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of established records of distinguished service; and (3) shall be so selected as to provide representation of the views of scientific leaders in all areas of the Nation.

The terms of eight Members of the National Science Board will expire on May 10, 1980. All Members of the 1980 class are eligible for reappointment except Dr. Norman Hackerman and Dr. Grover E. Murray who have been Members of the Board for two six-year terms. Section 4(d) of the Act states that: Any person, other than the Director, who has been a member of the Board for twelve consecutive years shall thereafter be ineligible for appointment during the two-year period following the expiration of such twelfth year.

The Board and the Director solicit and evaluate nominations for submission to the President. Nominations accompanied by biographical information may be forwarded to the Chairman, National Science Board, Washington, D.C. 20550, no later than August 15, 1979.

Any questions should be directed to Miss Vernice Anderson, Executive Secretary, National Science Board (202/632-5840).

Norman Hackerman,

Chairman, National Science Board.

[FR Doc. 79-17127 Filed 6-4-79; 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 June 14-16, 1979, in Room 1046, 1717 H Street, NW., Washington, D.C. Notice of this meeting was published on May 24, 1979 (44 FR 30177).

The agenda for the subject meeting will be as follows:

Thursday, June 14, 1979

8:30 A.M.-11:30 A.M.: Executive Session (Open/Closed)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities. A portion of this session will be closed to discuss classified information related to the operation of nuclear powered naval ships.

The Committee will hear and discuss the report of its Subcommittees on the status of the Three Mile Island Nuclear Station and the implications regarding nuclear power plant design of the Three Mile Island Nuclear Station Unit 2 Accident which occurred on March 28, 1979. A portion of this session will be closed as required to discuss Proprietary Information related to this matter.

11:30 A.M.-1:00 P.M.: Meeting with NRC Staff (Open)—The Committee will meet with members of the NRC Staff to hear reports on the status of the Three Mile Island Nuclear Station Unit 2 and the NRC Staff evaluation of industry replies to I&E Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06B, and 79-08, NRC orders to the operators of Babcock and Wilcox nuclear plants, and NRC Staff action in response to ACRS recommendations regarding the March 28, 1979 accident at the Three Mile Island Nuclear Station Unit 2 (TMI-2).

2:00 P.M.-2:30 P.M.: Executive Session (Open)—The Committee will consider items to be discussed during its meeting with the NRC Commissioners including ACRS Interim Reports No. 2 and No. 3 regarding implications of the March 28, 1979 accident at TMI-2, use of quantitative risk assessment as a regulatory basis, and the need for ACRS review of proposed regulations regarding transportation of spent nuclear fuel.

2:30 P.M.-3:30 P.M.: Meeting with NRC Commissioners (Open) (Room 1130)—The Committee will meet with the NRC Commissioners to discuss items noted above.

3:30 P.M.-4:00 P.M.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee on Operating Reactors and consultants who may be present regarding the request for an increase in power level for the Millstone Nuclear Power Station Unit 2.

Portions of this session will be closed if necessary to discuss Proprietary Information related to this matter.

4:00 P.M.—5:30 P.M.: Millstone Nuclear Power Station Unit 2 (Open)—The Committee will hear and discuss presentations by members of the NRC Staff and the applicant regarding the request for an increase in power for the Millstone Nuclear Power Station, Unit 2.

Portions of this session will be closed if required to discuss Proprietary Information related to this matter.

5:30 P.M.-6:30 P.M.: Executive Session (Open)—The Committee will hear and discuss reports of its Subcommittees on matter related to nuclear power plant safety including proposed revisions to NRC Regulatory Guides, use of probabilistic assessment as a regulatory requirement and return to operation of the Fort St. Vrain Nuclear Station.

Friday, June 15, 1979

8:30 A.M.-11:00 A.M.: Meeting with Metropolitan Edison Company (Open)—The Committee will hear presentations by and hold discussions with representatives of the Metropolitan Edison Company regarding the accident which occurred at TMI-2 on March 28, 1979, activities following the accident, and the current status of the plant. Portions of this session will be closed as required to discuss Proprietary Information related to this matter.

11:00 A.M.-1:00 P.M.: Meeting with Babcock and Wilcox Company (Open)—The Committee will hear presentations by and hold discussions with representatives of the Babcock and Wilcox Company regarding NRC-I&E Bulletins No. 79-05 and 79-05A, 79-05B; NRC orders to operating nuclear stations which make use of Babcock and Wilcox nuclear steam supply systems; and ACRS recommendations resulting from the March 28, 1979 accident which occurred at TMI-2.

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

2:00 P.M.-5:30 P.M.: Meeting with NRC Staff (Open)—The Committee will hear presentations by and hold discussions with members of the NRC Staff regarding recent operating experience and licensing actions including a failure of the condenser steam dump control valves to close following a load rejection at the Beaver Valley Nuclear Plant Unit No. 1, replacement of reactor pressure vessel nozzles in the Duane Arnold Nuclear Plant, and cracking in the feedwater piping at the D.C. Cook Nuclear Station.

The Committee will also hear a brief report from the NRC Staff regarding inadequacies in the design of the piping of several nuclear plants related to their ability to withstand seismic disturbances.

The NRC Staff will also report to the Committee on NUREG-0531 "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants" dated February 1979, and NUREG-0396, "A Modified Basis for the Development of State and Local Government Radiological Response Plans In Support of Light Water Nuclear Power Plants."

The future schedule for ACRS activities will also be discussed.

5:30 P.M.-6:30 P.M. Executive Session (Open)— Discuss proposed ACRS comments and recommendations regarding TMI-2; implications of the accident at TMI-2 on the design of nuclear facilities, and the proposed

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power level increase for the Millstone Point Nuclear Power Station Unit 2.

Saturday, June 16, 1979

8:30 A.M.–4:00 P.M.: Executive Session (Open)—The Committee will discuss proposed ACRS comments and recommendations regarding TMI–2: the implications of the accident which occurred at TMI–2 on March 28, 1979; and the requested power level increase for the Millstone Nuclear Power Station, Unit 2.

The Committee will discuss proposed ACRS reports to NRC regarding the request for a Construction Permit for the Palo Verde Nuclear Generating Station Units 4 and 5, and an Operating License for the Sequoyah Nuclear Plant Units 1 and 2.

The Committee will discuss proposed ACRS comments/positions regarding use of stainless steel fuel element cladding, stress corrosion cracking in nuclear power plant piping and a modified basis for development of radiological emergency response plans in support of light-water nuclear power plants.

The Committee will continue discussion of other items considered during this meeting including regulatory and other deficiencies identified as a result of the March 28, 1979 accident at TMI–2.

Portions of this session will be closed as necessary to discuss Proprietary Information, provisions for physical security of the facilities involved, classified Information and matters involved in an adjudicatory proceeding.

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on October 4, 1978 (44 FR 45926). 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.

