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MINUTES OF THE 344TH ACRS MEETING
DECEMBER 15-16, 1988

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DECEMBER 15-16, 1988

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For the Nuclear Regulatory Commission.

Jose A. Calvo,

Director, Project Directorate—IV, Division of Reactor Projects—III, IV, V and Special Projects Office of Nuclear Reactor Regulation.

[FR Doc. 88-28533 Filed 12-12-88; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Revised Meeting Agenda

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on December 15–17, 1988, in Room P–114, 7920 Norfolk Avenue, Bethesda, Md. Notice of this meeting was published in the Federal Register on December 1, 1988.

Thursday, December 15, 1988, Room P–114, 7920 Norfolk Avenue, Bethesda, MD

8:30–8:45 A.M.: Chairman's Comments (Open)—The ACRS chairman will report briefly regarding items of current interest.

8:45–10:45 A.M.: Sodium Advanced Fast Reactor (SAFR) (Open)—Report of ACRS subcommittee chairman regarding the review of this type of standardized nuclear plant (initial session). Meeting with representatives of NRC and DOE.

11:00–12:00 Noon: Equipment Qualification-Risk Scoping Study (Open)—Report of ACRS subcommittee chairman regarding review and comment of this Scoping Study. Meeting with representatives of NRC Staff, as appropriate.

1:00–2:00 P.M.: Equipment Qualification-Risk Scoping Study (Open)—Continue meeting/discussion of Equipment Qualification-Risk Scoping Study.

2:00–4:15 P.M.: Containment System (Open)—Comments by ACRS subcommittee chairman regarding proposed NRC recommendations for containment performance and improvements for BWR Mark I containment. Meeting with representatives of NRC Staff, as appropriate.

4:15–6:30 P.M.: NRC [illegible] Safety Goals (Open)—Remarks by ACRS subcommittee chairman regarding proposed NRC Staff [illegible] implementation of NRC's Safety Goal Policy. Meeting with representatives of the NRC Staff, as appropriate.

Friday, December 16, 1988, Room P–114, 7920 Norfolk Avenue, Bethesda, MD

8:30–9:30 A.M.: Meeting with Director, NRC Office of Nuclear Regulatory Research (Open)—Briefing and discussion of items of mutual interest.

9:30–12:00 Noon: Emergency Core Cooling Systems (Open)—Comments by ACRS subcommittee chairman regarding proposed NRC Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology proposed for use with best-estimate ECCS evaluation models. Meeting with representatives of NRC Staff.

12:00–12:30 P.M.: Future ACRS Activities (Open)—Discuss anticipated subcommittee activities and discuss items proposed for consideration by the full Committee, including Split of responsibilities between ACRS/ACNW.

1:45–3:00 P.M.: Meeting with Director, Office of Governmental and Public Affairs (Open/Closed)—Briefing regarding US–USSR Exchange of Safety-Related Information.

(Portions of this session will be closed as necessary to discuss information provided in confidence by a foreign source.)

3:15–4:45 P.M.: Operator Requalification (Open)—Briefing regarding lessons learned from implementation of revised operator requalification methodology (Draft Examiner Standard 601).

4:45–6:30 P.M.: Nuclear Power Plant Operations (Open)—Briefing regarding action taken in response to LaSalle Nuclear Station core power oscillation transient.

Saturday, December 17, 1988, Room P–114, 7920 Norfolk Avenue, Bethesda, MD

8:30–9:00 A.M.: Election of ACRS Officers for CY 1989 (Closed)—Discussion and election of ACRS chairman and vice-chairman for CY 1989 and Member-at-Large for ACRS Planning Subcommittee and report regarding Appointment of New Members.

(This session will be closed to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy.)

9:00–12:00 Noon: Preparation of ACRS Report (Open)—Discuss proposed ACRS reports.

1:00–2:00 P.M.: ACRS Subcommittee Activities (Open)—Report of the December 1, 1988 meeting regarding International Seminar on Quality and Quality Control and report of visit to E. Fermi Nuclear Plant [illegible]

2:00–3:00 P.M.: ACRS Activities (Open)—Discuss area of interest/activity.

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 discuss [illegible] the release of which would represent a clearly unwarranted invasion of personal privacy [5 U.S.C. 552b(c)(6)], to discuss information provided in confidence by a foreign source [5 U.S.C. 552b(c)(4)], and to discuss Proprietary Information applicable to matters being considered [5 U.S.C. 552b(c)(4)].

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley (telephone 301/492–4049), between 8:15 A.M. and 5:00 p.m.

Dated: December 7, 1988.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 88-28649 Filed 12-12-88; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50-334 and 50-412]

Duquesne Light Co. et al.; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. DPR-66 and NPF-73 issued to Duquesne Light Company, et. al. (the licensee), for operation of the Beaver Valley Power Station, Units 1 and 2, located in Beaver County, Pennsylvania.

Environmental Assessment

Identification of Proposed Action

The proposed amendments would exclude all containment isolation spring- and weight-loaded check valves not subject to Type-C testing from the lift test requirements of Specification 4.6.3.1. Other tests currently specified by regulation and the Technical Specifications (TS) are not affected by these amendments.

The proposed amendments are in accordance with Duquesne Light Company's application dated June 22, 1988.

The Need for the Proposed Action

The proposed changes to the TS are needed since the lift tests can be shown to be unnecessary for the valves involved. Performing unnecessary tests only increases resource expenditures and increases occupational radiological exposure for no gain in safety.

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The principal alternative would be to deny the requested amendment. This would not reduce environmental impacts of plant operation and would result in reduced operational flexibility.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the "Final Environmental Statement Related to the Operation of the Fort Calhoun Unit 1", dated August 1972.

Agencies and Persons Consulted

The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

Finding of No Significant Impact

The Commission has determined not to prepare an environmental impact statement for the proposed license amendment.

Based upon the forgoing environmental assessment, we conclude that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the application for amendment dated September 2, 1988, as supplemented November 22, 1988, which are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20555 and at the W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

Dated at Rockville, Maryland, this 30th day of November, 1988.

For The Nuclear Regulatory Commission.

Paul W. O'Connor,

Acting Director, Project Directorate—IV, Division of Reactor Projects—III, IV, V and Special Projects, Office of Nuclear Reactor Regulation.

[FR Doc. 88-28120 Filed 12-6-88; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Meeting Agenda

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on December 15–17, 1988, in Room P-114, 7920 Norfolk Avenue, Bethesda, Md. Notice of this meeting was published in the Federal Register on October 20, 1988.

Thursday, December 15, 1988

8:30 a.m.–8:45 a.m.: Comments by ACRS Chairman (Open)—The ACRS Chairman will report briefly regarding items of current interest.

8:45 a.m.–10:15 a.m.: Operator Requalification Program (Open)—Briefing regarding lessons learned from implementation of revised operator qualification methodology (Draft Examiner Standard 601).

10:30 a.m.—12:30 p.m.: Sodium Advanced Fast Reactor (Open)—Review of proposed standardized type of nuclear plant. Representatives of the Department of Energy and the NRC Staff will participate.

1:30 p.m.–3:30 p.m.: Equipment Qualification-Risk Scoping Study (Open)—Review and comment on NRC-sponsored Equipment Qualification-Risk Scoping Study including consideration of peer review comments.

3:45–6:30 p.m.: Quantitative Safety Goals (Open)—Review and comment on proposed NRC Staff plan for implementation of NRC Quantitative Safety Goals.

Friday, December 16, 1988

8:30 a.m.–9:00 a.m.: Future ACRS Activities (Open)—Discuss anticipated ACRS subcommittee activity and topics proposed for consideration by the full Committee.

9:00 a.m.–10:00 a.m.: Nuclear Safety Research (Open)—Briefing by and discussion with the Director, Office of Nuclear Regulatory Research, NRC, regarding aspects of the safety research program of interest to the ACRS and RES.

10:15 a.m.–12:15 p.m.: Emergency Core Cooling (Open)—Review and comment on proposed code Scaling Applicability and Uncertainty proposed for use with best-estimate ECCS evaluation models.

1:45 p.m.–2:45 p.m.: US-USSR Exchange of Information (Open/Closed)—Briefing regarding agreement to exchange safety-related information related to the design, operation, etc. of nuclear reactors.

Portions of this session will be closed as necessary to discuss information provided in confidence by a foreign source.

2:45 p.m.–4:45 p.m.: Containment Systems (Open)—Review and comment on recommendations for containment performance requirements and specific aspects of the BWR Mk I containment.

4:45 p.m.–6:15 p.m.: Reactor Operating Experience (Open/Closed)—Briefing and discussion of lessons learned from power oscillation transient at the LaSalle Nuclear Power Station.

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

Saturday, December 17, 1988

8:30 a.m.–9:00 a.m.: Selection of ACRS Officers (Closed)—Discuss qualifications of nominees proposed for election as Committee officers for Calendar Year 1989.

This session will be closed to discuss information the release of which would represent a clearly unwarranted invasion of personal privacy.

9:15 a.m.–12:00 Noon: Preparation of ACRS Reports (Open)—Discuss proposed reports to NRC regarding issues considered during this meeting.

1:00 p.m.–2:15 p.m.: ACRS Subcommittee Activities (Open)—Briefing and discussion regarding the status of activities assigned to cognizant subcommittees including thermal-hydraulic phenomena, E. Fermi plant visit and international conference on quality and quality assurance.

2:15 p.m.–3:00 p.m.: Regulatory Process (Open)—Discuss proposed review and discussion of NRC regulatory philosophy.

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on October 27, 1988 (53 FR 43487). 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 prepaid telephone call to the ACRS Executive Director, Mr. Raymond 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 discuss information the release of which would represent a clearly unwarranted invasion of personal privacy [5 U.S.C. 552b(c)(6)], to discuss information provided in confidence by a foreign source [5 U.S.C. 552b(c)(4)], and to discuss Proprietary Information applicable to matters being considered [5 U.S.C. 552(b)(4)].

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley (telephone 301/492-8049), between 8:15 a.m. and 5:00 p.m.

Date: December 2, 1988.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 88-28122 Filed 12-6-88; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 40-333]

James A. Fitzpatrick Nuclear Power Plant; Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR-59, issued to the Power Authority of the State of New York (the licensee), for operation of James A. FitzPatrick Nuclear Power Plant, located in Oswego County, New York.

By application dated November, 9, 1988, the licensee requested that the primary containment leak rate test requirements described in Technical Specification (TS) Section 4.7.A.2.a(10) and Section 4.7.A.2.f be amended for the 1988 refueling outage on an emergency basis under the provisions of 10 CFR 50.91(a)(5). The application stated that these TS changes were necessary to allow plant startup from the 1988 refueling outage without performing a Type A primary containment integrated lead rate test (ILRT) or a Type A, B, or C leak rate test (LRT) following replacement of the high pressure coolant injection (HPCI) system turbine exhaust line manual block valve, as explained below.

Section 4.7.A.2.a(10) of the TS and Section III.A.6(b) of Appendix J to 10 CFR Part 50 require that if two consecutive periodic Type A tests (ILRTs) fail to meet the acceptance criteria, a Type A test must be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria. When it was determined that the cause of the failure of tests, conducted in 1982, 1985 and 1987, to meet the acceptance criteria for the "As Found" condition was due to excessive combined leakage from several containment isolation valves, the licensee concluded that the most effective approach to eliminate the excessive leakage was to implement a Corrective Action Plan (CAP) using guidance given in Information Notice 85-71 dated August 22, 1985. In this CAP the licensee determined that 33 containment isolation valves, which previously were identified as having excessive leakage, should be replaced (21 during the 1988 refueling outage and 12 during the 1990 refueling outage). The 12 valves scheduled to be replaced during the 1990 refueling outage have acceptable leakage rates based on the test performed during the 1988 refueling outage.

As part of the CAP, the licensee replaced the HPCI turbine exhaust line manual block valve to the suppression chamber (23-HPI-11). TS 4.7.A.2.f and Section IV.A of 10 CFR Part 50, Appendix J require that following replacement of a component which is part of the primary containment boundary, either a Type A, Type B, or Type C LRT, as applicable for the area affected, must be conducted and the appropriate acceptance criteria met. Since an isolation volume for the resulting welds on the primary containment side of the valve could not be attained, the licensee conducted 100% radiography and dye penetrant tests on the welds to verify the structural integrity of the welds, in lieu of a Type A, B, or C test.

