this meeting as noted above to protect Proprietary Information (5 U.S.C. 552b(c)(4)) and to protect information the release of which would represent an unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)).

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

Dated: November 28, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-37091 Filed 11-30-79; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-348]

Alabama Power Co.; Granting of Interim Relief From ASME Section XI Inservice Testing Requirements

The U.S. Nuclear Regulatory Commission (the Commission) has granted relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" to the Joseph M. Farley Nuclear Plant, Unit No. 1 (the facility) located in Houston County, Alabama. The relief relates to the inservice testing program for the facility. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR Part 50. The interim relief is effective as of its date of issuance.

The relief is granted on an interim basis, pending completion of our detailed review from those inservice testing requirements of the ASME Code that the licensee has determined to be impractical within the limitations of design, geometry and materials of construction of components, because compliance would result in hardships or unusual difficulties without a compensating increase in the level of quality or safety.

The request for relief 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 letter granting relief. Prior public notice of this action was not required since the granting of this relief from ASME Code requirements does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the request for relief dated May 1, 1979, and (2) the Commission's letter to the licensee dated November 16, 1979.

These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the George S. Houston Memorial Library, 212 W. Berdeshaw Street, Dothan, Alabama 36303. A copy of item (2) 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, Maryland this 16th day of November, 1979.

For the Nuclear Regulatory Commission.

A. Schwencer,

Chief, Operating Reactors Branch #1, Division of Operating Reactors.

[FR Doc. 79-37088 Filed 11-30-79; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-261]

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

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 44 to Facility Operating License No. DPR-23, issued to the Carolina Power and Light Company, (the licensee), which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant Unit NO. 2 (the facility) located in Darlington County, Hartsville, South Carolina. The amendment is effective as of the date of its issuance.

The amendment establishes Technical Specifications to assure inspection and reporting requirements for a program of inservice inspection of steam generator tubing consistent with the requirements of Revision 1 of Regulatory Guide 1.83.

The application for the 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 pursuant to 10 CFR 51.5(d)(4) an 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 November 4, 1976, as supplemented June 30, July 29, 1977, June 4, August 9, 1978 and April 9, 1979, (2) Amendment No. 44 to License No. DPR-23, 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, N.W., Washington, D.C.

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

November 28, 1979

DETAILED SCHEDULE AND OUTLINE
FOR DISCUSSION
236TH ACRS MEETING
December 6-8, 1979
Washington, D. C.

Thursday, December 6, 1979, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 12:30 P.M. — Executive Session (Open)
   1.1) Discuss proposed ACRS report on Review of Regulatory Processes and Functions (MB/RFF et al.)

12:30 P.M. - 1:30 P.M. — LUNCH

2) 1:30 P.M. - 8:00 P.M. — Executive Session (Open)
   2.1) 1:30 P.M.-3:30 P.M.: Discuss safety implications of Report of the President's Commission on the TMI Accident (HL/JCM)
   2.2) 3:30 P.M.-5:30 P.M.: Discuss proposed ACRS report to NRC On Review of Regulatory Processes and Functions (MB/RFF et al.)
   2.3) 5:30 P.M.-6:30 P.M.: Discuss proposed ACRS report to NRC on the proposed pause in NRC licensing of nuclear facilities (DO/MWL)
   2.4) 6:30 P.M.-7:30 P.M.: Discuss proposed ACRS views/comments on the recommendations of the President's Commission on TMI regarding ACRS activities (MWC/RFF)
   2.5) 7:30 P.M.-8:00 P.M.: Report of ACRS Subcommittee on FNP core ladle and its application to land-based nuclear plants (DWM/GRQ)

Friday, December 7, 1979, Room 1046, 1717 H Street, NW, Washington, DC

3) 8:30 A.M. - 10:00 A.M. — Meeting with NRC Staff (Open)
   3.1) Discuss proposed NRC "action plan" for implementation of recommendations of the President's Commission on TMI and NRC Lessons Learned Task Force.

4) 10:00 A.M. - 12:00 Noon — Meeting with NRC Staff (Open)

4.1) Discuss NRC Final Report on TMI Lessons Learned

12:00 Noon - 1:00 P.M. — LUNCH*

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

5.1) Discuss proposed pause in nuclear power plant licensing

6) 2:30 P.M. - 3:30 P.M. — Executive Session (Open)

6.1) Discuss the President's plan to implement the recommendations of the President's Commission on TMI (Tentative depending on availability of plan) (HL/JCM)

7) 3:30 P.M. - 4:30 P.M. — Meeting with NRC Commissioners (Open)

Discuss:

7.1) Proposed pause in NRC Licensing of nuclear facilities

7.2) Questions of Commissioners regarding NUREG-0603, ACRS Comments on the NRC Safety Research Program Budget (tentative)

7.3) Discuss questions of Commissioners regarding NUREG-0572, ACRS Review of Licensee Event Reports (1976-1978) (tentative)

8) 4:30 P.M. - 5:45 P.M. — Meeting with NRC Staff (Open)

8.1) Discuss proposed revision of NUREG-0606, Unresolved Safety Issues

*Lunch on the table may be needed to permit a first reading of the proposed ACRS report on TMI lessons learned

Saturday, December 8, 1979, Room 1046, 1717 H Street, NW, Washington, DC

9) 8:30 A.M. - 4:00 P.M. Executive Session (Open)

9.1) 8:30 A.M.-12:30 P.M.: Discuss proposed ACRS reports to NRC on:

- Proposed Action Plans
- TMI Lessons Learned
- Regulatory Processes and Functions
- Proposed "Pause" in licensing nuclear facilities
- Recommendations of President's Commission on TMI regarding ACRS activities

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

9.2) 1:30 P.M.-3:00 P.M.: Discuss proposed ACRS letters/comments on:

- NRC Regulations which may need changing and related procedures (reply to Comm. Bradford) (DWM/RFF)
- Follow-up procedures regarding ACRS recommendations (reply to Commissioner Ahearne) (MWC/RFF)

9.3) 3:00 P.M.-3:15 P.M.: Election of ACRS Officers for CY 1980

9.4) 3:15 P.M.-3:45 P.M.: Subcommittee Report on Mk I containment, Long-Term Program (MP/AB)

9.5) 3:45 P.M. - 4:00 P.M.:

9.5.1) Future Subcommittee Activity

9.5-2) Future ACRS activity

Issue Date:

February 26, 1980

(FOIA EXEMPTION (b)5)

MINUTES OF THE
236TH ACRS MEETING
DECEMBER 6-8, 1979
WASHINGTON, DC

CERTIFIED

The 236th 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, December 6, 1979.

[Note: For a list of Attendees, see Appendix I. Mr. Bender was not present on Friday and Saturday, December 7 and 8.]

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 received from members of the public to present oral statements. He also noted that copies of the transcript of some of the public portions of the meeting would be available in the NRC's Public Document Room at 1717 H St., N.W., Washington, DC 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. Lewis and Mathis as reviewers for the 236th ACRS Meeting.

B. Role of ACRS

The Chairman noted receipt of a memorandum from the Secretary of the Commission (see Appendix IV) requesting the Committee's comments regarding the recommendation of the President's Commission's recommendations regarding the future role of the ACRS. He noted that time would be devoted to developing a response to this request.

II. Meeting with the NRC Staff Regarding Its Action Plan for Implementation of Recommendations of the President's Commission on TMI and the Lessons Learned Task Force and the "Final Report of the Lessons Learned Task Force" (Open to Public)

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

A. Subcommittee Report

Mr. Okrent, Chairman of the TMI-2 Accident Implications Subcommittee, noted that the Committee has available to it a second draft of the final report of the Lessons Learned Task Force, and that a detailed discussion of the contents of this report could be waived. He suggested that the time allotted to a discussion of the final report could better be applied to discussions regarding the NRC Staff's Action Plan for implementation of the recommendations of the President's Commission on TMI and of the Lessons Learned Task Force.

B. NRC Staff Presentation

R. Mattson, NRC Staff, informed the Committee that it was being presented with an outline of the NRC Staff's Action Plan for the implementation of the recommendations of the President's Commission and of the Lessons Learned Task Force, because the plan itself is still being written. The NRC Staff anticipates that it can present a draft of the plan to the Commissioners by Sunday morning, December 9, 1979, and that it planned to hold a public discussion with the Commissioners on this plan on Monday afternoon, December 10, 1979. He said that because the plan is, in fact, not completed, the discussion this afternoon would involve only the outline, and this in general terms. He discussed the first two sections of the outline, Operational Safety and Siting and Design.

D. Ross, NRC Staff, discussed the outline of the third part of the Action Plan, Emergency Preparations and Radiation Protection.

J. Scinto, NRC Staff, discussed the fourth and last portion of the Action Plan, Regulatory Structure and Process (for the outline of the Action Plan, see Appendix V).

III. Meeting with NRC Staff on Proposed Pause in Nuclear Power Plant Licensing (Open to Public)

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

A. Background

Mr. Okrent, TMI Accident Implications Subcommittee Chairman, discussed the background of the planned discussion, noting that the NRC Staff had, during the 235th ACRS Meeting, informed the Committee that the NRC Staff was preparing a pause in the licensing of nuclear power plants ready for operation, while it concentrated on the changes necessary to operating nuclear plants as a result of the studies made following the TMI-2 accident. At that meeting, several ACRS Members suggested that it might be useful if augmented low-power testing were to be permitted on these plants, whereby additional information could be obtained and additional training could be obtained for the operators of the new plants. Since the 235th ACRS Meeting, the NRC Staff has established a task force to consider the merit of low-power-licensing for test and training purposes. In the meantime, the Tennessee Valley Authority has proposed that its Sequoyah Nuclear Power Plant Unit 1 be licensed at low power for augmented testing and training. (For background material, see Appendix VI.)

B. NRC Staff Report

D. Vassallo, NRC Staff, noted that TVA has made a formal request for a low-power license for the Sequoyah Nuclear Power Plant Unit 1 to conduct tests beyond the current test requirements. He said that TVA has proposed that these tests would be conducted by all five of TVA's operating crews to give them training and hands-on experience prior to actual operation.

D. Vassallo noted that 15 of the 24 short-term TMI-2 Lessons Learned requirements have been resolved, and that the NRC Staff would require the resolution of the remaining 9 before issuing the low-power license. He said that the NRC Staff would also require that TVA's emergency procedures be reviewed by Westinghouse. TVA will be required to establish an onsite technical support center. TVA has offered to perform some probabilistic assessment studies. The NRC Staff also intends to review improved emergency plans for the State of Tennessee. The Applicant claims that the reactor will be ready for fuel loading around December 20. The NRC Staff expects that it will be able to make a recommendation to the Commissioners for issuance of a low-power license for this special testing by the end of this year.

S. Varga, NRC Staff, said that issuance of a low-power license for the augmented test program is not a precursor for issuance of an operating license.

Mr. Okrent noted that the Committee is only considering, at this time, the application for a low-power license for the proposed augmented testing program.

C. TVA Position

J. Green, TVA, discussed the TVA application for an augmented low power test program, and the detailed tests that TVA is proposing (see Appendix VII). He said that safety analyses and operating procedures are currently being developed, but are not available yet. He said that the proposed testing program will have no adverse effect on the equipment — nor will it damage the equipment.

D. NRC Staff Evaluation

R. Baer, NRC Staff, said that the Staff has formed an ad hoc group to review the concept of low-power-testing on Sequoyah and other reactors. He said that this testing could expand the evaluations of the TMI-2 accident. He said that the NRC Staff agrees that the tests must be conducted safely and with low risk at low power. The work that has been done so far is only preliminary. The NRC Staff view is that the proposed tests would be useful to train operators and verify procedures; however, the NRC Staff has not reached a consensus. He noted that the proposed tests are only a simulation of decay heat removal, and that, because of reactivity feedback, there may be some variances between the results of the test program and those that would be reached under real conditions.

E. Caucus

The Committee indicated that it believed it could write a report favorable to the proposed low power testing program.

IV. Meeting with Members of the NRC Staff Regarding the Proposed Revision of NUREG-0606, Unresolved Safety Issues (Open to Public)

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

S. H. Hanauer, NRC Staff, informed the Committee that the NRC Staff had a problem in that it could not write a meaningful section on the resolution of generic items for the annual 1979 report to Congress with the current state of affairs in the NRC. He said that they have proposed, instead, to report that there are no new issues in 1979, and explain that in the aftermath of the TMI-2 accident, efforts have been expended toward resolving the direct problems observed from that accident. He said that in order to write a meaningful report, it will be necessary to review the Action Plan and the report of the Rogovin group, neither of which are available at this time. In addition, he proposed to write a special report during 1980, on a schedule to be determined, which would take into account the lessons learned and other decisions made regarding generic items. He provided the Committee a general outline of the items to be covered in this proposed report (see Appendix VIII).

S. H. Hanauer asked the Committee either for their concurrence or additional advice.

C. Michelson, ACRS Consultant, requested that the NRC Staff examine the following postulated scenario from both a qualitative and risk assessment point of view: the secondary steam system fails in containment, either a PORV or a safety valve opens on the primary side and fails to close. What is the overall effect and risk to the containment system?

V. Executive Sessions (Open to Public)

A. Subcommittee Reports

1. Floating Nuclear Plant

Mr. Moeller, Subcommittee Chairman, discussed the background for the development of a core ladle, and the status of this design, as well as other unresolved aspects of the FNP design (see Appendix IX).

R. Baer, NRC Staff, requested the Committee's input on the core ladle regarding the concept, the design, and the criterion that provides for a two day delay in the migration of a molten core. He requested these comments for the 237th ACRS Meeting. He said that the NRC Staff was reluctant to try to finalize the safety requirements and the review of the application for a manufacturing license without the Committee's input.

R. Baer said that the design essentially replaces concrete with a magnesium oxide refractory. During the two day hold up of the postulated molten core, other interdiction methods can be taken. B. Haga, Offshore Power Systems, said that the Applicant supports the NRC Staff position. He said the Applicant is ready for consideration by the Committee of all aspects of the Floating Nuclear Plant, including the upperhead injection systems performance and core melt accident problems, that the Applicant would meet on any of the issues at the Committee's pleasure, and that the Applicant can design and provide a filtered vent system for the FNP containment if so required by the Committee. He noted that the Applicant is required to meet all post-TMI requirements.

Mr. Okrent suggested that the Committee is not ready to give its comments at this time for the following reasons:

- The Committee has not reviewed in detail NUREG-0502 regarding the core ladle.
- It is not clear what the two day delay of molten core migration accomplishes.
- The full effect of a core ladle has not been studied.
- The question of heat loads on the structures and their affect on failures has not been addressed adequately.
- The issue of treatment for evolved hydrogen from severely damaged cores is going to rulemaking, and is currently an open issue.

He noted that in 1976, the Committee requested a study regarding increased containment design pressure.

Mr. Etherington said that the Applicant would be happy if the Committee could find that the conceptual design of the crucible was reasonable. This matter could be addressed alone.

Mr. Shewmon suggested that it would be easier for the Committee if it reviewed each item separately as the information became available.

R. Baer said that it was his understanding that the two day delay was intended for estuarine sites so that time could be obtained to take other interdictive action. For an ocean site, it has been proposed that in the event of a major accident, the barge be scuttled and allowed to settle on a layer of clay beneath the barge.

The Committee agreed to proceed with the review of the conceptual design of the core ladle for the floating nuclear plant, but this review was deferred later until the 238th ACRS Meeting (February).

B. President's Commission Report

The Committee discussed briefly the President's Commission Report, especially those portions dealing with the role of the ACRS in the regulatory process. An interim report to Chairman Ahearne was prepared on this matter (see Appendix X). Mr. Lewis agreed to draft a proposed report regarding the recommendations beyond those dealing with the activities of the ACRS, and also to draft a proposed report regarding those items directed to the role of the Committee in the regulatory process.

The Committee named an ad hoc working group to examine and evaluate methods to implement the recommendations of the President's Commission to strengthen the ACRS. Named to this group were Messrs. Carbon, Fraley, Lewis and Plaine.

C. Review of NRC's Action Plan

The Committee named an ad hoc subcommittee to review the NRC's Action Plan for upgrading the NRC functions. Mr. Etherington was named Chairman, with Messrs. Bender, Lewis, Mathis and Okrent as members. R. K. Major is the cognizant staff engineer.

D. Future Agenda

1. Schedule for 237th ACRS Meeting

The Committee approved a tentative schedule for subjects to be considered at the 237th ACRS Meeting (see Appendix II).

2. Schedule of ACRS Subcommittee Meetings and Tours

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

E. Recognition of Chairman

The Committee extended its gratitude to its outgoing Chairman, Max W. Carbon.

F. ACRS Reports and Letters

1. Interim Comments on Recommendations of President's Commission Regarding ACRS Activities

In response to NRC's Secretary Chilk's memorandum of November 9, 1979 (see Appendix IV), the Committee provided its interim comments to Chairman Ahearne regarding the recommendations of the President's Commission to strengthen the ACRS (see Appendix X).

2. Interim Low Power Operation of Sequoyah Nuclear Power Plant Unit 1

The Committee prepared a report endorsing low power operation of the Sequoyah Nuclear Power Plant Unit 1 (see Appendix XI).

3. Report on TMI-2 Lessons Learned Task Force Final Report

The Committee provided its comments on NUREG-0585, Report on TMI-2 Lessons Learned Task Force Final Report (see Appendix XII).

4. Comments on Pause in Licensing

The Committee prepared a report providing its comments on the NRC's pause in the licensing of nuclear power plants (see Appendix XIII).

Adequacy of Procedures for Communication and Interaction with the NRC Staff

5. Response to Chairman Ahearne's Inquiry dated Oct. 3, 1979

The Committee prepared a response to Chairman Ahearne's inquiry of October 3, 1979 regarding the adequacy of procedures for communications and interactions between the ACRS and the NRC Staff (see Appendix XIV).

6. Letter to Representative Morris K. Udall Regarding the Proposed NRC FY-80 Supplemental Research Budget

The Committee prepared a letter to Representative Morris K. Udall providing its comments regarding the proposed NRC FY-80 Supplemental Research Budget (see Appendix XV).

7. Identification of NRC Regulatory Requirements Which Need Changing

The Committee prepared a letter to Commissioner Bradford identifying NRC regulatory requirements needing changing (see Appendix XVII).

8. Review of the NRC Regulatory Processes and Functions

The Committee completed its report, Review of the Regulatory Processes and Functions (see Appendix XVI).

VI. Executive Sessions (Closed to Public)

A. Election of Officers

The Committee elected Mr. Plesset to be its Chairman and Mr. Mark to be its Vice-Chairman for Calendar Year 1980.

The 236th ACRS Meeting was adjourned at 2:40 p.m., Saturday, December 8, 1979.

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

ACRS STAFF

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

CONSULTANT

C. Michelson

NRC ATTENDEES

236TH ACRS MTG.

Dec. 6, 1979 (Thursday)

Div. of Project Management

R. Baer

Dec. 7, 1979 (Friday)

Div. of Project Management

H. Clayton
R. L. Baer
R. A.Benedict
J. F. Stolz

Operating Licensing Branch
Div. of Operating Reactors

P. F. Collins

Light Water Reactors Br. 4
T. C. Houghton

Inspection and Enforcement

J. C. LeDoux

Nuclear Reactor Regulation

D. Ross
R. Mattson
S. H. Hanauer
D. B. Vassallo
J. P. Knight
A. W. Dromerick
F. Pagano
J. Martin
L. Reiter
R. E. Jackson
J. D. Kane
C. Graves
T. Dunning

PUBLIC ATTENDEES

236TH ACRS MEETING

December 6, 1979 (Thursday)

P. C. Higgins, AIF
T. Martin, NUTECH
R. Smith, McGraw Hill
H. B. Piper, PMC
J. Dann, McGraw Hill
R. J. Ross, Doub & Muntzing
M. Laggart, Jersey City Power and Light
G. C. Sorensen, Washington Public Power Suply System
O. K. Earle, Washington Public Power and Supply System
M. B. Whitaker, South Carolina Electric and Power Company
C. P. Amyot, Consultant
P. B. Haga, Westinghouse, OPS
D. H. Walker, Westinghouse, OPS
D. P. Ormsby, Tennessee Valley Authority
L. M. Mills, Tennessee Valley Authority
H. Hamada, TEPCO
R. D. Guthrie, Tennessee Valley Authority (TVA)
R. Alsup, TVA
D. C. Craven, TVA
H. Green, TVA
M. Siano, Westinghouse
R. J. Sero, Westinghouse
W. J. Johnson, Westinghouse
M. R. Harding, Westinghouse
C. R. Tuley, Westinghouse
H. Hamada, TEPCO
K. Ota, KEPCO

PUBLIC ATTENDEES

236TH ACRS MEETING

December 7, 1979 (Friday)

C. Grochmac, Stone and Webster
P. Galuszka, The Va-Pilot, Portsmouth, VA
R. Jansen, Westinghouse, Pittsburgh, PA
M. B. Whitaker, South Carolina Electric & Gas, Columbia, SC
B. Horin, Deberoise & Liberman, Wash., DC
S. Kast, Wash. Star, Arlington, VA
J. Szwejkowski, CEI, Cleveland, OH
G. Kann, VEPCO, Richmond, VA
S. R. Phelps, EEI, Wash., DC
C. Brinkman, CE, Gaithersburg, MD
B. Horin, Deberoise & Liberman, 2500 Wisc. Av., Wash., DC
R. Bala, EBASCO Services, New York City, NY
J. Gormly, Pacific Gas and Electric, Alameda, CA
K. C. Fortino, LNRA&T, D.C.
J. Newman, LNRAT, Worthington, Bethesda, Md.
G. A. Blanc, Pacific Gas & Electric Co., Moraga, CA
B. Miller, Richmond Times-Dispatch, Wash., DC
G. Sorensen, WPPSS
J. Szwejkowski, CEI
L. S. Gifford, General Electric Company
E. A. Baum, Va. Electric and Power Company
J. R. Bgnum, Tennessee Valley Authority
J. Lyons, TVA
D. Zeringue, TVA
R. B. Borsum, Babcock and Wilcox
M. W. Laggart, Jersey Central Power and Light
R. Morgan, TVA
E. A. Liden, Pacific Electric and Gas
D. L. Lambert, TVA
J. A. Raulston, TVA
J. Ballentine, TVA
G. Wilson, TVA
J. R. Walker, TVA

APPENDIX II

FUTURE AGENDA

ACRS annual report to Congress on the NRC research program	8 hours
Proposed NRC action plan to implement the recommendations of the President's Commission on TMI	8 hours
Methods to implement recommendations of the President's Commission to strengthen the ACRS	2 hours
NRC Bulletins and Orders resulting from TMI/Small Break LOCA Analyses	2 hours
Proposed revisions to NRC siting criteria	2 hours

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

December 8, 1979

APPENDIX III

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS AND TOURS

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

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

M W 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

DECEMBER - 1979

13	Power & Elect. Sys. (GRQ) - WK, JE, CM, JR
19	Waste Mgmt/Fuel Cycle (RM/PT) - SL, WK, CM, WM, DWM, JR
20	Site Evaluation (RM/PT) - DWM, SL

JANUARY - 1980

3-4	ECCS/Bull. & Ord. -Los Angeles, CA (PB/AB) - WM, MP, HE, DO(tent.)
7	TMI Action Plan (RKM) - HE, WM, MB(tent.), HL
8	B&W (RM) - HE, JE, JR
9	RSR (TGM) - DO, HE, WK, SL, CM, MP, PS, CS
9	Procedures & Admin. (1:00 p.m.) (tent.) (RFF) - MP, MC, SL, WK, DWM
10-12	237th ACRS Meeting
23-24	Metal Components (EI) - PS, MB, HE
25	ATWS (PB) - WK, JE, JCM(tent.)
31(tent.)	TMI-1(restart) Harrisburg, PA (RM) - HE, SL, DWM, MP(tent.)