I have determined in accordance with Subsection 10(d) P.L. 92–463 that it is necessary to close portions of this meeting as noted above to protect Proprietary Information (5 U.S.C. 552 b(c)(4)), to preserve the confidentiality of classified information and the arrangements for physical protection of the Palo Verde, Sequoyah and Millstone nuclear plants (5 U.S.C. 552b(c)(1)) and to permit discussion of matters involved in an adjudicatory proceeding (5 U.S.C. 552 b(c)(10)).

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 202/634–1371), between 8:15 A.M. and 5:00 P.M. EDT.

Dated: May 31, 1979.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 79-17333 Filed 6-4-79; 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. 38 to Facility Operating License No. DPR–23, to 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, near Hartsville, South Carolina. The amendment is effective as of the date of its issuance.

The amendment deletes pressurizer level as an input to safety injection actuation, and requires actuation of safety injection based on two out of three channels of low pressurizer pressure.

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 licensee's submittal dated May 18, 1979, (2) Amendment No. 38 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. and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 24th day of May, 1979.

For the Nuclear Regulatory Commission.

A. Schwencer,

Chief, Operating Reactors Branch No. 1, Division of Operating Reactors.

[FR Doc. 79-17337 Filed 6-4-79; 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. 37 to Facility Operating License No. DPR–23, issued to Carolina Power and Light Company, which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant, Unit No. 2 (the facility) located in Darlington County, South Carolina. The amendment is effective as of its date of issuance.

The amendment incorporates Commission requested changes regarding the qualifications of the Environmental and Radiation Control Supervisor.

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

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MINUTES OF THE
230TH ACRS MEETING
JUNE 14-16, 1979
WASHINGTON, D. C.

The 230th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N. W., Washington, DC, was convened at 8:30 a.m., Thursday, June 14, 1979.

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

The Chairman noted the existence of the published agenda for this meeting, and the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been 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 Street N. W., Washington, DC, within approximately 24 hours.

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

I. Chairman's Report (Open to Public)

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

A. Reviewers

The Chairman named Messrs. Lawroski and Mathis as reviewers for the 230th ACRS meeting.

B. William H. Zimmer Nuclear Power Station Unit 1: Alleged False Statement During ACRS Review

The Chairman noted the matter identified in the memorandum from J. G. Keppler, Director, Region III, IE, to D. Thompson, IE, dated April 10, 1979, regarding alleged erroneous information provided to the Committee during its OL review of the Zimmer Nuclear Station (see Appendix IV). The Committee requested that the NRC Staff report to the ACRS regarding resolution of this matter (see Appendix XXXIV).

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C. Visits of Foreign Nuclear-Powered Vessels

The Chairman noted the planned visits to US ports of 2 foreign nuclear-powered naval vessels. The Committeedecided not to review these visits.

D. Scope and Timing of ACRS Report on NRC Research Programs

The Chairman reported on a discussion with Representative M. K. Udall, Chairman of the House Subcommittee on Energy and the Environment, Committee on Interior and Insular Affairs, and H. R. Myers, Special Consultant to the Committee on nuclear matters, regarding the scope and timing of the annual ACRS Report on the NRC's Research Program. The ACRS' plans for an early report to the NRC Commissioners in July, to be followed by the full annual report in January, appeared to be satisfactory.

E. Bailly Generating Station: Proposed Change in Piling Design

The Chairman noted the receipt of a request from the Commissioners for the Committee to consider the safety implications of the use of shorter than originally proposed pilings under major structures at the Bailly Generating Station, dated June 8, 1979 (see Appendix V). The Committee agreed to consider this matter during the 231st ACRS Meeting (July) (see Appendix XXXIV).

II. Meeting on the Investigation and Implications of the March 28, 1979 Accident at Three Mile Island Nuclear Plant Unit 2 (TMI-2) (Open to Public)

[Note: Richard K. Major and Ragnwald Muller were the Designated Federal Employees for this portion of the meeting.]

A. Three Mile Island Subcommittee Report

Mr. Etherington, Subcommittee Chairman, reported on the Subcommittee Meeting held in Harrisburg, PA, June 6-7, 1979. He discussed the current status of TMI-2, the interface between the NRC Staff and the Licensee, Metropolitan Edison Company, the details of the accident and the initiating events, and general operating problems that were experienced at the plant (see Appendix VI). He noted that the Subcommittee members visited the TMI site. He noted also, that although it had been suggested, no press conference was held because the Subcommittee Chairman believed that the full Committee should first determine policy regarding NRC's press conferences.

B. TMI-2 Implications Subcommittee Report

Messrs. Etherington and Siess reported for the TMI-2 Implications Subcommittee.

Mr. Etherington noted that the TMI-2 Subcommittee met on May 31 and June 1, and that 8 Committee Members and 4 Consultants attended. Proposed studies of hydrogen problems, including hydrogen generation and removal, detonation of hydrogen-oxygen mixtures inside the reactor vessel, combustion data, and a possible program of selected experiments were discussed. A question was raised concerning the possibility of hydrogen embrittlement of the TMI-2 primary system components; the NRC Staff plans to investigate this matter.

The status of boiling water reactors with respect to TMI-2 accidents, and the responses from IE Bulletin 79-08 were discussed. This Bulletin requires lic nsees of BWRs to review and report the following items:

- the applicability of TMI-2-type accidents to their plants,
- containment isolation,
- consequences of loss of main feedwater,
- level indication systems,
- operating and training procedures,
- valve positioning indication,
- the possibility of inadvertent release of radioactive fluids,
- maintenance and test procedures,
- procedures for notification of NRC of unexpected conditions,
- procedures for dealing with abnormal hydrogen generation, and
- appropriate changes in technical specifications.

The applicability of NUREG-0560, Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company, ACRS recommendations were requested of GE and of all BWR licensees. The NRC Staff plans to issue a generic PWR report similar to NUREG-0560 in July, and to take appropriate actions.

Lessons learned from the TMI-2 accident, including the combination of human error, equipment malfunction, and design deficiencies were discussed; detailed discussions will be presented to the full Committee at this meeting.

Babcock and Wilcox (B & W) discussed their efforts immediately following the TMI-2 accident and their current efforts to support the operations of B & W plants currently operating, including the development of modifications to these plants and the retraining of operators. A device, newly installed on the B & W simulator, to read out margins of saturation from pressure and temperature imput, was demonstrated. Actions taken to reduce the prbability of accidents, such as the TMI-2 accident, were discussed. Small-break LOCA phenomenology and operating guidelines were also discussed.

In response to a request by the NRC Staff, B & W provided an analysis, dated May 12, Small Break in the Pressurizer PORV with no Auxiliary Feedwater and Single Failure of the ECCS with Realistic Decay Heat. This analysis concluded that the B & W system can survive this extreme condition indefinitely with no feedwater. However, during the early stages of heat removal by boiling, the safety valves would have to pass water of equal volume to the steam generation. B & W has undertaken to verify that these valves have sufficient capacity to pass this quantity of water without significantly over-pressurizing the system.