Based on an evaluation of the licensee's CAP, the alternate tests performed to ensure system integrity, and the implementation of an improved valve maintenenace program, an exemption to the requirements of Section III.A.6(b) and Section IV.A of Appendix J to 10 CFR Part 50 was issued to the licensee by letter dated November 16, 1988. The exemption was noticed on November 25, 1988 (53 FR 47784).

When it was recognized that the licensee had inadvertently failed to identify that a TS amendment would be required in addition to the exemption, the licensee submitted the necessary amendment request dated November 9, 1988. Based on an evaluation of the amendment application (which is virtually identical to the exemption), a temporary waiver or compliance from the provisions of TS Section 4.7.A.2.a(10) and Section 4.7.A.2.f was issued by the NRC staff to the licensee by letter dated November 18, 1988. This allowed plant startup from the refueling outage without compliance with these TS requirements pending the NRC staff's review of the licensee's amendment request. In order to complete its review in an expeditious manner, yet allow for public comment, the NRC is processing the licensee's amendment proposal on an exigent basis under the provisions of 10 CFR 50.91(a)(6).

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

These proposed changes do not increase the probability or consequences of an accident previously evaluated. The containment leakage rates assumed in the Final Safety Analysis Report (FSAR) require that the valves which perform containment isolation functions, as well as the primary containment itself, exhibit superior leak rate characteristics. When the licensee found that the limit was frequently being exceeded, a CAP was initiated. The CAP involved a detailed analysis of the causes for exceeding the allowable limit, determination that the primary cause was valve seat leakage, identification of the valves which were causing the problems, determination of the best method to correct the problem valves, and implementation of the resulting plan to ensure that the leak limits are not exceeded in the future. It was determined that over time some of these valves exhibited gradual degradation to the point where their combined seat leakage rate, when added to the leakage rate resulting from the previous Type A test, caused the limit to be exceeded. This resulted in the determination that many valves needed to be replaced, some during the 1988 refueling outage and other during the 1990 refueling outage. All of these valves were tested prior to the end of the outage with satisfactory results. Using this program, the intergrity of the primary containment has been restored so that it is reasonable to assume that the design leakage rate limits of the

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

Revised: December 12, 1988

SCHEDULE AND OUTLINE FOR DISCUSSION
344TH ACRS MEETING
DECEMBER 15-17, 1988
BETHESDA, MARYLAND

Thursday, December 15, 1988, Room P-114, 7920 Norfolk Avenue, Bethesda, Md.

1) 8:30 - 8:45 A.M. — Chairman's Comments (Open)
- 1.1) Opening remarks (WK)
- 1.2) Items of current interest (WK/RFF)

2) 8:45 - 10:45 A.M. — Sodium Advanced Fast Reactor (SAFR) (Open)
- 2.1) Report of ACRS subcommittee chairman regarding the review of this type of standardized nuclear plant (initial session) (DAW/MME)
- 2.2) Meeting with representatives of NRC and DOE

10:45 - 11:00 A.M. — BREAK

3) 11:00 - 12:00 Noon — Containment Systems (Open)
- 3.1) Comments by ACRS subcommittee chairman regarding proposed NRC recommendations for containment performance and improvements for BWR Mark I containment (DAW/MDH)
- 3.2) Meeting with representatives of NRC Staff, as appropriate

12:00 - 1:00 P.M. — LUNCH

3) 1:00 - 2:00 P.M. — Containment Systems (Open)
- 3.3) Continue meeting/discussion of Containment Systems

4) 2:00 - 4:15 P.M. — Equipment Qualification-Risk Scoping Study (Open)
(3:00-3:15 - BREAK)
- 4.1) Report of ACRS subcommittee chairman regarding review and comment of this Scoping Study (CJW/SD)
- 4.2) Meeting with representatives of NRC Staff, as appropriate

5) 4:15 - 6:30 P.M. | NRC Quantitative Safety Goals (Open)

5.1) Remarks by ACRS subcommittee chairman regarding proposed NRC Staff plan for implementation of NRC's Safety Goal Policy (DAW/MDH)

5.2) Meeting with representatives of the NRC Staff, as appropriate

Friday, December 16, 1988, Room P-114, 7920 Norfolk Avenue, Bethesda, Md.

6) 8:30 - 9:30 A.M. | Meeting with Director, NRC Office of Nuclear Regulatory Research (Open)

6.1) Briefing and discussion of items of mutual interest (CPS/SD)

7) 9:30 - 12:00 Noon (10:00-10:15 - BREAK) | Emergency Core Cooling Systems (Open)

7.1) Comments by ACRS subcommittee chairman regarding proposed NRC Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology proposed for use with best-estimate ECCS evaluation models (DAW/PAB)

7.2) Meeting with representatives of NRC Staff

7.3) Discuss ACRS report to NRC (tentative)

8) 12:00 - 12:30 P.M. | Future ACRS Activities (Open)

8.1) Discuss anticipated subcommittee activities (MWL/RFF)

8.2) Discuss items proposed for consideration by the full Committee (WK/RFF)

8.3) Split of responsibilities between ACRS/ACNW (WK/RFF)

12:30 - 1:45 P.M. | LUNCH

9) 1:45 - 3:00 P.M. | Meeting with Director, Office of Governmental and Public Affairs (Open/Closed)

9.1) Briefing regarding US-USSR Exchange of Safety-Related Information

	(Note: Portions of this session will be closed as necessary to discuss information provided in confidence by a foreign source.)
3:00 - 3:15 P.M.	BREAK
10) 3:15 - 4:45 P.M.	<u>Operator Requalification</u> (Open) 10.1) Briefing regarding lessons learned from implementation of revised operator requalification methodology (Draft Examiner Standard 601) (FJR/HA)
11) 4:45 - 6:30 P.M.	<u>Nuclear Power Plant Operations</u> (Open) 11.1) Briefing regarding action taken in response to LaSalle Nuclear Station core power oscillation transient (WK/PAB)

<u>Saturday, December 17, 1988, Room P-114, 7920 Norfolk Avenue, Bethesda, Md.</u>

12) 8:30 - 9:00 A.M.	<u>Election of ACRS Officers for CY 1989 (Closed)</u> 12.1) Discussion and election of ACRS chairman and vice-chairman for CY 1989 (WK/NSL) 12.2) Discussion and election of Member-at-Large for ACRS Planning Subcommittee (WK/NSL) 12.3) Report regarding Appointment of New Members (Note: This session will be closed to discuss information the release of which would constitute a clearly unwarranted invasion of personal privacy.)
13) 9:00 - 12:00 Noon	<u>Preparation of ACRS Report</u> (Open) 13.1) Discuss proposed ACRS report on: 13.1-1) Equipment Qualification Risk-Scoping Study (CJW/SD)
12:00 - 1:00 P.M.	LUNCH

14) 1:00 - 2:00 P.M.	ACRS Subcommittee Activities (Open) 14.1) Report of the December 1, 1988 meeting regarding International Seminar on Quality and Quality Control (CPS/EGI) 14.2) Visit to E. Fermi Nuclear Plant (WK/PAB)
15) 2:00 - 3:00 P.M.	ACRS Activities (Open) 15.1) Discuss areas of interest/ activity in W. Kerr memo dated 10/4/88

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MINUTES OF THE
344TH ACRS MEETING
DECEMBER 15-16, 1988

The Advisory Committee on Reactor Safeguards (ACRS) met on December 15-16, 1988 at 7920 Norfolk Ave., Bethesda, Md. The purpose of this meeting was to conduct the discussions and perform the actions described in the attached agenda. The meeting was chaired by Dr. Kerr.

All of the discussions were held in open session except for brief discussions during which the ACRS officers for CY 1989 were elected and the appointment of new members was discussed.

A transcript of selected portions of the meeting was kept and is available in the NRC Public Document Room. [Copies of the transcript are also available for purchase from the Heritage Reporting Corporation, 1220 L St., N.W., Washington, D.C. 20005.]

I. Chairman's Report (Open)

[Note: Mr. R. F. Fraley was the Designated Federal Official for this portion of the meeting.]

Dr. Kerr began the meeting with a brief summary of the planned agenda and the procedures under which the meeting discussions were being conducted. Dr. Kerr noted that Mr. Lockard would be retiring in the near future and expressed the Committee's appreciation for Mr. Lockard's excellent work.

II. Sodium Advanced Fast Reactor (SAFR) (Open)

[Dr. M. El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Mr. Ward, Advanced Reactor Designs Subcommittee Chairman, briefed the Committee regarding the Subcommittee's discussions of the SAFR design. He indicated that the NRC Staff has reviewed a preliminary safety information document (PSID) which was provided by DOE. The Staff's review is considered a preapplication review for the purpose of providing guidance early in the design process on the acceptability of the SAFR design. The issuance of the draft safety evaluation report (SER) does not constitute an approval of the SAFR design. The Staff's review has been performed under the guidance of the Commission's advanced reactor, severe accident, safety goal, and standardization policy statements. Mr. Ward indicated that he expects the ACRS will write a report to the Commission on this subject at the January 1989 ACRS meeting. Mr. Ward also noted that DOE has made the selection between SAFR and Power Reactor Inherently Safe Module (PRISM) designs and has selected the PRISM concept with GE as the prime contractor.

Mr. Landry, NRC/RES, indicated that the staff review is expected to be completed by January 1989 with the CRGR review in January 1989 and recommendations to the Commission by end of January 1989.

The SAFR has been designed by Rockwell International in cooperation with Bechtel, Inc. and Combustion Engineering, with Argonne National Laboratory providing major analytical and testing support. The SAFR conceptual design utilizes one or more independent power paks with each power pak consisting of four modules. The individual modules produce 900 MWt (350 MWe). The design consists of a sodium-cooled reactor system that transports heat via the primary coolant through two intermediate coolant loops to two steam generators (i.e., a two-loop design). The power pak reactor system employs a compact pool-type design. It is fueled by a stainless-steel clad, sodium-bonded metallic alloy of U-Pu-Zr. The design relies on passive reactor shutdown and decay heat removal systems.

SAFR will be designed for an SSE of 0.3g and OBE of 0.1g. DOE believes that on this basis the design will be acceptable for about three-quarters of the currently identified potential sites. DOE has proposed a siting design basis source term based on radioactive materials released from melting a single fuel subassembly rather than the traditional TID-14844 releases. No conventional containment building and no requirements for preplanned off-site emergency evacuation are proposed.

Dr. Remick questioned the seismic specifications for the SAFR design (OBE = 0.1g and SSE = 0.3g) from the perspective of the NRC regulations for the establishment of the OBE at 50% of the SSE value. Dr. Siess noted that there has been an attempt to eliminate the 50% requirement based on probabilistic analysis.

Mr. Landry stated that the overall objective of the SAFR program is to develop a conceptual design that minimizes plant cost and maximizes inherent safety. Other objectives are minimum potential for severe accidents and the elimination of the need for off-site evacuation planning by demonstrating low risk. The SAFR concept proposes fewer systems, components, and structures classified as safety-related than is the practice with licensed LWRs. The main control room and balance-of-plant items as well as many other items associated with the nuclear island (such as diesel generators and cooling water systems) are specified as commercial industrial grade.

Two shutdown (scram) systems are utilized in SAFR, neither of which are classified as safety-grade. The Automatic Plant Trip Systems (APTS) can drive in all six primary control rods, and can interrupt power to the electromagnetic latch and drop three secondary control rods into the core. In addition, the three secondary rods can be dropped in by the Self-Actuated Shutdown System (SASS). The SASS is based on a magnet with a curie point at about 1050°F (higher than operating temperatures). The secondary safety rods are released whenever the core outlet temperatures exceed the curie point temperature.