FEBRUARY - 1980

6	RSR (TGM) - DO, HE, WK, SL, JCM, MP, CPS, PGS
7-9	238th ACRS Meeting
22	GETR - San Francisco, CA (EI) - DO, JCM, WK

RF

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

2.4

November 9, 1979

RECEIVED

OFFICE OF THE
SECRETARY

APPENDIX IV
REQUEST FOR ACRS COMMENTS ON PRESIDENT'S
COMMISSION'S RECOMMENDATIONS REGARDING
FUTURE ROLE OF ACRS

MEMORANDUM FOR: Max Carbon, Chairman, ACRS

FROM: Samuel J. Chilk, Secretary

SUBJECT: REPORT OF THE PRESIDENT'S COMMISSION ON THE ACCIDENT AT TMI

The Commission would be aided in its consideration of the Presidential Commission's recommendations if ACRS would provide the Commission with its views and analysis of the role of ACRS as contained in the recommendations of the report. Your comments are requested at the earliest practicable date.

cc:
Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne
General Counsel
Acting Director, OPE
Exec Dir for Operations
Roger Mattson, NRR

TMI-2

Mattson et al

TMI ACTION PLAN INDEX

I. OPERATIONAL SAFETY

I.A.1 Operating Personnel and Staffing

1. Shift Technical Advisor
2. Shift Manning
3. Amended 10 CFR 50.54(k)
4. Shift Supervisor Duties
5. Codification of Requirements

I.A.2 Training and Qualifications of Operating Personnel

1. Immediate Upgrading of Training and Qualifications
2. NRR Audit Vendor Training Program
3. NRR Participate in IE Inspector Training
4. Plant Drills
5. Long Term Upgrading of Trainning and Qualifications
6. Accredit Training Institutions

I.A.3 Licensing and Requalification of Personnel

1. Revised Scope and Criteria for Licensing Exam
2. Selection of Shift Operators, Supervisors, and Technical Advisors (revise and move to I.A.1)
3. Internal NRC Operator Licensing Reforms
4. Operator Fitness

I.A.4 Simulator Use and Development

1. Study of Training Simulators
2. Upgrade Training Simulator Standards
3. Regulatory Guide on Training Simulators
4. Review of Simulators for Conformance to Criteria
4. NRC Engineering Simulator

I.B.1 Licensee's Responsibility and Accountability for Safety

1. Establish Acceptance Criteria
2. Inform Utilities
3. Obtain Commitments/Submittals
4. Issue Reg Guides
5. I E Develop Program-Evaluation
6. Implement

I.B.2 Independent Review of Plant Operating Activities

1. Establish Acceptance Criteria
2. Inform Utilities
3. Obtain Commitments/Submittals
4. Issue Reg Guides
5. IE Develop Program-Evaluation
6. Implement

I.B.3 Radiation Protection Organization and Staffing

1. Establish Criteria
2. Implement Criteria

I.B.4 Management and Technical Qualifications of Licensee

1. Establish Criteria
2. Inform Utilities
3. Obtain Commitments
4. SD issue Proposed RG

I.B.5 Radiation Protection Technician Training and Qualification Program

I.C.1 Extension of Technical Specifications to All Plant Items Having a Safety Func[illegible]
and to Management and Administrative Controls

I.C.2 Operating Procedures

I.D.1 Control Room Design

1. Indentification of and Recovery from Conditions Leading to Inadequate core cooling
2. Additional Post-Accident Instruments
3. Improved Control Room Information Display and Alarm Systems
4. Control Room Review
5. Display Plant Safety Status
6. Control Room Design Standard

I.D.2 Operational Aids

1. On-line Reactor Surveillance System
2. Improved Instrumentation for Power Plant Application
3. Enhanced Operator Capability
4. Human Error Rate Analysis
5. Engineering Simulator
6. Environmental Status Monitoring
7. Improved Control Room Display and Diagnostic Systems

I.E. Analysis and Dissemination of Operating Experience

1. Task Force
2. Establish Office
3. Program Office Support
4. Establish Formal Program
5. Coordinated Network-Industry & Licensees
6. NPRDS
7. Reporting Requirements
8. IREP
9. Review of Licensee Programs
10. Foreign Sources

I.F. Quality Assurance

I.G. Preoperational and Low Power Testing of Near Term OLs

II SITING AND DESIGN

II.A. Siting

1. Rule Making
2. Site Evaluation
3. High Population Density
4. Fission Product Release Research
5. Natural Phenomenon Research

II.B. Degraded Core

1. Primary System Vent
2. Shielding
3. Sampling
4. Training
5. Research
6. Rulemaking

II.C.1 Non-Category I Structures

1. Investigation
2. Implement Phase 1
3. Implement Phase 2

II.C.2 Systems Engineering and Reliability

1. IREP - General
2. Systems Interation
3. IREP - Plant Specific
4. System Interaction - Implement Changes
5. Reliability Assurance
6. Reliability Assurance

II.D.1 Relief and Safety Valve Testing

1. Requirements and Program
2. Testing

II.E.1 Auxiliary Feedwater

1. Simplified Reliability Analysis
2. Flow Capacity

II.E.2 ECCS

1. Frequency of Challenge
2. Research on Performance/Small Break

II.E.3 Decay Heat Removal

1. Natural Circulation
2. Shut Down Heat Removal Performance and Reliability – ~~to degraded core~~
3. Cold Shutdown (Regulatory Guide 1.139) – to degraded Core
4. Alternative Concepts

II.E.4 Containment Design

1. Penetrations
2. Water Level
3. Isolation
4. Integrity Check
5. Purge
6. Research Alternative Designs

II.F. Instruments and Controls

1. Accident Monitoring
2. Recording of Critical Plant Parameters

II.G. Electrical Power

1. Power Supplies for Pressurizer Relief Values, Block Valves, and Pressurizer Level Indication.
2. Essential Control Room Indicators - Del. from action plan .. still "to do".

II.H TMI 2 Cleanup and Examination

II.J.1 Vendor Inspection Program

II.J.2 Construction Inspection Program

III. EMERGENCY PREPARATIONS AND RADIATION PROTECTION

III.A.1 Improve NRC Capability to Respond to Emergencies

III.A.2 Improve Utility Facilities for Responding to Emergencies

III.A.3 Upgrade Emergency Planning and Preparedness

III.A.4 Emergency Planning & Rulemaking

III.B.1 Training of State and Local Government Personnel

III.B.2 Funding of State and Local Government Emergency Planning & Preparedness

III.B.3 FEMA Role in State & Local Government Emergency Planning.

III.C.1 Federal Response Planning

III.C.2 Meteorological Information for Emergency Response

III.C.3 Hydrological Monitoring

III.D.1 Distribution of Potassium Iodide

III.D.2 Radiation Levels for Protecting the Public

III.D.3 Evacuation Study

III.E.1.a Educational Opportunities for the News Media

III.E.1.b Informing the Public

III.E.2 Emergency Status Briefings and Procedures

Now part C

III.F.1.a Radiation Protection Emergency Analytical Laboratory

III.F.1.b Radiation Monitoring and Surveillance of Workers

III.F.1.c Dose Calculation Manual for Emergency Situations

III.F.1.d Radiation Protection

III.F.1.e Control Room Habitability

III.F.2.a Radiation Control-Accident Mitigation and Cleanup Design Features for Radwaste Systems

III.F.2.b Radiation Control - Vent Gas Inadequacies

III.F.2.c Radiation Control - Secondary Side Radiological Hazard

III.F.2.d Radiation Control - Improvements in Radioiodine Adsorbers

III.F.2.e Chemical Behavior of Radioactive Iodine in Water

III.F.2.f Vital Area Access and Sample Collection

III.F.2.g Liquid Pathway Interdiction Requirements

III.F.2.h Respiratory Protection Equipment

III.F.2.i In-Plant Source Term measurements of ventilation system Performance

III.F.2.j Accident Source Distribution At TMI-2

III.F.3.a Health Physics Measurements

III.F.3.b Off Site Instrum. to Actively Measure Dose & Dose Rates

III.F.3.c In Plant Rad. and Airborne Radioactive Mon. Instrumentation

III.F.3.d Radiation Monitoring

IV. REGULATORY STRUCTURE & PROCESS

IV.A. Overall NRC Organization and the Functions of the Commission

1. Achieving Single Location -- Long-Term
2. Achieving Single Location -- Interim
3. Commission Role in Adjudication
4. Strengthening Authority of Chairman and EDO
5. Delegate Authority to Single Commissioner for Emergency Response Study
6. Elimination of Non-Safety Responsibilities
7. Study of NRC Top Management
8. Revise Delegations to Staff

IV.B. NRC Staff Organization & Functions

1. Increase Emphasis on Human Factors
2. Strengthen Enforcement Process
3. Increase I&E Organizational Effectiveness
4. Fully Implement Resident Inspection
5. Direct Observation of Construction
6. Upgrade Vendor Inspection
7. Extend Lessons Learned to Non-Reactor Program
8. Upgrade NRC Training Programs

IV.C. Advisory Organizations & Functions

1. Strengthen ACRS Role
2. Improve Follow-Up ACRS Advice
3. Study of Need for Additional Advisory Committees

IV.D. NRC Safety Goals & Assessment Process

1. Developing NRC Safety Policy Statement
2. Expanded Research on Methodology for Safety-Cost Trade-Offs
3. Develop More Integrated & Systematic Approaches to Safety Assessments
4. More Effective Safety Issue Resolution

IV.E. Improvements in Adjudicatory and Rulemaking Process

1. Public and Intervenor Participation in Hearing Process
2. Improved Safety Rulemaking Procedures
3. Study of Construction-During-Adjudication Rules

DELETION I
pages A-22 - 23

KNOXVILLE, TENNESSEE 37902

December 3, 1979

APPENDIX VII
TVA PROPOSAL FOR AUGMENTED TEST AND TRAINING PROGRAM FOR SEQUOYAH 1

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

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 = 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 ≈ 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

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 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 ~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.

Establishment of natural circulation from stagnant conditions

Purpose

Demonstrate that natural circulation can be established by simulated decay heat buildup in the reactor core. In addition, this will simulate re-establishment of interrupted natural circulation.

Initial Conditions

Reactor Power $\simeq$ Hot zero power test range

Reactor coolant pumps tripped steam generators.

Isolated.

Pressurizer Heaters controlling pressure.

Normal primary system temperature and pressure.

Test Description

With stagnant conditions existing throughout the primary system, operator will initiate feed flow to all steam generators and will slowly increase reactor power into the nuclear heating range to establish a driving head for natural circulation. Establishment of natural circulation will be verified by observing the response of the hot leg and cold leg instrumentation in each loop. Core exit thermocouples will be monitored to access core flow distribution.

Boron Mixing and Cooldown

Purpose

Verify boron mixing and the capability to safely cooldown and depressurize the reactor coolant system on natural circulation.

Initial Conditions

Reactor power ≈ 1%

Natural circulation established

Steam generators being fed by normal feedwater supply

Pressurizer heaters controlling pressure

Normal primary system temperature and pressure

Test Description

Test will be initiated by boration of the reactor coolant system and concurrent control bank withdrawal to maintain ≈ 1% power. Primary coolant samples will be taken from each loop at specified time interval to evaluate boron mixing in the primary system. Boration will be terminated when boron concentraiton is increased ≈ 100 ppm. Primary system temperature and pressure will be reduced at a controlled rate in the natural circulation mode. Cooldown will be terminated before primary system temperature reaches 450° F.

PRESENT STATUS

NEW UNRESOLVED SAFETY ISSUES

- TMI RELATED ISSUES NOT YET REVIEWED FOR USI
- CURSORY REVIEW OF ISSUES FROM OTHER SOURCES
  - NO RISK ASSESSMENT
- 1979 NRC ANNUAL REPORT
  - NO NEW ISSUES REPORTED
  - IN-DEPTH/SYSTEMATIC REVIEW IN 1980
  - SPECIAL REPORT TO CONGRESS IN 1980

IN DEPTH REVIEW

SOURCES

135 GENERIC TASKS

ACRS GENERIC ITEMS

ABNORMAL OCCURRENCE REPORTS

ACRS LER ANALYSIS

NEW NON-TMI ISSUES

TMI-RELATED ISSUES

- NRC INSPECTION REPORT
- NRR LESSONS LEARNED
- ACRS REPORTS
- PRESIDENT'S COMMISSION
- SPECIAL INQUIRY
- EPRI ANALYSES
- NRC ACTION PLAN

IN DEPTH REVIEW

PRIORITY COMPONENTS

SAFETY SIGNIFICANCE

ENVIRONMENTAL SIGNIFICANCE

LICENSING EFFECTIVENESS OR EFFICIENCY

URGENCY

GENERIC APPLICABILITY

TMI RELATED PROGRAM AREAS THAT WILL LIKELY INCLUDE USI TASKS

- MAN-MACHINE INTERFACE AND CONTROL-ROOM DESIGN
- QUALIFICATION AND TRAINING OF OPERATION, MAINTENANCE, AND SUPERVISORY PERSONNEL
- OFFSITE EMERGENCY RESPONSE, EMERGENCY PLANNING, AND ACTION GUIDELINES
- SITING POLICY, INCLUDING COMPENSATORY DESIGN AND OPERATING PROVISIONS FOR PLANTS IN AREAS WHERE EVACUATION WOULD BE DIFFICULT
- SYSTEMS RELIABILITY AND INTERACTIONS
- CONSIDERATION IN LICENSING REQUIREMENTS OF ACCIDENTS INVOLVING DEGRADED OR MELTED FUEL

OTHER USI CANDIDATE ISSUES

- RADIATION EFFECTS ON REACTOR VESSEL SUPPORTS
- SAFETY RELATED PUMP AND VALVE OPERABILITY AND RELIABILITY
- BWR WELD FAILURE OF JET PUMP RETAINER BOLT
- EQUIPMENT AND OPERATOR RESPONSE TO EARTHQUAKES
- PIPE CRACKS IN PRESSURIZED WATER REACTORS

ABNORMAL OCCURRENCES REPORTED IN 1979

- ANO-1 INCIDENT - DEGRADED ESF
- 5 PLANT SHUTDOWN - SEISMIC DESIGN
- THREE MILE ISLAND UNIT 2 ACCIDENT
- OYSTER CREEK LOW WATER LEVEL INCIDENT
- DAMAGE TO NEW FUEL ASSEMBLIES AT SURRY
- ANO-1 - UNIT 2 - MISPOSITIONED AFW CONTROLS
- PALISADES - LOSS OF CONTAINMENT INTEGRITY

discussion at 6-8 Dec 79
ACRS Meeting

HIGHLIGHTS OF FNP SUBCOMMITTEE MEETING
LOS ANGELES, CALIFORNIA
NOVEMBER 17, 1979

APPENDIX IX
FNP CORE LADLE

The Subcommittee met with representatives of the NRC and OPS to discuss the proposed design of the core ladle and the implications of the TMI-2 Accident on the FNP design. A primary purpose of the meeting was to discuss the OPS response to the questions raised by the ACRS (Fraley ltr of July 25, 1979).

Although the Subcommittee concluded that progress is being made and that several questions concerning the core ladle have been resolved, additional questions remain relative to certain aspects of the FNP design. To assist the full Committee in deciding how to continue the review and evaluation of this application, these questions are summarized below.

1. There are several studies underway that may produce data and/or information directly applicable to the acceptability of the FNP design. These studies include:
   a. A WASH-1400 type evaluation of ice condenser plants;
   b. A study of the importance of liquid pathway releases to the impact of accidents in nuclear power plants;
   c. A Safety Evaluation Report assessing the safety significance of the core ladle proposed for incorporation into the FNP. Although such a ladle should delay the entrance of a molten core into the sea, it could be that it will also exacerbate airborne releases.

   The status of these studies raises the question whether it would be wise for the ACRS to await the results of these efforts prior to its review of the assessment of the core ladle as well as the acceptability of the overall FNP design.

2. Although incorporation of a core ladle into the FNP appears to offer advantages, a number of additional questions remain.

a. The proposed ladle is being designed to provide a minimum of a two-day delay in the release of a molten core. On what basis was the two-day figure selected? What does it accomplish? How difficult would it be to extend this time? What are the criteria for judging the overall acceptability of a ladle?

b. What happens if water should gain access to the ladle prior to and after the molten core reaches it? How serious a problem would this be? Has it been evaluated?

c. Will the proposed ladle change the nature and quantities of the radionuclides that become airborne and available for release? If so, how will this affect the acute and long-term impact of such releases?

31. There is a variety of questions concerning the containment proposed for the FNP. In the opinion of several members of the Subcommittee, this subject is in need of evaluation on a systems basis. Such an assessment should include consideration for the following items:

a. Would there be advantages in increasing the containment design pressure? To what degree could it be increased and at what cost?

b. Are there better approaches that should be developed for handling the production of hydrogen gas? Is controlled burning feasible? Is inerting a practical approach?

c. Is filtered venting a useful approach for handling excessive pressure buildup? Should experiments be conducted to evaluate the removal efficiencies of sea water for specific airborne radionuclides vented through it? Are there other approaches that should be considered?

d. Are accident assessments based solely on WASH-1400 scenarios adequate? Should the ACRS require an evaluation of accident impacts associated with other scenarios?

4. At the present time, the NRC Staff has not developed criteria for judging the adequacy of the control of liquid radionuclide releases from nuclear power plants.
   a. Should ACRS review of the FNP design be delayed until criteria for limitations on accidental liquid releases are developed?
   b. Should review of the design be delayed until criteria are developed for judging the adequacy of various methods of mitigating such releases?

5. Finally, there is a number of procedural questions that will impact on this application. Some of these may need to be answered prior to final ACRS action.
   a. How will the results of the TMI-2 Lessons Learned studies be incorporated into such facilities?
   b. How and what time schedule do the NRC Commissioners plan to act relative to the FNP?

The ACRS may want to discuss these and other related questions with the Commissioners prior to completion of its review.

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

December 10, 1979

APPENDIX X
INTERIM COMMENTS TO CHAIRMAN AHEARNE
REGARDING THE RECOMMENDATIONS OF THE
PRESIDENT'S COMMISSION TO STRENGTHEN
THE ACRS

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

Subject: RECOMMENDATIONS OF PRESIDENT'S COMMISSION REGARDING ACRS ACTIVITIES

Dear Dr. Ahearne:

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 recommendation of the President's Commission, we have the following comments.

The ACRS agrees that its role should be strengthened. The Committee also agrees that it is important to maintain its independence. The Committee believes however that measures taken to strengthen its role should not jeopardize that independence.

Although the Committee agrees with the intent of certain recommendations of the President's Commission which are meant to strengthen its role, it is not ready to endorse these recommendations until it has had the opportunity to study alternatives which might be more appropriate and effective.

The ACRS will give this matter early attention so that its views can be formulated promptly.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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

December 11, 1979

APPENDIX XI
INTERIM LOW POWER OPERATION OF SEQUOYAH
NUCLEAR POWER PLANT UNIT 1

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

SUBJECT: INTERIM LOW POWER OPERATION OF SEQUOYAH NUCLEAR POWER PLANT, UNIT 1

Dear Dr. Ahearne:

During its 236th meeting, December 6-8, 1979, the Committee considered a proposal for interim, low power operation of the Sequoyah Nuclear Power Plant, Unit 1. At its 229th meeting, May 10-12, 1979 and also at its 228th meeting, April 5-7, 1979 the Committee had considered aspects of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Power Plant, Units 1 and 2. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on March 12, 1979 and on November 5, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this plant on February 11, 1970.

The Sequoyah Nuclear Power Plant is located on the west bank of the Tennessee River in Hamilton County in southeastern Tennessee approximately 17 miles northeast of the center of Chattanooga, Tennessee. Construction on Unit 1 is essentially complete and construction of Unit 2 is about 90% complete. Each unit will utilize a four-loop pressurized water reactor nuclear steam supply system having a power level of 3411 MWt and an ice condenser system enclosed within a free-standing steel containment vessel which is surrounded by a reinforced concrete shield building. The ice condenser system is similar to that used in the McGuire Nuclear Station and the Donald C. Cook Nuclear Plant. The Applicant has modified the ice condenser system as a result of the operating experience gained in the Donald C. Cook Nuclear Plant. The Applicant and the NRC Staff have made plans to monitor the performance of the ice condenser containments at the Sequoyah Nuclear Power Plant (Generic Item 63 in the ACRS report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979). The Committee recommends that such plans be implemented.

The Sequoyah Nuclear Plant will utilize 17x17 fuel assemblies. A surveillance program has been developed by the NRC Staff to follow the behavior of these assemblies, and data are being obtained from several plants now in operation in which such assemblies have been installed for test. Experience to date has been satisfactory. The Committee wishes to be kept informed of the results of the various 17x17 assembly inspections and test programs now under way.

The Sequoyah site is considered by the NRC Staff to be within the Southern Valley and Ridge tectonic province. The maximum historic earthquake within this tectonic province is the 1897 Modified Mercalli Intensity (MMI) VIII earthquake in Giles County, Virginia. During the construction permit review, the NRC Staff concluded that a modified Housner response spectrum anchored at 0.18g was acceptable as the safe shutdown earthquake. Since that time, the NRC Staff has adopted methods which would characterize an MMI VIII earthquake with the more conservative response spectrum specified in Regulatory Guide 1.60 anchored at 0.25g.

The Applicant, in response to NRC Staff recommendations, has evaluated the Sequoyah design using a site-specific safe shutdown response spectrum developed from North American and Italian strong motion records of appropriate magnitude and epicentral distance and has compared the probability of the safe shutdown earthquake being exceeded at Sequoyah to that at other Tennessee Valley Authority plants that meet the Standard Review Plan. It has been concluded that the risk of exceeding the present design spectrum and the risk of exceeding the site-specific spectrum are comparable and that the probability of exceeding the safe shutdown earthquake is not appreciably different from that for other plants in this region. The NRC Staff has reviewed the Applicant's evaluation and has concluded that the Sequoyah plant is adequate to withstand the effects of the safe shutdown earthquake without loss of its capability to perform required safety functions. The NRC Staff, to verify their judgments regarding structural and component design margins, has performed an audit of the design margins in representative critical sections of the reactor and auxiliary building structures and in representative components required for safe shutdown.

The Committee recommends that this program for the quantification of the seismic design margin be continued and expanded to the extent necessary to ensure that all structures and equipment necessary to accomplish safe shutdown do indeed have some margin. Similar recommendations have been made by the Committee for the North Anna Power Station, Units 1 and 2, and the Davis-Besse Unit 1 in its reports dated January 17, 1977 and January 14, 1979. This matter should be resolved on a schedule and in a manner satisfactory to the Staff.