Representatives of B & W responded to the ACRS recommendations:

- Additional analyses have been made and reported on both small break LOCAs and natural circulation.
- Recommendations have been made to B & W customers to use wide-range hot-leg temperature indicators, installation of the new B & W indicator to show the margin of saturation, and improved displays of key operating data.
- A supplementary training program is being offered.
- Matters of reactor vessel level indication, reactor coolant venting, and expanded safety research are being reviewed. (For a consultant's report regarding natural circulation, the steam generator model, pressurizer sizing, and operator awareness, see Appendix VII.)

Mr. Siess noted that the NRC Staff identified research needs relating to the TMI-2 accident, and proposed that this research be initiated in Fiscal Year 1980. This research would require a supplemental budget of approximately $29 million. This supplemental research would include

- Transient small-LOCA events, $13 1/2 million. of which $9 1/2 million would be for tests and approximately $4 million for analyses,
- Enhanced operator capability, $3.5 million,
- Plant response under accident conditions, $5 million,
- Post-mortem examination and plant recovery, $2.7 million,
- Improved risk assessment, $2.4 million, and
- Improved reactor safety, $2.2 million.

If budget supplements are not feasible, this research could be financed by reallocation of existing research funds. Commission and/or Congressonal approval would be required. If the reallocation option is adopted, lower priority programs, such as the breeder reactor, advanced converter reactor, safeguards, and most of the fuel cycle environmental research programms would be eliminated. A combination of a supplementary budget and reallocation could be adopted. A fourth alternative, reorientation of current programs to the maximum extent practicable and requesting the Electric Power Research Institute (EPRI), Department of Energy (DOE), and industry to find the remainder of the research funds, has not been given much consideration by the NRC Staff.

It is possible, when consideration is given of the lessons learned from the TMI-2 accident, that the NRC's entire research program may be redirected, and the supplementary research funds obtained in this manner.

Mr. Siess noted that when the Committee included in its Interim Report Number 3 that there was a need for additional research brought out by the TMI-2 accident, he believed that that research would be of an exploratory, rather than confirmatory, nature.

Mr. Siess noted that the NRC Staff is considering the Semiscale Facility for some of the confirmatory research; several members expressed the view that the Semiscale Facility was inadequate for this type of research.

C. Status of TMI-2

R. H. Vollmer, NRC Staff, discussed the current status of the TMI-2 plant (see Appendix VIII).

D. Evaluation of Responses to NRC Orders, IE Bulletins, and ACRS Recommendations

D. Ross, NRC Staff, provided a status report on the responses from NRC Orders and IE Bulletins emanating from the TMI-2 accident (see Appendix IX).

E. Lessons Learned from TMI-2

1. Overview

R. Mattson, NRC Staff, discussed the functions of the Lessons Learned Task Force and provided a flow chart describing the information processed by the Lessons Learned Working Group and the Bulletins and Orders Task Force with their interfaces (see Appendix X). He also identified the personnel that make up the Lessons Learned Working Group. He informed the Committee that its recommendations with regard to the TMI-2 accident would be answered by the Task Force in writing. Responsibility has been assigned to specific individuals for each recommendation.

R. Mattson said that the working group is still attempting to define the long-term objectives. These objectives will be discussed with the Committee when they are defined.

The Chairman quoted from the Committee's 1976 report on TMI-2, and noted that Metropolitan Edison has indicated that no work has been done on these recommendations or on Regulatory Guide 1.97, which deals with instrumentation to follow the course of an accident. He said that the representatives from Metropolitan Edison have indicated that no pressure was exerted from the NRC Staff to see that the recommendations were complied with.

Mr. Bender suggested that the NRC Staff note the recommendations that the Committee made with regard to TVA's Hartsville Plant. He recommended that the time to make modifications in nuclear power plants is at the Construction Permit stage, not the Operating License stage.

R. Mattson said that one of the lessons learned is that the NRC Staff can no longer stall on such matters. (For a current listing of the currently identified areas for investigation for the Lessons Learned Task Force, see Appendix XI.)

2. Operations Lessons Learned

J. Milhoan, NRC Staff, identified the areas relating to personnel and operator procedures and training for study by the Lessons Learned Operations Subgroup (see Appendix XII). He said that one of the objectives is to assure that, in the future, operators are trained adequately to cope with unanticipated transients.

S. Levine, NRC Staff, pointed out that one of the reasons that the TMI-2 accident went on for such a long time was that there were no operators present who were knowledgeable of both the engineering and the systems involved. A member suggested that one of the difficulties encountered may be that there is too much direction to detail emanating from the NRC Staff, and that one must face the reality that the people who operate a plant must be provided adequate latitude so that they can properly operate that plant.

Another member suggested that in making its determinations regarding recommendations for operation, the NRC Staff also look at the near-accidents that have occurred, such as at Brown's Ferry, Oyster Creek, and Davis Besse, as well as TMI-2.

3. Systems and Equipment Lessons Learned

R. Tedesco, NRC Staff, itemized a preliminary list of reactor systems and equipment that are being reviewed in the light of the TMI-2 accident (see Appendix XIII).

F. Metropolitan Edison's (Met. Ed.) Presentation

1. Introduction

R. C. Arnold, General Public Utilities (GPU), discussed the total experience and nuclear experience of its employees (see Appendix XIV).

2. Possible Generic Improvements

R. Keaten, Met. Ed., discussed possible generic improvements for the TMI-2 plant, including operational instrumentation, diagnostic equipment, computer capability, and equipment design (see Appendix XV). He noted that the lists of items he presented are neither complete nor are the recommendations firm. He also noted that improvements were being planned for the computer system when the accident took place.

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3. Operation of the Unit

J. Herbein, Met. Ed., noted that shift checklists and log keeping are maintained throughout the plant for a variety of support systems and equipment, including many non-safety-related systems. These logs are used to monitor equipment performance as well as plant status for indications and warning of off-normal conditions, possible equipment malfunctions, and changes in plant status (see Appendix XVI). Some of the plant equipment parameters are required by technical specifications to be read and documented. These procedures have been reviewed by the plant on-site review committee and are approved by the station superintendent. The shift relief and log entries procedure requires the shift foreman and control room operator to keep records relative to the hourly log, the control room log, and the shift foreman's log. This procedure provides guidance for individuals relieving the shift, who are required to become familiar with operations in progress, any special instruction that has been left to log duty personnel, and plant status. Operators are required to acknowledge and have an understanding and awareness of changes in plant status by signing the control room log prior to assuming the shift duty. Both the shift foreman and the control room operator keep turnover notes relative to specific actions that have occurred or will occur on their shift. These notes, passed from shift to shift, enable the operators to focus at watch relief on the other-than-normal conditions. Copies of these notes are passed daily to the station manager for his review. In the future, this process will be formalized to a greater degree.

Met. Ed. is considering the following additions to the checklists and logs:

- a critical valve and component checklists added to the shift and daily check procedure,
- a former control room operator turnover checklist and/or status board that will precisely delineate the status of such items as the major equipment out of service, primary and secondary system parameter abnormalities, system tests currently in progress, instrumentation out-of-service, electrical and mechanical maintenance in progress, off-normal indication for positions in key monitoring and control systems such as the ICS and the RPS, and

- the location of caution tags, as well as the status of key core reactivity and heat transfer parameters, and the status of significant alarms.