Mr. G. Van Tuyle, Brookhaven National Laboratory (BNL), presented results of analyses that were performed by BNL. The accidents analyzed are loss of heat sink (LOHS), loss of flow (LOF), transient overpower (TOP), and unprotected single-pump-seizure accidents. For the LOHS, the feedwater pumps providing water to both of the two steam generators are assumed to lose power, causing the steam generator to dry out, with the resulting loss of heat rejection. BNL assumed loss of normal cooling and of the Direct Reactor Auxiliary Cooling System (DRACS), with the outside surface of the reactor vessel well insulated. The calculation indicated that the combination of negative reactivity feedback from radial expansion and the Reactor Air Cooling System (RACS) heat removal will prevent damage to the reactor. If RACS air flow is stopped, major fuel damage starts to occur in about 18 hours. Sodium boiling occurs in about 36 hours. In another scenario, an LOF event is initiated by an instantaneous loss of power to the primary, the intermediate loop, and the steam generator pumps. Scram does not occur. The inertially controlled coastdown of the primary pump is modeled by an initial six-second flow "halving" time. The fuel temperature increases as the flow decreases, and, as a result, the power level decreases. The negative reactivity feedbacks decrease power but the core remains at an elevated temperature. Scram must occur to bring the plant to cold shutdown. It is believed that the operator would have sufficient time to take action before significant core damage occurs.

For the TOP event, the calculations indicated that the radial expansion is the largest of the negative feedback mechanisms. BNL concluded that no fuel damage is expected during this event.

BNL analyzed an unprotected single pump seizure event in which one of the two centrifugal pumps would seize during full-power operation. The other pump continues to operate and the plant protection systems fail to scram. The seizure of one pump causes a drop in primary system flow impedance. As a result, the unfailed pump will experience a large flow increase (up to 128% of its rated condition) and will cavitate. The reactivity feedback reduces the power level to a point where the maximum full centerline temperature and the maximum sodium temperature in the core are within acceptable limits. The conclusion is that this event would be mitigated by the reactivity feedbacks in the core, with no fuel damage or requirements for immediate operator action.

Overall conclusions are: 1) the SAFR passive cooling via RACS is effective and fault tolerant, 2) the SAFR inherent shutdown systems are similar to Power Reactor Inherently Safe Module (PRISM) design, and 3) transients result in higher temperatures in the SAFR design and, as a result, control rod driveline expansion is enhanced.

Mr. Landry described some of the SAFR design events that were considered in the NRC staff's review. These events are intended to bound the LMR design basis accidents, as well as some beyond design basis accidents, with margins to account for uncertainties. These bounding events are also expected to provide conservatism in selecting a suitable site source term.

The staff has concluded that the SAFR design has the following general safety advantages:

- A slow response to core heat-up events
- Inherent beneficial reactivity feedback effects associated with the fuel and core expansion of the metal fuel pins
- Characteristics which will make it possible to demonstrate by test the significant safety features and performance of the plant over a wide range of events.

The staff also concludes that the design has potential vulnerabilities, such as:

- The large positive reactivity effects associated with sodium voiding
- The use of a relatively new metallic alloy in a sodium-bonded fuel design, with the potential for relocation of the fuel following melting or eutectic formation and resulting reactivity-induced power excursions.

The Staff's conclusion is that the SAFR design has the potential to achieve a level of safety at least equivalent to current generation LWRs provided the design and research and development needs are resolved.

Dr. Kerr questioned the validity of the high reliability the NRC Staff is associating with nonsafety-grade systems. Dr. Siess questioned the Staff's argument regarding the reliability of safety-grade systems vs. nonsafety-grade systems. He added that it is not clear, and has never been demonstrated to the ACRS that safety-grade components are more reliable than nonsafety-grade components. He noted that sometimes the only difference between the safety-grade and nonsafety-grade components is different quality assurance requirements.

Dr. Kerr wondered if the long time that this SAFR design allows for the operator to take action could increase the likelihood that the operator might do the wrong thing.

Mr. Carroll commented that the term "inherently safe" is confusing and misleading. Dr. Remick agreed and added that use of the term "walk away" reactor is similarly confusing.

Mr. Michelson expressed concern regarding the definition of external events and the fire hazards from sodium, especially from pipe-break events in piping around one inch in diameter.

Dr. Remick expressed concern regarding analysis of the wetting of the inside surface of the reactor vessel in sodium spill-over accidents. He thinks that this type of accident has not been analyzed satisfactorily. Dr. Shewmon shared the concern and added that consequences of sodium leakage from the reactor vessel should be evaluated.

Mr. Ward commented that fuel rod bowing has been inadequately analyzed.

III. Containment Systems (Open)

[Note: Mr. Gary Quittschreiber was the Designated Federal Official for this portion of the Meeting.[

Mr. Ward provided a brief report on the December 6, 1988 Containment Systems Subcommittee meeting noting that a consultant report from that meeting was available. A draft report had been prepared and distributed to the members. He requested that members review this draft and provide comments to him prior to the January 1989 ACRS meeting.

Dr. Themis Speis, Deputy Director for Generic Items of the Office of Nuclear Regulatory Research, provided a condensed version of his December 6th subcommittee meeting presentation on the Mark I Containment Improvements Program. This program was initiated as a result of concerns about the ability of Mark I containments to perform adequately during some severe accidents. The program is intended to complement the Individual Plant Examination (IPE). It focuses on the ability of the containment to withstand severe accident challenges. Dr. Speis noted that the primary objective is to determine what actions, if any, should be taken to reduce vulnerability of containments to severe accident challenges. Early efforts will be focused on Mark I containments. Several early containment failure modes are associated with large scale core melt. The problem is exacerbated by the small size of these containments.

Members of the Committee questioned the staff on the conclusion that molten core material would exit the cavity and attack the liner. The staff stated that there was no door seal (dam) which would delay the exit of the core material from the vessel cavity. Mr. Michelson indicated that he knew of some plants that had such a door seal. The staff is collecting information from the utilities as to the design configuration of individual Mark I plants. It did not appear, however, that the staff has asked for this information.

In response to a question from Mr. Michelson concerning whether the staff considered putting a dam on the flat floor below the Peach Bottom vessel, Dr. Speis said that the staff does not see much benefit in doing this. Mr. Beckner said they may not have specifically looked at this but their generic conclusion is that there are problems with the technical feasibility of this approach to mitigation. The material may be undercut, or if it is a ceramic it may shatter upon contact with the core material. He also believes that construction in an operating reactor will result in significant man-rem exposure.

Dr. Speis discussed insights obtained from the 12 PRAs which were performed for Mark I containments (6 by industry and 6 by the NRC) in the area of dominant accident initiators (station blackout, ATWS, and loss of decay heat removal). PRAs have predicted wide variation in the accident frequency for different plants. It appears that implementation of venting procedures can reduce the core melt frequency for the TW sequence (loss of the ultimate heat sink with containment failure prior to core melt) by an order of magnitude or more. In response to a question from Dr. Kerr concerning whether a similar survey of accidents for Mark II and III containments would lead to the same dominant accident initiators, Dr. Speis said he would obtain this information for the Committee.

Dr. Speis summarized the discussions carried out at a workshop held on February 24-26, 1988 with about 150 representatives of industry, research groups, NRC staff, and the public. There was a variety of views on the probability of liner melt-through. However, there was general agreement that water in the drywell is useful to delay/prevent shell failure and to reduce fission product releases. Industry emphasis was on prevention. The industry position is that the backfit requirements should be plant specific, and subject to the backfit process. Dr. Speis discussed the staff approach in this area. An effort is made to achieve a balance between accident prevention to reduce the likelihood of an accident occurring, and mitigation to reduce the challenge to containment and the magnitude of radioactive releases to the environment. This approach produced the following recommendations for the Mark I containments:

1) Accelerate implementation of the station blackout rule (ATWS implementation will be essentially complete by January 1989).

2) Require an alternate water supply for drywell spray and vessel injection, with a pumping capability that is independent of both normal and emergency AC power.

3) Require hardened (i.e., able to withstand severe accident pressures) venting capability from the wetwell.

4) Require enhanced ADS reliability with additional power and/or nitrogen supplies and improved cable reliability.

5) Require the implementation of the improved EPGs (Revision 4 of the BWR Owners Group).

Dr. Kerr noted that the Station Blackout Rule must not provide for an adequate electrical power supply if the modification proposed in Item 2 was needed. Mr. Thadani noted that Item 2 goes well beyond the scope of the Station Blackout Rule but that a utility could combine their station blackout modifications into the proposal. Dr. Speis suggested that it may be possible for many plants to use the modifications provided as a result of the Station Blackout Rule to take care of this matter. Dr. Kerr suggested that the staff reexamine the Station Blackout Rule to assure that the AC power supply was sufficiently reliable.

Mr. Michelson questioned the staff as to whether new equipment installed as a result of the Mark I Containment Improvement Program would be environmentally qualified for the expected accident environment conditions. Dr. Speis said the equipment would be qualified to function during an accident.

In response to a question from Dr. Kerr concerning whether the Station Blackout Rule implementation was considered in the evaluations which concluded that the TW sequence was an important contributor, Mr. Thadani said that it was not. The Station Blackout Rule has not yet been implemented, and backfits will vary from plant to plant. Some plants may not be affected very much since they may already be meeting most of the requirements of the rule. Mr. Thadani added that the risk associated with the TW sequence is not driven by the availability of electrical power. Mr. Beckner stated that the TW sequence can be considered separately.

In response to a question from Mr. Carroll concerning how one assures that water gets into the vessel cavity for those plants that do not have a "door seal," Mr. Soffer said that their observation at Peach Bottom was that two spray headers had spray angles of 180 degrees and that the water drains into the sump at the center of the pedestal. Mr. Michelson added that he believes that the pedestal area would eventually fill with water.

Dr. Speis discussed several industry efforts which have already been proposed and/or are taking place in this program, including the following:

1) The BWR Owners Group has proposed Revision 4 to the EPGs which includes venting of containment. The NRC staff has recently approved this revision.

2) Vermont Yankee is planning changes during their 1989 refueling outage associated with this program.

3) Pilgrim has developed a safety enhancement program which implements several recommendations from this program, including the addition of a hardened vent from the torus to the stack, adding a third onsite diesel, adding a backup nitrogen supply for ADS, additional inerting, and use of fire protection diesel pumps for decay heat removal.

Dr. Speis discussed the cost/benefit analysis of the suggested improvements, noting that the major benefit is core melt reduction resulting from venting. Venting can also prevent containment failure resulting from slow overpressure. The industry cost estimates range from about $48 million to $176 million with the result that the cost/benefit does not always meet the $1,000 per man-rem criteria. However, the staff believes that some of the enhancements can be justified from an engineering analysis. Consideration is being given to implementing some changes via rulemaking.

Dr. Speis asked the Committee to provide comments to the Commission from a technical perspective as to whether the proposals recommended by the staff are appropriate.

Dr. Speis provided the following NRC staff conclusions and recommendations:

1) The proposed enhancements provide substantial increase in the overall protection of public health and safety.

2) The proposed enhancements are generally cost beneficial.

3) It is proposed to implement the enhancements via rulemaking.

4) Confirmatory research should be performed in the areas of phenomena relevant to in-vessel and ex-vessel accident progression, the effect of cavity water on the probability of liner melt-through, and the associated source terms.

Dr. Speis mentioned several comments provided by the CRGR as a result of their review of the proposed enhancements.

Mr. Wayne Hodges, Chief of the Reactor Systems Branch in the Office of Nuclear Reactor Regulation, discussed the Emergency Procedure Guidelines (EPGs) provided in the latest Revision 4 (recently approved) as they relate to the Mark I Containment Improvement Program. The criteria used to develop the EPGs for BWRs are symptom-based (not on specific scenarios) and specify appropriate actions for all emergencies including severe accidents. The EPGs are not limited by licensing or design basis assumptions. They apply to plants as currently built using all available plant equipment, and not just safety-related equipment. Operators are guided to take the best action possible, including using nonsafety systems. Mr. Hodges stated that the guidelines are developed by the BWR Owners and not the NRC.

Mr. Hodges discussed the major improvements that Revision 4 incorporated, including the restructuring and simplification of the general form of the guidelines, and significant changes in the guidelines for ATWS mitigation, which acknowledge that large power oscillations might occur. Revision 4 has also added hydrogen control guidance for Mark I and Mark II containments and has provided improved containment venting criteria.

Mr. Hodges discussed the improvements in Revision 4 in the area of ATWS mitigation. He noted that alternate rod insertion guidelines have been restructured and simplified, and that the reactor pressure vessel water level control band has been extended to below the top of the active fuel during ATWS. There have been changes made to lower the reactor vessel water level limit which initiates isolation. This keeps the condenser available as a heat sink for a longer time.