The Emergency Core Cooling Systems (ECCS) for the Sequoyah Nuclear Plant incorporate the Upper Head Injection (UHI) system. The NRC Staff has completed its review of the Westinghouse Electric Corporation ECCS evaluation model for plants equipped with UHI, and the Committee in its April 12, 1978 report on the McGuire Nuclear Station has concurred with the

Staff's conclusions. The NRC Staff has completed its review of the application of this approved evaluation model to the Sequoyah Nuclear Plant and concurs with the Applicant.

The Committee has been reviewing the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2 and has made recommendations for improvements in plant design and operating procedures which should be considered for all pressurized water reactors. The Committee is continuing its review of the implications of this accident and expects to provide additional recommendations. It is expected that these recommendations will be considered and implemented as appropriate by the NRC Staff. The Committee wishes to be kept informed.

The NRC Staff has identified a number of outstanding issues, confirmatory issues, and licensing conditions, not related to TMI-2 accident considerations, which have not been specifically addressed in this report. These issues should be resolved in a manner satisfactory to the NRC Staff.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. Those problems relevant to the Sequoyah Nuclear Plant should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: 54-60, 63-65, 69, 71, 72, 74, and 76.

The NRC Staff has not completed its review of the Sequoyah Nuclear Power Plant application for a normal operating license at full power, and various implications of the Three Mile Island accident on the Sequoyah Plant remain to be decided. The ACRS has not completed its own review in regard to these matters.

The Applicant has proposed a program of interim low power operation to provide improved operator training and the development of additional experimental information on the behavior of a nuclear unit and its systems under transient conditions. The Applicant has proposed a special test series which includes the following:

1. Natural circulation following a simulated reactor trip.
2. Natural circulation following a simulated loss of offsite power.
3. Natural circulation with loss of pressurizer heaters.
4. Effect of steam generator isolation on natural circulation.
5. Natural circulation at reduced pressure.
6. Cooldown capability of the charging and letdown system.

7. Heat removal following a simulated loss of onsite and offsite AC power.

8. Establishment of natural circulation from stagnant flow conditions.

9. Boron mixing and cooldown.

The NRC Staff plans to review the proposed experimental program in detail to assure itself that all safety-related aspects are being dealt with appropriately. The Committee wishes to be kept informed.

The NRC Staff advised the Committee that it will require that TVA's emergency procedures for Sequoyah be reviewed by Westinghouse. The NRC Staff also stated that an acceptable emergency plan will exist prior to reactor operation.

The Committee believes that there is reasonable assurance that the Sequoyah Nuclear Power Plant, Unit 1 can be operated on an interim basis up to power levels of about five percent of full power without undue risk to the health and safety of the public. Subject to approval of the detailed test program by the NRC Staff, the Committee recommends approval of an interim low power license for the purposes proposed.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

References:

1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1 to 13, and Amendments 1 to 61.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 31, 1979, containing revised responses to the Lessons Learned Requirements.
4. Letter, L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 30, 1979, containing responses to ACRS questions.
5. Letter from L. M. Mills, TVA, to L. S. Rubinstein, NRC, dated October 23, 1979, containing information on natural circulation in Sequoyah, Unit 1, and Diablo Canyon, Unit 1.
6. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated October 12, 1979, containing responses to ACRS recommendations.

7. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated September 7, 1979, containing responses to the Short-Term Recommendations of the Lessons Learned Task Force.
8. Letter from L. M. Mills, TVA, to D. B. Vassallo, NRC, dated July 12, 1979, containing responses to NRC-I&E Bulletin 79-06A and ACRS recommendations.

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

December 13, 1979

APPENDIX XII
REPORT ON TMI-2 LESSONS LEARNED TASK FORCE FINAL REPORT

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

SUBJECT: REPORT ON TMI-2 LESSONS LEARNED TASK FORCE FINAL REPORT

Dear Dr. Ahearne:

The TMI-2 Lessons Learned Task Force has issued its Final Report, NUREG-0585. The ACRS provides comments herein both on the specific recommendations made by the Task Force and on related subjects. The Committee will first address the recommendations made in NUREG-0585.

1. Personnel Qualifications and Training.

The ACRS gives general support to the recommendations made in this category.

The ACRS believes that, although a broader technical background should be required of Shift Supervisors, it may be neither necessary nor practical to require that all Shift Supervisors have a Bachelor of Science Degree. The Committee recommends that the NRC define its criteria for "equivalent training and experience in engineering or the related physical sciences." The ACRS believes that a training program tailored to the requirements of reactor operation, possibly of less than four years duration, may provide a practical alternative to a formal degree program. The Committee believes that the NRC should define the scope and duration of a training program that may be considered as an acceptable alternative to a degree curriculum. The ACRS also recommends that, if the Technical Advisor system proves satisfactory, consideration should be given to offering licensees the option of retaining that system instead of upgrading the academic education of Shift Supervisors to the specified level.

The ACRS recommends that the adequacy of staffing in the NRC Operator Licensing Branch be reevaluated with respect to the number of personnel and breadth of their background.

The Committee believes that additional emphasis must be given to the determination of what constitutes an adequate degree of in-house technical capability for each licensee and assurance of the continuing development of such capabilities. The ACRS also believes that attention must be given to providing, on a continuing basis, technical backup to review safety-related design changes or to provide assistance under accident conditions by a group having the depth of technical knowledge which exists in the organization of the nuclear steam system supplier and a well-qualified architect-engineer during the period while the plant is being designed.

2. Staffing of Control Room.

   The ACRS supports this recommendation.

3. Working Hours.

   The ACRS supports this recommendation.

4. Emergency Procedures.

   The ACRS, in general, gives strong support to this recommendation. However, the Committee believes that the emergency procedures at licensed power reactors should receive priority. The ACRS recommends that the licensees should give priority to the development of improved emergency procedures with the aid of expert, interdisciplinary review groups and that the NRC Staff should review, in depth, the existing and proposed, emergency procedures for a large sample of licensed reactors on a priority basis.

   The knowledge developed from the concurrent industry and NRC efforts should be used to revise, in a timely fashion, the emergency procedures of all operating plants.

5. Verification of Correct Performance of Operating Activities.

   The ACRS gives general support to this recommendation.

6. Evaluation of Operating Experience.

   The ACRS gives general support to these recommendations.

   Additional Committee comments on this subject are contained in NUREG-0572, "Review of Licensee Event Reports (1976-1978)."

7. Man-Machine Interface.

   The ACRS gives general support to these recommendations.

   In addition to the nine items listed in NUREG-0585, Appendix A, Section 7.1 , the Committee recommends that the licensee should include in his evaluation the data recording requirements and recall capabilities of the minimum set of plant parameters that defines the safety status of a nuclear power plant.

8. Reliability Assessments of Final Designs.

The ACRS strongly supports the application of reliability assessments to final designs. The Committee supports the Integrated Reliability Evaluation Program (IREP) which is being initiated by the Office of Nuclear Regulatory Research. However, the Committee does not agree that the proposed IREP will fully satisfy the need. The ACRS recommends that the NRC develop a program in which licensees acting individually or jointly develop reliability assessments of their plants, in addition to the NRC IREP, which should be performed concurrently.

If the reliability assessments were performed in the manner proposed above, it would accelerate obtaining potentially significant safety information and expedite the development of the basis for changes, should they be necessary. It would also provide the operating organizations with better technical insight into the safety of their plants and would provide the benefits to be derived by separate studies of system reliability.

9. Review of Safety Classifications and Qualifications.

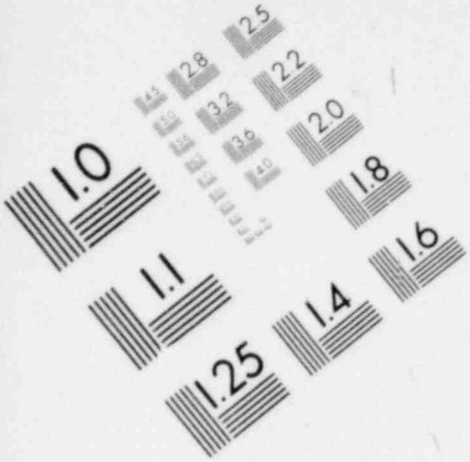
The ACRS supports this recommendation. A particular problem warranting early attention is the qualification of operator information systems. More generally, the Committee believes that more than a year will be needed to accomplish the overall task, partly because of its breadth and depth, and partly because of the very considerable number of knowledgeable personnel which would be needed.

The Committee agrees that completion of the overall task should not be made a condition for the licensing of new plants.

10. Design Features for Core-Damage and Core-Melt Accidents.

The ACRS supports this recommendation. However, the Committee believes that the recommendation should be augmented to require concurrent design studies by each licensee of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk.

The ACRS agrees with the recommendation made by the Lessons Learned Task Force in NUREG-0578 that the Mark I and Mark II BWR containments should be inerted while further studies are made of other possible containment modifications in accordance with the general recommendations in this category. The ACRS also recommends that special attention be given to making a timely decision on possible interim measures for ice-condenser containments.



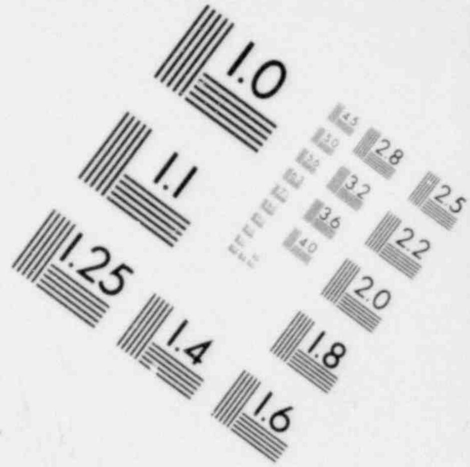
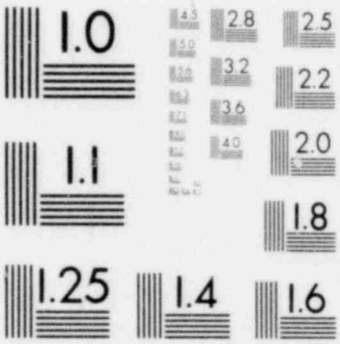


IMAGE EVALUATION TEST TARGET (MT-3)



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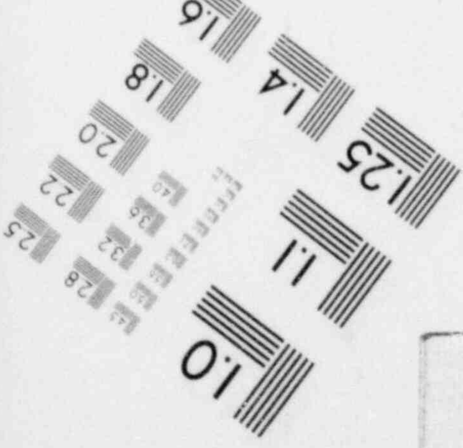
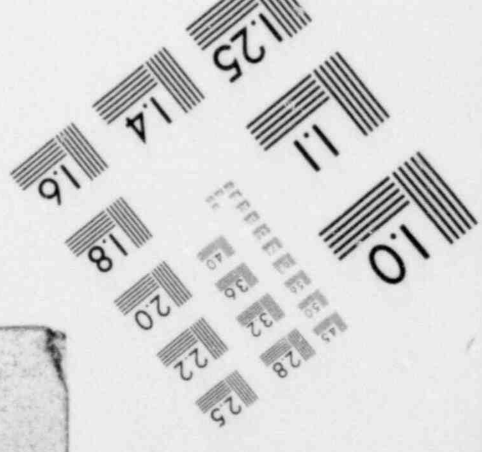
MICROCOPY RESOLUTION TEST CHART





The Committee also recommends that special attention be given to operating reactors located at densely populated sites.

11. Safety Goal for Reactor Regulation.

    The ACRS supports this recommendation.

12. Staff Review Objectives.

    The ACRS supports this recommendation. However, the ACRS believes that there is a need for review of NRC safety rules, regulations, guides and philosophy on a regular basis in order to ascertain various matters including the following:

    a. Does an appropriate balance exist in the expenditure of NRC financial and manpower resources among the various research areas, on the resolution of safety issues, on the legal requirements of licensing, and on inspection and enforcement?

    b. Is there an appropriate division of effort and responsibility between industry and the NRC?

    c. Has an undesirable inflexibility in the approach to safety developed due to previous decisions, or for other reasons?

    d. Are there any important gaps in the existing safety review process? Is there a mechanism for searching out such gaps?

13. NRR Emergency Response Team.

    The ACRS gives general support to these recommendations. The Committee believes that the timing of implementation should be more flexible. The Committee believes that better definition of the NRC role and responsibilities in an emergency will have an influence on the determination of the makeup, training and abilities of an NRC emergency response team.

The ACRS wishes to make some comments and recommendations on several matters not directly addressed in NUREG-0578 or NUREG-0585.

1. The ACRS believes that the lessons learned from the TMI accident should be viewed in a broader perspective. The Committee agrees that the TMI accident shows a need for considerable improvement

in reactor operations and in knowledge of the behavior of a plant during a wide range of transients. However, the Committee believes that there are other potentially important contributors to the probability of a reactor accident, and they should also receive priority attention.

Reliability assessments and systems interactions studies, as discussed under recommendations 8 and 9 above, should serve this function in part. However, there is a need also to consider, in some more systematic way, methods to uncover significant design errors, to detect system or component degradation, and to test systems under conditions more closely simulating the range of situations which might result from transients and accidents.

2. The Task Force has not addressed the need to reexamine the adequacy of the current design basis for emergency cooling recirculating systems, as recommended by the ACRS in its report of August 14, 1979 on "Studies to Improve Reactor Safety."

   There are several other specific recommendations made by the ACRS in its interim reports Nos. 2 and 3 on Three Mile Island both dated May 16, 1979 and in its report of August 14, 1979 on studies to improve reactor safety. The Committee believes that the NRC Staff should address each such recommendation in formulating its overall action plan.

3. The ACRS recommends that a reevaluation should be made of the potential influence of a serious accident involving significant atmospheric release of radioactive materials from one unit of a multiple unit site on the ability to maintain the other units in a safe shutdown condition.

4. The ACRS recommends that the industry and the NRC Staff undertake studies to ascertain what contingency design measures, beyond those covered in the Task Force recommendations, may ensure improved capabilities for recovering from or mitigating the effects of accidents beyond the design basis. For example, in some cases, it may be possible to provide alternative measures in the event of loss of the safety grade ultimate heat sink for an extended period of time.

5. The ACRS recommends that the NRC Staff give attention to the seismic implications of TMI, for example, the seismic qualifications of auxiliary feedwater supplies, the acceptability of failure of nonseismic Class 1 equipment, and the suitability of emergency procedures for earthquakes.

6. The ACRS recommends that greater consideration be given to the provision of dedicated shutdown heat removal sytems, and to the potential merits of having a shutdown heat removal system capable of operating at normal system pressure.

The ACRS expects to address other considerations of reactor safety and the regulatory process in a separate report.

Sincerely,

Max W Carbon

Max W. Carbon
Chairman

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

December 11, 1979

APPENDIX XIII
COMMENTS ON PAUSE IN LICENSING

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

Subject: COMMENTS ON THE PAUSE IN LICENSING

Dear Dr. Ahearne:

The President's Commission on the Accident at Three Mile Island has recommended that:

> "Because safety measures to afford better protection for the affected population can be drawn from the high standards for plant safety recommended in this report, the NRC or its successor should, on a case-by-case basis, before issuing a new construction permit or operating license: (a) assess the need to introduce new safety improvements recommended in this report and in NRC and industry studies; (b) review, considering the recommendations set forth in this report, the competency of the prospective operating licensee to manage the plant and the adequacy of its training program for operating personnel; and (c) condition licensing upon review and approval of the state and local emergency plans".

Since issuance of this report, the Nuclear Regulatory Commission has stated that there will be a pause of many months before the NRC will license any of the reactors now nearing readiness for operation while safety improvements are worked out for the reactors already in operation. Longer delays are anticipated for new construction permits.

The ACRS agrees with most of the recommendations made by the President's Commission. The ACRS supports the basic recommendation of the President's Commission which is quoted above, but with some qualifications which are discussed below.

The ACRS believes that the risk to the public health and safety which is posed by the operating nuclear power plants is comparable to or probably smaller than the risk posed by other existing methods of generating the same quantity of electricity. The ACRS also believes that this risk is comparable to or less than that posed by many other technological activities of society.

The ACRS has, in the past and again since the Three Mile Island accident, recommended that the NRC and the nuclear industry take major steps to improve the safety of nuclear power reactors. The ACRS believes that it is proper that nuclear power be safer than other comparable technologies. The Committee has sought this goal. It believes that the country wants a higher level of safety for nuclear reactors and is willing to pay for it. The ACRS also believes that the country wants a higher degree of assurance as to the level of safety which is being attained.

While the ACRS believes that interim licensing of the next six to twelve nuclear power reactors for operation on the same basis as is now being accepted for currently operating reactors would not pose undue risk to the public health and safety, the ACRS favors the consideration of additional improvements in their safety on a case-by-case basis, as recommended by the President's Commission. Nevertheless, the following additional considerations can and should have a strong bearing on the specific NRC approach and actions in this regard:

1) For those reactors which are ready for power operation, there exists the possibility that a considerable body of experimental information having either a plant-specific or a general safety significance can be obtained by performing appropriate tests on systems or the entire plant at powers up to about 5% of full power. These are tests which are not usually run because of the time they consume. They would afford essentially no risk to the public health and safety. There also exists the possibility of providing more than the normal training of operators.

2) If the NRC pause becomes relatively extended, there may arise a more severe national need for additional electric power. The ACRS recommends that consideration be given to permitting newly completed nuclear power plants which meet the requirements of NUREG-0578 to start up and undergo testing at power levels up to 50% or 75% of full power, after which they could be placed in a shutdown condition, available for call in the event of national need, while the NRC reaches a resolution as to the additional safety requirements it will impose before permitting normal commercial operation.

3) The ACRS believes that the safety improvements recommended by the President's Commission, the various NRC Task Forces and the ACRS itself, should be evaluated and acted upon expeditiously. However, the ACRS believes that a judicious choice is required as to which recommendations require implementation promptly, which require implementation on a specified time scale during which reactors are permitted to operate, and which warrant study and resolution on some specified and achievable time scale. The ACRS supports the rapid steps being taken by the NRC to develop an action plan and will expedite its review of the plan.

4. Although the ACRS believes that operating reactors should receive priority, and that reactors under construction also require emphasis from the NRC Staff, the Committee recommends that the NRC Staff take steps in timely fashion to redirect, as appropriate, the design of reactors for which a construction permit has not been granted or for which construction has not been initiated. General guidance, as well as requests for studies of design alternatives could be useful in this regard. The Committee believes that the initiation of possible design changes need not await the complete development of a final NRC position on changed or additional requirements for reactors which have not yet received a construction permit.

The ACRS is available to work with the NRC Staff to help achieve these actions.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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

December 11, 1979

APPENDIX XIV
ADEQUACY OF PROCEDURES FOR COMMUNICATIONS AND INTERACTIONS BETWEEN ACRS AND NRC STAFF

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

Dear Dr. Ahearne:

The following comments are provided in response to your letter of October 3, 1979 regarding the adequacy of procedures for transmitting recommendations and questions generated by the ACRS, its subcommittees, and its individual members to the NRC Staff and the subsequent staff responses.

The ACRS realizes that its procedures and practices for transmitting recommendations and questions to the NRC Staff, and to the Commission itself, have been deficient in some respects. Steps have been and are being taken to correct this, both by the ACRS itself and in cooperation with the NRC Staff. For example, in 1976 the Committee and the then Director of Licensing agreed on a procedure to obtain clarification of ACRS recommendations when needed. This procedure has been used only sparingly by the NRC Staff. The Committee is aware of the need to indicate priorities more specifically and to describe more clearly the basis for its concerns and questions and the degree of importance that it attaches to them.

Although there have been significant problems with the nature and timeliness of the NRC Staff's response to ACRS concerns, the Committee believes that changes in its procedures, together with one or more of the changes in the NRC Staff's procedures now being considered, will be of help in improving the present situation.

It must be noted, however, that many of the ACRS recommendations are formally addressed to the Commission itself, in accordance with the statutory requirement that the ACRS advise the Commission. In many cases, these reports are simply referred by the Commission to the NRC Staff for action or response. In most of these cases, this procedure is appropriate. However, there are some circumstances in which the recommendations involve matters of policy or are such that action or specific attention by the Commission itself, particularly an indication of priority and authorization of appropriate resources, is required. The degree to which these reports receive the attention of the Commission has not always been apparent.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

Sec OP 1.6

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

OFFICE OF THE
COMMISSIONER

October 3, 1979

Dr. Max W. Carbon, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Carbon:

There is a growing concern about the adequacy of transmitting recommendations and questions generated by the ACRS, its subcommittees, and its individual members to the NRC staff and about the subsequent staff responses. I share these concerns. The present procedures should be reviewed and revised as necessary or new procedures developed. I would appreciate ACRS comments on the adequacy of the present procedures and the areas in which they need to be strengthened.

I am concerned about a related area as well, namely, that in generating recommendations and questions, the ACRS is not providing sufficient guidance to the NRC staff as to the priority or degree of importance the ACRS assigns to each. For example, some inquiries may be of such a nature that the ACRS does not recommend the staff follow up. Therefore, I would appreciate knowing to what extent and by what method questions generated by the ACRS, its subcommittees, and its individual members are screened and prioritized before transmittal to the NRC staff.

I would appreciate suggestions as to an appropriate mechanism whereby the ACRS and the Commission could work together to revise or redraft the procedures and priorities.

Sincerely,

John F. Ahearne
Commissioner

cc: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
EDO
SECY
OPE
Ray Fraley, Exec. Director, ACRS ✓

OP 1.6
REG-1
OP 2.1
A-63

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

December 10, 1979

APPENDIX XV
LETTER TO REP. M. K. UDALL REGARDING
PROPOSED NRC FY-80 SUPPLEMENTAL RESEARCH
BUDGET

The Honorable Morris K. Udall, Chairman
Committee on Interior and Insular Affairs
U. S. House of Representatives
Washington, DC 20515

Dear Congressman Udall:

This letter is in response to oral requests for ACRS comments on the amendment for supplemental appropriation for the NRC FY-80 Authorization.

The NRC proposed supplemental funds for research are, for the most part, in good agreement with those previously reviewed by the ACRS and discussed in Part 2 of its report NUREG-0603, "Comments on the NRC Safety Research Program Budget." However, the ACRS strongly recommended a supplemental request for \$3.4 million for Research to Improve Reactor Safety, and stated that an FY-80 budget of \$4.4 million was barely sufficient to begin work on the initial program proposed in NUREG-0438. The ACRS continues to support strongly its recommendation for an additional \$3.4 million supplement for Research to Improve Reactor Safety. The ACRS considers it essential that the NRC significantly increase the pace of this program. In its letter of July 18, 1979 to NRC Chairman Hendrie, the ACRS recommended that there be strong programs of research to improve reactor safety both in the NRC and DOE. In that letter, the ACRS stated that a level of \$4.4 million within the NRC for FY-80 would be less funding than desirable.

If an additional \$3.4 million cannot be added to the supplemental FY-80 budget for the NRC, the ACRS recommends that money be reprogrammed from other areas to provide the recommended funding for Research to Improve Reactor Safety.

The ACRS also wishes to note that it places considerable importance on its recommendations for new directions in research as made in NUREG-0603. The ACRS recommends that the NRC be given sufficient reprogramming authority to address these ACRS recommendations vigorously in FY-80.