To increase capabilities, Met. Ed., in the future, will assign a degreed engineer on shift during plant operations.

J. Herbein said that currently monthly computer summaries of all nuclear plant LERs are received by the training department. reviewed, and those that are pertinent, are included in lectures that are part of the operator's requalification program. In the future, the summary provided by the NRC will be reviewed by the licensing group, and appropriate LERs will be sent to engineering and/or the licensing group. Following the review of the LERs, both the engineering and the licensing groups will forward recommendations to the training manager for incorporation into the operator training program. On the initial review, high priority items LERs will be forwarded to the plant operations review committee and the unit superintendent for their information and action. The licensing group will manage the task tracking system and will keep a record of the actions taken by the engineering, licensing, and training groups. Past experience has indicated that the complete LERs, rather than summaries, are needed to adequately access the significance.

Mr. Okrent requested that Met. Ed. identify the publication which lists the LERs that Met. Ed. uses to identify trouble areas.

J. Herbein discussed proposed changes to emergency administrative surveillance and operating procedures. With respect to emergency procedures, an objectives section will be added to indicate an overall direction to the operator. These procedures will also be made to reflect the NRC Bulletin items that were issued following the TMI-2 accident. In addition, a multiple plant parameter philosophy will be used in emergency procedures so that a reactor's coolant systems conditions can be judged. Further, a separate natural circulation procedure will be provided. With regard to administrative procedures, shift turnover checklists will reflect safety features, component depressurization accidents, small break LOCAs, and Unit I system change modifications.

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J. Herbein said that the training program will consist of three parts:

- The operators will receive 32 hours of instruction at a B & W simulator.

- Operators will participate in a proctored and evaluated classroom training program at TMI consisting of 108 hours of instruction, including lectures on such topics as heat transfer and fluid dynamics, reactor coolant system elevations and manometer effects, the TMI-2 transient, TMI emergency plan and procedures, simulated instrument failure drills in the Unit I control room, instructions in the proper use of the new direct NRC phone lines to Bethesda and King of Prussia, PA, procedures for surveillance and corrective maintenance, and procedures for the sign, switch, and valve alignment sheets to restore emergency equipment to the normal lineup following maintenance, and

- Each licensed operator will receive a company written and oral examination, in addition to both oral and written examinations originating with the NRC.

W. Zewe, Met. Ed., the shift supervisor at TMI-2 at the time of the accident, discussed with the Committee the conditions at the plant during the accident, and his perceptions of what was going on during the accident.

J. Herbein noted that the B & W site manager has been at the TMI site since the construction phase of Unit I, and has acted as the liason between Met. Ed. and the B & W home office in Lynchburg. He has supported Met. Ed. in preparaton of operating procedures and their review, initial start-up and test coordination, physics testing, refueling outage preparation, and has helped in activity coordination and scheduling. Throughout the day of the TMI-2 accident, he participated in technical discussions, and concurred with command team decisions. He also established communications between the TMI site and B & W , Lynchburg, VA.

4. Key Decisions Processes

R. C. Arnold, GPU, used a post-TMI-2 organization chart to outline the current decisons making process for the clean-up and recovery of the stricken unit (see Appendix XVI).

In answer to a question regarding recommended changes to handle post-accident problems, R. C. Arnold suggested that the reactor heat removal system should be inside containment rather than leading to the auxiliary building, and that it might be helpful is one of the steam generators had auxiliary piping so that it could be available to be used in the manner in which it is now being used at TMI-2 following piping modifications.

5. Additional Questions

   a. What were the initiating events leading to water hammer and plant trip?

      W. Zewe noted that the initiating event of the TMI-2 accident was the closure of the condensate polisher valves on the condensate feedwater system, and starving the suction to the condensate booster pumps and also to the feedwater pumps, which resulted in their automatic trip and an ensuing automatic trip of the turbine generator, which led to the high pressure trip of the reactor. Following the valve closure there was water present in the control air system. The water came into the system as the result of a check-valve failure in the instrument air line. The instrument air was being used, along with high pressure sluicing water, to "fluff" a clog of demineralizer resins. This water is at approximately 50 pounds higher pressure than the control air. It is believed that the water hammer was caused by the closure of the valves.

      C. Michelson, ACRS Consultant, suggested that this problem might be eliminated if the licensee were to eliminate the use of control air for service purposes.

   b. What problem in the control room ventilation caused the need for respirators in TMI-1?

      R. Dubiel, Met. Ed., said that the Unit 2 control room ventilation system does not provide automated controls. Once the system is placed in an emergency line-up, it remains so until manually changed. On the morning of the accident, this system was placed in an emergency line-up early in the transient, and should have provided for outside make-up air. While the flow rate of the make-up air was not measured, Met. Ed. believes that the control room was not at positive pressure continually through the day, and as a result, there were, from time to time, introductions of air-borne radioactivity into the control room. This occurred in both units. The primary reason for the introduction of air-borne radioactivity can be tied to the stagnant meteorological conditions that existed. Levels of some of the particulate isotopes reached 10-100 times the maximum permissible concentration. He said that he believes there was a significant safety factor if the operators wore respiratory equipment, but that the control room could not have been rendered uninhabitable. However, no analysis has been made.

c. Natural Circulation

R. Keaten discussed the conditions for loss of natural circulation at TMI-2 operating in its current mode of natural circulation heat removal (see Appendix XVII). He said that no analysis has been made for the condition tht would exist when the power-operated relief valve is open.

d. PORV Block Valve Operation

R. Keaten discussed the block valve operations for the power-operated relief valve for 4-5 hours and 10 hours after plant trip. He said that an inspection of the data obtained from the TMI-2 accident infers that the block valve was open during the first 4-5 hours of the accident. The operating staff believes that the valve was closed during the majority of this time, but there is no computer data to directly indicate the actual conditions. At 10 hours after trip, the operators were deliberately intending to open the block valve as a method of depressurizing the primary system so that the decay heat removal system could be operated. The shift supervisor gave instructions to the operator to open that valve while the supervisor was watching the building pressure indicator, and that when the block valve was opened, the pressure spike in the reactor containment building was obtained.

e. Hydrogen Bubble Calculations

R. Keaten briefly discussed the reduction of the hydrogen bubble volume in the reactor coolant system during the later stages of the TMI-2 accident (see Appendix XVIII). He noted that no correction was made for the presence of steam in the bubble. The initial calculations were made for a temperature of 280° F and a pressure of 1000 psi, which approximated the reactor conditions at the time the bubble was first noticed.

G. Babcock and Wilcox's (B & W) Presentation

1. Chronology of Events and B & W Support for TMI-2

J. H. Taylor, B & W, discussed the chronology of events and B & W actions with regard to the TMI-2 accident, an organization chart showing the relationships of the offices involved in the support, the level of B & W support chronologically, and the tasks in which B & W was involved (see Appendix XX). He discussed the communications during the accident, and noted the difficulties that were encountered because of busy telephone lines.