Mr. Hodges discussed containment venting as a means to prevent core melt and reduce dose. This strategy was discussed in NUREG-0737 (Item I.C.1) and required procedures which allowed for events with multiple failures and operator errors. The BWR Owners' Group developed EPGs to comply with this early requirement by calling for venting to prevent failure of the containment. It later became evident that venting was an effective means for preventing core melt, and for mitigating the consequences of core melt.

In response to questions from Mr. Michelson concerning the possibility that leakage of steam to equipment areas during venting (through low-pressure ducting) might make things worse instead of better, Mr. Hodges said that the staff feels that if the ducting is ruptured recovery may be complicated. The staff believes that this is still better than allowing core melt to occur. Installation of the hardened vent in the Mark I program is being proposed to prevent this problem. Current EPG guidelines do not require a hardened vent. Mr. Thadani added that the staff is firmly convinced that the Mark I plants need a hardened vent to take full advantage of the benefits of venting. He noted that the licensee for the Pilgrim Plant estimates that the most likely use of the vent would be for prevention before any fuel damage occurred.

Mr. Hodges stated that earlier versions of the guidelines recommended that venting be performed at the time that the primary containment design pressure limit was reached. The new guidelines recommend venting prior to reaching that limit. Neither set of guidelines specifies how long to leave the vent open. This will be defined in the plant-specific procedures.

In response to questions from Mr. Michelson concerning what assurance the NRC has that the EPGs can actually be implemented with the equipment and procedures in place, Mr. Hodges said at this point they do not have any assurance. The NRC has audited the implementation of the EPGs and found cases where the implementation was not satisfactory. Mr. Thadani added that this is a problem with EPGs in general, and not just for the Mark I plants. Mr. Hodges noted that the NRC has audited the implementation of the EPGs for 15 plants (including both PWRs and BWRs) over the past several months and has found equipment-based implementation problems in all of them.

Mr. Hodges discussed the changes in the primary containment pressure limit (PCPL) from earlier revisions of the EPGs. Revisions 2 and 3 required opening vents upon reaching the PCPL, which was twice the design pressure. Revision 4 provides better guidance for determining PCPL. The PCPL is set at a pressure limit which bounds the pressure capability of the containment. This may be determined by the containment vent valve operability limit, the steam relief valve operability limit, or the reactor pressure vessel vent valve operability limit.

Mr. Hodges discussed advantages and disadvantages of containment venting. Advantages include maintaining low reactor pressure vessel pressure to allow core cooling by low pressure systems, cooling of the core with systems external to containment and removal of decay heat by pool steaming, use of the pool for scrubbing of fission products, and prevention of containment failure due to overpressure. Disadvantages of venting included the possibility of increasing dose if not coordinated with evacuation, possible loss of some equipment due to the harsh environment caused by venting, the potential for inadvertent venting, and the potential for the bypassing of pool scrubbing due to improper venting.

Mr. Hodges stated that the NRC has taken the position that the decision on venting should be made by the senior utility manager on site at the time venting is needed.

Dr. Speis promised that the staff will update the Committee on this program at the January 1989 ACRS meeting.

IV. Equipment Qualification (EQ)-Risk Scoping Study (Open)

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

Dr. Siess, Acting Chairman of the Reliability Assurance Subcommittee, stated that for several years the Office of Nuclear Regulatory Research (RES) of the NRC has funded EQ research to study the methods for qualifying safety-related electrical equipment and to demonstrate their survivability during and following design basis accidents (DBAs) that produce harsh environments. The EQ research was terminated by the end of FY 1986. At that time, the ACRS, in its February 16, 1986 report to the Congress and in its June 11, 1986 report to the Commission, recommended that the EQ research be funded to assess the survivability of electrical equipment under hostile environmental conditions resulting from accident, including severe accidents. In response, the RES Staff stated that they plan to perform a risk-based prioritization study on EQ in FY 1987 to determine the need for further research in this area. Accordingly, the EQ-Risk Scoping Study was initiated and performed by the Sandia National Laboratories (SNL). He stated that the purpose of the EQ-Risk Scoping Study was to use the information from existing PRAs and assess the risk significance and risk uncertainties associated with current EQ requirements for safety-related electrical equipment.

Dr. Siess stated that the Reliability Assurance Subcommittee discussed the EQ-Risk Scoping Study during the meetings on December 16, 1987, June 14, 1988, and December 12, 1986. This matter was also presented to the full Committee by representatives of RES and SNL during the 339th ACRS meeting, July 14-16, 1988. Since a set of peer-review comments was provided to the ACRS the day before the July 1988 ACRS meeting, the Committee decided to defer action on this matter pending a detailed review of the peer-review comments and the associated SNL responses by the Reliability Assurance Subcommittee. Accordingly, the Reliability Assurance Subcommittee met with the representatives of RES and SNL on December 12, 1988 and discussed the peer-review comments and SNL's responses to them.

Dr. Siess stated that the peer review was performed by the following personnel:

Mr. Kenneth Canady, Duke Power Company
Mr. George Sliter, Electric Power Research Institute

Mr. Andrew Wolford, Idaho National Engineering Laboratory
Mr. Sal Carfagno, Franklin Research Center

The peer-review panel members met twice with representatives of SNL to discuss their comments on the preliminary results and conclusions of the Scoping Study. The peer-review comments were generally favorable. In response to the peer-review comments, several clarifications and changes have been made to the draft report of the Scoping Study. Dr. Siess stated that, in his opinion, the review of the Scoping Study by the peer-review panel has contributed to the credibility of the conclusions and the quality of the final report of the Scoping Study.

Mr. Michelson asked who decided to do a peer review of the EQ-Risk Scoping Study. Dr. Siess stated that the Staff is trying to do peer review on most of the studies similar to this one. He is not sure whether there is a general RES policy related to peer review. He suggested that when the Committee meets with Mr. Beckjord, Director of RES, on December 16, 1988, it may want to find out about the RES policy regarding peer review.

After further discussion, the Committee discussed the comments and recommendations proposed by the Subcommittee on the EQ-Risk Scoping Study and approved them, with minor changes, for transmittal to the Commission.

<u>V. NRC Quantitative Safety Goals</u> (Open)

[Note: Mr. Gary Quittschreiber was the Designated Federal Official for this portion of the meeting.]

The Committee held a discussion with Mr. Wayne Houston, Director of the Division of Safety Issue Resolution in the Office of Nuclear Regulatory Research, on the staff's presently proposed plan for implementation of the Safety Goal Policy. The plan being proposed to implement the program is said to clarify the role that safety goals, the quantitative objectives, and the use of PRAs will have in future regulatory decisions. It includes discussion on implementation of the Backfit Rule. The NRC staff's resolution of ACRS recommendations made in reports on May 13, 1987, April 12, 1988, and July 20, 1988 was also discussed.

Mr. Houston noted that the staff is and has been using PRAs to make decisions, but does not have any clearly stated practices, goals, or criteria as to what purpose PRAs serve in the regulatory process. He hopes that the implementation plan will bring this into clearer focus. He suggested that the hierarchy for use of the safety goals would be used first by the designers, then by the operators, and, last of all, by the regulators. With regard to regulation, he suggested that the quantitative objectives incorporated in the safety goal should be regarded as targets for generic regulatory requirements and not as criteria for individual licensing decisions.

Mr. Houston discussed the words "substantial increase in overall protection" as stated in the Backfit Rule as a statutory standard and suggested that this could be used in conjunction with the Safety Goal and PRAs in the context of developing regulatory analyses which determine if a "substantial increase" is achieved. The staff's proposed plan provides a discussion of a potential approach to developing a relationship between the Safety Goal Policy's quantitative health objectives (QHO) and the Backfit-Rule definition of an adequate protection standard. Mr. Houston suggested that the relationship might fall between two and ten times the QHO. The Backfit Rule discussion does state that compliance with the Commissions rules, regulations, and positions are presumptive evidence of meeting an adequate protection standard. However, under the Backfit Rule, no matter how safe a plant is, if the cost benefit criterion can be satisfied by a proposed change, it can be required.

Mr. Houston stated that the specification of a core damage frequency as a safety goal objective provides a goal against which one can measure and make design judgments. The NRC staff's proposed plan will recommend to the Commission that the existing policy statement be supplemented to include design objectives for core damage frequency and a specific definition for a large release.

Mr. Houston discussed recommendations by the OGC for dealing with averted onsite costs. The staff will continue to use the $1,000 per person-rem criteria (for dose received within 50 miles from the site) and will recommend that averted onsite costs not be treated as a benefit but rather as a factor which offsets the utility's costs. When used in this way (appearing in the numerator of the cost/benefit equation as a negative cost) it often does not make a large difference when dealing with perspective modifications since it affects only the cost side of the ratio and not the benefit side.

Mr. Houston described the principal elements of the NRC staff's recommendations in the plan: 1) establishing a hierarchy of quantitative objectives, 2) reviewing PRAs to assess effectiveness of regulatory requirements, 3) integrating risk reduction modifications and testing proposed modifications, and 4) using subsidiary goals for generic safety issue resolution in conjunction with less than full scope PRAs. The staff is proposing a five-level hierarchy, starting with the qualitative safety goals and working down to the last level of regulatory requirements. Mr. Ward noted one significant difference from the ACRS proposal was that the staff is not establishing guidelines for defense-in-depth. The staff is dealing with the core melt issue by prevention and has not established criteria which set goals for mitigation.

In response to questions from Mr. Michelson concerning the scope of the PRAs being suggested, Mr. Houston said they are recommending full scope PRAs which consider both external and internal events and exclude only sabotage and other fuel cycle matters.

In response to questions from Dr. Kerr concerning allocation of fractions of risk to certain safety issues, Mr. Houston noted that the staff is already doing this for core melt frequency but not with regard to a containment performance design objective.

The staff's definition of a large release is a release that has a potential for causing an off-site early fatality which is said to be a "natural threshold" effect. The staff believes that this avoids the problem of arbitrarily choosing a release of a given number of curies which has no direct relationship to a particular consequence. The staff believes that the word "potential" has a significant meaning in this definition. The staff has defined the term "core damage" as the potential threshold associated with the loss of adequate core cooling.

Mr. Houston noted that the major thrust of the safety goal policy is defined in the plan as being directed to light water reactors; however, many of the principles could be applied to advanced reactor designs.

ACRS members questioned Mr. Houston on the terms and frequencies being proposed by IAEA, EPRI, and others and whether there was any need for them to be consistent with the NRC's proposal. Mr. Houston discussed some problems associated with having different requirements in the areas of siting, emergency planning, and need for containment when core damage frequency is very low. He suggested that establishing lower targets for future plants might result in better designs. He believes that one cannot have as much confidence in a PRA for new unproven designs as compared to a PRA on an operating plant. In addition, as the number of plants increases one will need lower risk per reactor to have the same overall risk.

Mr. Houston discussed the information which can be obtained from existing PRAs and the use of this information in PC-based codes. Dr. Kerr questioned the ease of using these codes by those other than the PRA analysts.

Mr. Houston stated that he believes NUREG-1150 will play a dominant role in the assessment of regulatory requirements over the next several years.

Mr. Houston stated that he believes the staff does plan to deal with the question of whether one can give a probabilistic interpretation of the meaning of "credible." He noted that this question has come up in hearings and in the use of the Standard Review Plan. The staff proposes to deal with this term in such a way as not to change the large release guideline definition.

The staff intends to use partial scope PRAs in those cases where a PRA exists for a particular type event and will use the same numerical objectives for the overall safety goal. This gets away from allocation of risk.

The Committee decided to try to write a report on this matter at its January 1989 meeting. No further NRC Staff presentations were requested for the January 1989 meeting.

VI. Meeting with the Director of the NRC Office of Nuclear Regulatory Research (Open)

[Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

Mr. Beckjord, Director of the NRC Office of Nuclear Regulatory Research (RES), discussed the items given below.

RES Reorganization

Mr. Beckjord stated that the RES reorganization became effective on July 17, 1988. The main objectives of the reorganization are to:

- Consolidate the efforts related to implementing the Commission's Severe Accident Policy and the resolution of Unresolved Safety Issues (USI) and Generic Safety Issues.

- Restructure severe accident research so as to provide earlier input to the decision-making process on severe accident issues as well as longer term confirmatory research needed for closure of severe accident issues.

- Make most effective use of limited RES resources.