Sincerely,

Max W. Carbon
Chairman

NUREG-0642

A Review of NRC Regulatory Processes and Functions

Manuscript Completed: December 1979
Date Published: January 1980

Advisory Committee on Reactor Safeguards

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

December 17, 1979

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

Dear Dr. Ahearne

The experience at Three Mile Island, Unit 2 was a dramatic reminder that improvements in the nuclear regulatory process are needed. This is not to overlook the fact that the existing process has so far been quite effective in protecting the health and safety of the public and provides a solid base for the needed improvements. The experience of twenty-five years of nuclear power stands as evidence of that statement.

In this context, and while continuing its review of the TMI-2 accident implications, the ACRS has been reexamining the regulatory process, and submits herewith the results of this study. We had a dual objective. First, we wanted to provide a single source to describe our understanding of how the system has functioned up to now. The many investigations of TMI have revealed considerable confusion about the structure of this complex and interactive process, and we have tried to describe it and its geneology. Second, we wished to call out weaknesses, as we see them, and to make appropriate recommendations for change.

You will find that we have not separately listed our recommendations in any "executive summary" so that a reading of the document is necessary, but it is our view that recommendations for change should be contained in the description of the existing system to make them meaningful. Nonetheless, some of the more important recommendations appear in Chapter 8. We have found this exercise instructive to ourselves.

We are, of course, aware of the recommendations of the President's Commission, the President's response to those, and of the other reviews now in progress. We hope that this document will be generally useful, and submit it with that intent.

Sincerely,

Max W. Carbon
Chairman

FOREWORD

Any important government function deserves periodic examination to determine whether it is serving the public need in an appropriate manner. The Nuclear Regulatory Commission (NRC) has been in existence since 1975 to regulate nuclear matters affecting the health and safety of the public through a government licensing process. The recent accident at Three Mile Island, Unit 2 (TMI-2) has made the public extremely sensitive to nuclear regulatory activities. The Congress is giving serious consideration to alterations in the regulatory structure, anticipating that such changes may enhance the national public safety. The President appointed a Commission to examine the TMI-2 event and to make recommendations concerning the regulatory process and functions as a result of information derived from that accident. These actions all point toward a need for prompt reexamination of the United States nuclear regulatory system.

While both the NRC and the President's Commission are developing independent assessments of the regulatory process, nuclear regulation cannot be examined in the context of a single event or a single point of time. The process has been evolving over a period of about 25 years and has the advantage of thoughtful and probing review over that entire period, much of it broadly displayed through the communications media to the entire population. Hence, it is appropriate at this time to understand well what has developed over the 25-year period before considering changes that materially affect the current regulatory processes. Changes are needed urgently in some areas. Many are already being effected or planned by the NRC organization and its licensees. However, care must be taken to assure that the changes under consideration or to be identified in the future will, in fact, strengthen the regulatory process and functions.

The Advisory Committee on Reactor Safeguards (ACRS) has spent much time over many years observing and examining the NRC licensing process. The Committee is, consequently, in a position to comment on the situation, and it believes this review will be helpful to those examining the regulatory process by discussing how it works, where it is weak, and the opportunities for improvement. The Committee's review may also help put current proposals and discussions in perspective.

CONTENTS

1. INTRODUCTION

The Congress of the United States established the NRC along traditional regulatory lines, wherein the Commission sets regulatory criteria and requirements for industrial participants who are bound to meet the regulatory requirements as a condition of licensing. The law places the onus on the licensee to show compliance and on the regulatory Commission to determine compliance. The Commission has authority to impose both legal restraints and monetary penalties on those who fail to comply with the regulatory requirements. The Commission's authority generally transcends that of state and local governments, but it has acted to establish a cooperative relationship with all levels of government in order to maximize public acceptance of the regulatory process.

The operation of the NRC has some unusual aspects, including the way in which the Commission itself functions, the statutorily defined functions of the regulatory operations staff, the hearing process of the Atomic Safety and Licensing Boards (ASLB) and the review by the ACRS. Much of this is unique among United States regulatory processes, but the principles are similar to those of other regulatory systems.

The Congress has assigned to NRC the responsibility for regulating the construction and operation of nuclear power plants operated by privately financed public utilities and publicly owned power agencies. The NRC discharges this responsibility by imposing technical and administrative requirements as a condition of issuing construction permits and operating licenses, and by monitoring the performance of licensees. However, the prime responsibility for safe design, construction, and maintenance of nuclear power plants rests with the licensees.

Insofar as safety is concerned, the system has a number of advantages, but the primary one is that the user groups have both financial and legal incentive to operate the power plants in a safe fashion. The regulatory organization can act as a "watchdog" to make certain that the conditions of the license are satisfied. The system suffers from unevenness of application that leads to shallow audits of some areas of safety interest and overly detailed review of others. The present system also puts grave responsibility on licensees to make certain that the nuclear technology is used in a way which minimizes the potential for harm to the public even though they have counteractive pressures to minimize costs and improve profitability.

Other regulatory systems can be visualized. One such system would involve operation of a plant built with private or public funds by a governmental organization while a second governmental organization served as a "watchdog" over the first. Some countries use this arrangement. However, the advantages of one system over another can be discussed only in qualitative terms. The present system has a substantial base of experience developed over a quarter of a century; hence, attention in this review is directed mainly to the existing regulatory concept, its strengths, its weaknesses, and the need for improvements.

The NRC functions under the requirements of the Atomic Energy Act and its subsequent modifications, although the Commission was created by the Energy Reorganization Act of 1974. The Commission staff numbers more than 2000 people. By comparison with some others, it is a large regulatory agency. The Atomic Energy Act specifies the duties of the Commission as they apply to regulation of the use of radioactive and fissionable material, with the main emphasis on nuclear power plants and the nuclear fuel cycle, in the interest of public safety. However, as a spinoff of the National Environmental Policy Act (NEPA) and the Calvert Cliffs decision, the NRC has directed a large portion of its activities to the NEPA evaluation of licensing actions. The NEPA review requires a significant commitment in terms of manpower, perhaps 50-75 per cent as much effort as does the safety review. Thus, when examining the nuclear regulatory process, it is important to recognize the regulatory Commission's response not only to its own legislative mandate but to the related responsibilities derived from NEPA. The agency's functions are further complicated by its overlapping responsibilities with the Department of Energy (DOE). These, too, have to be taken into account when considering the regulatory process.

Although the review of the nuclear regulatory process presented herein was performed from the vantage point of the TMI-2 experience, the entire history of nuclear power regulation was considered. The reference time for this discussion of the state of the regulatory process is that period just prior to the TMI-2 accident. Since that time, changes have been made or are being planned by the NRC and by its licensees. Changes of which we are aware are noted herein.

2. REGULATORY GOALS

The Atomic Energy Act and its subsequent amendments and the Energy Reorganization Act of 1974 provide a broad charter for the NRC, its staff, its hearing boards, and its advisors to regulate nuclear energy processes and products, as needed, to protect the health and safety of the public. NEPA, as interpreted in the Calvert Cliffs decision, includes requirements for balancing of costs and benefits and evaluation of environmental impacts as conditions for nuclear licensing. These statutory requirements must comprise the basis for judgments about the effectiveness of nuclear regulatory processes. While identifying important organizational participants, the legislation does not specify the regulatory process in great detail, thereby allowing the Commission latitude in establishing the methods for satisfying statutory requirements.

The Commission has not set forth specific objectives or goals in any official document or statement, but they can be inferred from the types of activities in which the NRC is involved and the resultant decisions. Although not stated formally, the goals of the Commission should be kept in mind when judging the organization and programmatic thrust of the NRC. A list of these goals should include:

1. establishment of regulatory policies, standards, practices, and procedures that, while recognizing the societal need for energy and the associated economic considerations, make due allowance for public safety and moral obligations to present and future generations,

2. provision of criteria for public safety or other regulatory decisions set forth in understandable form and, where practical, with the use of quantitative risk evaluation methods which permit the relating of nuclear risks to other societal risks,

3. provision and maintenance of a regulatory staff to establish requirements and enforce regulations,

4. establishment of a regulatory system such that license compliance with the requirements can be demonstrated,

5. provision of evidence through documentation, and regulatory actions that the goals of the regulatory process are being met, and

6. establishment of procedures for keeping the public informed on all matters of public interest, both from a societal and a technological point of view.

One of the purposes of this review is to determine whether these goals can be met. Safety, environmental protection, and economics can have conflicting demands; the NEPA acknowledged this in its requirement for environmental balancing. The balance may be altered as industry grows, technological understanding broadens, or political circumstances change (2.1). Public acceptance of the regulatory process depends upon conveying to the public an accurate and fair representation of regulatory effectiveness with respect to established regulatory goals.

The regulatory process is discussed in this report with these goals in mind. The report provides evidence as to how nearly the goals are being attained, but no attempt is made to establish a grading system because the standards for judgment will always be influenced by time and circumstances. It is important that the process include the capabilities needed to achieve the goals if it is to serve the public adequately.

(2.1) When nuclear-generated electric power was originally introduced as a source of energy in the United States, the main consideration was its economic competitiveness with other forms of energy, such as coal, gas and oil. Recently its availability has become a matter of strategic importance to our national defense and international policy. Public safety and national or world economic investment can also influence political circumstances. These matters can have a bearing on how, whether, and where to use nuclear power.

3.0 THE CHANGING STYLE OF THE REGULATORY PROCESS

The NRC institutional arrangement has developed over about a 25-year period. Initially, the regulation of nuclear power plants was carried out by an arm of the now defunct Atomic Energy Commission (AEC). The regulatory function became more active in the mid-1950s when the first commercial nuclear installations were being planned. During that period, the AEC participated in the development of a number of nuclear power concepts.

The ACRS was established in late 1947 by the AEC to review safety-related aspects of the AEC-owned research, test, experimental, and production reactors. In 1955, the AEC established a small, full-time hazards evaluation staff to perform safety reviews with technical guidance and oversight provided by the ACRS. The AEC staff and the applicants for licenses were responsive to the recommendations of the ACRS, which was in many respects the ultimate reactor safety authority. In 1957, the ACRS was established as a statutory body. At the same time the licensing process was opened to public participation by the establishment of the AEC public hearing process conducted by a "hearing examiner." The hearing process was a procedural mechanism to demonstrate on the public record that the review was complete and to adjudicate differences between parties. Although the ACRS was not a party to the hearing, its recommendations were given serious attention by all parties, including the hearing examiner.

By the early 1960s, the nature of the hearing process had changed and the hearing examiner was asked to make technical decisions regarding interpretations of AEC regulations, the scope of the regulations, and the technical basis for the regulatory licensing process. The AEC Regulatory Staff had to develop its own expertise to address these issues and began to make its own independent judgments, which were tested along with those of license applicants during the review process. In 1962, the ASLB was established to conduct licensing hearings. The ASLB consisted of three members: two with technical backgrounds and one skilled in the conduct of hearings. A small overlap of ACRS and ASLB functions may have resulted, but the primary functions of the ASLB were to adjudicate disagreements between parties concerning the licensing action and to provide a public forum for discussing the adequacy of the safety review. The ASLB was not expected to conduct an independent review which duplicated that of the AEC Regulatory Staff or the ACRS, although an occasional test for comprehensiveness was considered within the ASLB review scope. An ACRS report was required before safety related aspects of the ASLB review could begin, but the ACRS report was not a formal part of the record, and the ACRS did not present testimony to the hearing board. The hearing boards relied on the AEC Staff for an interpretation of the ACRS recommendations and relied on the testimony of the Staff, the applicant, and the intervenors as the principal basis for judgment.

In the early 1970s, the regulatory organization was extensively revised by the AEC. NEPA, as a consequence of the Calvert Cliffs decision, required more attention to environmental issues extraneous to the nuclear safety evaluation process. At the same time, the AEC Regulatory Staff was substantially expanded and its capability enhanced in response to public concern for the adequacy of some nuclear power plant safety features. This was the situation at the time of the split of the AEC into the NRC and the Energy Research and Development Administration (ERDA) under the Energy Reorganization Act of 1974.

The creation of the NRC did not materially change nuclear power plant licensing, but the new Commission did provide a different perspective on regulatory management. The Regulatory Staff began to act more autonomously with regard to the ACRS. While it continued to review each case and to provide broad safety guidance, the ACRS now began to function primarily as a sounding board where the staff judgments could be tested and tuned, with the Staff accepting ACRS recommendations selectively.

The NRC has now become an independent government unit judging nuclear regulatory matters by a set of rules that it has generated internally. When so disposed, the NRC Staff responds to ACRS recommendations. When it deems such action inappropriate, it will defer the action or set it aside by making a brief record of such action in the NRC Safety Evaluation Report. The ASLBs have become the principal judges in determining whether NRC regulatory actions are in accord with NEPA and the Atomic Energy Act. It is with this style of operation in mind that the organization of the NRC must be examined.

4. REGULATORY ORGANIZATION

When the NRC was created by the Energy Reorganization Act of 1974, a large part of the regulatory organization was already in existence. The Reorganization Act created the Commission consisting of five members, and assigned to it the regulatory responsibilities of the AEC. The form of the regulatory process was already established as a combination of safety regulation and a review to determine compliance with NEPA requirements. The regulatory process was expected to continue under the guidance of a regulatory commission unfettered by previous commitments to the development of atomic energy. Nevertheless, a new administrative operation had to be established, the offices created by the Reorganization Act had to be staffed, and the regulatory functions had to be apportioned among these offices. The regulatory documents also had to be reviewed, gaps filled, and plans for extension of the document preparation program had to be developed to provide an adequate documentary basis for regulation. In a number of areas, notably waste management and material safeguards, there was no regulatory precedent of substance and a new regulatory program had to be created. The development of an effective regulatory organization is one of the major goals of the NRC, and this effort is still in progress. A review of its present status will indicate where further development is needed.

4.1 Regulatory Documents

The NRC adopted the regulations developed by the AEC as the basis on which nuclear power plant licensing would be processed. The basic regulations were in existence and identified in the Code of Federal Regulations. They had been extended by other internally developed documents prepared by the Staff when it was still a part of the AEC. The basic documents consist of:

1. rules established as a basis for regulation and published in the Code of Federal Regulations, providing policy and technical guidance for licensing purposes,

2. Regulatory Guides which describe methods acceptable to the NRC Staff for implementing specific parts of the Commission's regulations, and

3. a Standard Review Plan which sets forth internal review procedures followed by the NRC Staff in evaluating documents and other information submitted for licensing review.

These constitute an extensive set of requirements and practices, many of which are used throughout the world. They are further expanded by various technical documents prepared by the NRC technical Staff, government laboratories, NRC approved industrial reports, and well known national standards (4.1). Some of these documents, particularly some regulatory guides, are excessively prescriptive, while some other types of documents tend to identify objectives without establishing a basis for determining conformance with the requirements.

Even though there is a need for changes, improvements, and additions in many portions of the documentation, on the whole the present documentary base is substantial and has provided an effective regulatory tool. The preparation of new regulatory documents would benefit from a thorough review of precise needs and intentions and an analysis of the existing information to establish where serious gaps exist and where upgrading of the quality of information in the documents would be beneficial to the regulatory program.

4.2 The Nuclear Regulatory Commission

The five members of the NRC are appointed by the President of the United States with the concurrence of the Senate. The Commissioners are appointed for terms of five years and not more than three may be members of the same political party. The NRC Chairman is selected by the President.

The Commissioners must approve the NRC rules published in the Federal Register and all mandatory requirements of the Commission. They review and approve the budget and manpower levels submitted to the President and the Congress, and may review the decisions of the ASLB and the Atomic Safety and Licensing Appeal Panel (ASLAP) on their own initiative or because of appeal from within or outside the Commission. They select and appoint the heads of the five independent offices and the Executive Director for Operations as well as members of the ACRS, ASLB Panel, and ASLAP. They may direct the regulatory staff to proceed along specified lines to satisfy

(4.1) Section III, "Nuclear Components," of the ASME Boiler and Pressure Vessel Code is the best known standard applied to nuclear plants but most of the professional engineering societies have contributed useful standards through the American National Standards Institute, Inc. These professional societies include the Institutes of Electrical and Electronics Engineers, American Concrete Institute, American Society of Civil Engineers, American Society for Testing Materials, American Nuclear Society, the American Society for Nondestructive Testing, and the American Society of Mechanical Engineers.

regulatory objectives. For the most part, the Commissioners have avoided direct involvement in the regulatory decision process to assure their independence when called upon to review regulatory decisions.

Because of their professional backgrounds, political allegiance, and individual attitudes, the Commissioners can have widely divergent views concerning nuclear power plant regulation. They do, however, act as a collegial body operating on a majority rule basis. The individual regulatory offices often have to work out plans for implementing their duties with the intent of obtaining continuing support for their activities from a Commission majority. The Congress evidently intended the regulatory process to function under this democratic style of control, but this approach does not always lead to the development of a clear regulatory position on important public safety matters.

Many styles of operation could be envisioned for the Commission, but so far it has chosen to function as a referee in determining whether the regulated industry was conforming to the rules set by the Commission, and to enter the adjudicatory process only when regulatory actions were challenged. This choice left the regulatory functions to the NRC Staff and the initial judgments concerning the appropriateness of regulatory licensing actions to the ASLBs.

Conceivably, the Commission could become the determining body in licensing actions, accepting opinions from the ASLB, the NRC Staff, the ACRS, or other sources as part of the bases for its judgments. While the licensing rules would still have to be considered, other judgmental factors might be introduced into the licensing process. In its determinations, the Commission might be responsive to public attitudes existing on local, regional, and national levels. Alternatively, the Commission could leave the judgments related to technical safety matters to the regulatory Staff and direct its attention to the requirements of NEPA.

Administration of the licensing process and enforcement of licensing rules would require a different type of involvement. Actions involving inspection, technical review, and conformance reporting would have to be delegated to subordinates who would need authority to enforce the regulations. An administrative executive would be essential to provide a point of authority. If they were adequately equipped by training and experience, the Commissioners could evaluate whether specific regulatory functions were being performed appropriately. The present Commission has a broad distribution of capability, ranging from training in law to nuclear physics, but the individual background of each Commissioner is different, raising the question of whether each opinion deserves equal weight in other than broad policy matters. In-depth knowledge of the subject matter by each Commissioner should be required for equal weighing of their opinions on technical matters beyond policy judgments.

As conceived under the Energy Reorganization Act of 1974, the Commission is intended to be responsive to public attitudes as influenced by the prevailing political environment. If the law were changed to emphasize technological background as a requirement for Commission appointment, then the qualifications of the Commissioners might justify more intimate involvement in licensing decision making with respect to rules, inspection, enforcement, and technical specifications. If the law were changed to put the primary emphasis on health and environmental impacts, the Commissioners could become more intimately involved in the NEPA matters. If the law required that they have legal training, the Commissioners could have more intimate involvement in legal interpretation of the regulations and could judge directly how the regulation satisfies the requirements of the Energy Reorganization Act. Since none of these is presently a dominant requirement and the collective background of the Commissioners encompasses all of them, a policy-making role for the Commission seems to be appropriate.

There could be some advantage gained by designating one Commissioner as the executive officer of the Commission. Alternatively, there could be some advantage in assigning individual areas of decision authority to each Commissioner in addition to his overall policy-making role. Another option, which would be consistent with the present structure of the Commission, could give appreciable technical management power to the Executive Director for Operations who could also serve as the spokesman for the Commission. This position would then require considerable technical skill in addition to management experience, and his relationship with the Commission would have to be carefully defined. These options should be considered as alternatives, depending on public needs and interests, if the present Commission form of regulation is to be retained.

4.3 Atomic Safety and Licensing Boards

Each three-member ASLB is drawn from a panel of board members preselected by the Commissioners. These members have a range of capabilities, and all have a reputation for significant professional accomplishment. They are expected to have understanding of the hearing process and technical knowledge of the regulatory approach and the requirements of NEPA. They are expected to make technical judgments and to evaluate the evidence available to assess whether the regulatory process conforms to the requirements set forth in the law.

Board decisions may be appealed to an ASLAP if the license applicant, the regulatory Staff or the intervening groups challenge the ASLB rulings. The ASLB hearings are adversary in nature, with matters argued before the boards in a quasi-legal format, and the decisions of the boards are recorded and used as precedents in subsequent hearings. The legal staff of the NRC is, to a major extent, occupied with the preparation of cases to be presented before the hearing boards. Members of the regulatory Staff develop their safety reviews in a form suitable for use in this quasi-legal environment.

4.4 Regulatory Operating Functions

The NRC Staff, under its Executive Director for Operations, is divided into five statutorily established and equally ranked offices: Regulation, Inspection and Enforcement, Nuclear Materials Safety and Safeguards, Standards, and Research. In addition, the Office of the Executive Legal Director establishes and implements legal procedures. Each statutory office has explicit duties in response to the organizational plan set forth in the Energy Reorganization Act of 1974. The NRC has established documented rules and regulations under which its operational staff functions. The discussions which follow are intended to show how the organization currently works and where redirection might be of value.

4.4.1 Office of Nuclear Reactor Regulation

The Office of Nuclear Reactor Regulation (NRR) is the focal point for defining licensing requirements. Licenses are granted when the NRR has determined that the necessary documentation has been submitted, that the plant is to be designed or to be operated in accord with established rules and regulations, and that the licensee has shown the required competence to meet the regulatory requirements (4.2).

The NRR staff includes personnel with backgrounds in many aspects of nuclear technology, including such topics as nuclear physics, radiation protection, chemistry, fluid mechanics, thermal analysis, structural design, seismology, hydrology, mechanical engineering, chemical engineering, and electrical and instrument engineering. To evaluate NEPA requirements, some economics and social science skills are also provided. To evaluate a license application, they use NRC Regulations, Standards, Standard Review Plans, preapproved submittals of vendors, recognized engineering practice, and comparable information as bases for judgment. The NRR Staff reviews for compliance with both NEPA and NRC requirements. Prior to granting an Operating License (OL), the NRR Staff requires that the licensee provide a set of proposed technical specifications to which he will conform when operating the plant. Technical Specifications approved by the NRC Staff are incorporated in the license as requirements.

(4.2) The documentary evidence of regulatory compliance is usually covered by a Final Safety Analysis Report (FSAR), a set of technical specifications, a preoperational test program, and a qualification program for operating personnel. This is required for an operating license, which must be granted before a licensee can load nuclear fuel. A construction license is granted prior to plant construction and is based upon a Preliminary Safety Analysis Report (PSAR) to show that the design and construction will comply with regulations.

Since the Staff that performs reviews cannot be large enough to examine every detail of every design, the NRR Staff to a large degree relates each new license to some previously approved plant and focuses its attention on the differences. Some standardization has naturally evolved from this process. The NRR Staff tries to concentrate on what is new in the license application and to accept without reexamination features which have been previously accepted. When new information, operating experience, or regulatory prudence indicate the need, the NRR Staff will reexamine an area that has been previously reviewed, even if previously accepted practice is being followed.

The technical strength of the NRR Staff is critically important. The Staff must have a good understanding of the basis for licensing, the subtleties of engineering variations between plant designs, and must recognize operating circumstances that may challenge the safety feature performance of a plant. The Staff reinforces its own skill with expert consultants and technical assistance contracts. Where necessary, it draws upon the Office of Nuclear Regulatory Research to develop new or supportive information to aid in licensing evaluation. Over the years, this mode of operation has built a very extensive store of knowledge on which the NRC Staff can draw. However, the extremely broad range of characteristics and performance which may have important consequences and the complicated interrelationships between them, invite concern for the ability of the NRC to cover the entire range of technology. Staff attention to conformance with regulatory logic, and the ability of the NRC Staff to relate its regulatory requirements to proper construction of the plant and to its control by the human operators under circumstances that might lead to accidents are paramount considerations.