2. Actions Taken to Reduce the Probability of the TMI-2 Initiating Sequence

E. A. Womack, B & W, discussed the actions taken by B & W to reduce the probability of a recurrence of a TMI-2 type initiating sequence. He noted that B & W has issued a bulletin regarding high pressure system injection operation, and the conditions for maintenance of that operation. This bulletin was subsequently sent to all pressurized water reactor plants by the NRC. He discussed also the second NRC bulletin, which focused on the challenges to the pressurizer pilot-operated relief valve which is used both in B & W plants and other plants, but used differently in the B & W control system. He said that B & W believes these actions have significantly contributed to mitigating the possibility that such an event will occur again. He said that B & W is continuing to learn, and expects to continue to learn, more about the events that took place at TMI-2. He focused on the operation of the pilot-operated relief valve, showing part of the event tree that described the situation in B & W plants prior to April 21, 1979. He discussed the expected system behavior with all systems working normally, the loss of main feedwater event analysis carried out for the FSAR, the system pressure transient following the loss of main feedwater, and compared the aforesaid descriptions with those of system behavior expected after the operating changes were made on April 21, 1979 (see Appendix XXI).

3. Small-Break LOCA Phenomenology

B. M. Dunn, B & W, discussed the work done recently by B & W on small-break LOCA phenomenology (see Appendix XXII).

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During this discussion, Mr. Plesset requested that the NRC Staff provide information during the June 19-20 ECCS Subcommittee Meeting regarding the potential mechanical forces that can be developed by injecting cold water into steam, steam/water mixtures, and hot pipes.

4. Small-Break LOCA Operational Guidelines

E. R. Kane, B & W, discussed the small-break LOCA operational guidelines (see Appendix XXIII).

C. Michelson suggested that B & W also consider the maximum high pressure injection effect if a small break occurs during a semi-cooled or start-up condition. He also suggested that B & W address the problem of a small-break LOCA while the reactor is in a shut-down cooling mode.

5. Transient Experience Review and Failure Modes and Effects Analysis

E. A. Womack discussed B & W's transient experience review and failure modes and effects analysis (FMEA) including the scope for a reliability analysis of the integrated control system (ICS), the format for FMEA reporting, progess made in the FMEA system, a list of power-operated relief valve actuations for various anticipated transients, a list of reactor trips, a list of challenges to the reactor protection system, ICS hardware failures, an analysis of 246 reactor trips, ICS control response, failures of inputs to the ICS, and abnormal transient operation guidelines (see Appendix XXIV).

6. Other B & W Actions to Date

D. H. Roy, B & W, discussed other B & W actions since the TMI-2 accident, including accident prevention, accident mitigation, and accident recovery (see Appendix XXV). He also discussed B & W actions regarding ACRS recommendations.

W. Lipinski, ACRS Consultant, suggested that there is a need to consider the criteria for the primary system boundary.

D. H. Roy said that B & W is currently receiving and evaluating all LERs originating with PWRs. He said the company has enhanced its sensitivity to problems since the TMI-2 accident.

During this discussion, Mr. Siess requested that the NRC Staff provide information regarding the capacity of the emergency power supply for dual plants with swing diesels to supply power for ECCS to both units, simultaneously, in the event of loss of off-site power.

III. Meeting on Millstone Nuclear Power Station Unit 2 (Power Increase) (Open to Public)
[Note: Elpidio G. Igne was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

Mr. Etherington, Subcommittee Chairman, reviewed the background for the Licensee's request to increase the power level of the Millstone Nuclear Station Unit 2 from 2560 to 2700 MWt (see Appendix XXVI). He stressed that the issue here is that, while the site and containment have been reviewed for 2700 MWt, the ECCS analyses performed for the FSAR were carried out for a power level of 2560 MWt. However, subsequent ECCS analyses have been performed at 2700 MWt.

B. Status of the NRC Staff Review

E. L. Conner, Jr., NRC Staff, discussed the safety reviews for both Millstone 2 and Calvert Cliffs, similar plants, including the FSAR reviews, the ACRS review, environmental review, the NRC Staff actions regarding power level increases over the past 3 years, actions regarding the unresolved issues identified in the ACRS report of June 11, 1974 on Millstone 2, the original license conditions, system changes for operation at the increased power level, the systems requiring re-analysis, the Licensee Event Reports statistics for 1978, the abnormal occurrences at Millstone 2, and the occupational doses (see Appendix XXVII).

C. Licensee Presentation

R. Harris, Northeast Public Utilities, described the Millstone Plant and its site, its licensing and operating history, an overview of the request for a power increase, the cycle 3 core design, power increase methodology changes, transient and accident analyses, cycle 3 power increase modifications, shielding arrangements, and gamma surveys (see Appendix XXVIII).

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D. Committee Action

The Committee voiced no objection to the proposed increase in power level for the Millstone Nuclear Power Station Unit 2 from 2560 to 2700 MWt and instructed the ACRS Executive Director to so notify the NRC Executive Director for Operations (see Appendix XXXIV).

IV. Meeting with NRC Staff on NUREG-0531, "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants, February, 1979" (Open to Public)
[Note: Elpidio G. Igne was the Designated Federal Employee for this portion of the meeting.]

W. Hazleton, NRC Staff, discussed the history of previous NRC Staff stress corrosion cracking studies, and the current study leading to NUREG-0531, Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants, February, 1979. He discussed the scope of the study, the participants, summaries of German and Japanese experience, the causes of intergranular stress corrosion cracking, the major conclusions reached, recommendations of the study group, and NRC Staff actions and follow-on plans (see Appendix XXIX).

L. C. Shao, NRC Staff, said that within 2 years the results of the Office of Nuclear Regulatory Research studies to address the problems of stress corrosion cracking, pipe leaks, and breaks should be available.

V. Meeting with NRC Staff on Recent Operating Experience, Licensing Actions, Generic Matters, and Future Agenda (Open to Public)
[Note: Gary R. Quittschreiber was the Designated Federal Employee for this portion of the meeting.]

A. Duane Arnold: Replacement of Recirculation Nozzles

1. Summary of Allegations

a. Welder Qualifications

C. Williams, NRC Staff, said that principally there were two series of allegations regarding welding of pipe at Duane Arnold. The first involved alleged non-conformance with respect to welder qualification documentation control. There were instances where the welder's qualifications could not be established clearly. The IE investigation demonstrated that, in each of those instances, adequate related documentation resolved any serious

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problem, i.e., no circumstances were found where a welder actually had failed to meet the qualification requirements of ASME Sec. 9. In association with the repairs, there was an additional requirement that any welder involved in this work must demonstrate his proficency in a simulated restricted environment. The recirculation inlet piping, in some instances, is bounded by supports and other equipment, so that a welder had only about 15-16" clearance. The additional requirement was that he weld at least 18" of a similar joint configuration in that environment. In many instances the record was not clear whether he had established this requirement.

b. Radiographic Film Quality

C. Williams said that the second major category of allegations involved alleged misinterpretations of radiographic film quality relating to the examination of welds 2 and 6. Three radiographic inspectors, who were contracted by the licensee, alleged that they had previously rejected radiographs during their end-process accumulation data, and ultimately had found that the licensee, in further considerations, had accepted these radiographs. Two inspectors from Region III reviewed the subject radiography after the fact of acceptance by the Licensee, and concluded that in several instances the welds did not, in their best judgement, meet the requirements of ASME, Sec. 3, paragraph NB4424. IE engaged consultants to review the work, and although there is not complete agreement with the IE inspection, there is substantial agreement between the consultant's interpretation of the rejectable conditions of certain of the welds, Region III's interpretations, and those of the alleger. The Licensee has maintained that the radiographs, though having certain anomalies, are not questionable to the extent that they should be rejected. These issues were not resolved, and the matter was passed to IE Headquarters for resolution.