- Clarify the responsibilities of the RES Deputy Directors.

Mr. Beckjord stated that, under the current organization, the four divisions of RES fall into two major categories:

- Divisions Responsible for Research

  - Division of Engineering
  - Division of Systems Research

  These divisions will come under Dr. Ross, Deputy Director for Research.

- Divisions Responsible for Resolution of Issues, and Development of Rules and Requirements

  - Division of Safety Issue Resolution
  - Division of Regulatory Applications

  These divisions will come under Dr. Speis, Deputy Director for Generic Issue Resolution.

Mr. Michelson asked which Division is responsible for research related to equipment qualification and fire protection. Mr. Beckjord responded that such research will be the responsibility of the Division of Engineering.

In response to a question from Dr. Siess about manpower allocations, Dr. Ross stated that the Division of Engineering and the Division of System Research have about 60 people each, and the Divisions of Safety Issue Resolution and Regulatory Applications have about 50 people each.

Dr. Siess noted that, under the reorganization, although resolution of USIs and generic issues has been consolidated under one division, prioritization of generic issues is handled by the Advanced Reactors and Generic Issues Branch under a different division. He asked whether the same personnel work on both advanced reactor issues and generic issues. Mr. Beckjord responded that these issues are handled by two separate sections in that branch.

Mr. Michelson commented that the Staff seems to be reluctant in addressing which generic issues will be applicable to future plants. Dr. Speis responded that the Staff recently had a two-day workshop to discuss several matters, including the applicability of generic issues related to future plants.

Mr. Michelson stated that in the resolution of each generic issue the Staff should specify clearly whether that specific issue will be applicable to future reactors. Dr. Speis said that he would discuss this matter in detail with the ACRS at a future meeting.

Dr. Siess requested a copy of the summary report related to the workshop held recently on advanced reactors. Dr. Speis agreed to send a copy of that report.

In response to a question from Dr. Siess, Dr. Speis stated that implementation of the resolution of USIs and generic issues is a joint effort between RES and NRR.

Status of Implementation of the National Research Council's Recommendations on Revitalizing Nuclear Safety Research

Mr. Beckjord stated that, as requested by the Commission, the National Research Council's Committee (ad hoc) on Nuclear Safety Research performed a study regarding the future course of nuclear safety research in the U.S. The conclusions and recommendations of the study are documented in a report entitled "Revitalizing Nuclear Safety Research," dated December 8, 1986. There are several recommendations specifically addressed to RES, some to the EDO, and some to the Commission.

Mr. Beckjord discussed the recommendations specifically addressed to RES and the status of their implementation:

- The NRC should bring in high-caliber researchers to bolster management.

  Mr. Beckjord stated that to accommodate this recommendation he tried to hire an experienced research person. He made job offers to 11 qualified individuals, but none of them accepted the job. He then tried to accommodate this recommendation by means of visiting fellowships and staff exchanges. Under the visiting fellowship program, he had one professor from Stanford University to work on the severe accident program, one professor from MIT to work on PRA studies, and a person from Brookhaven National Laboratory to work on various aspects of severe accidents. He believes that this approach has been very constructive and brought some outside experience and new ideas.

Dr. Siess asked about the role of the people who came to RES under the visiting fellowship program. Mr. Beckjord responded that it depends on the needs of RES. The people who came on board thus far helped in reviewing the existing research, planning of new research, and performing some PRAs.

- The NRC should consider separating the function of standards development and research.

Mr. Beckjord stated that this recommendation was not accepted either by RES or by the Commission. He believes that they have to have additional people and resources to accommodate this recommendation effectively. Owing to budget constraints, he does not believe it is possible to get additional people for RES.

- The NRC should develop a cogent philosophy of safety research.

Mr. Beckjord stated that RES had already prepared a statement of research philosophy and was approved by the Commission in May 1988. It is included in NUREG-1325, Disposition of Recommendations of the National Research Council in the Report Revitalizing Nuclear Safety Research, and also in the NRC's Five-Year Plan. He stated that, in this statement of philosophy, RES has set forth the key principles that should govern the definition, planning, conduct, use, and closure of nuclear regulatory research projects.

Dr. Siess asked whether the statement of research philosophy has been applied so far in making decisions, settling arguments, or assigning research priorities. Mr. Beckjord stated that it is being applied.

- The NRC should establish a research program planning process involving all of the relevant offices within the NRC, as well as representatives from industry and the university research community acting as participating advisors.

Mr. Beckjord stated that to accommodate this recommendation they have established research review groups that include representatives from various offices of the NRC. These groups meet periodically to plan and develop research programs to support NRC programs and strategies. They also plan to meet with industry groups, such as EPRI, IDCOR, and NUMARC, twice a year to discuss coordinated research efforts.

- The NRC should impanel an independent advisory group reporting to the Director of RES.

  Mr. Beckjord stated that the Commission has approved the creation of an independent advisory committee called the "Nuclear Safety Research Review Committee." RES had already met with this Committee twice. This Committee plans to establish some subcommittees to look at various research programs in the areas of human factors, severe accidents, etc.

  Dr. Kerr asked about the membership of the subcommittee for severe accidents. Mr. Beckjord responded that it consists of the following members:

  Salomon Levey, S. Levey, Inc. (Chairman)
  Richard Wilson, Harvard University
  Cordell Reed, Commonwealth Edison
  David Morrison, IIT Research

  Dr. Siess asked whether RES has a general policy for setting up peer-review panels to review research results. Mr. Beckjord responded that they have a general policy that requires contractors performing research for the NRC to publish the results in scientific journals. As far as peer review of other research studies are concerned, they do not have a written policy. However, he strongly encourages the use of the peer-review process.

  Mr. Beckjord discussed briefly other recommendations of the National Research Council related to creating a fair and competitive process for contracting research, and performing more research at universities. He stated that in FY 1988 they spent about $13 million for research at universities and private contractors; in FY 1989, they expect to spend about $16 million.

  Dr. Kerr asked how much money is spent at universities for research. Mr. Beckjord and Mr. Bartlett committed to provide this information at a later date.

- Status of Implementation of the National Research Council's Recommendations on Human Factors Research

  Mr. Beckjord stated that in response to another request by the Commission, an ad hoc committee of the National Research Council performed a study on the need for additional research in the human factors area. The results and conclusions of this study are documented in a report entitled "Human Factors Research and Nuclear Safety," dated February 19, 1988.

  In summary, the National Research Council recommended that the NRC facilitate the capability for conducting human factors research by:

- Staffing and maintaining continuity of the program at the branch level

- Adopting a systems-oriented or socio-technical perspective to the research

- Utilizing independent peer reviews to enhance the quality of research products

- Establishing improved mechanisms for transferring research results to user communities by means of annual written reviews and a bibliographic search system.

- Increasing the timely transfer of knowledge to industry.

In addition, the National Research Council recommended that research be conducted in the following major areas:

- Human-system interface design

- Research on the personnel subsystem (training, qualifications, etc.)

- Human performance (human error)

- Management and organizational performance

- Studies on the regulatory environment.

Mr. Beckjord stated that the human factors program plan and the Staff's responses to the recommendations of the National Research Council are included in SECY-88-141, "Human Factors Initiatives and Plans," dated May 23, 1988. Subsequent to reviewing the contents of SECY-88-141, the Commission directed the Staff to address several issues related to the NRC's human factors programs. The Staff's responses to the issues raised by the Commission are included in SECY-88-294, "Human Factors Program," dated October 13, 1988. Mr. Kaufman stated that an updated document related to human factors programs is being prepared and is expected to be submitted to the Commission during January 1989.

The Committee suggested that the ACRS Subcommittee on Human Factors review the human factors program plan and other related issues.

<u>VII. Review of NRC/RES Code Scaling Applicability and Uncertainty (CSAU) Methodology</u> (Open)

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

Mr. Ward, Chairman of the Thermal-Hydraulic Phenomenon (T/H) Subcommittee, briefly summarized the history of the CSAU effort. He noted that CSAU was developed to evaluate the effectiveness of the best-estimate ECCS codes. The T/H Phenomena Subcommittee last discussed this issue at a meeting on December 7, 1988. Mr. Ward said the Subcommittee believes the CSAU program is a success and requested comments from ACRS members at the conclusion of this discussion. Mr. Ward said he would provide a draft letter on this topic for the Committee's consideration during the January meeting.

Dr. Shewmon requested information during this presentation that addresses how much margin would be available by using the CSAU methodology as measured against the Appendix K PCT. He said that recent reactor vessel fluence data indicates that some plants' vessel design life may be shortened due to higher-than-expected fast neutron (pressurized thermal shock limitations) fluence. He wondered if the additional margin available could offset the expected design life penalty. Mr. Ward said the margin obtained by use of best-estimate ECCS codes could indeed be used to increase power peaking, thus reducing the fluence at the vessel wall. To quantify the margin, one must do a best-estimate ECCS evaluation. The CSAU methodology is an acceptable method for use in the required uncertainty analyses.

Dr. Zuber, RES, described the development of the CSAU methodology and its application to an LB LOCA code calculation. He also identified the members of the Technical Program Group (TPG) that developed and applied the CSAU method. Dr. I. Catton has acted as an observer to the TPG on behalf of the ACRS. While the TPG effort consumed 117 man-months of time, Dr. Zuber indicated that to apply the technique to some other area would probably require only 36-48 man-months.

The objectives of the CSAU methodology are to:

1. Provide a technical basis for quantifying uncertainty within the context of the revised ECCS rule.

2. Provide an auditable, traceable, and practical method for combining quantitative analyses and expert opinion to arrive at a computed value of uncertainty.

3. Provide a systematic and comprehensive approach for:

   a. Defining scenario phenomena
   b. Evaluating code applicability
   c. Assessing code "scale-up" capability
   d. Quantifying code uncertainty related to:

      ° Code and experiment inaccuracies
      ° Code scale-up capabilities
      ° Plant state and operating conditions.

Representatives of RES said that the CSAU was reviewed during its development by the ACRS and an international peer review group chaired by N. Todreas (MIT).

The TPG also developed three simple physical models designed to calculate: 1) the PCT, 2) the effect of ECC bypass on PCT, and 3) the effect of steam binding on PCT. Dr. Zuber indicated that the central reason for developing these simple models was to provide a practical means of summarizing the information/knowledge obtained via CSAU. He said simple physical models are the best method for such a knowledge transfer to future users. These physical models were not developed for use in the licensing process.

RES has calculated the LB LOCA PCT bound, using CSAU to be 1572°F at the 95% certainty limit. In response to Dr. Shewmon, Dr. Zuber indicated that the BE approach provides about 600°F margin in PCT vis-a-vis the "old" Appendix K requirements.

The key summary points noted by Dr. Zuber were:

- We have an extensive experimental data base for LB LOCAs.
- We have systems codes that are very detailed.
- We have a methodology that meets the three objectives stated above.
- We understand the physics of an LB LOCA very well.
- The LB LOCA issue is resolved.

Dr. Zuber said the "moral" of the LOCA research effort is that given: 1) an extensive experimental data base from well-scaled facilities and well-designed tests, and 2) close integration of experiments and analyses, closure of the LB LOCA issue was possible. Dr. Zuber also indicated that CSAU could be applied to other parts of the severe accident research effort.

The details of the CSAU methodology were provided. The method is subdivided into three subelements: 1) scenario requirements and code capability, 2) assessment and ranging of parameters, and 3) sensitivity and uncertainty analysis.

Details of the above subelements were reviewed. Figure 1 outlines the methodology subdivided pursuant to the subelements and is summarized below.

- "Requirements and Capabilities" in which scenario modeling requirements are identified and compared to computer code capabilities to determine their applicability to the particular scenario and to identify potential limitations.

- Assessment and Ranging of Parameters" in which code capabilities to calculate processes important to the scenario are assessed against experimental data to determine code accuracy, scale-up capability and ranges of parameter variations needed for sensitivity studies.