4.4.2 Office of Inspection and Enforcement

The Office of Inspection and Enforcement (I&E) is the regulatory control arm of the NRC. It investigates licensed installations for conformance with regulations. It establishes whether licensees and their agents are conforming to licensing requirements. The I&E organization uses the rules published in the Code of Federal Regulations, NRC Regulatory Guides, and Technical Specifications as bases for regulatory enforcement. The capabilities of the I&E staff were for many years concentrated on assuring that construction practices, such as material control, welding, equipment storage, and pressure testing, conformed to regulatory requirements. Experience had shown this to be the main source of nonconformance. Attention in the public press to reports of poor workmanship and worker malefaction intensified this interest. There was always, however, general attention to other areas of regulatory compliance.

The I&E staff uses a system of audits to examine both plant records and physical installations. Members of the staff visit supplier factories periodically to establish qualifications and obtain written reports from the licensees to determine compliance with regulations. More recently, the NRC has added a staff of in-plant construction and operation inspectors. Primarily, however, the I&E organization relies on a set of "quality assurance" practices established by the owner in compliance with NRC regulations to assure that installed quality meets the regulations. Preoperational test programs are used to verify the needed operational capabilities wherever practical. The I&E Staff monitors these programs. Oftentimes, the tests require engineering analysis. Analytical methods and the operational results are both usually channeled to the NRR Staff for technical review.

With the existing type of capability in the I&E organization, regulatory evaluation of operational adequacy is information oriented. Operational procedures are reviewed by the I&E Staff, but the intent is mainly to show that procedures conform to technical specification requirements. The actual efficacy of the procedures is left to the judgment of the licensees. The I&E Staff has developed an outline of study to be employed in the licensees' training program to assure operator competence. A group of training examiners, by observation and testing, determine the competence of operators.

To review operational matters not identifiable in procedures would require a level of technical understanding available only in those who have a background in design logic and system performance. The NRR Office evaluates this broad subject matter as a basis for licensing approval, but the I&E Office uses the information in a condensed form suited only to the information checking actions it must perform. With additional emphasis now being directed to simulator training, fundamental system behavior, symptomatic analysis of instrumentation signals, and similar matters, the current style of review of operational matters by the I&E organization will need alterations in order to allow a new technical role in the licensing process for I&E. When asked, the NRR Office through its Division of Operating Reactors works in support of I&E to provide broader expertise on an as-needed basis. While the present arrangement could work in principle, an improvement in the I&E organization's ability to address unusual technological matters through reorganization, training, staff additions, or by other approaches seems to be required (4.3).

4.4.3 Office of Standards Development

The Office of Standards Development (SD) develops the regulatory documents which form the basis for regulations. All radiation exposure standards, regulatory guides, and many of the rules published in the Code of Federal Regulations emanate from this office. This office is primarily a coordinator

(4.3) The loose coupling of these capabilities seen in the TMI-2 experience does not serve the regulatory function adequately. Too much time elapsed between the identification of difficulty and the effective use of the NRR expertise. Recently, there has been discussion of setting up a technical review function separate from both I&E and NRR to provide service to both.

of information and acts as the Secretariat for the NRC Staff in the preparation of material for use in the regulatory process.

The SD has created a substantial body of documents which define acceptable engineering practice. These have been most effective when addressing design, construction, and installation types of subject matter. The standards associated with operating procedures, instrumentation, emergency response, radionuclide cleanup, and comparable matters have tended, with a few exceptions, to be general in form and oriented to performance goals rather than to explicit requirements. Such standards serve a useful purpose in directing the interested organizations to the proper objectives, but they do not provide the type of regulatory definition needed as a basis for rule enforcement. Technical specifications provided by licensees and approved by the NRC Staff are the main regulatory controls.

While the present organization of SD adequately serves its assigned purposes, this office should also have additional capability in the operational areas in order to provide more effective documents for I&E purposes. Some additional skills relevant to operational procedures in emergencies are an urgent need (4.4).

4.4.4 Office of Nuclear Materials Safety and Safeguards

The Office of Nuclear Materials Safety and Safeguards (NMSS) is primarily concerned with the nuclear fuel cycle external to the power plant. It is responsible for public safety regulation with respect to accountability of fissionable materials, safety of fuel manufacturing and reprocessing, spent fuel storage, and waste management and security provisions of all licensed facilities. Problems of material diversion and industrial sabotage are also under its jurisdiction.

The NMSS office has concentrated its interest on material accountability, protection and industrial security. Its rules and regulations, except for material accountability, have a base of practice that developed during the AEC era and at least until recently very little has been done to realign this base in accord with current public interests. Not until the last few years has the NMSS Office organized itself to direct the NRC's waste management regulatory program in an effective manner. Previously it appeared to have adopted a reactive style of regulation directed toward correction of problems exposed in the public press and to providing inputs to DOE and the Environmental Protection Agency (EPA), both of whom are attempting to establish a national posture in this area.

(4.4) Thus far, operating standards have consisted mainly of test procedures and listings of required tests. Standards for measuring capabilities of operating organizations in meaningful terms seem lacking.

The NRC's jurisdictional responsibility in waste management is sufficiently vague to make the regulatory program difficult to implement, but the matter of nuclear power safety cannot be divorced from either nuclear waste management or spent fuel handling. The nature of the problem suggests that the NRC needs to expedite its own regulatory approach to these matters rather than waiting for other agencies to offer solutions. Since certain aspects of the assignment of federal responsibility are vague (4.5), new legislation may be needed to enable the NRC to accomplish these tasks.

4.4.5 Office of Nuclear Regulatory Research

The confirmation of safety bases used in the regulatory process has always been a fundamental requirement for ensuring the health and safety of the public. The safety research programs, first initiated under the direction of the AEC, have been continued at a substantial level under the direction of the Office of Nuclear Regulatory Research. This office acts as a research manager by contracting the research work to national laboratories, universities, private contractors, including nuclear industry organizations and other sources. Probabilistic analysis methodology also comes under this office. The major part of the research program funding is assigned to operation of the emergency core cooling (ECC) and fuel-failure-mechanisms experimental facilities. Other important work under this office includes pressure vessel reliability, core melt behavior, advanced reactor safety, steam generator degradation phenomena, and a number of miscellaneous studies. The need for research to improve safety has recently been recognized, but so far it has been funded at a minimal funding level.

The effectiveness of the Office of Nuclear Regulatory Research has to be considered in relation to its preestablished obligations. The prior commitments to ECC system investigations and fuel failure experiments leave little latitude for other types of safety research within the funding limits. The "confirmatory" approach which the Office of Nuclear Regulatory Research is expected to follow, allows very limited opportunity for new safety initiatives. While the work underway is well managed in an administrative sense, its contribution to overall reactor safety is mainly through enhancing confidence in current practice rather than by providing strong technical innovation.

(4.5) EPA has been designated to set environmental standards for radionuclide releases and DOE is assigned the responsibility for establishing waste isolation techniques. Until DOE has a definitive technology that is consistent with EPA environmental standards, NRC cannot establish meaningful regulations.

4.5 Advisory Committee on Reactor Safeguards

The 15-member ACRS, appointed by the NRC under the requirements of the law, reports on the public safety adequacy of specific licensing actions. The Committee reports directly to the NRC, and its budget and staff support are provided as an item within the overall NRC authorization and appropriation. The Committee is careful to assure that its membership is free of financial influence that might affect its regulatory review and also that it is free of all NRC Staff involvement.

During the early nuclear power era, the ACRS established safety criteria on an ad hoc basis as questions arose during licensing reviews. It was during this period that containment requirements were established, design practices developed, design basis accidents (DBAs) identified, and the engineering methodology for accident evaluation was established. The ACRS became the principal body for identifying supportive research and development work to establish safety adequacy of nuclear power plant design, although the sources of information on which such recommendations were based often came from the national laboratories and the nuclear industry. Such important experimental investigations as the nuclear shutdown characteristics of water-cooled reactors under reactivity excursions, pressure vessel integrity, BWR pressure suppression containment characteristics, nuclear fuel failure properties, and ECC system performance grew out of ACRS reviews. The ACRS was the principal motivating force in establishing the importance of reliable emergency core cooling and shutdown heat removal capability for large nuclear power installations. Many of these requirements have since been embodied in the NRC Regulations under 10 CFR Part 50, Appendix A, and are generally covered by Standard Review Plans or Regulatory Guides in connection with other reference documents.

The ACRS has, with the support of the Commissioners and the NRC Staff, maintained an active review of NRC rules; Regulatory Guides dealing with design, construction, and operational experience; experimental programs; and analytical studies. In 1977 the Congress asked the ACRS to review the Safety Research Program of the NRC on an annual basis and report its findings to the Congress. Implicit in these assignments is the expectation that the ACRS will provide carefully weighed advice and that it will not passively accept Staff action or inaction that reflects deleteriously on safety recommendations concerning licensing actions.

In the early 1960s, the ACRS began to concentrate its attention on siting guidelines with the intent of looking beyond the literal interpretation of the regulations. Siting near high population centers, behavior of the reactor core under degraded cooling conditions, including potential core melts, seismic design methodology, and instrumentation to follow the course of accidents beyond the design basis were regularly discussed with the NRC Staff. More recently probabilistic analysis methodology for safety assessment has been actively encouraged by the ACRS.

The emphasis on such sophisticated technological questions may have diverted the attention of the ACRS and the NRC Staff from many of the more routine safety-related problems that often precede major accidents. The Committee tends to assume that once it has identified a safety problem, the problem will be investigated in detail by the NRC Staff. An individual member often must be extremely persistent before his colleagues will devote extended attention to his safety concerns. Except for transcripts and minutes of meetings, there is no record of the differences of opinions expressed by Committee members during formulation of a Committee position unless one or more members dissent from the collegial view.

The ACRS has identified many matters needing safety attention because of their accident potential, but it has not devoted serious attention to the effectiveness of operator training or to the behavior of control systems under accident conditions. In calling attention to common-mode failure problems, electrical reliability questions, probabilistic analysis and system interactions studies, the Committee has tended to express its interests in fairly general terms without attempting to determine how those matters would be pursued or what personnel capabilities are needed by the licensees or the NRC Staff to respond to these inquiries. The ACRS could have done more to help the Commission identify NRC Staff weaknesses so that Staff enhancement would have produced more valuable safety analysis results.

The ACRS is often passive in its response to Staff work, thus sanctioning work to proceed in areas in which the Committee does not expect the results to be useful. The Committee could respond more actively in such instances. The ACRS serves on a part-time basis, and most of its members have other duties and responsibilities. To perform its work, the Committee relies on the knowledge and experience of its membership, the assistance of well-qualified consultants, a small supporting staff, and a recently added group of short term "Fellows." Because of the limited time available, the Committee could not effectively review all Staff work. There is a need to determine whether the Committee's attention is being directed to the correct areas. Certainly an independent committee cannot be constrained in its review actions, but the level of detail to which it pursues some matters and the cursory level of attention which it addresses to others does raise some questions. It may be appropriate for the ACRS to undertake a serious review of how its functions could be made more effective, and the Committee would benefit from a thorough introspective examination of the manner in which it performs its role.

During development of the early reactors it was essential that the ACRS review license applications in as much detail as feasible, and the Committee has continued such review in areas where new designs or new technologies have appeared. The ACRS is required under the law to report on each nuclear power plant license. This it does through prereview by subcommittees, followed by full Committee action when the NRC Staff license review has reached an appropriate point. When a large number of license applications were being processed, this represented a major part of the ACRS workload. The Committee has recommended to the Congress that it be given the latitude to review plants on a selective basis in order to improve its effectiveness and minimize the time spent on matters already having acceptable safety precedent. The Committee needs to establish more order in its review functions so that important matters will not be overlooked and the Committee work will provide optimum benefit to public health and safety.

5. NUCLEAR INDUSTRY ORGANIZATION

The nuclear power industry is an outgrowth of the electrical utility industry, and its organizational structure is similar. The suppliers of electricity to the public, using conventional methods of raising capital, procure the funds to design and construct nuclear power plants, to purchase the nuclear power and turbine generator equipment, and to buy the nuclear fuel. In most cases, the electrical utility organization provides the plant operating staff. The organizational structure of the whole electrical utility and supply industry is directed toward a regulated mode of business. The industry must establish a service rate structure for the sale of electrical power to the public before it can arrange financing or proceed through the licensing process. It is therefore important that the industry know the regulatory requirements and be able to translate them into a plant design that can be built and operated in accord with its electrical supply schedule and cost commitments.

The utility organizations make use of many service and supply sources on a purchase contract basis to supplement their own capabilities. In reviewing the regulatory process, it would be unrealistic to evaluate the adequacy of the industry on the premise that each utility has within its own corporate structure the capability to meet all of the requirements of public safety. The collective industry capability must be evaluated.

5.1 Plant Licensing Responsibilities of the Owner

The plant owner is the designated license holder under NRC rules. He has to show both financial and technical competence to meet the licensing obligations. The plant owner, usually either a private electrical utility corporation or a public power organization, is responsible to the NRC for defining the nuclear steam supply system (NSSS) to be licensed, for identifying and showing the adequacy of the site on which it is to be placed, for providing appropriate engineered safety features for the system, for coupling the system to a turbine generator and electrical distribution network, for establishing a fuel supply, for showing compliance with spent fuel and radioactive waste disposal requirements, and for providing a competent organization to design, construct, and operate the plant. Normally, an owner can satisfy only a portion of this capability with his own organization. The remainder is provided through contract agreements with other organizations. Nevertheless, the owner is ultimately held responsible by the NRC for the safety of his plant.

Normally, the plant owner employs his own operating staff, which is qualified in accordance with NRC regulations. The system of operator training includes simulator training under the guidance of the NSSS vendor's technical staff, hands-on operational experience, and direct training programs dealing with the owner's licensed facilities. Operators are effectively trained to respond to events encountered in normal operation and to representative events of unusual nature which can occur in emergencies. However, the training programs should be expanded to include a broadened spectrum of emergency events (5.1). There is a particular need to include unusual events which at the outset are minor in nature, but if not adequately controlled can escalate into major emergencies. The effectiveness of this preparatory program depends upon the dedication of the owner's operating staff and its initial level of skill. Many licensees have benefited from the United States nuclear Navy program by hiring personnel trained in that program. These operators are well versed in the nuclear operational disciplines of the Navy, but their limited technical background makes it difficult to translate their Navy experience directly to commercial nuclear equipment. Enhanced capability is a recognized need.

The plant owner is expected to provide a technical support organization as well as the operating staff for the plant, and these groups are sometimes supported by a centralized technical service group. The technical organization usually prepares operating procedures, establishes technical specifications to assure that the plant is operated in accord with design intent, evaluates malfunctions and failures, maintains an awareness of technical problems in other plants that may influence the operating facility under its technical surveillance (5.2), does trouble shooting, plans shutdowns, and carries out other functions appropriate to the installation. The technical skill of the supporting staff is crucial to successful plant operation. Recent changes in regulatory requirements for operators have been directed toward enhancing in-plant capability of the owner's technical staff. In addition, owner groups are developing plans for operating support centers to enhance existing capabilities. This effort is aimed toward upgrading operating capabilities at licensed power plants to reduce the likelihood of such accidents as the one at TMI-2.

Operating organizations are expected to have internal emergency response capability and to establish a working relationship with governmental organizations designated to handle emergencies extending beyond plant boundaries. Operators are also expected to control within regulatory limits the handling and discharge of radionuclides and other radiation sources. The plant

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(5.1) The simulator training is intended to provide this understanding, but no simulator equipment can cover all operational circumstances. Simulator equipment can be set up to address peculiar operating conditions, and this type of training is now receiving priority attention by licensees.

(5.2) A recent study by ACRS has established that such awareness is not as widespread as desirable in the industry. In many companies there is a need for the owner's technical organization to establish a systematized effort to insure being informed of unusual events in other plants and to determine the applicability of such events to their facilities.

to assure that his operating staff will handle such matters in accord with regulatory restrictions. These specialized operational functions are still being developed (5.3) in many operating organizations.

5.2 Architect-Engineers' Role

Some large utility organizations have sufficient capability to develop a complete plant design once a NSSS has been purchased, but most use outside architect-engineering organizations to prepare a design in accord with the plant owner's wishes. The architect-engineer (A-E) may be brought in to help select the NSSS or after its purchase, but in either case, the A-E will normally design the balance of the plant around the system selected. This effort will include the design for the containment, the fuel storage facilities, waste disposal and effluent systems, offsite power supply systems, electrical distribution and emergency power systems, the foundations, the secondary piping systems, and other related equipment and facilities. The A-E often serves as the plant owner's agent in developing responses to licensing requirements related to plant design but is not normally a party to the licensing commitments. Although not directly licensed, the A-E is treated by the NRC as an integral part of the owner's licensed capability. Hence, many A-E firms have obtained approval from NRC for their engineering practices and have had these approvals extended to cover a number of installations.

The range of A-E work includes design of many highly complex safety features, such as emergency power systems, secondary heat removal systems, and radionuclide effluent cleanup systems. Foundations and other structures designed to accommodate earthquakes, tornadoes, floods, and fluid system rupture are particularly sensitive nuclear safety areas handled by A-Es. Earlier designs were found to have numerous minor design faults that required correction, but the A-Es have learned through experience. The capability of A-E organizations is being strengthened through experience gained by personnel with repeated application of their nuclear plant designs.

Some A-E firms have elected to develop standardized design concepts (5.4) to be preapproved by NRC in order to expedite the licensing process. Even if "custom" designs are used, the practices followed are intended to minimize the amount of new licensing review once a design has been approved for licensing. The desire to minimize licensing review and to use designs that have a well-established cost basis has inhibited design innovation in standardized plants

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(5.3) The TMI-2 experience showed operating weaknesses in these areas in the aftermath of the accident. The NRC had not emphasized the need for such capability sufficiently, but current actions should correct the deficiency.

(5.4) The advantage of standardized design approval is that it precludes further NRC staff review of these systems unless some new safety problem appears.

as well as "non-standardized" or custom designed plants even though such changes might enhance public safety provisions and improve reliability. Even when there are opportunities for substantial cost savings coincident with other advantages, the problems brought about by a delay in licensing because of extensive reviews of the new design features usually discourage design innovation.

The importance of establishing a "licensable design" is emphasized by plant owners to their A-Es. This approach has tended to stabilize the design process so that recent designs have corrected most of the deficiencies observed in earlier submittals and minimized innovations requiring further review. Nevertheless, the scale of the engineering effort for a nuclear plant is so broad that no plant can be completely error free. Normally, the A-Es provide continuing engineering services to the plant owner in evaluating errors in design and construction or in new licensing matters that may arise over the plant lifetime. When errors are exposed, a design review may show that the conservatism incorporated in the design will accommodate the errors safely. However, this error tolerance has not immunized nuclear plants from difficulties introduced by design mistakes. On occasion, misapplication of recognized design practice has resulted in serious engineering flaws, as for example improper summing of directional forces from earthquakes. The architect-engineering organizations are expected to maintain quality assurance systems to provide satisfactory design quality, but there is still room for considerable improvement in the design quality assurance practices in nuclear installations. Attention is needed to proper use of design methodology and to assurance that equipment is fabricated and installed in accord with design intent.

5.3 Nuclear Steam Supply System Vendors' Role

In a business sense, the NSSS vendor is an equipment supplier selling a system to be installed as part of a licensed power plant. As a practical matter, the NSSS vendors have, through licensing negotiations with the NRC when each system was initially submitted, established a licensing basis that is used repetitively in subsequent applications. NSSS vendors have offered explicit standardized designs for licensing under the NRC standardization program, but these are normally variations of previously licensed installations where some of the "standardization" had already been established. The NSSS vendors' obligation to the plant owner is to furnish a licensable system, and usually his contractual agreement includes handling, as the plant owner's technical representative, the NSSS licensing negotiations with the NRC. This has often created confusion concerning placement of the licensing responsibility. In most cases, the NSSS vendors' licensing obligations are limited to those he accepts as a contractor of the plant owner.

In spite of this limited responsibility, the NSSS vendors have most of the nuclear system safety expertise associated with licensing the equipment they supply. The plant owner relies heavily on this capability for advice and training and anticipates its availability for the life of the plant. To insure public safety, the NSSS vendor organization must be maintained at a high level of technical etence as a backup to the plant owner organization since the owner may ot have adequate capability to respond to emergencies on his own. The NSSS v dors, as shown by TMI-2 experience, do not study every safety aspect of their systems because they consider some matters outside the bounds of he licensing requirements. Yet, their involvement in prompt resolution safety questions which do arise is mandatory.

5.4 Nuclear Fuel Suppliers' Role

Normally, the NSSS vendor provides the first loading of fuel for a reactor core and may also contract to provide subsequent fuel loadings. The plant owner may elect to obtain reactor fuel independently of the NSSS vendor. In any case, the nuclear fuel supplier must show that the fuel he will supply is compatible with the reactor system in which it will be used. This requires both experimental and analytical evaluation of the fuel. The NRC has now developed a set of analytical procedures to be followed to show that the fuel is acceptable. The supplier is also required to show that his manufacturing processes will produce the needed fuel quality. The plant owner then accepts responsibility for the fuel as a purchased item to be used in the nuclear plant. The NRC limits its relations with the nuclear fuel supplier to accountability, performance verification, and manufacturing control questions relevant to the regulatory process.

5.5 Special Nuclear Support Services

Such matters as in-service inspection, pressure system evaluation, radioactive effluent disposal, fuel management strategies, and similar matters are often handled through service contracts to outside organizations. The plant owner usually contracts for such services on an ad hoc basis when they are needed. They are important operational elements of the plant owner's licensed capability. The qualifications of such specialty organizations are generally not determined by formal procedures; but with rare exception, those performing the services have established a high level of expertise through long participation in nuclear power industry activities.

5.6 The Nuclear Plant Constructors' Role

The nuclear power plant owner will sometimes act as the constructor of the plant by purchasing all materials, subcontracting for conventional building and erection services, and hiring his own labor force to perform installation work, including piping, electrical distribution, special service systems, and other work normally outside the scope of his contracts. Alternatively, the plant owner may elect to contract for a turn-key installation. There may also be intermediate arrangements between these two extremes. The owner sometimes acts as his own construction manager, and at other times he may hire an outside service organization, such as an architect-engineering firm, to perform that service. The owner is expected to have a quality assurance organization to establish that the work is being performed in accord with nuclear regulatory requirements. The owner will require that each portion of the constructor-installer organization have a related quality assurance organization to meet regulatory requirements. The owner's quality assurance responsibility also covers the adequacy of the quality assurance procedures of the A-E, the NSSS vendor, and the equipment and materials vendors to insure adequate design, engineering, and testing. There will normally be an understanding between the owner and the constructor-installer as to what will be provided to the operating organization. In any case, this entire construction program is required to conform to the drawings and specifications prepared by the A-E, the NSSS vendor, or other engineering organizations that have participated in developing the licensed plant design.