In answer to a question, C. Williams said that IE does not believe that there is a question of bad faith, but rather a difference in judgement as to what is acceptable under the code.

2. Resolution

V. Noonan, NRC Staff, said that his group focused on the safety of the welds from a standpoint of stresses, rather than

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the acceptability of the welds with respect to codes. In all cases, the safety of the welds was shown to meet the current FSAR allowables. He also discussed the weld designs, and the modifications of the designs of the safe ends to eliminate crevices (see Appendix XXX).

B. D. C. Cook: Cracking in Feedwater (Carbon-Steel) Pipes

E. Jordan, NRC Staff, reported on cracks in carbon-steel piping of the feedwater lines of D. C. Cook Unit 2. The plant was shut down because of unidentified leakage of 3 gpm. Inspection of the source of the leakage revealed that there were through-wall, circumferential cracks in 2 feedwater lines near the nozzle welds. Non-destructive testing in other Unit 2 nozzles identified cracking in these nozzles also. Unit 1 was shut down for refueling; inspection found that all 4 nozzles in the feedwater system of Unit 1 also had cracks. Testing was by both ultrasonic and radiographic means. Unit 1 has operated for approximately 4 years, and Unit 2 for 1 year.

E. Jordan described the piping and welds involved, the cracks discovered, and also identified the plants for which feedwater nozzles had been inspected since this matter was identified on May 25, 1979 (see Appendix XXXI). He noted that all the piping involved is Class 2 piping, radiographed at time of installation, but not subject to periodic, in-service inspection.

E. Jordan said that the Licensee is making repairs, including replacement of the elbow, rebuilding wall thickness, and grinding and cleaning the interior surface of the nozzle. The cause of the problem has not been identified. The Licensee and the vendor, W, have performed stress analyses which do not clearly indicate that the material was over-stressed. There are some evidences of fatigue. The piping on the Unit 2 steam generators is being instrumented with strain guages and accelerometers to measure differential motion to characterize the fatigue aspects. The Licensee has committed to reinspect the refueling outage. E. Jordan noted also that San Onofre has cracks in 3 nozzles.

C. Inadequacies in the Design of Nuclear Power Plant Piping to Withstand Seismically-Induced Loads

W. Russell, NRC Staff, discussed the problem in seismic analysis which led to the NRC Staff's shutting down of 5 nuclear power plants pending reanalysis and possible modifications to meet codes. The specifics of the problem addressed the methodology of

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algebraic summation and intramodal responses. The code involved in the original analysis was Shock II, a proprietary code of Stone and Webster. The safety concern was that systems that could both cause an accident and/or mitigate an accident were affected. The specific concern was that the use of an algebraic method without a time sequence could cancel loads in the analysis.

W. Russell discussed the NRC Staff's order to show cause, the status of the piping reanalysis as of June 14, 1979, the information requested by the NRC Staff, and the response of operators of reactors, both those shut-down and others (see Appendix XXXII). He noted that at Surry, the licensee is using soil-structure interaction techniques in its current analyses.

D. Beaver Valley 1: Failure of Steam Dump Valve to Close

D. A. Beckman, NRC Staff, reported on an incident occuring at Beaver Valley 1, on January 18, 1979, during which a steam dump valve failed to close (see Appendix XXXIII).

Mr. Bender offered the opinion that freezing had a part in the event, and recommended that the NRC Staff should require all plants, located in areas where freezing may be a problem, to qualify its equipment for low temperature environments.

In answer to a question, A. Schwencer, NRC Staff, said that no safety issue has been identified in this matter.

E. Future Agenda

The Committee agreed on a future agenda for ACRS reviews (see Appendix II).

With respect to the Committee's proposed report on Palo Verde, R. Baer, NRC Staff, said that he does not believe that this report is on the critical path for issuance of a Construction Permit. He said that at this time the NRC Staff resources are limited for case work because of the TMI-2 accident, and that NRC Staff members are not available now to address the open items. Construction is not planned to start until January, 1981.

In answer to a question, R. Baer said that currently the NRC Staff is dealing with applications for seven Operating Licenses, six of them already having been reviewed by the Committee, for which the NRC Staff is currently writing supplements dealing with the matters raised by the TMI-2 accident. He also said that the NRC Staff has not begun to write supplements for Construction Permits yet.

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Mr. Okrent recommended that the Committee review the NRC Staff's resolutions of TMI-2 issues. He suggested that this could be done either on a plant-by-plant basis, or as a generic review.

Mr. Fraley recommended that the Committee write a report on the lessons learned from the TMI-2 accident after it reviews the NRC Staff proposals on these matters. He suggested that the Committee require supplementary staff reports on the TMI-2 issues for both OLs and CPs.

D. Karner, Arizona Public Service Company, offered his company's opinion that as replicate plants, the Palo Verde plants are unique. He said that the company plans to file an FSAR for all 5 units in October, 1979, in preparation for the NRC Staff's OL review for Palo Verde Unit 1. Any modifications made for Units 1-3 would be incorporated in Units 4 and 5. He said that the Committee would have an opportunity to review these modifications for Unit 4 and 5 at the time of the FSAR/SER review. He said also that the company has resolved the open items in accordance with NRC Staff wishes so that the NRC Staff input could be reduced.

Mr. Okrent noted his opposition to the proposal that the Committee review modifications to Palo Verde Units 4 and 5 during the OL review of Palo Verde Units 1-3.

F. Subcommittee Activities

A schedule of future subcommittee activities was distributed to members (see Appendix III).

VI. Executive Sessions (Open to Public)
[Note: James M. Jacobs was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Reports

1. Regulatory Activities

The Committee concurred in the regulatory position of Regulatory Guide 1.9 (Rev. 2), Selection Design and Qualification of Diesel Generator Units Used as Standby (On-site) Electric Power Systems at Nuclear Power Plants.

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2. Meeting with Representatves of Japanese Regulatory Agencies

Messrs. Lawroski and Plesset reported the highlights of their meeting, in Japan on May 21-25, 1979, with representatives of Japanese regulatory agencies, including the Committee on the Examination of Reactor Safety. Discussions centered on reactor safety policy and practice, and the cooperative Emergency Core Cooling Systems (ECCS) Program.