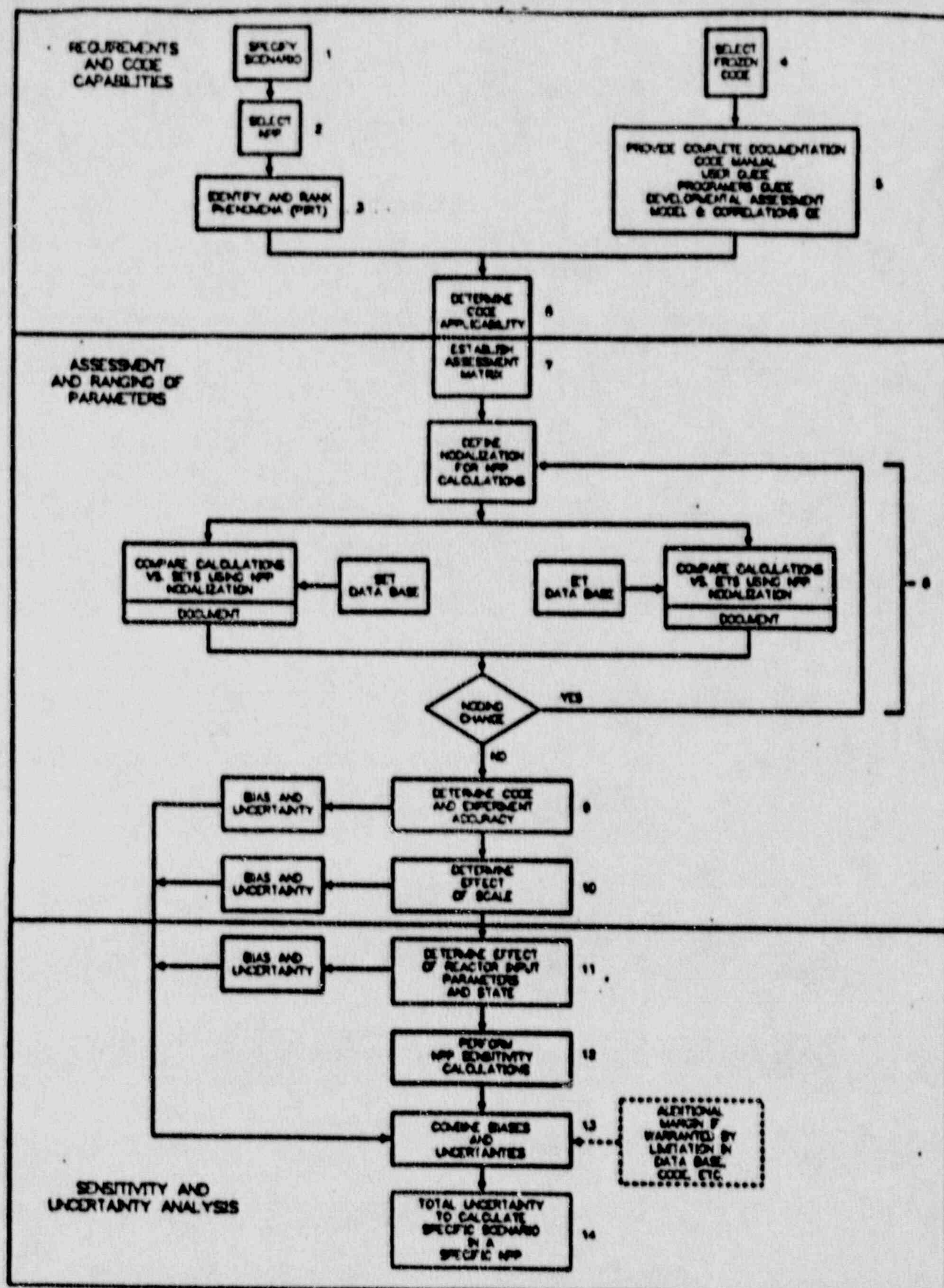


Figure . Code scaling, applicability and uncertainty (CSAU) methodology.

FIGURE 1

- "Sensitivity and Uncertainty Analysis" in which the effects of individual contributors to the total uncertainty are obtained, and for which the propagation of uncertainty through the transient is properly accounted.

Dr. Zuber stressed that the key advantages to use of CSAU is that the methodology is systematic, comprehensive, traceable, and auditable. The last two characteristics are important for its use by a regulatory agency.

In response to questions from Mr. Ward as to the possible benefits of this research, Dr. Zuber stated that, as an example, using BE LOCA analyses could result in a savings of about $8 billion to utilities owning Westinghouse reactors.

RES representatives said that the CSAU method will be applied to the scenario of an SB LOCA in a B&W plant, using the RELAP-5/MOD-3 code.

In response to questions from Drs. Remick and Kerr concerning applying CSAU methodology to severe accident research, Dr. Zuber indicated that the use of CSAU methodology would drive the researchers to focus on issues/phenomena that are of central importance to resolution of their concerns.

Dr. Ward said he would provide the ACRS with a draft letter on this topic for the Committee's consideration during its January meeting.

VIII. Meeting with the Director, Office of Governmental and Public Affairs (Open)

[Note: Mr. Herman Alderman was the Designated Federal Official for this portion of the meeting.]

Mr. Harold Denton, Director of the Office of Governmental and Public Affairs, described some of the recent NRC contacts with the Soviet Union. Mr. Denton reminded the Committee of his previous briefing in which he told the Committee about his first trip to the Soviet Union. He noted that the Soviets made a reciprocal visit to the United States in the fall of 1987 and at that time both parties agreed to proceed to attempt to develop an agreement covering civilian nuclear power safety.

Chairman Zech and Alexander Pxotsenko signed, in April 1988, an agreement to cooperate in the field of civilian nuclear power plant safety. One of the features of that agreement was the establishment of a joint coordinating committee which selects specific topics in which cooperation would take place and develop the framework under which the cooperation would occur. Mr. James Taylor was named chairman of the U.S. delegation and Dr. PonomarevStepnoy was named chairman of the Soviet delegation. The first meeting of the joint committee was held in the Soviet Union in August 1988.

Mr. Taylor discussed the results of that meeting. He noted that they had set out to select topics where they could be of benefit to the Soviet Union and the United States. One area of cooperation that was agreed to was to exchange inspections, on a trial basis, at operating nuclear stations.

Dr. Shewmon asked if the Soviets have a set of regulations comparable to the U.S. regulations. Mr. Taylor replied that the Soviets are developing a body of regulations.

Mr. Taylor said that the Soviets are very interested in having the U.S. participate in a joint analysis of the level of safety of a Soviet nuclear power plant design. An agreement has been reached whereby the Soviets will provide the safety analyses for Zaporsia and the U.S. will provide the complete safety analysis for the South Texas plant.

One other area of exchange was in the area of nuclear safety research. The United States outlined the broad areas of safety research conducted by the NRC. The Soviets, in turn, provided elements of their ongoing research. The objective was to determine if there were areas of possible joint cooperation in research that could be carried out under the US/USSR agreement.

Of interest to the United States was the area of radiation embrittlement and in-place vessel annealing. The Soviets have annealed several reactor vessels in place. An area in which the Soviets expressed considerable interest was the work that the United States has performed on fire safety.

Mr. Taylor noted that the Soviets had a fire at their Ignalena plant. The fire was in the cable area and affected some of the reactor control circuits. Mr. Taylor said the NRC expects to get more of the details concerning this fire in the future.

Dr. Denton said the Soviets have formed a ministry called the State Committee for the Safety of Nuclear Installations. It employs about 900 people and was formed from existing organizations. He noted that the Soviets do not have guides, rules, or regulations comparable to those which the United States has developed over the past two decades.

Mr. Taylor noted that the Soviets are interested in how the United States conducts the decision-making process on backfits and in severe accident analysis. The performance of and use of PRAs are also of interest.

The NRC and DOE will both participate in discussions with the Soviets on the health and environmental effects of Chernobyl. It is expected that a large body of information on the biological effects will be developed. The Soviets are trying to set up the mechanisms to give health and physical examinations to the population that has been involved. The United States hopes to share in the biological and environmental information that will be derived from this study.

Chairman Kerr asked to what extent the dose to the people involved in Chernobyl was measured. Mr. Taylor replied that he did not know how accurate the exposure records were.

Mr. Taylor said that the Soviets and the United States had discussed how the United States works to gather operational data and attempts to benefit from operational data and events.

The Soviets have had considerable experience in the erosion/corrosion area. They are willing to share their experience and data with the United States.

Mr. Edward Shomaker has been appointed as the NRC project manager to handle interagency coordination.

Chairman Kerr asked if the ACRS could participate in the April meetings with the Soviets. Mr. Taylor replied that the ACRS would be welcome to participate with the working groups or in discussions of any topics that are on the agenda.

Dr. Remick asked about the effects of the recent Soviet earthquake on nuclear reactors. Mr. Taylor replied that there were two reactors in the area and they are both operating. He said the Soviets had announced that these two reactors will be shut down in two years and that the shutdown was related to concern as to possible effects of the earthquakes.

IX. Operator Requalification (Open)

[Note: Mr. Herman Alderman was the Designated Federal Official for this portion of the meeting.]

Dr. Remick noted that when Part 55 was modified, one of the modifications was that reactor operator licenses had to be renewed every six years instead of every two years. Requalification examinations had a negative impact on the operators because the tests were not always performance based (i.e., not related to what operators had to do in the plant). NRC decided to look at new methods of administering the requalification exams. These new methods were developed and tests conducted in five pilot programs. Dr. Remick said that his understanding was that the pilot programs have been very successful.

Mr. Ken Perkins, Chief, Operator Licensing Branch, discussed the requalification program for operator licensing. He said that the staff developed and tested a new methodology for assessing the effectiveness of facility requalification training programs and, at the same time, for assessing the proficiency of operators in maintaining the goal of enhancing plant safety.

The new requalification methodology utilized existing industry training program standards to develop and administer the examinations. By administering requalification examinations that are consistent with the existing facility developed training programs, NRC reduces the impact on the facilities and their operators while improving the program assessments.

Each NRC requalification examination included an operating test and a written examination. Each of these was comprised of two distinct parts.

The first part of the operating test was conducted in a simulator facility. This allowed the examiners to observe selected control room crews during simulated transients and accident scenarios. The focus of this portion of the examination was on crew performance rather than on individual performance. The second part of the operating test was conducted during a plant walk-through.

During the plant walk-through, individual operators were evaluated on their ability to correctly perform plant tasks that are important to safety. The emphasis of this mode of testing was to ensure that the operators have maintained their understanding of and proficiency in performing selected system tasks. The walk-through was conducted by facility-appointed evaluators. The NRC examiners evaluated the examination process, asked questions of the operators as necessary to ensure adequate system knowledge and job performance measurement coverage, and made independent assessments of the operator's performance and the evaluator's examination as they were administering it.

The written examination is administered in two parts. The written exam is an open reference examination administered to assess the operator's knowledge of plant systems, procedures, and operating limits. The plant operations section is administered in a control room environment and was designed to evaluate the operator's knowledge of plant systems, integrated plant operations, and instruments and controls. This section was also used to evaluate the operator's ability to diagnose postulated events and to recognize limiting conditions of operation as defined in technical specifications.

The procedures section of the examination has an open reference format and is administered in a classroom setting. It was designed to evaluate the operator's ability to analyze a given set of conditions and determine the proper procedural steps and administrative practices to follow. The operators were given access to the same abnormal emergency and administrative procedures that would be available to them in dealing with similar real-world situations in the control room.

The written examination was developed from the examination question bank proposed and provided by the licensee. NRC reviewed and modified the proposed items, as necessary, to ensure accuracy, clarity, importance to safety and appropriateness for an open reference format.

NRC and the facility licensees worked together in developing, administering, and grading the examination.

Mr. Perkins said it takes about nine months of training to become an examiner. Examiners take a number of courses at the NRC Training Center and get on-the-job training working with certified examiners. Upon successful completion of this process they become certified examiners.

Mr. Michelson asked what the passing grade was for this type of examination. Mr. Perkins replied the minimum was 80% for the two written sections combined. For the walk-through, there are 10 job performance tests. Eight out of the 10 have to be performed satisfactorily for a passing grade. Critical tasks (about five) have been defined for the simulator test. If an individual cannot perform one of those tasks he may fail and will fail if he cannot perform two of the tasks. Mr. Perkins pointed out that, although the simulator testing involves teams, it evaluates individual performance.