Although much emphasis is placed on establishing qualification standards for craft skills, there is always some residual concern as to whether the quality of the workmanship will meet anticipated regulatory standards and whether the work will be done in accordance with the requirements stipulated by the drawings and specifications. Many construction faults have been reported over the 25-year nuclear power plant history, and in spite of the quality assurance requirements, there is still evidence that some organizations do not exercise adequate control over the construction work. The NRC Office of Inspection and Enforcement can identify such matters early in the construction program, but the regulatory action is often of such limited forcefulness that constructors fail to respond adequately. The need for high-quality construction must be further emphasized in the regulatory program.

5.7 Assessment of Collective Industry Capability

The licensing of nuclear installations obviously requires consideration of all of the industrial elements upon which the owner-licensee depends. The industrial system limits the liability of the industrial participants to those established by the owner through contracting.

Many A-E organizations do not have independent self-audit procedures to check drawings adequately to insure that they reach the field with a minimum of errors. They rely excessively on construction forces or test personnel to expose and correct errors in the field. While more systematic than the A-Es in their design and manufacturing controls, the NSSS organizations would also benefit from improved review processes. The interfacial relations between nuclear steam supply systems and the balance of plant systems are especially deserving of attention.

The quality assurance system on which the NRC depends to assure adequate quality in the licensed installation needs to be strengthened in the areas of design methodology and installation conformance with design intent. The opportunities for engineering blunders affecting public safety are too numerous to allow this matter to continue in its present management style.

Under the present arrangement, the regulatory process needs to have more control over the licensees' contractors since the owner-licensee cannot assure that he will have access to all of these capabilities if they should be required for public safety reasons. Alternatively, the regulations could require that the owner establish capabilities equivalent to those of his contractors whenever they are important to safety. In particular, the capabilities of the NSSS vendor and the A-Es in system behavior, trouble shooting, and performance analysis could be required to be a part of the owner's capabilities.

6.0 MAJOR TECHNOLOGICAL ISSUES

Limiting damage to the core and restricting the dispersal of radionuclides resulting from accidents in nuclear power plants are primary functions of the "engineered safety features."(6.1) By design, the latter features are required to meet more stringent standards than equipment provided only for power production and special attention is directed to their reliability under accident circumstances. To establish the adequacy of the engineered safety features, "design basis accidents" are postulated which are supposed to bound the accident contingencies having a probability large enough to require consideration. Some engineered safety features are evaluated in relation to features of the site, in particular the size of the site and its distance from the nearest population center. All engineered safety features are designed to function properly in the face of severe natural phenomena, including earthquakes, tornadoes and floods.

6.1 Engineering Methodology for Public Safety Protection

In considering the capability of engineered safety features, each DBA is related to a range of failure contingencies. Some of these are concerned with how failures are initiated, some with how they propagate, and some with the conditions prevailing when failure occurs. Although the TMI-2 accident did not exceed the bounds of the postulated accident conditions with regard to release of radioactive materials, it did lead to core damage greater than that predicted in the analysis of DBAs, and the TMI-2 event has raised renewed interest in how accident bounds should be defined. The objective of the engineered safety features is to control the consequences of failure in such a way that the health and safety of the public are not jeopardized. Unless there is a precise definition of what is meant by "failure," the effectiveness of the regulatory approach cannot be evaluated. Therefore, attention must be directed toward the meaning of failure as it affects public safety.

Among the important nuclear safety technology matters highlighted by the TMI-2 accident is the question of whether there is an effective way to separate the safety related features of the plant from those intended for normal operational use and not considered essential to public safety protection. The assumption of separation of safety from non-safety

(6.1) Engineered safety features are defined as the systems and equipment needed to assure that DBA consequences do not exceed the site radiation exposure limits specified in 10 CFR Part 100. However, many other systems are important for preventing or mitigating accidents.

features has had a profound influence on the manner in which nuclear safety regulations are imposed, and the separation philosophy must be understood and used properly. Of special significance is the interaction between the "safety" and "non-safety" portions of the plants. The accident conditions themselves may cause interaction, or the initiating events can involve unexpected interactions that alter the performance of the engineered safety features. System interaction questions are complicated further by man-machine relationships associated with operator actions in nuclear plants. Many of these issues are amplified further in the following subsections.

6.1.1 Design Basis Accidents and Probabilistic Analysis

Since the early history of nuclear power plant regulation under the AEC, the design basis accident conditions used for the purpose of design of the containment and of the features intended to remove radioactive materials from the containment atmosphere assumed the release of very large amounts of fission products in a containment building whose basic integrity was assumed to be maintained intact. The assumed radionuclide release is derived in part from core melting experiments, but the containment design pressure is based on the assumption that core cooling will be maintained and that no fuel melting will occur (6.2). The containment does not include provisions to cope with a molten core or the heat, hydrogen, and other aspects of an accident in which the whole core melts.

On the other hand, the engineered safety features have been designed to prevent severe core damage for a large number of design basis events including earthquakes, a pipe rupture in the primary system, a ruptured steam line, a loss of offsite power, etc.

(6.2) The TMI-2 incident involved accident conditions very much like the DBA except that containment pressurization did not extend over a long period of time and fuel probably did not melt. Core cooling was disrupted for short but significant periods of time, leading to core damage and gaseous fission product release after the nuclear reaction had been halted. Cladding damage also exposed the bare fuel pellets to the reactor coolant, leaching out some solid radionuclides. The containment did not maintain its leak tightness perfectly, but the type of leakage experienced did not result in damaging radionuclide release to the public environment. The extent of the TMI-2 failure and the manner in which the core cooling system was operated heightened interest in DBA assumptions, but the subject was not new.

In connection with the establishment of design basis events, the regulatory process should take account of both the probability and consequences of the event in order to establish a risk evaluation basis. Much could be learned by examining the possible differences in behavior of existing plants compared to those studied in the "Reactor Safety Study" (WASH-1400) that result from design variations, site conditions, and a host of other variables known to exist because of changes in technology and engineering judgments among plants and systems.

The Reactor Safety Study showed that the probabilities of accidents involving core melting without adequate core cooling were high enough to deserve attention. Since 1966, ACRS had urged the AEC, NRC, and the nuclear industry to look beyond the design basis accident for circumstances that might warrant mitigation by design. More recently, the floating nuclear plant vendor had been required, in response to an environmental impact evaluation, to provide features to reduce the consequences of a core melt.

Design basis events, such as earthquakes, are usually examined in the design of nuclear plants to show that they can occur without resulting in accidents, but these and other events, unless dealt with adequately, could subsequently lead to an accident of greater severity. For example, continuing loss of offsite power without the provision for long-term continuity of the emergency in-plant power supply could eventually interrupt core cooling enough to permit core damage or even core melt. Some of the events such as large double ended pipe breaks have a low probability of occurrence but nevertheless are now dominant considerations in safety evaluations concerning design basis accidents.

Other more likely events might be identified as deserving greater emphasis if probabilistic analysis were used instead of the DBA approach.

The DBA approach to safety analysis has been useful and relatively effective in the analysis of reactor systems. However, the experience gained with its use, the continuing development of probabilistic methods, and experience in power plant behavior that has been accumulated all suggest that the approach should be modified to include increasing use of probabilistic considerations.

Severity of the DBA is one of the crucial technological issues. Should core melt be assumed, and if so, how completely? If not, is the core damage experienced at TMI-2 the appropriate basis for establishing containment leak tightness? Are the previous design bases for containment, which allow for large scale fission product release but not the other phenomena associated with core melt, adequate to protect the public health and safety? The technical basis for the previously used accident assumptions involves a compromise which tries to cope with most accidents. The logic does not always involve totally consistent assumptions (6.3).

A more logical method for establishing severity levels is to use the Reactor Safety Study approach. The method would have to include consideration of both consequence uncertainty and engineering reliability questions involving applications where little experience exists and quantitative safety goals would be needed.

Probabilistic methods are not presently developed to the point where they can be substituted completely for consideration of DBAs in the traditional way, however, and it appears necessary for the immediate future to continue the current policy of specifying arbitrary accidents as a basis for regulation (6.4). The present umbrella of DBAs may need modification, and a study should be made to determine which if any additional accidents should be added to those now considered.

The regulatory process should be able to show the public and the regulated industry how safety requirements are established and to clarify inconsistencies when they appear.

6.1.2 Failure Definition

The primary interest of nuclear safety regulation is to prevent the spread of radionuclides to the external environment, thereby protecting the health and safety of the public. Therefore, the failure mechanisms that might result in a release of radionuclides are the first safety considerations. The failure boundaries in a nuclear power plant have been described as: (1) the fuel cladding, (2) the primary system pressure boundary, and (3) the containment boundary. Each has some independence from the others, but they are not three truly independent barriers. It would have been desirable if the regulatory safety approach could have minimized the interdependence

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(6.3) Self-consistency has been an issue before the ASLAP. The NRC Staff once required a BWR containment to be inerted because of H_2 combustion potential, but the ASLB ruled that the assumed hydrogen generation potential was inconsistent with other assumptions.

(6.4) Arbitrary accident definitions can take several forms. Current practice assumes core melt level fission product releases but perfect containment and core cooling. Other combinations such as partial melting with degraded core cooling could be selected. Containment leakage could be an accident variable.

of these boundaries so that failure of one boundary would not lead to failure of the others. However, this failure approach cannot be fully realized. Under some circumstances, failure of the primary pressure boundary could cause fuel cladding failure, but the reverse is unlikely. Similarly, failure of the primary system could cause failure of the containment under some circumstances.

The NRC has nevertheless placed great reliance on these separate lines of defense and has developed its requirements for engineered safety features consistent with this failure protection concept. The engineered safety features are expected to function independently of the normal plant equipment affecting the primary coolant boundary even when postulated failures of the primary boundary are considered. Failure of the primary system is therefore permissible from a public safety standpoint because the separate lines of failure protection provide defense in depth. However, a definition of acceptable failure involves a number of controversial matters.

One aspect of that definition is establishing failure tolerance. Piping systems, for example, have suffered stress corrosion cracking but the extent of the cracking has never resulted in a loss-of-coolant accident that would actuate the ECC system. One failure concern is whether the cracks could propagate uncontrollably, creating a rupture that excessively challenges the ECC system. Another possible concern is that some severe condition, such as an earthquake, might cause a set of cracks to propagate into a failure of uncontrollable character. Hence, failure can be defined as acceptable only if it is controllable within public safety limits under the transient conditions stipulated for consideration by regulatory practice.

A second aspect of the definition is the influence of the operating environment on the failure. A failure may be initially acceptable under regulatory requirements, but if its control requires the continuing integrity of equipment that cannot survive the operating environment after failure, then it may eventually become uncontrollable. For instance, severe fuel failure which released radionuclides to the primary containment and, within a short time, through excessive heat or ionizing radiation, caused a failure of a containment seal would not have been an adequately controlled failure.

The third aspect is the question of how many failure events must be considered when defining an acceptable failure. The current approach is to use the single failure criterion which assumes that an initial system or equipment failure occurs and then postulates one more equipment failure, usually associated with the mitigation actions intended to control the initial failure consequences. This "single failure criterion," adopted from electrical circuitry design practice, has been used in the NRC regulations as a way of defining acceptable failure, but it is more likely to be applicable only to very simple systems. For complex systems, multiple failures may be experienced subsequent to the initial failure and some other standard of acceptability is needed.

These several aspects of failure are sufficient to illustrate why an understandable definition of acceptable failure is needed to provide a basis for regulatory practice. With a well-founded definition, it would then be possible to show which types of failure would not constitute a cause for public safety concern; which types of failure known to be unacceptable, if allowed to run their course unchecked, could be controlled within acceptable consequences by mitigating features such as physical restraints or backup operational features; and which types of failure are clearly outside the bounds of acceptability, even with mitigating features, unless further failure control provisions such as emergency evacuation are provided.

Much of the safety research program sponsored by the NRC is aimed at establishing the nature of failure and showing that the consequences are acceptable within regulatory limits. However, the tolerance of equipment for failure and the distinction between important and unimportant failure events are not yet adequately defined and more work is needed.

6.1.3 System Interactions

In the prior discussion of failure, reference was made to the interactions between various operating systems and how they might lead to significant failures from a public safety standpoint. As currently used, the term "system interaction" refers to all of those circumstances that could arise where there is a possibility of the events occurring in one system imposing safety related stresses on another system. For example, actuation of a fire water sprinkler system that damaged the electrical controls could invalidate the capability of all engineered safety features.

System interaction questions involve such matters as (1) the relationship between the normal control systems and the so-called protection systems that are presumed to be isolated from each other but could have interactive effects; (2) the release of radionuclides or heat into the operating environment of engineered safety features to degrade their short- or long-term performance and possibly negate their safety function (6.5); and (3) a crossover of a short circuit fault from one circuit to another that could destroy redundant electrical equipment provided for public safety reliability purposes. Most of these matters are given some consideration in the licensing process. The regulations are intended to avoid deleterious system interactions, but recent experience suggests that the whole subject should be under constant surveillance by personnel who have insight into potential system interaction difficulties.

(6.5) The Browns Ferry fire was an illustrative circumstance. The fire destroyed the electrical control circuitry, and it was necessary for the operators to find an alternate power supply for actuating certain valves to depressurize the system in order to establish the core cooling safety function.

6.1.4 Man-Machine Interactions

Nuclear power stations cannot be operated solely by human action or by machine automation. Operators are needed to establish a state of readiness for the plants, to relate them to the external electrical demands and to provide fuel, maintenance, and similar service activities. One way to minimize human mistakes is to automate the plants or to provide better computerized analysis so the likelihood of human thinking errors will be minimized.

None of the older plant designs have sufficient computerized analysis capability to be useful in analyzing most operational symptoms quickly. Some newer designs have improved computerized analysis capability but still provide only a limited set of automated functions such as the emergency power supply systems, reactor safety protection systems, pressure relief, containment isolation valves, and a few basic mechanical equipment functions.

There may be advantages to expanding the automated plant features to reduce the need for operator action during transient operating periods, but how and whether this should be done deserves considerable thought. Most of the more modern plants are providing additional computerized control capability that could by computer initiated control signals ease the knowledge requirements put on the operators, but concern has been expressed about such systems causing undesirable operational actions through computer malfunction. The safety threat from such malfunction offsets to some extent the desirability of computerized response.

There is need to improve the information displays in control rooms. These have been developed along lines that follow customary display practice for non-nuclear steam power stations combined with the now-traditional display scheme for nuclear controls. This display has considerable merit because operating personnel are accustomed to it. But it may not draw operator attention adequately to the crucial instrumentation needed in emergencies. The alarm systems may be excessively confusing and some information displays could be better located (6.6).

Even if information displays are improved, the diagnostic needs for accident control purposes will not be met. In order for either operating personnel or automated controls to respond to instrumentation signals, there must be less ambiguity of interpretation that could lead to erroneous safety ac-

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(6.6) This is not to say that the existing control rooms are unacceptably poor. The experience at TMI-2, although justifiably drawing criticism for the quality of the instrument displays, did not show that operators were unable to identify operating conditions or to determine whether control equipment was functioning. Some valve closures and the condition of the steam relief quench tanks were not adequately displayed but minor design changes could correct these problems. The real concern is whether the diagnostic burden on operating personnel is excessive.

tions of the sort that occurred at TMI-2. Attention will have to be concentrated on integrating information from diverse sensors and combining the information in such a way that the accident symptoms lead the operators to initiate correct safety control actions. Symptom correlation with instrument signals to direct operator action to the appropriate safety procedures could eliminate much of the concern about man-machine interfacial response. Not enough attention has been addressed to this matter.

In addition to information needed for diagnostic purposes, operating personnel must have some emergency instrumentation provisions to maintain cognizance of accidents that do not proceed along anticipated lines. An example is instruments that show whether fuel has failed and what type of failure may have occurred. Without such provisions, the operating personnel are less able to correct unforeseen events that may have been overlooked during accident analysis even though the corrective action might be easily performed.

6.1.5 Separation of Safety from Non-Safety Systems

The NRC regulations are generally founded on the idea that if the systems important to safety are reviewed carefully and the plants are properly constructed with suitable features taking into account the plant site, then the public will be protected adequately. The NRC review practice has been one which separates safety from non-safety systems, with primary attention given to the safety systems.

The initial intent of the separation philosophy was probably to avoid conflict between demands from normal operating modes and those peculiar to safety functions. As the scope of reactor licensing broadened, the separation philosophy permeated the design process but not with consistent logic. One typical example is the removal of decay heat. In what is perceived as an "emergency," the ECC system is classified as a system important to safety and receives commensurate treatment and attention. On the other hand, those aspects of decay heat removal associated strictly with normal shutdown, a much more frequent need, do not receive the same emphasis.

Thus, this separation philosophy has resulted in the creation of two systems which are treated differently in the safety reviews. The safety system is scrutinized carefully, but the non-safety system may be totally ignored in the review process. Important safety matters could be excluded from review if improperly classified. In some cases, the concept of separation results in overdependence on a specialized safety provision whose safety capability would be better realized if considered as a part of the whole operating plant. Feedwater systems to steam generators cannot for example, be uniquely separated into safety and non-safety categories (6.7).

(6.7) The TMI-2 Auxiliary Feedwater Systems obviously had safety related functions that had to be integrated with normal feedwater supply capability.

As now applied, the philosophy is also used to distinguish between safety related and non-safety related functions with respect to their quality and reliability. An advantage of a properly implemented separation philosophy is that safety related systems requiring very high reliabilty can be designed specifically to meet their requirements without imposing these requirements on those non-safety related features which require less rigorous design. A disadvantage of the separation philosophy is that it cannot be implemented perfectly and is therefore sometimes arbitrary and artificial. For example, a control system and a shutdown protection system could be considered an integrated control system because they are interactive (6.8).

The separation of safety from non-safety functions is necessary when the functions have contradictory requirements. It is desirable in some cases to make them independent to prevent the circumstances which interfere with normal functions from also destroying the safety protection function. For example, an operating electrical power system might be damaged by a lightning strike, and if the emergency power system were tightly coupled, it also might be damaged by the same lightning. This type of separation has been encouraged in the regulatory process, and in some parts of the world, deliberate "bunkering" of some engineered safety features has been introduced to assure the integrity of the safety function. In recent years, concern has been expressed about the use of engineered safety features to perform other normal plant functions although such optional use could be desirable since, under some circumstances, such arrangements might enhance the reliability of the safety features by providing a means for monitoring their operability. Care still needs to be taken to assure that the non-safety functions cannot interfere with the capability of the engineered safety features at the time of need.

Because it is impractical to impose all of the safety stringencies on every plant detail, the separation concept must be used. A few very important features with extremely high public safety protection value will need special quality, redundancy, and testability properties that cannot be extended to every plant element. The extent of this type of treatment may need to be greater than has been provided in the past. Alternatively, new design approaches could be developed wherein the safety treatment placed less dependence on such safety related features. Higher reliability may be attained in some cases if the separation concept is discarded so that the entire system can be considered as responding to the safety requirements. Credit for the capability of features previously considered outside the public safety

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(6.8) Detailed consideration of anticipated transients without scram (ATWS) showed that current power reactor designs routinely depend on "scram" protection for shutdown systems in certain "anticipated transients" to provide needed corrective actions to prevent overpower. Thus, the "shutdown system" is made a part of the control system. Nevertheless, Appendix A, Criterion 24, of 10 CFR Part 50 requires that control systems be separated from protection systems.

provisions may also be justifiable. Indeed, the review process itself cannot be permitted to follow arbitrary lines of separation between safety and non-safety features since this could easily result in overlooking important system interactions or malfunctions that have public safety importance. The whole principle of safety separation needs to be redefined with the intent of developing a more logical and more effective result.

6.2 Siting Aspects of Public Safety Regulations

An established precept of nuclear safety practice is to seek sites with acceptable public safety characteristics including remoteness from population centers. The NRC Reactor Site Criteria, 10 CFR Part 100, use the site properties as a reference basis and require the engineered safety features to be designed to limit the release of radioactive materials to acceptable limits under postulated accident conditions. However, the containment is not designed to cope with core melt, and the use of currently employed engineered safety features to permit reactor siting in more populated areas has been questioned. Certain types of accidents could create conditions beyond the engineered capability of such features. It is therefore necessary to reevaluate the criteria for siting, including the accident conditions under which site safety is judged, when establishing regulatory requirements.

6.2.1 Siting Criteria

Under early safety practices, the criteria for nuclear power plant siting revolved around the definition of power plant exclusion areas, low population zones, and the dependence to be placed on engineered safety features to assure the health and safety of the public in the event of accidents. At one time during the most active period of power plant licensing, use of engineered safety features to mitigate accident contingencies was a major consideration in determining how close to population centers a power plant could be sited. More recently, there has been a tendency to discount this dependence on engineered safety features. Nevertheless, containment leak tightness is still a determining factor in establishing the rate and quantity of radionuclides that could escape to the environment if an accident were to release large quantities of radionuclides from the core. The direction of the dispersal, the dilution of gaseous radionuclides, and the settling-out of particulates are determined by analyzing site-related meteorology.

The TMI-2 accident resulted in conditions well below 10 CFR Part 100 limits even though radionuclide releases into the containment were close to design basis assumptions and the containment leak tightness was not equivalent to that assumed by design. There were compensatory factors since the opening from the containment allowed some radionuclides to escape, but only through a route which included an array of piping, tanks, and filters where water, steam, and surface contact could capture some of the releases. Thus, factors in addition to the usual engineered safety features associated with containment were beneficial to public safety protection.

Not all accidents involving design basis radionuclide release rates would have the benefit of these mitigating factors if containment integrity were to be lost. For example, at TMI-2, the hydrogen generated from the zirconium-water reaction evidently resulted in combustion within the containment that caused pressures higher than those provided by design in some low-pressure containments associated with other commercial plants.

The Reactor Safety Study showed that the likelihood of a core melt was high enough to deserve consideration in reactor siting. The study also indicated that the hydrological path for radionuclide dispersal was generally long enough to eliminate it as a short-term threat to the public in the event of a melt-through accident. However, more attention should be directed to the ultimate consequences of such events. Siting criteria should be aimed toward establishing sites best able to accommodate core melting contingencies over the long term. In particular, the hydrological considerations involving potable water systems should not be ignored. Practical methods for protecting such systems from radionuclide contamination should be available for all nuclear power plant sites.

These siting matters have been considered by the AEC and the NRC for many years, but the circumstances surrounding the TMI-2 accident have placed new emphasis on them. The initial public safety protection considered for nuclear reactor systems was primarily the selection of sites remote from highly populated regions, and this remains a valuable public safety protection feature if other lines of defense are not adequate. Where practical, maximum advantage should be taken of remote siting as a public safety provision.

6.2.2 Multi-Unit Sites

The selection of sites for nuclear power stations and related facilities has to include consideration of fuel and waste transportation, electrical power supply distribution, waste heat dispersal, and accident interactions between units, as well as the environmental surroundings, including population distribution. Most nuclear power plant sites involve only one or two nuclear reactor units, but a number of installations have been planned involving several reactors, and others have been discussed that extend the sites to as many as ten 1000-Mwe units. There are advantages in multiple unit sites in concentrating installations where the best siting conditions prevail and, at the same time, establishing a large enough power complex to justify an adequate technological support capability to enhance operating skill. The disadvantage of multiple unit siting is that an accident at one unit could jeopardize all others, and multiply the property risk and vulnerability of the power system from a single accident. There is no clear basis for the selection of one approach over another at this stage in the technological development. Whether large multiple-unit sites would be desirable depends very much on whether an accident at one site of the type that occurred at TMI-2 could be isolated in such a way that the remaining facilities could be operated in a mode acceptable from a public safety standpoint. However, before the latter approach could be accepted, a number of matters would need to be resolved. They include:

1. showing that an accident involving one unit at a site could be isolated in a manner that would eliminate its effect on other units,

2. defining the technological skills needed to make the site acceptable in terms of operational capability, and

3. identifying the physical arrangements of nuclear power plant support facilities, emergency control, transportation resources, and plant orientation to optimize the risk considerations introduced by the multiple unit approach.