3. Reliability and Probabilistic Assessment

Mr. Kerr, Subcommittee Chairman, noted that in January the Commissioners asked the NRC Staff to prepare detailed guidance on the use of reliability and probabilistic assessment in the regulatory process. The NRC Staff provided a draft report on this subject, that did not provide much guidance. Mr. Kerr was of the opinion that the RES interpretation of the Commissioner's request does not coincide with that of the Committee. He said that the Subcommittee will be happy to review a detailed guidance document when it is received.

4. LER Subcommittee Report

Mr. Moeller, Subcommittee Chairman, identified the general areas that have been involved in the Subcommittee's review of the LERs generated during the period 1976-1978. He said that the Subcommittee plans to have a draft report available for Committee consideration at the 231st ACRS Meeting (July). He said that the Subcommittee and its consultants have tried to critique the significant LERs and to evaluate the corrective actions that have been taken by the utilities and/or the NRC Staff.

R. F. Fraley suggested that the possible use of LERs to identify precursors of potential accidents should be investigated.

B. TMI-2 Implications

Members discussed the TMI-2 accident, identifying those areas of concern that should be included in the Committee's next report on the implications of the accident. Items identified included hydrogen formation, hydrogen explosions, hydrogen-oxygen recombination, steam-water explosions, steam-fuel cladding interactions, core melt implications, and analytical studies to consider scenarios beyond Class 8 accidents.

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Mr. Okrent offered to prepare a draft report for consideration at the 231st ACRS Meeting (July).

Mr. Kerr suggested tht discussions of this type should be scheduled for Thursday and Friday sessions of the meeting, not Saturday.

C. Action on Plants Reviewed by ACRS but for which no OLs Have Been Issued

The Committee agreed to request that the NRC Staff brief the Committee regarding additional requirements that the NRC will impose on those plants for which the Committee has written favorable OL reports but for which Operating Licenses have not been issued (see Appendix XXXIV).

D. Sequoyah Nuclear Power Plant Units 1 and 2

The Committee agreed to defer further action on its review of the application for an Operating License for the Sequoyah Nuclear Power Plant Units 1 and 2 until the NRC Staff makes known its plans (via a SER supplement) regarding this plant (see Appendix XXXIV).

E. Palo Verde Nuclear Generating Station Units 4 and 5

The Committee agreed to defer further action on its review of the application for a Construction Permit for the Palo Verde Nuclear Generating Station Units 4 and 5 until the NRC Staff makes known its plans regarding this plant (see Appendix XXXIV).

F. Three Mile Island Nuclear Station Unit 1: Return to Power

The Committee expects that the NRC Staff will keep it informed regarding the Applicant's proposals and the NRC Staff's intentions regarding the conditions that must be satisfied prior to the return to operation of Three Mile Island Nuclear Station Unit 1. The Committee agreed to review this matter with special attention to the interactions between Unit 1 and the recovery operations at Unit 2. This review is scheduled for the 232nd ACRS Meeting (August) (see Appendix XXXIV).

G. TMI Implications Subcommittee Assignment

The Committee agreed that the TMI Implications Subcommittee should study NRC Nuclear Power Plant Siting Practices in the near future, but did not set a date for the study. Included in the study will be the interrelations between siting and Class 9 accidents, and consideration of realistic analyses for greater than Class 8 accidents.

H. Testimony Before TMI-2 Investigatory Groups

The Chairman noted the possibility that the Committee may be asked to testify before one or more of the groups investigating the TMI-2 accident. It was the consensus of the Committee that it would be appropriate to consider the underlying causes and factors leading to this accident. The Committee agreed to devote part of the 231st ACRS Meeting (July) to consideration of the underlying factors relating to the TMI-2 accident.

I. Subcommittee Press Conferences

The Committee agreed that Subcommittee Chairmen should not schedule press conferences following subcommittee meetings. It was proposed that comments to the press by Subcommittee Chairmen et. al. should be limited to factual information concerning the Committee's review process.

The Chairman brifly discussed a proposed letter (see Appendix XXXV) regarding the scope of interviews, speeches, etc. by ACRS Members.

J. ACRS As a Research "User"

The Committee deferred action on a proposed request that the ACRS be considered a "user" of RSR information and agreed to discuss at the 231st ACRS Meeting (July) the NRC Staff requirements for an identified user in order to obtain authorization for research.

K. NRC Research Program Mid-Year Report

Mr. Siess noted that this proposed report is being prepared at the request of the Commissioners to be used as a basis for the Fiscal Year 1981 research budget request. July 1 will be the deadline for information or comments regarding this report. The report will be discussed, finalized, and transmitted both orally and in writing to the Commissioners during the 231st ACRS Meeting.

The Committee requested that the ACRS Staff obtain H. Denton's comments on the proposed supplement to the Fiscal Year 1980 research budget for technical assistance programs related to the TMI-2 accident. Comments on the proposed supplement will be included in the above report.

L. Pipe Crack Studies

It was agreed that the Committee will give further consideration to a letter to the NRC recommending reexamination of primary system piping integrity, taking into account recent nuclear plant experience (e.g. stress corrosion cracking, etc.). Mr. Okrent was requested to prepare a draft of this proposed report for consideration during the 231st ACRS Meeting (July).

M. TMI-2 Bulletins and Orders Subcommittee

The Committee agreed that a new Subcommittee should be named to consider the responses of vendors/utilities to IE Bulletins and NRC Orders resulting from the TMI-2 accident; Mr. Mathis, Chairman, and Messrs. Bender, Etherington, Lawroski, Plesset, and Shewmon have been assigned.

The ACRS Staff was requested to obtain the NRC Staff's "report" on auxiliary feedwater system reliability for nuclear power plants.

N. Use of Consultants

The Chairman informed the Committee that its consultant, C. Michelson, can devote up to 1/2 of his time to TMI-2 matters for the next several months.

It was the consensus of the Committee that formal, final reports from its TMI-2 consultants need not be required, but that written comments on appropriate matters identified by the consultants, should be prepared.

O. Scope of ACRS Activities

The Committee offered no comments regarding R. F. Fraley's memo of June 12, 1979, Follow-Up Items for ACRS Consideration (see Appendix XXXVI).

Mr. Siess noted his concern that the Committee may be getting outside its expertise when opinions of consultants have to be depended upon entirely.

P. ACRS Reports and Letters

1. Summary Comparison of Stainless Steel and Zircaloy Fuel Rod Cladding

The Committee approved a reply to Commissioner Gilinsky's letter regarding a comparison of the merits of stainless steel vs. Zircaloy for nuclear fuel cladding (see Appendix XXXVII).

2. ACRS Action on Proposed Revisions of Regulatory Guides

The Committee approved a memorandum to the Executive Director for Operations informing him that the Committee concurs in the regulatory position of Regulatory Guide 1.9 (Rev. 2), Selection, Design, and Qualificatgion of Diesel-Generator Units Used as Standby (On-Site) Electric Power Systems at Nuclear Power Plants (see Appendix XXXVIII).