X. Nuclear Power Plant Operations - Actions Taken in Response to Core Power Oscillation Event at LaSalle Unit 2 (Open)

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

Dr. Kerr, as Chairman of the Core Performance Subcommittee, introduced this topic to the Committee. He noted that the oscillation event at LaSalle was unexpected, given the stability analyses supplied by GE. He said the Committee would hear presentations from the BWR Owners Group (BWROG), NRC/NRR, and NRC/RES, reporting on the status of actions taken and planned to address the implications of this event.

Mr. Tom Rausch, Commonwealth Edison, spoke on behalf of the BWROG. Points made by Mr. Rausch as background included:

- BWROG initiated generic studies following the March 1988 LaSalle event and BWROG plans to resolve this issue were discussed with the NRC on June 24, 1988.

- Preliminary results from the BWROG analyses are now available and interim corrective actions have been implemented by all BWR utilities. These findings and interim actions were discussed with the NRC on November 9, 1988.

- BWROG stability program now includes investigations of viable long-term solutions to the stability issue.

The BWROG has two main programs underway. These are:

- Phase 1A - Perform an assessment of postulated large amplitude oscillations without scram (ATWS) to 1) determine the response of average core power and 2) confirm that current ATWS analyses are not affected.

- Phase 1 - Perform an assessment of plant response to postulated regional thermal-hydraulic instability. Identify existing mitigation capability and develop appropriate guidance for operators.

The analyses supporting the above work were performed with the TRAC-G code which includes full three-dimensional capability.

The Phase IA analyses did not reveal any concerns. For Phase I the analyses showed that the safety limit (SL) minimum critical power ratio (MCPR) could be exceeded in some cases. GE, with the support of the BWROG, has issued interim corrective actions which ban operation in regions of the power/flow map where instabilities are known to occur (Figure 2). In addition, licensees were instructed to scram if oscillations are seen or if the more restrictive operating regimes are entered. In response to Dr. Kerr, Mr. Rausch said all BWR plants have taken actions to assure that the operator always knows where the plant is operating on the power/flow map.

In response to Dr. Shewmon, Mr. Rausch said the BWROG has the "authority" to require all BWR utilities to adhere to their directives even if a given utility is not a formal member of the BWROG. This authority exists when a given issue being addressed is designated "generic."

Future plans of the BWROG include:

- EPRI has been requested to provide critical review of the BWROG analysis to assure no important issues have been overlooked.

- The BWROG Executive Oversight Committee has authorized a study of intermediate and long-term solutions to this issue.

- The BWROG Stability Committee met December 13-15, 1988, to formulate plans and initiate studies. The BWROG plans to review progress with the NRC in 6 months, and to identify viable long-term solution within 12 months.

Mr. L. Phillips, NRR, discussed the status of the NRR review of the BWR stability issue. In response to Mr. Michelson, Mr. Phillips said NRC believes the power oscillations can range up to 3-500% of nominal power level.

Mr. Phillips indicated that NRR is preparing a commission paper that will provide a status of the actions taken on this issue. NRR sees the following safety concerns:

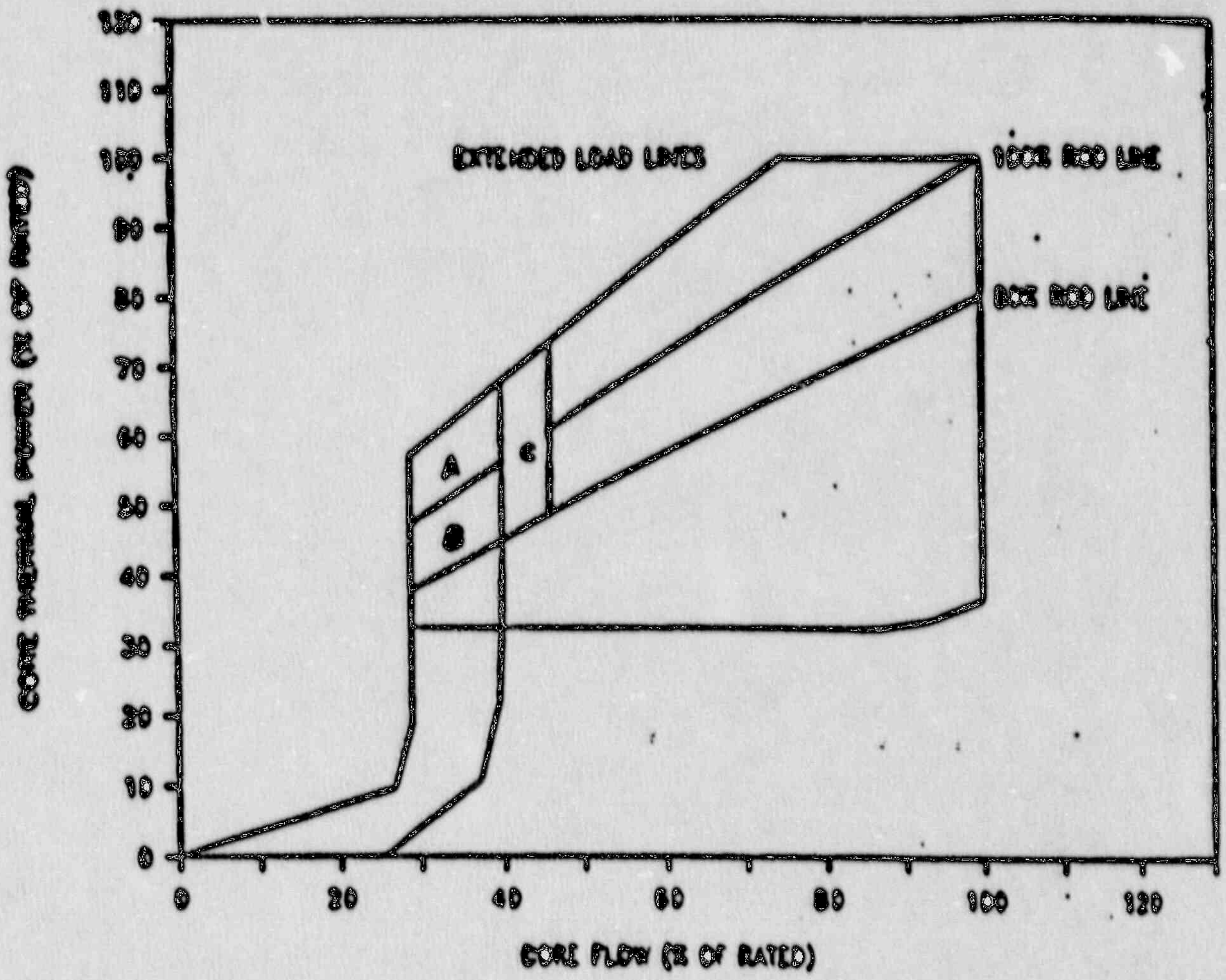
- For region-wise/out-of-phase oscillations fuel design limits (MCPR) may be exceeded prior to detection.

- The maximum amplitude of oscillations during ATWS is not known for core-wide/in phase oscillations. The effects of continuing large amplitude oscillations on core thermal power and the effects on pressure during ATWS need to be evaluated.

NRR noted that BNL has been performing analyses of oscillation phenomena using the RAMONA 3-B code. Mr. Phillips discussed the main contributors to instability and the actions taken by Japanese and European BWR operators to prevent/mitigate the phenomenon.

INDUSTRY ACTIONS

RECOMMENDED INTERIM CORRECTIVE ACTIONS



- O REGION A/B - EXIT IMMEDIATELY
- O REGION C - MINIMIZE OPERATION
- O SCRAM IF OSCILLATIONS ENCOUNTERED

FIGURE 2

NRR is calling for more calculations to evaluate the effects of large amplitude oscillations on core thermal power. In response to Dr. Kerr, Mr. Phillips indicated that the codes available to the Staff cannot at this time adequately model oscillation behavior; however, he expects the codes will be improved in order to do so.

NRR plans to issue a supplement to NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors." The supplement will approve the use of the BWROG interim corrective actions with modifications. The principal additional requirement is that operators manually scram a plant if a dual recirculation pump trip occurs. NRR expects to receive the BWROG's proposals for long-term corrective actions within one year.

Mr. D. Bessette, NRC/RES, discussed a proposed program and schedule for research on BWR instability. He said the issues that RES sees for this program are:

- What is the minimum critical power ratio a plant may experience during an instability event?

- What effect do control systems and operator actions have?

- What is plant response during ATWS, including effect of ATWS procedures?

Dr. Kerr said he believes that the key safety issue here is the impact of core power oscillations given an ATWS. He asked if RES agrees with this point. Mr. Bessette replied in the affirmative. Dr. Kerr asked RES to focus the remaining presentation on this point.

RES is conducting an assessment of the relevant codes of interest by use of the Swedish FRIGG test facility data and plant data obtained during the LaSalle event. RES will apply the CSAU methodology to this program.

In response to Dr. Kerr, Mr. Bessette said he believes the codes will eventually be able to adequately model oscillation events. RES also indicated that it should take about one year to complete this effort.

Dr. Kerr asked if the Committee had any comments on this issue or if they believed some additional action is required. After some discussion, it was conluded that the current activities to address this concern appear reasonable and the Committee would follow the resolution effort.

XI. Executive Sessions (Open)

A. Subcommittee Reports

There were no subcommittee reports given during this meeting.

B. Reports, Letters and Memoranda (Open)

1. Equipment Qualification - Risk Scoping Study (Report to Chairman Zech dated December 20, 1988)

The Committee recommended that the requirements of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," be reevaluated in light of the conclusions of the study as to the overemphasis on the importance of radiation dose in equipment qualification. It also noted that the failure rates used in PRAs may not be appropriate for accident environments and recommended that the implications of this observation be studied further. The Committee commented favorably on the use of a peer-review process to review the research work and the results and conclusions of risk scoping studies.

C. Other Committee Conclusions (Open)

1. Vogtle Electric Generating Plant

The Committee had, in its report dated August 13, 1985, on the Vogtle Electric Generating Plant, Units 1 and 2, stated that the ACRS would consider the need for further review of Unit 2 if there was a significant delay in the schedule for start-up. The projected start-up date for Unit 2 has slipped about eight months (from June 1988 to February 1989). The Committee decided that it was not necessary to conduct additional review of Unit 2 because of this delay in the projected start-up date.

2. Implementation of Severe Accident Policy for Future LWRs

The Committee decided that time should be scheduled at a future ACRS meeting for an NRC staff briefing on the implementation of the Severe Accident Policy for future LWRs. Mr. Fraley subsequently scheduled discussions for a future (tentatively February 1989) ACRS meeting. (Dr. El-Zeftawy has the action on this item.)

3. Application of Leak-Before-Break Technology

The Committee decided to review the NRC staff's proposal for a Commission policy statement on additional applications of leak-before-break technology and provide comments on this matter. The Subcommittee on Thermal Hydraulic Phenomena (D. Ward/P. Boehnert) was given this assignment. Mr. Fraley informed the Commission and the Office of the Secretary of the Commission of the Committee's decision on Thursday, December 15, 1988. A subcommittee meeting will be scheduled during February 1989 for discussion of this matter.

4. B&W Steam Generators

The Committee decided that it wished to continue to be kept informed of the NRC/industry plans for conducting experimental research on B&W once-through steam generator thermal-hydraulic performance. The Committee requested a briefing from the NRC staff at a future ACRS meeting. (See letter from R. Fraley to V. Stello, dated December 20, 1988.)

D. Future Activities (Open)

1. Future Agenda

The Committee agreed to the tentative future agenda as shown in Appendix II.

2. Future Subcommittee Activities

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

The 344th ACRS meeting was adjourned at 6:15 p.m., Friday, December 16, 1988.

ACRS/ACNW COMMITTEE & SUBCOMMITTEE MEETINGS

5th ACNW Meeting, December 21, 1988, Bethesda, MD, Room P-422, REINSTATED.