Specific site development plans of this type have not been studied adequately. The criteria for acceptability should include not only the capacity to handle a large number of units but also the characteristics that minimize jeopardy of population centers. Further work is needed before a policy for evaluating large multiple unit sites can be established.

6.2.3 Site-Related Safety Improvements

Nuclear power stations have incorporated many features intended primarily to enhance their safety as the result of direct regulatory requirements. These features have included off-gas filtration, automated containment isolation, and hydrogen recombiners for containment. Further improvement in some areas may be desirable. A comprehensive study should be made to define the most urgent needs. The discussion which follows illustrates the types of safety improvement that could be of value.

An important safety contribution would be a system which could remove radioactive materials from the containment atmosphere after an accident so that the remaining gases could be vented to the atmosphere. Specification of the details of such a system and the needed performance reliability would involve research and experimental work. If such a system could be provided, public safety actions after a TMI-2 type of event would be easier.

More versatile and more reliable core cooling capability is another area that might enhance public safety. The experiences at Browns Ferry and TMI-2 both point to the desirability of being able to provide reliable core cooling capability from multiple sources. Diversity of the capability, its independence from accident circumstances, its resistance to deliberate sabotage, and its ability to directly cool the core under a range of circumstances could directly reduce the likelihood of a TMI-2 type of accident as well as other accidents offering the potential for core damage and even fuel melting. Conceptual engineering studies would be valuable in determining how this capability could be realized.

The ACRS has supported the investigation of both of these features as part of the NRC research program to improve reactor safety. Other types of safety improvements might be envisioned. These include different means for primary system pressure relief, changes in materials of construction, techniques for minimizing accumulation of radioactive materials that directly interfere with in-service inspection, and modifications in existing containment concepts. However, more independent initiative is needed by the nuclear industry in identifying safety improvements.

6.2.4 Nuclear Power Plant Waste Management

A problem that had, until the TMI-2 accident, received virtually no attention is the matter of radionuclide cleanup following such an event. Similar problems pertain to the decommissioning processes for nuclear installations. The NRC has, in the past, left these responsibilities to its licensees. As a result, the associated planning and supporting research have been inadequate. This is clearly shown by the inability to handle the large volumes of radioactive gaseous and liquid wastes that were generated by the TMI-2 accident. Neither the industry nor the involved federal agencies nor their advisory groups adequately envisioned or planned for accident situations in which the character and magnitude of the waste management problems would be significantly different from those of routine nuclear power plant operations. The associated consequences included increased personnel exposures, an inability to collect adequate samples to assess the situation, and a delay in restoration activities. The accompanying public opposition to plans for the disposal of the decontaminated-waste fluids, even though these involve risks no greater than those associated with similar wastes resulting from normal operations, has also delayed cleanup of the plant.

The need for usable low-level waste disposal technology that meets established criteria, policies, procedures, and regulations is apparent. Meaningful regulatory action directed toward opening and operating new low-level waste disposal facilities might reduce public concern over this matter.

6.2.5 Emergency Response

Questions concerning nuclear industry capabilities for handling off-site emergency-response problems associated with accident situations have been of interest since the beginning of nuclear power development. Those responsible for the safety of nuclear installations, beginning with the AEC, recognized the need to develop such capabilities, but it was not pursued vigorously, partly because of industry concerns and partly because of a lack of sufficient interest on the part of state and local authorities. As a result, even though the NRC has required licensees to establish emergency plans in cooperation with state and local governments, this planning has been inadequate because the state and local government units have not had either the funds or the personnel to participate on an effective basis. Also contributing to these problems is the fact that, as implied above, the NRC has had no regulatory authority over state and local governments. As a result, the NRC Staff could only indirectly review the radiological emergency plans of such agencies.

In the past the AEC and NRC considered evacuation primarily in terms of the controlled releases of radionuclides which would occur if containment integrity was maintained. Only in recent years has the NRC Staff begun to examine emergency preparedness in terms of more serious accidents where evacuation might be considered at distances of ten or more miles.

With the occurrence of the accident at TMI-2, there has been a substantial alteration in this situation, particularly with respect to the interest of state and local governments in such matters. In addition, several bills now pending before the Congress hold promise of correcting certain aspects of these problems. These actions are necessary to implement needed changes in the regulatory process.

6.2.6 Accident Recovery

The degree of difficulty encountered in restoration of a nuclear power plant which has been subjected to severe accident conditions is dependent in large part upon the forethought given such a probability during the design phase. When a significant amount of radioactive material escapes from the primary coolant system, its confinement within the containment minimizes the immediate jeopardy to the public. However, as the TMI-2 experience has shown, the ultimate recovery from such an accident is impeded greatly if the containment cannot be entered and there is no effective way to remove the radioactive materials.

A thorough study of accident recovery methods is needed to ease the problems associated with handling this type of situation should it recur.

The options include addition of internal decontamination water sprays or comparable cleanup systems, robot type equipment that could be used to reduce the concentration of radioactive material to a level suitable for human access, or possibly secondary types of enclosures intended mainly to limit the spread of radionuclides from unanticipated accidents. Ultimately, even previously molten fuel may need to be removed from the containment and transformed to a more suitable condition for long term isolation. Attention is now being devoted to these problems as they apply to TMI-2, but the question is of sufficient general interest that it should be a part of the longer term contingency planning.

7. REGULATORY MANAGEMENT MATTERS

Public understanding and acceptance of nuclear power as a beneficial source of energy depends to a large measure upon effective regulatory management. In establishing the NRC, the intent of the Congress was to create a regulatory agency which was free from promotional bias. It was believed that such an agency could oversee the safe use of nuclear energy and improve public confidence in the regulatory process. The law implied by its sanctioning of nuclear plant licensing that nuclear power was an acceptable source of energy but that the policies and practices under which it was regulated needed modification.

Any such regulatory process, however, is extremely complex. It has legal, economic, social and political aspects, and it involves very complex technology. The regulatory process must be stable in the eyes of the industry, it must be vigilant in protecting the safety of the public, and it must handle safety questions intelligently, responsively, and expeditiously.

To satisfy these regulatory obligations, the competence and responsibility of those involved in the regulatory process must be shown to be suited to regulatory purposes. If they are then able to develop a format which is understood by all the participants, a suitable regulatory system should result. The effectiveness of the regulatory process should be evident from the regulatory reporting system, the regulatory actions involved in correcting safety problems, and the communications releases through which the regulatory agency provides information to the public. These matters are not all handled satisfactorily in the current regulatory system. Attention is directed to some of the most urgent matters in the following discussion.

7.1 Organizational Issues

As discussed in Chapters 4 and 5, the regulatory organization and the nuclear industry have both structured their organizations for interactive response to regulatory demands. However, the organizational structure is not set forth with such clarity that every need can be identified and shown to be met. The responsibilities of the organizations, their competence, and the manner in which they perform their duties determine whether the organizational structure is adequate. In many cases, as subsequent discussions show, organizational problems exist that need attention.

7.1.1 Staff Competence

Taken as a whole, the professional competence of the NRC Staff is impressive because of its varied talents and the high level of academic training and experience which its members have attained. Nevertheless, each time a significant new safety problem appears, it usually points to a weakness in the regulatory process. This is particularly true with respect to the designation of problem areas for attention. Areas that now seem to need the most attention are systems analysis and plant operations. With respect to systems analysis, the NRC Staff, which has been highly compartmentalized, needs to build a stronger capability to understand and anticipate the interactions between plant systems, including the effects on such systems of accident environments and external phenomena. Relative to plant operations, the I&E staff needs to be able to understand better the behavior of operating systems, to assess the capabilities of the operating staff, and to assure that their activities do not jeopardize public safety because of design, construction, or operational errors.

The recent organization of a systems engineering group within the NRC Staff will be helpful in reducing the compartmentalization of technical skills and may ultimately satisfy the systems analysis need. The operational aspects of nuclear power plants have not yet been examined sufficiently to clarify how the NRC Staff capability should be altered. Areas in need of attention include a better understanding of methods for training nuclear power plant personnel, improved procedures for analyzing systems interactions, a broad capability for accident simulation, improved methods for the control of radionuclide effluents, and upgraded procedures for inservice inspection of plant safety features. All of these examples suggest a need for reorientation of existing review procedures rather than the addition of new staff skills. However, if the present staff is preoccupied with existing tasks, new sources of manpower may need to be obtained.

One possible way of expanding the I&E capability is through the use of third party review. The development of outside sources to review other plant features on a systems basis might be a useful approach. This approach is already accepted by the NRC for the Primary Coolant Circuit and Containment Structures under the <u>ASME Boiler and Unfired Pressure Vessel Code, Section III, Nuclear Components</u>. The qualifications of such reviewers would need to be established, but in principle this approach could extend the capabilities of the NRC Staff in matters pertaining to nuclear quality assurance.

To provide an independent assessment of its capabilities, the NRC Staff should consider the establishment of ad hoc review groups. While the ACRS could contribute to this activity, it does not appear to be an effective use of the Committee's limited time. Other arrangements should be sought. Individual ACRS members might be able to lead ad hoc review groups composed of consulting experts. It is important that such reviews be conducted by people who have an understanding of administrative as well as technological matters.

7.1.2 Industry Competence

The nuclear industry infrastructure is broad enough to satisfy most licensing requirements, given financial support and management backing. Thus far, however, segments of the industry have tended to limit their interests to complying with specific requirements of licensing, while managing the engineering aspects of nuclear power plants along the lines of conventional utility practice. Following this approach, many utilities have relied heavily on outside consulting services for technical guidance, although some of the larger utilities have established substantial nuclear engineering competence. Recent events indicate that nuclear power plant licensees need more basic capability to prepare for accident contingencies, to diagnose and respond to such events as they evolve, and to provide backup resources when needed.

The operating organizations cannot become totally knowledgeable about all nuclear steam system transient characteristics, but they can strengthen their understanding through training programs and professional staff additions. The organization of this additional capability will have to be adapted to existing operating situations, but it is extremely important that each licensee or license applicant establish direct top level managem nt interest in this capability on a continuing basis. The nuclear steam ystem suppliers and the architect-engineers also need to strengthen their capabilities in support of the operational organizations.

It would be appropriate for the NRC to encourage each of the major participants in the nuclear industry to commit themselves to an aggressive program for the development of safety improvements. Regulatory action alone will not satisfy the interest of public safety. The industry needs to demonstrate not only a commitment to the task, but also a methodology and a timetable for its accomplishment.

7.1.3 ACRS Effectiveness

The ACRS is assigned the responsibility for reviewing nuclear installations prior to licensing, and reporting the results of their deliberations to the NRC. In the Committee's view, some monitoring of current license applications and of operating experience will always be needed to assure up to date and comprehensive treatment of safety matters. Similarly, ACRS review of NRC's safety requirements, as embodied in regulations, standards, and standard review plans, must be continued since these requirements provide the basis for Staff judgment on such matters. The ACRS also needs to keep itself currently informed of safety research and international nuclear safety matters. When specific safety issues arise, the ACRS will frequently be asked to use its range of expertise to assist the regulatory administration in defining a path for minimizing public safety risk. All such matters are important and would appear to deserve priority over other demands on the Committee (7.1). This is especially true since the time of ACRS members is limited by their part-time status.

7.1.4 Clarification of Responsibility

Within the regulatory organizational structure, there are five line offices under the direction of the Executive Director for Operations (EDO). Because the law provides for direct access to the Commissioners by the Directors of three of these Offices, the authority of the EDO for public safety decisions may be diluted. Further, offices have sometimes acted independently of each other when their action should have been coordinated. The result is apparent confusion concerning the source of authority for regulatory positions. This has adversely affected public confidence in the regulatory process. Integrated and identifiable authority is needed to correct this situation.

The Commissioners also do not at present have a well-defined role. Legislative action should be taken to establish how the Commissioners as a collegial body and as individuals should meet their responsibilities and display appropriate regulatory leadership. If some other form of regulatory management approach is ultimately established, similar definition of the regulatory management role is needed.

A matter of equal concern is whether the NRC has delegated too much responsibility for public safety to the licensees. The NRC could interject itself more into operational planning and training. The presence of an NRC representative at a plant site offers NRC the prerogative to decide when and whether plants should be started up or shut down. In addition, the NRC could set more explicit requirements with respect to plant design, operating procedures, and effluent discharges, and it could require all applicants to follow these NRC directions. Thus far, the NRC has avoided this because it would essentially relieve the licensees of any responsibility for design and operational

(7.1) The ACRS in the past has reviewed radionuclide shipping cask design and verification programs, waste management plans, and other comparable matters of lesser safety significance. The Committee can continue to handle such matters when licensing activities are slow but it could not carry a heavy extra load concurrently with intensive licensing.

It is noted that in Japan, the advisory functions are divided between two committees, one for power plants and the other for the balance of the fuel cycle.

decisions. Such an approach might also result in the loss of the objectivity of the NRC review since the agency would be defending its own designs and operating initiatives. There is a crucial need to establish that licensees who accept such responsibilities are capable of meeting them.

7.2 Regulatory Format

The conduct of the regulatory process requires a well understood format in which the technological matters are presented and the quasi-legal public review is effected. No system as complicated as the nuclear regulatory process could have a detailed prescription for every regulatory requirement. Much that exists in the regulatory process is a result of continual development of review documents, and adversarial discussion between license applicants and safety reviewers, as well as the application of recognized conventional engineering methodology to important safety matters in every technological area. The application of this well understood base and the manner in which "standardization" is used to assure public safety must be appreciated by those concerned with regulatory management. The legal framework, itself, depends upon this format, but its use may be distorted if conventional legal processes are applied to safety areas. The ensuing discussion will show where some adjustment of the regulatory format is justified and desirable.

7.2.1 Preservation of Regulatory Base

The good safety record of the nuclear power industry is largely attributable to the regulations of the NRC, and its predecessors, and to the efforts of the nuclear industry. In considering the need for change in the regulatory process, care must be taken to preserve the good qualities of the regulatory system while seeking improvements. The current approach, based on the use of regulatory documents, is well understood even though some of them may be subject to misinterpretation, some may need to be more definitive and some may need to be expanded. It is important to work with the existing base to the maximum extent practical. If a new set of documents were introduced, the interpretation process, itself, could lead to regulatory chaos.

The experienced personnel involved in the regulatory process in both the regulatory and licensee organizations are also an important part of the base. Although management changes are needed and the definition of responsibility should be improved, those knowledgeable about the safety logic and the implicit but unstated cost-benefit balance must be permitted to function in a system not overly encumbered by procedural requirements or arbitrary management edicts.

7.2.2 Standardization

The c ncept of "standardization" was originally envisioned as a way to accelerate the licensing process by minimizing review time. Most NSSS vendors have established basically uniform configurations. All major equipment is standardized in manufacture and performance. The thrust of recent standardization has been to obtain "design approval" on a system basis so that system review will not have to be performed repetitiously.

Balance-of-plant designs by A-Es have followed a similar trend. The level of detail provided in standardized designs is not as complete as might be seen, for example, in air transport systems. The adequacy of the system definition, including the level of detail to be provided for final approval of standardized design has not yet been established. Insufficient experience is available to confirm the benefits anticipated from standardization. Up to now, it seems to extend further the variability of designs from those of existing plants.

A variation of standardization that has received considerable support is the "replication" of existing designs. This approach does reduce the design variability since the intent is to follow closely what has been done before. As applied in recent licensing actions, replication approaches have, unfortunately, tended to restrict initiatives for safety improvements on the premise that they violated the principle of "design stability" which standardization is intended to promote as a means of streamlining the approval process. This restriction might also be interpreted as a mechanism for circumventing requirements for public safety improvements.

There are certainly advantages to standardization that could be realized if many nuclear plants needed to be licensed rapidly. It is not certain that the present NRC approach really brings forth the advantages of standardization. The mode in which "standardization" is being used should be reexamined to determine whether alterations (7.2) would enhance nuclear plant reliability and safety without loss of the streamlining effects on licensing that it is intended to provide.

The range of reliability and safety in current designs can be measured in part by the current study of the critically required PWR auxiliary feedwater systems wherein a range of 100 or more in apparent reliability between various designs has been discovered. Comparable ranges of reliability may well be found in each of the other functional systems required

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(7.2) The concept of a standard LWR design for national use has been suggested. Such a design could evolve from careful sifting of the current designs to determine the most reliable and economical means by which functions common to all plants are accomplished.

for safe shutdown and accident mitigation. These range from the service water system through component cooling (including considerations of whether such a system is necessary), the secondary steam system (again, if necessary) environmental and equipment cooling systems, and the like. These systems, as exemplified by the PWR auxiliary feedwater systems, may all satisfy the minimum requirements of present regulations, yet still show an extreme range from very poor to generally excellent practice. In the final analysis it may well be argued that study would show that some BWR or PWR design features should be eliminated from a future standardized design.

A concept of standardization could be established that would be based almost entirely on the LWR experience over the last 20 years plus consideration of comparative accident vulnerability as determined by careful study of critical systems design under all modes of operation. Unproven extrapolations of nuclear technology might be excluded although evolution of design improvements within a few developmental plants could be part of the overall effort.

7.2.3 Legal Framework

A sound legal basis is essential to the regulatory process. One of the mechanisms in this process is the review of a license application by an ASLB. Such a review is intended to establish that the NRC has a basis for its rules and regulations, that it is following its own regulatory requirements and policies, and that it has satisfied the intent of the NEPA.

Since the NRC staff has satisfied itself as to the adequacy of the safety of a given facility prior to such a review, its legal staff generally supports the licensing actions before the ASLB. The NRC's legal staff also serves as a channel through which the Boards can probe the NRC staff positions on licensing actions.

There are some significant advantages to the public in this process. It sometimes provides an opportunity for further examination of legitimate safety concerns not fully exposed in the previous reviews. It also provides a valuable forum for discussing NEPA issues of concern to the public. Nevertheless, the hearing process leans more toward legal maneuvering than to a position supportive of public safety and environmental concerns. In addition, it seems to have discouraged discussion of safety issues in the Safety Evaluation Report (SER) and in other documentary evidence intended for Hearing Board review. It also leads to legally oriented oral statements by NRC Staff members. Most importantly, this approach discourages the NRC Staff from discussing controversial subjects of safety concern in open meetings, including those with the ACRS. These restraints are probably intended to eliminate extraneous matters that might unnecessarily delay the hearing process. Unfortunately, they may also prevent full exploration of some significant safety issue.

Under these conditions, the Staff SER appears to be prepared mainly to provide information for the quasi-judicial ASLB hearing. As a result, the SER consists primarily of repetitive "boiler plate" which often tends to obscure and provide little amplification of safety issues. The result is that the SER has become a document of little value to those people responsible for safety reviews of nuclear facilities. This includes members of the ACRS.

Public safety is not well served by this legal style of safety issue presentation. If the SER included discussion of the various aspects of each significant safety issue, together with a judgment basis for the NRC Staff conclusions, the report would serve a more appropriate role at the ASLB hearing. The reasoning of the NRC Staff could be examined by the ACRS and the ASLB without the need for advocacy by the NRC legal staff. Where a basis for ruling on a particular safety issue had been previously established, it could readily be identified. The public would then be able to see why, where and how the NRC Staff's safety conclusions were drawn.

ASLB rulings on specific safety issues have sometimes, because of legal considerations, adversely affected public safety interest as the following example illustrates. The ASLB has on occasion ruled that the NRC could not require planning for emergency action beyond the low population zone (LPZ). It has also ruled in some cases that the radius of the LPZ must be reduced because of population growth near a plant site. These two rulings have combined to permit a high population density adjoining some sites without commensurate planning for emergencies.

The ASLB hearings are also used as a mechanism for determining whether the NRC Staff has an appropriate basis for rulemaking. Although the hearing provides an opportunity for open debate, the subject matter is sometimes outside the context of specific licensing actions. Whether such hearings provide the proper forum for establishing technological validity is not entirely clear. For example, adversary proceedings lasting more than a year were needed to develop rulemaking (7.3) for analytical techniques to demonstrate the performance adequacy of ECC systems. Even so, some reliability aspects were never adequately addressed during this hearing process. If such a process is to be used as the basis for rulemaking, the manner in which the issues are to be addressed and the rules established needs further study.

The attention directed to NEPA may be indirectly interfering with public safety reviews by diverting attention to other interests, such as power system load growth, cost-benefits of alternate power sources, anti-trust considerations, and other environmental matters. These are concerns of major public interest, and the NRC is probably justified in its diligent attention to them. However, there has sometimes been a tendency to move

(7.3) Published in 10 CFR Part 50 as Appendix K.

NEPA matters ahead of public safety matters. The selection of a power plant site, for instance, is weighed carefully by NRC with respect to its economic benefits, social impacts, and power system demand, but in most cases, safety alternatives are weighed only with respect to whether a particular site meets the minimum safety requirements (7.4).

The Public Hearings are an important aspect of the nuclear regulatory process, but some consideration needs to be given to changing the style of the hearings so that the safety issues can be exposed fully without unnecessarily delaying licensing actions. The combining of NEPA and Safety Reviews in the ASLB hearings may be a contributing complication. To the extent practical, it would be desirable to separate these two issues in the hearing process.

7.3 Regulatory Actions

Public perception of regulatory actions will be improved if safety problems are reported on a timely basis and actions are implemented promptly when needed to assure the protection of the public. Since the accident at TMI-2, the NRC staff has been reexamining the manner in which public safety problems are identified and how it implements corrections. Specific changes to be proposed are still under discussion. The areas where alteration in the regulatory style could be of immediate value are noted below.

7.3.1 Reporting of Safety Problems

New safety problems will appear in nuclear installations, and it is unrealistic to assume that all will be predictable. The NRC requires all licensees to report safety-significant happenings promptly so that necessary regulatory actions can be taken. The comprehensiveness of the reporting requirements, however, may not be adequate to cover all areas of interest nor to include all participants who might make a safety contribution. Action should be taken to make certain that nuclear plant owners and operators, constructors, NSSS and other equipment suppliers, inspection and service organizations, craftsmen, operating personnel, and even the public at large report matters of public safety significance as soon as they are known. While this may occasionally cause unnecessary reaction to minor safety matters, it will assure that maximum time is available to correct serious difficulties.

(7.4) An exception is noted in the case of the Hope Creek Nuclear Station whose site was changed from Newbold Island after NEPA review focused attention on the less than desirable population distribution in the proximity of the previously selected site but only after the earlier site had carried through an extensive licensing review including ACRS hearings.

At the same time, the reporting system should not be excessively burdensome. The informational requirements should be defined in such a way that those involved in reporting can, without excessive effort, provide whatever information is necessary to assess the safety significance of such matters. Of particular importance is the need to avoid a prosecutory environment(7.5) for those who report errors, faults, and maloperations, particularly when deliberate wrongdoing is not evident. Only in this way can the regulatory system assure a positive response from licensed participants, their contractors, and their employees.