3. Letter to D. L. Basdekas

The Committee prepared an acknowledgement to a letter received from D. L. Basdekas, NRC Staff, regarding the safety implications of the TMI-2 accident regarding reactor control systems (see Appendix XXXIX). This matter was referred to the ACRS Power and Electrical Subcommittee for follow-up in considering the implications of the TMI-2 accident.

The 230th ACRS Meeting was adjourned at 4:00 p.m., Saturday, June 16, 1979.

APENDIXES TO THE
230TH ACRS MEETING
JUNE 14-16, 1979

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230th ACRS Meeting

Meeting Dates:
June 14-16, 1979

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
James M. Jacobs, Technical Secretary
Herman Alderman
Andrew L. Bates
David E. Bessette
John Bickel
Paul A. Boehnert
Sam Duraiswamy

Elpidio G. Igne
David H. Johnson
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
Peter Tam
Dwight W. Underhill
Hugh E. Voress
Harold Walker
Gary Young

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230th ACRS Meeting

Meeting Dates:
June 14-16, 1979

CONSULTANTS

I. Catton
C. Michelson
W. Lipinski

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NRC ATTENDEES

230TH ACRS MEETING

Thursday, June 14, 1979

Div. of Project Management

Robert Baer
Walter Hauss
Fred Allenspach
John F. Stolz
I. A. Peltier
D. O'Reilly
C. Heltemes, Jr.

Div. of Systems Safety

T. M. Novak
R. P. Denise
B. W. Slercn
M. L. Boyle
R. Mattson
R. Tedesco

Standards Development
E. C. Wenzinger

Research
R. DiSalvo

SCSB
A. Taboada

S&P
W. T. Russell

J. M. Grant

Div. of Operating Reactors

R. H. Vollmer
P. S. Kapo
D. C. Iann
T. A. Kevevn
W. S. Hazelton
C. Y. Cheng
V. S. Noonan
T. H. Liu
J. R. Fair
J. Martore
R. A. Hermann
T. A. Kevevn
R. A. Hermann

Nuclear Reactor Regulation

R. A. Caora
D. Ross
M. W. Hodges
D. Skovholt

Management & Program Analysis

C. E. Shortt

Office of Inspection and Enforcement

H. S. Wong
E. L. Jord
C. Williams, Region III
J. J. Burns, Region 6

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APPLICANT ATTENDEES

230TH ACRS MEETING

June 14, 1979

Northeast Utilities
R. M. Kacich
F. Farrell
R. D. Hart
P. V. Gurnex
E. Randolph Foster

APS

A. Gelin
D. B. Kauman

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PUBLIC ATTENDEES

230TH ACRS MEETING

June 14, 1979

J. East, VEPCO
A. R. Rozy, NUSCO, 423 Baldwin Av., Meriden - Conn
F. L. Carpentino, Combustion Engineering, Durham, Conn
R. T. Harris, NUSCO, Weathersfield, Conn.
R. R. Mills, Bomcustion Engr., Bloomfield, CT
Mr. Layer, BBR, 7735 Old Georgetown Rd., Bethesda, MD
N. Shirley, GE, Gaithersburg, MD
A. Kimmins, Wash. Public Power Supply Systems, Richland, WA
J. B. Hoch, Pacific Gas & Electric Co., San Francisco, CA
F. Stetson, AIF, Rockville, MD
E. Fuller, S. Levy, Inc., Campbell, CA
R. Adamson, McGraw-Hill, Skillman, NJ
Kunihiro, Ota, KEPCO, 1725 K St., NW Wash., DC
M. H. Furbush, PG&E, 14190 Amherst St., Los Altos Hills, CA
Gautman Sen, Public Service Electric & Gas, Newark, NJ
G. Adamantiades, EPRI, Wash., DC
K. Tortino, Lowenstein-Newman, Wash., DC
W. H. House, II, Bechtel Power Corp., Frederick, MD
R. Borsum, B&W, Derwood, MD
G. A. Blanc, PG&E, 11513 Falls Road, Potomac, MD
N. M. Johnson, S&W, Boston, MA
P. Seiffer, Dave, Purcell, Jefferson, NJ
P. J. Kochis, Bechtel Power Corp., 12242 Erhison Rd., Ellicott City, MD
Ed. Fuler, S. Levy, Inc., Campbell, CA
J. East, VEPCO, Richmond, VA
A. L. Millet, Ottawa News Service, Wash., DC 20003

W. Williams, Jr., S.C. Public Service Auth.
Osmund W. Dixon, S.C. Elec & Gas Co.
B. Boatright, Pickard, Lowe & Garrick

J. E. McEwen, Stafco
M. An. Stafco, Inc.
James Bloom

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NRC STAFF ATTENDEES

230TH ACRS MEETING

June 15, 1979

Div. of Project Management

R. Baer
F. Williams
S. Varga

Div. of Operating Reactors

D. Allison
S. MacKay
D. Wigginton

I&E
D. Beckman, Region I

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INVITED ATTENDEES

230TH ACRS MTG.

June 15, 1979

General Public Utilities Service Company

E. Wallace
R. Reaten
R. Arnold
H. Dieckamp

Metropolitan Edison

G. Miller
R. Dubrit
W. Zlu
J. Hertium

Public Service Electric & Gas

G. Seu

SCPSA

W. Williams, Jr.

B&W

J. H. Taylor
D. H. Roy
E. A. Womack
B. Durm
J. J. Cudlin
E. R. Kane

1623 ~~152~~ 184

1623 ~~217~~

PUBLIC ATTENDEES

230TH ACRS MTG.

June 15, 1979

A. Yuspeh, Shaw, Pittman, et al
S. Wynkoop, McGraw Hill, ARlington, VA
J. Maffre, AIF
K. Layer, BBR, 7735 Old Georgetown Road, Bethesda, MD
Kunihiro, Ota, KEPCO, 1725 K St, NW, Wash., DC
A. Kimmins, WPPSS, Richland, WA
Hiroyoshi Hamada, The Tokyo Electric Power, Wash., DC
R. Adamson, McGraw Hill, Skillman, NJ
R. G. Cockrell, WPPSS, Richland, WA
D. B. Karner, Arizona Public Service Co., Phoenix AR
M. H. Schwartl, Pickard, Lowe & Garrick, 1200 18th St., NW, Suite 612, Wash., DC 20036
Bob Adamson, McGraw-Hill, Skillman, NJ
D. Harbrecht, Pittsbrugh, Press, Wash., DC
B. Montgomery, Bechtel

June 16, 1979
R. H. Leyse, EPRI, Rockville, MD

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APPENDIX II

ACRS FUTURE AGENDA

ACRS MEETING PROJECT	TYPE OF REVIEW	REACTOR VENDOR	SER ISSUE DATE
July			
Bailly Generating Sta.	Piling Design		
August			
Three Mile Island 1	Restart		
September			
None			
October			
None			
November			
Shoreham	OL	GE	10/1/79
LaSalle 1&2	OL	GE	10/1/79

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1623 ~~219~~