7.3.2 Resolution of Generic Problems

Some years ago, the ACRS developed a list of safety matters that, although requiring attention, were not of such urgency that they required final resolution for all specific license applications. It was intended that these matters be covered by the NRC and its licensees over the long term and that the problems be corrected as solutions were found. The rate at which these "generic safety items" were being examined and resolved, however, was relatively slow and this has caused considerable public concern (7.6). In the past two years, the NRC Staff has established a more complete Generic Items list of its own, that incorporates all of the ACRS items, and has recommended priorities for addressing each item. Although the NRC Staff list is more extensive than the ACRS list, there is agreement on most of the high priority matters. Action plans for resolving the items of highest priority have been established and an "unresolved Safety Issue Task Force" was recently formed by the NRC Staff to assure that high priority matters are given adequate attention.

Although the NRC Staff actions in the past have not appeared to be aggressive in addressing generic problems, or timely in implementing their solutions, current efforts appear to be more acceptable. Some matters cannot be readily resolved by physical changes and will require surveillance or other

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(7.5) Although it is difficult to excuse mistakes and unintended violations of regulations, the threat of legal jeopardy in such instances can only create an environment of protective cover-up among the threatened that tends to hide important safety information. If the legal threat is sufficiently serious, career-minded professionals will seek other employment areas, weakening the industry's capability.

(7.6) The need for "instruments to follow the course of accidents" is a generic item whose resolution was planned through issuance of Regulatory Guide 1.97. The guide was excessively vague in some areas and overly demanding in others. The NRC was never able to reach an understanding with the industry concerning implementation. In a similar vein, the ATWS issue has been debated for more than 10 years, but an agreed upon implementation plan for resolving the issue has not yet been established.

types of contols to minimize public risk. Others may involve implementation of major plant changes during planned outages. The correction of generic problems can be handled on a longer term basis if the risks are well understood and appropriate defenses are maintained. The current staff actions appear to be responsive to regulatory needs, and they should be continued in an aggressive mode. Establishing positive implementation plans once resolution actions are known is essential to maintaining public confidence in the regulatory process.

7.3.3 Back- and Forward-Fitting of Safety Improvements

The public risk associated with omitting or delaying desirable safety improvements or correcting safety deficiencies may be quite small if only a few plants are involved and operating organizations can provide compensating surveillance, for example. Changes in existing plants are often costly, and redesign sometimes delays the licensing process. These factors must be taken into account when the NRC imposes new requirements. Nevertheless, a limit must be established with respect to the cumulative risk from delaying such actions. Some matters(7.7) currently under consideration have been deferred for such a long time that they might be viewed as the object of deliberate procrastination. The NRC needs to show how its judgments concerning backfit or forward fit actions are established. Cost and schedule cannot be overriding considerations if there is real concern for public safety.

7.3.4 Public Communications

The public anticipates that the NRC will keep it informed in an intelligent and responsible way concerning safety problems, licensing actions, regulatory deficiencies, health effects, waste disposal, and similar matters. The public, as well as the NRC licensees, often have difficulty in determining which sources of information are authoritative and whether information provided by NRC staff members is fact or opinion, official or private, preliminary or final. Clearly, as was recognized in connection with the accident at TMI-2, a single well-informed spokesman is essential to avoid confusion in responding to an emergency. The NRC organization should be prepared through such a spokesman to explain, clarify, correct, modify, amplify or otherwise inform the public of matters appearing in the public information media in a timely fashion so that the public can identify the authoritative regulatory voice and discern the public safety significance of the information.

7.7 The Recirculation Pump Trip provision intended to alleviate concerns for ATWS consequences in BWRs is not yet fully implemented even though this has been a recognized need for about a decade. Also, increased pressure relief capacity in PWRs seems to be meeting high industry resistance even though recent ATWS reviews show that such capability will eliminate most concerns for this safety issue.

The provision of a designated spokesman to express the official NRC viewpoint, however, should not be a mechanism for stifling expression of divergent views. Indeed, some Commissioners and some members of the NRC Staff may differ with the official position and they should be encouraged to express those views. Speakers should state that they are expressing personal views if they do not represent the collective NRC viewpoint. When appropriate, the NRC may even wish to have its spokesman discuss divergent positions that are under consideration. The benefit from having a designated spokesman is that the press and the public can see the regulatory thought processes at work in both the official and the independent positions and can have some understanding of their bases.

8. OVERALL ASSESSMENT

The regulatory base being used by the NRC is substantial. Over the 25-year period of development, the regulatory process has evolved a methodology for accident assessment in the interest of public safety that covers virtually all of the major issues. It has many imperfections, but the goals outlined in Chapter 2 of this report have all been addressed. As has been indicated in preceding sections of this report, there is considerable unevenness in the effectiveness of the regulatory activities, and in some cases, the capability does not measure up to the need.

There are a number of strong points in the current regulatory process. They include an established review methodology that is commonly understood and used by the regulatory staff and the regulated industry, a regulatory staff on the whole of high caliber that handles the technological issues knowledgeably and with dedication, and a system for identification of problem areas that draws attention to safety matters. These are valuable assets of the current regulatory system, and they should not be jeopardized by changes in the management structure or in the scope of the regulatory authority.

There are also shortcomings in the regulatory process that need improvement. The President's Commission appointed to investigate the TMI-2 accident made a number of recommendations in this respect.

The ACRS concurs with many of these recommendations and offers the following seven recommendations as its interpretation of the needed actions pertaining to the regulatory process:

1. The nuclear regulatory function requires strong leadership. This could be provided by one of several options such as a Regulatory Commission Chairman having full executive authority, a single administrator to whom all regulatory functions report, an administrator with full executive responsibility reporting to the Commissioners on policy matters or a Commission formed from the chief technical, legal and enforcement executives of the regulatory organization with one of them designated to be the chief executive officer. The essential requirements of the leadership assignment are a knowledgeable understanding of the regulatory processes, a sound technological background, and the ability and authority to act decisively on regulatory questions including the handling of nuclear safety emergencies.

2. The President's Commission proposed that an oversight committee be established to examine the performance of a nuclear regulatory organization headed by a single administrator. The ACRS is not persuaded of

the need for such a part-time oversight committee specific to nuclear energy, and believes that, if such a committee were to be created, it should have a much broader charter with regard to technological issues in society.

3. Except for a few limited cases considered during the past few years, the staff has been unwilling to investigate potentially significant safety matters if they were not identified as part of the "design basis." Its consideration of the ramifications of accidents involving degraded safety feature performance or other circumstances leading to accident consequences beyond those covered by the "design basis" was too restrictive, causing both the industry and the regulatory staff to be inadequately prepared for unanticipated accident circumstances. There has been a salutary change in the NRC Staff views of such matters since the TMI-2 accident that seems responsive to the need. Future organizational arrangements should assure that this interest will be sustained.

4. There is a need to strengthen some NRC Staff functions, including those related to (a) provision of a systems approach to safety review, (b) a better audit of design, and (c) improved regulatory monitoring of licensee performance including operations and technical support.

5. The role of the ACRS should be strengthened by establishing the necessary arrangements for assuring that timely and adequate attention to ACRS concerns is given by the Commissioners as well as the NRC Staff.

6. The nuclear industry must strengthen its ability to handle safety matters. A strong technical and managerial capability in this area on the part of all licensees and their contractors is very important. The industry has taken some positive steps in this direction since the TMI-2 accident, but further changes are still needed.

7. The relevant knowledge and expertise gained during plant design and construction must be transferred to those responsible for plant operations. The licensees, individually and cooperatively, should take an active rather than a passive role in a design decision making process. The utility licensees must show they have effective and timely access to the technical resources of their contractors and suppliers or the equivalent over the plant lifetime.

In addition to the preceding seven general recommendations, the ACRS recommends that the following nine technological matters be considered at the earliest opportunity.

8. Accidents beyond the current design bases should be considered in deciding on the future approach to siting, to reactor design, and to emergency measures. Future reactors should not be located at sites with high population densities. Using a risk-benefit evaluation basis, design and other measures should be considered to further reduce the probability of serious accidents and to mitigate their consequences.

9. The ACRS believes that the fundamental safety goal of both the NRC and the nuclear industry should be to achieve a degree of safety that is as good as reasonably achievable, taking into consideration appropriate technical, social, and economic factors.

10. Where practical, a quantitative approach should be used in establishing safety criteria, in assessing potential enhancement of safety, and in providing well qualified comparative risk assessments relating nuclear power to other technological aspects of society. Publicly stated goals with regard to acceptable risk, the levels of safety which are thought to have been achieved, and the uncertainties inherent in such estimates of risk should be available to provide a basis for judgment by the public.

11. The "single-failure criterion" and other failure control design bases should be modified as necessary to encourage more consideration of progressive, common cause, and multiple failures arising from a single initiating event. A systematic evaluation should be made of the needed reliability for components, systems, or groups of systems, commensurate with the impact of their failure on accident consequences affecting the public health and safety.

12. Separate and dedicated safety systems can and should be used where appropriate to enhance reliability; however, future safety review and evaluation should consider not only safety-designated items, but also the potential safety influence of all portions of the plant.

13. Substantially increased attention should be given by the nuclear industry and the regulatory staff to potentially adverse system interactions. A method for studying system interactions needs to be developed and used for this purpose.

14. Much more attention must be given to man-machine interactions with respect to the manner in which they can affect public safety.

15. Regulatory and industry organizations should aggressively investigate such safety improvements as filtered vented containment, dedicated shutdown heat removal systems, and design changes to reduce the probability of successful sabotage, and implement those found appropriate. The nuclear industry should be more aggressive in seeking safety improvements beyond those required by the regulations and the regulatory process should provide incentives for this purpose.

16. Where practical, the techniques of probabilistic analysis should be applied to operating plants and to plants under construction to ascertain whether there are design improvements whose implementation would reduce the overall risk to the public.

With regard to the regulatory and industry organizations there is a need for skill enhancement in some areas, improved quality assurance arrangements for design, and greater industry initiative to improved safety. The actions to satisfy these needs are outlined in the following eight recommendations:

17. A procedure is needed whereby operating nuclear plants are periodically reexamined taking into account current nuclear criteria and standards. The performance of the operating organization and the technical support available to it should also be examined during these periodic reviews. The existing systematic review program should be restructured and expedited, with responsibility placed on licensees to periodically evaluate and report on the safety acceptability of continued plant operation.

18. The basic orientation of the NRC safety research program should be shifted from overemphasis on "confirmatory research" to substantial effort in research intended to improve nuclear power safety by assisting in the resolution of identified safety concerns, by examining possible safety improvements and by exploring for issues or problems of potential significance. The probabilistic techniques developed for risk assessment should be made an active working tool in the safety improvement effort.

19. It is recommended that the NRC use its powers vigorously under 10 CFR Part 21 to require that NSSS vendors, A-Es, and licensees promptly report safety concerns that may be raised within their organizations, including submittal of pertinent internal documents.

20. It is important to public safety that the nuclear steam system vendor organizations be maintained at a high level of competence or that an equivalent source of expert knowledge of the performance and function of the nuclear steam supply systems be developed and maintained as a direct support available to licensees when needed during the plant life time.

21. A fundamental change in approach by both architect-engineer and plant owner must be developed in which the objective of the architect-engineer is to make the safety of the plant as good as reasonably achievable, rather than merely meeting existing regulatory requirements at minimum cost. For example, the use of probabilistic techniques and systems engineering studies, performed jointly by the A-E and the owners' staff, should help to determine where significant

gains in system reliability or safety margin can be obtained at reasonable cost. A-Es should be required to demonstrate that appropriately safe design has been attained.

22. Methods should be developed and implemented to provide a meaningful, more extensive design check and audit of the balance-of-plant than has been the general custom previously. This might be partially achieved through appropriate, certified third party organizations which are independent of both the nuclear industry and the NRC Staff. However, the internal review functions of the owner and the A-E must also be improved.

23. As stated in its recent Review of Licensee Event Reports (NUREG-0572), the Committee believes that operating experience can provide an important source of safety guidance for commercial power plants. The Committee encourages the NRC to continue to develop a program under which benefits of the lessons learned from LERs can be fed back into the design, construction, operation, and maintenance of nuclear plants.

24. The development of a limited number of standard LWR plant designs using an as good as reasonably achievable design philosophy would provide guidance in judging public safety adequacy and should be encouraged. Where appropriate, these designs should include ideas that depart from previous practice.

The safety of operating nuclear power plants and of those nearly ready to be licensed can be improved during the current licensing "pause" adopted by the NRC. The ACRS agrees that some of the safety improvements could be significant. However, the Committee does not believe that the absolute or incremental risk from operation of several more newly completed nuclear power plants will pose unusual or unacceptable individual or societal risks. Serious consideration should be given to permitting startup tests for plants ready for licensing that have safety features at least equivalent to those now required for currently operating plants. These plants could then be placed on standby as being ready for operation if required in the national interest while the NRC is deciding on the needed changes in safety requirements beyond those already announced.

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16. ABSTRACT (200 words or less)

A reexamination by the ACRS of the Regulatory Process has been made. Objectives were to provide in a single source, ACRS' understanding of the Regulatory Process and to point out perceived weaknesses and to make appropriate recommendations for change.

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

December 13, 1979

APPENDIX XVII
IDENTIFICATION OF NRC REGULATORY
REQUIREMENTS WHICH NEED CHANGING

The Honorable Peter A. Bradford
Commissioner
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: IDENTIFICATION OF NRC REGULATORY REQUIREMENTS WHICH NEED CHANGING

Dear Mr. Bradford:

Your memorandum of September 28, 1979 requests ACRS views on whether the lack of a specific procedure for identifying rules and regulations which need revision has inhibited the Committee. In addition, you asked that the Committee identify any rules and regulations which need to be addressed promptly in order to ensure public health and safety.

In the evaluation of nuclear safety, ACRS review is normally directed to an area of technical interest or concern rather than a specific rule or regulation which may need changing. Resolution of the topic in question may warrant a change in regulatory policy or requirements which in turn can lead to the need for a change in NRC regulatory guides, the NRC Standard Review Plan, or Branch Technical Positions as well as the regulations themselves. The Committee is therefore not in a position to identify specific regulations which need changing. We can however cite several examples of safety related areas which have a direct bearing on the adequacy of NRC regulatory requirements. To the extent practical, we believe that resources should be directed toward making appropriate modifications in relevant regulatory requirements consistent with the recommendations of the Committee in previous reports, letters and memoranda which have addressed these matters in some detail. References to these previous documents are included in the Attachment.

Examples of safety related areas include:

I. Consideration of Accidents Beyond the Limits of the Regulatory Design Basis (including emergency procedures beyond the LPZ);

II. More Widespread Use of Probabilistic Analysis in Decision Making within NRC (including reexamination of the single-failure criterion);

III. Prompt Resolution of ATWS;

IV. The Requirement of Plant Design Changes to Reduce the Possibility and Consequences of Sabotage;

V. Changes in Commission Regulations Providing a Well-Defined Basis on which to Withhold Plant Security Information;

VI. The Introduction into the Licensing Process of Systems Interaction Analysis, not Limited by the Constraints of the Single Failure Criterion, and Including the Effect of Malfunctions of Equipment Heretofore Considered as Non-safety-Related;

VII. Additional Consideration of Small Break LOCA and the Reliability of ECCS;

VIII. Improved Monitoring and Display of Plant Status Information to Assist the Operators in Evaluating and Dealing with Anomalous or Unanticipated Situations;

IX. Reliability Requirements for Systems Important to Plant Safety Including those not Identified as Engineered Safety Features;

X. Systematic Evaluation Program;

XI. Backfitting Criteria;

XII. Feedback of Operating Experience;

XIII. Improvements in Operator Training and Licensing.

Although the ACRS is not inhibited by the lack of a specific procedure from making recommendations it considers appropriate, the lack of adequate "follow-up," including allocation of appropriate NRC resources to do related analysis, evaluation, and research, has in some cases had an inhibiting effect on the Committee in its ability to formulate recommendations regarding specific technical issues. A well defined procedure for ACRS participation in rulemaking would be useful in understanding the roles and responsibilities of the ACRS and the NRC Staff in this area.

The Committee would be pleased to discuss these items in more detail if you desire.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

Attachment:
List of Significant References in Support of ACRS Recommendations

LIST OF SPECIFIC REFERENCES IN SUPPORT OF ACRS RECOMMENDATIONS

I) Consideration of Accidents Beyond the Limits of the Regulatory Design Basis

A. Instrumentation to follow the course of a serious accident

1. Letter, Mangelsdorf to Muntzing, August 14, 1973, "Instrumentation to Diagnose the Course of a Serious Accident."

2. "Status of Generic Items Relating to Light-Water Reactors: Report No. 4", April 16, 1976 - Item II-11, Instruments to Follow the Course of an Accident.

3. Report, Moeller to Rowden, October 22, 1976 "Report on Three Mile Island Nuclear Station Unit 2."

4. "Report on Selected Safety Issues Related to Light Water Reactors - Issues 16-27", February 23, 1977, Process Variables During Accidents.

5. Report, Carbon to Hendrie, August 13, 1979, "Short-term Recommendations of TMI-2 Lessons Learned Task Force."

B. Means for Retaining Molten Fuel or a Molten Core

1. Report, Okrent to Seaborg, October 12, 1966, "Report on Reactor Safety Research Program."

2. Memo, Fraley to Shaw, January 11, 1971, "ACRS Comments on a Core Retention System to Mitigate the Consequences of a Core Meltdown."

3. NUREG-0496, "1978 Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," December 1978, page 3-3.

4. NUREG-0603, "Comments on the NRC Safety Research Program Budget," July 1979, page 1-5.

C. Consideration of Evacuation Plans Outside the Part 100-Defined LPZ

1. Reports to the Chairman, AEC on: Newbold Island 1 and 2, July 17, 1973; Seabrook 1 and 2, December 10, 1974.

2. Letter, Kerr to Gossick, December 10, 1975, "Report on Review of Siting Policies for Licensing Nuclear Facilities."

D. Means to Deal with Large Accumulations of Hydrogen or Other non-Condensible Gases

1. Memoranda on "Status of ACRS Recommendations," as follows:

Fraley to Rusche, March 14, 1977, page 4
Case to Fraley, July 1, 1977, page 6
Fraley to Case, January 18, 1978, page 1
Case to Fraley, April 18, 1978, page A-2, A-3

2. Report, Carbon to Hendrie, April 7, 1979, "Interim Report on Recent Accident at the Three Mile Island Nuclear Station Unit 2."

II) <u>More Widespread Use of Probabilistic Analysis in Decision Making Within NRC</u>

A. Letter, Moeller to Gossick, July 14, 1976.

B. NUREG-0392, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," December 1977 - page 9.6.

C. Report, Carbon to Hendrie, May 16, 1979, "Interim Report No. 3 Three Mile Island Nuclear Station Unit 2."

D. NUREG-0603, "Comments on the NRC Safety Research Program Budget," July 1979, Section 1.2.11.

III) <u>Prompt Resolution of ATWS; in Particular, the Installation of Recirculation Pump Trips in BWRs</u>

A. Report, Hendrie to Seaborg January 27, 1970, "Report on Palisades Plant."

IV) <u>The Requirement of Plant Design Changes to Reduce the Possibility and Consequences of Sabotage</u>

A. Report, Moeller to Rowden, August 17, 1976; "Design Provisions for Protection Against Sabotage."

B. Report, Moeller to Rowden, September 16, 1976, "Clarification of August 17, 1976 ACRS Report on Design Provisions for Protection Against Sabotage."

C. NUREG-0496, "1978 Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program " - page 10-3.

V) <u>Changes in Commission Regulations Providing a Well-Defined Basis on Which to Withhold Plant Security Information</u>

A. Report, Bender to Hendrie, August 18, 1977, "Nuclear Plant Security."

VI) The Introduction into the Licensing Process of Systems Interaction Analysis, not Limited by the Constraints of the Single Failure Criterion, and Including the Effect of Malfunctions of Equipment Heretofore Considered as Non-safety Related

A. Letter, Stratton to Muntzing, November 8, 1974, "System Analysis of Engineered Safety Systems."

B. Letter, Bender to Gossick, June 17, 1977, "Review of Systems Interaction."

C. Report, Carbon to Hendrie, May 16, 1979, "Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2."

D. Letter, Carbon to Gossick, October 12, 1979, "Systems Interactions Study for Indian Point Nuclear Generating Unit No. 3."

E. Memo, Igne to ACRS Members, October 18, 1979, "Evaluation of Potential Interactions Due to High Energy Line Breaks at Salem 2."

VII) Additional Consideration of Small Break LOCA and the Reliability of ECCS

A. Report, Carbon to Hendrie, April 7, 1979, "Interim Report on Recent Accident at the Three Mile Island Nuclear Station Unit 2."

VIII) Improved Monitoring and Display of Plant Status Information to Assist the Operators in Evaluating and Dealing with Anomalous or Unanticipated Situations

A. Memo, Fraley to Commissioners, April 18, 1979, transmitting "Recommendations of the NRC Advisory Committee on Reactor Safeguards regarding the March 28, 1979 Accident at the Three Mile Island Nuclear Station Unit 2."

B. Report, Carbon to Hendrie, May 16, 1979, "Interim Report No. 2 on Three Mile Island Nuclear Station Unit 2."

IX) Reliability Requirements for Systems Important to Plant Safety Including those not Identified as Engineered Safety Features

A. Letter, Moeller to Gossick, July 14, 1976.

B. Letter, Bender to Gossick, March 15, 1977, "Reliability of Power Supplies."

C. Letter, Bender to Gossick, May 15, 1977, "Auxiliary System Reliability."

D. Memo, Fraley to Gossick, March 14, 1979, "Requirements for Shutdown and Decay Heat Removal Using Safety Grade Equipment."

E. Report, Carbon to Hendrie, May 16, 1979, "Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2."

X) Systematic Evaluation Program

A. Report, Okrent to Seaborg, June 14, 1966, "Periodic, Comprehensive (Ten-Year) Review of Operating Power Reactors."

B. Report, Bush to Seaborg, November 17, 1970, "Safety of Operating Reactors."

C. Report, Carbon to Hendrie, October 11, 1979, "Systematic Evaluation Program."

XI) Backfitting Criteria

A. Report, Carbon to Hendrie, May 16, 1979, "Report on Quantitative Safety Goals."

XII) Feedback of Operating Experience

A. "Review of Licensee Event Reports (1976-1978)," NUREG-0572.

XIII) Improvements in Operator Training and Licensing

A. Report, Carbon to Hendrie, May 16, 1979, "Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2."

APPENDIX XVIII

Additional Documents Provided for ACRS' Use

1. Report of The President's Commission on the Accident at Three Mile Island, p 61-79, "Commission Recommendations"

2. Letter, Chairman J. M. Hendrie, NRC to Dr. Frank Press, Director, Office of Science and Technology Policy, Preliminary Analysis and Views of the Nuclear Regulatory Commission on the Recommendations of the President's Commission on the Accident at Three Mile Island

3. Summary of testimony before the Nov. 14, 1979 hearing of the Subcommittee on Energy Research and Production, House Committee on Science and Technology

4. NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, July 1979

5. NUREG-0585, TMI-2 Lessons Learned Task Force Final Report, Oct. 1979

6. OMB Circular A-63, Advisory Committee Management, March 27, 1974