Regional Programs, January 5-6, 1989, Region IV Office, 611 Ryan Plaza Drive, Arlington, TX (Boehnert), 8:30 a.m. The Subcommittee will review the activities under the control of the Region IV Office. Attendance by the following is anticipated, and reservations have been made at the Hawthorn Suites (telephone: 817/640-1188), 2401 Brookhollow Plaza Drive, Arlington, TX for the nights of January 4 and 5:

Dr. Remick (5th only)
Mr. Carroll
Dr. Kerr
Mr. Michelson
Mr. Ward

Improved Light Water Reactors, January 10, 1989, 7920 Norfolk Avenue, Bethesda, MD (Alderman), 8:30 a.m., Room P-114. The Subcommittee will review the proposed final version of 10 CFR Part 52, Early Site Permits, Standard Design Certification. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Wylie (tent.)
Mr. Michelson
Dr. Siess
Mr. Ward

Auxiliary and Secondary Systems, January 11, 1989, 7920 Norfolk Avenue, Bethesda, MD (Duraiswamy), 8:30 a.m. - 12:00 noon, Room P-422. The Subcommittee will discuss Control Air System Design and Operating Experience, and the proposed resolution of Generic Issue 43, "Air Systems Reliability." Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson
Mr. Carroll
Dr. Siess

Mechanical Components, January 11, 1989, 7920 Norfolk Avenue, Bethesda, MD (Igne), 1:00 p.m., Room P-422. The Subcommittee will discuss Air Operated Valve Testing and Operating Experience (including Solenoid Air Control Valves) and other related matters. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson
Mr. Carroll
Dr. Siess
Mr. Wylie
Mr. Wohld

345th ACRS Meeting, January 12-14, 1989, Bethesda, MD, Room P-114.

6th ACNW Meeting, January 23-24, 1989, Bethesda, MD, Room P-114.

344TH ACRS MEETING MINUTES
(Rev.)

APPENDIX II

TENTATIVE ACRS AGENDA ITEMS

January 12-14, 1989

Sodium Advanced Fast Reactor (SAFR) (Open) (DAW/MME) Estimated Time: 2 hrs. Complete ACRS discussion and preparation of ACRS report on the preapplication review of this standardized plant.

Fitness for Duty (Open) (FJR/HA) Estimated Time: 1 hr. - Review and report on proposed NRC rule regarding fitness for duty of nuclear power plant operators.

Standard Design Certification and Combined Licenses for Nuclear Power Plants (Open) (CJW/HA) Estimated Time: 2-3/4 hrs. - ACRS review and report regarding proposed final version of 10 CFR Part 52 regarding Early Site Permits, Standard Design Certifications, and Combined Licenses for nuclear power plants.

Meeting with NRC Commissioner James E. Curtiss (Open) (WK/RFF) Estimated Time: 1 hr. - Discuss items of mutual interest regarding ACRS/NRC activities.

ECCS (Open) (DAW/PAB) Estimated Time: 1-3/4 hr. - Prepare ACRS report on proposed NRC Code Scaling Applicability and Uncertainty Evaluation Methodology.

Containment Systems (Open) (DAW/MDH) Estimated Time: 2½ hrs. - Complete ACRS discussion and preparation of report to NRC regarding proposed recommendations for BWR Mark I containment performance and improvements.

NRC Quantitative Safety Goals (Open) (DAW/MDH) Estimated Time: 2 hrs. - Complete ACRS discussion and preparation of ACRS report to NRC on the proposed plan for implementation of the NRC's Safety Goal Policy.

Generic Issue 43, "Air Systems Reliability" (Open) (CM/SD) Estimated Time: 2 hrs. - Review and comment on proposed resolution of Generic Issue 43, "Air Systems Reliability."

Nuclear Safety Research Program (Open) (CPS/SD) Estimated Time: ½ hr. - Discuss proposed ACRS annual report to the U.S. Congress on the NRC safety research program.

Anticipated ACRS Activities (Open) (WK/RFF/MWL) Estimated Time: ½ hr. - Discuss topics proposed for consideration by the Committee.

ACRS Subcommittee Reports (Open) (FJR/RFF) Estimated Time: 1½ hr. - Discuss anticipated ACRS subcommittee activities and hear and discuss the status of assigned subcommittee and designated members activities.

New Members (Closed) (FJR/NSL) Estimated Time: ½ hr. - Discuss the qualifications of candidates proposed for consideration as nominees for appointment to the ACRS.

Accident Management (Open) (WK/MDH) Estimated Time: 1 hr. - Briefing regarding NRC staff development of a program plan on severe accident management.

APPENDIX I
MINUTES OF THE 344TH ACRS MEETING
DECEMBER 15-16, 1988

THURSDAY, DECEMBER 15, 1988

Public Attendees	NRC Attendees
Stephanie Sharron, SERCH-Bechtel	R. Landry, RES
R. T. Lancet, Rockwell International	L. Soffer, RES
C. L. Allen, SAIC	D. Persinko, NRR
G. J. Van Tuyle, Brookhaven Natl. Lab.	W. Beckner, RES
N. Suttora, NUS	R. W. Houston, RES
G. Sherwood, DOE	
Gil Brown, NUMARC	
L. Gifford, GE	
W. P. McCaughey, BG&E	
F. T. Stetsom, SAID	
J. Russell, DOE	

FRIDAY, DECEMBER 16, 1988

Public Attendees	NRC Staff
Wolfgang Wulff, Brookhaven Natl. Lab.	E. Beckjord, RES
Ali Tabatabai, PNL	C. Bartlett, RES
M. E. Waterman, INEL/EG&G Idaho	F. Coffman, RES
G. S. Lellouche, SLI	D. Persinko, NRR
F. C. Phifer, SERCH/BECHTEL	E. Shomaker, OGC
Art Bivens, NUMARC	J. Shea, GPA
Tom Tausch, Commonwealth Edison & BWROG	Dave Lange, R I, DRS
	L. Wiens, NRR
	K. Perkins, NRR
	H. Scott, RES

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ACRS MEETING

DATE DEC. 15-16, 1988

ATTENDEES	Thursday	Friday	Saturday
Dr. William Kerr, Chairman	✓	✓	
Dr. Forrest J. Remick, Vice Chairman	✓	✓	
Mr. James C. Carroll	✓	✓	
Dr. Harold W. Lewis	✓	✓	
Mr. Carlyle Michelson	✓	✓	
Dr. Paul G. Shewmon	✓	✓	
Dr. Chester P. Siess	✓	✓	
Mr. David A. Ward	✓	✓	
Mr. Charles J. Wylie	—	—	

APPENDICES
MINUTES OF THE 344TH ACRS MEETING
DECEMBER 15-16, 1989

Human Factors, January 26, 1989, 7920 Norfolk Avenue, Bethesda, MD (Alderman), 8:30 a.m., Room P-422. The Subcommittee will review the Human Factors Research Program Plan. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Remick	Mr. Michelson
Mr. Carroll	Mr. Ward
Dr. Kerr	Mr. Wylie

Auxiliary and Secondary Systems, January 27, 1989, 7920 Norfolk Avenue, Bethesda, MD (Duraiswamy), 8:30 a.m. - 1:00 p.m., Room P-422. The Subcommittee will review the adequacy of the proposed Staff's plans to implement the recommendations resulting from the Fire Risk Scoping Study. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson	Dr. Siess
Mr. Carroll	Mr. Wylie

Mechanical Components, January 27, 1989, 7920 Norfolk Avenue, Bethesda, MD (Igne), 2:00 p.m., Room P-422. The Subcommittee will review the proposed resolution of Generic Issues 70, "PORV Reliability," and 94, "Low Temperature Over-Pressure Protection," and other related matters. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson	Dr. Siess
Mr. Carroll	Mr. Wylie (tent.)

Babcock & Wilcox Reactor Plants, February 1 & 2, 1989, Sacramento, CA (Igne), 8:30 a.m. The Subcommittee will discuss the lessons learned from the approximately 2-year shutdown of Rancho Seco that occurred following the December 16, 1985, overcooling event. Topics for discussion include monitoring extended start-up program as well as plant and organization changes as a result of the restart effort. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Wylie	Mr. Michelson
Mr. Carroll	Mr. Ward
Dr. Kerr	

Safety Research Program, February 8, 1989, 7920 Norfolk Avenue, Bethesda, MD (Duraiswamy), 8:30 a.m., Room P-114. The Subcommittee will discuss the ongoing and proposed NRC Safety Research program and budget. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Siess	Dr. Shewmon
Dr. Kerr	Mr. Ward
Mr. Michelson	Mr. Wylie
Dr. Remick	

346th ACRS Meeting, February 9-11, 1989, Bethesda, MD, Room P-114.

7th ACNW Meeting, February 22-23, 1989, Bethesda, MD, Room P-114.

Occupational and Environmental Protection Systems, March 1-2, 1989, 7920 Norfolk Avenue, Bethesda, MD (Igne), 8:30 a.m., Room P-114. The Subcommittee will discuss the general status of emergency planning for nuclear power plants. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Remick
Mr. Wylie
et al.
Mr. Kathren
Dr. Shapiro

347th ACRS Meeting, March 9-11, 1989, Bethesda, MD, Room P-114.

Materials and Metallurgy, March 15-16, 1989, Columbus, OH (Igne), 8:30 a.m. The Subcommittee will review the degraded piping program, including NDE and aging of centrifugally cast stainless steel piping material. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Shewmon
Dr. Lewis
Mr. Michelson
Mr. Ward
Mr. Etherington
Dr. Hutchinson
Dr. Thompson

8th ACNW Meeting, March 22-23, 1989, Bethesda, MD, Room P-114.

Limerick 2, March 28, 1989, Philadelphia, PA (Quittschreiber), 8:30 a.m. The Subcommittee will review Limerick 2 for a low power operating license. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Kerr
Dr. Lewis
Dr. Siess

Maintenance Practices and Procedures, March 30, 1989, 7920 Norfolk Avenue, Bethesda, MD (Alderman), 8:30 a.m., Room P-114. The Subcommittee will review the proposed maintenance rule. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson
Mr. Carroll
Mr. Wylie

Materials and Metallurgy, April 27, 1989, Palo Alto, CA (Igne). The Subcommittee will discuss the status of the following matters: erosion/corrosion of pipes, hydrogen water chemistry, zinc addition to primary coolant loop and its effects on materials, decontamination effects on materials, and other related matters. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Shewmon
Dr. Lewis
Mr. Michelson
Mr. Ward
Mr. Etherington

International Conference on Quality, May 14-18, 1989, San Diego, CA (Igne). Attendance by the following is anticipated:

Dr. Remick
Dr. Siess
Mr. Ward
et al.

Advanced Pressurized Water Reactors, Date to be determined (January/February), Bethesda, MD (El-Zeftawy). The Subcommittee will review the licensing review bases document being developed by the Staff for Combustion Engineering's Standard Safety Analysis Report-Design Certification (CESSAR-DC). Attendance by the following is anticipated:

Mr. Carroll
Dr. Kerr
Mr. Michelson
Dr. Remick
Dr. Shewmon
Mr. Wylie

Decay Heat Removal Systems, Date to be determined (January/February), Bethesda, MD (Boehnert). The Subcommittee will continue its review of the proposed resolution of Generic Issue 23, "RCP Seal Failures." Attendance by the following is anticipated:

Mr. Ward
Dr. Kerr
Mr. Wylie
Dr. Catton
Mr. Davis

General Electric Reactor Plants (Peach Bottom Restart), Date to be determined (January/February), Bethesda, MD (Alderman). The Subcommittee will review the proposed restart plan for the Peach Bottom Plant. Attendance by the following is anticipated:

Dr. Kerr
Dr. Lewis
Mr. Michelson
Dr. Siess

Joint Core Performance/Thermal Hydraulic Phenomena, Date to be determined (January/February), Bethesda, MD (Boehnert/Houston). The Subcommittee will review the implications of the core power oscillation event at LaSalle, Unit 2. Attendance by the following is anticipated:

Dr. Kerr
Mr. Ward
Mr. Michelson
Dr. Shewmon
Mr. Wylie
Dr. Catton
Dr. Lee
Dr. Lipinski
Dr. Plesset
Mr. Schrock
Dr. Sullivan
Dr. Tien

AC/DC Power Systems Reliability, Date to be determined (February), Bethesda, MD (El-Zeftawy). The Subcommittee will review the proposed resolution of Generic Issue 128, "Electrical Power Reliability." Attendance by the following is anticipated:

Mr. Wylie
Mr. Carroll
Dr. Kerr
Dr. Lewis
Mr. Davis
Dr. Lee

Instrumentation and Control Systems, Date to be determined (February/March), Bethesda, MD (El-Zeftawy). The Subcommittee will review the proposed resolution of Generic Issue 101, "Break Plus Single Failure in BWR Water Level Instrumentation." Attendance by the following is anticipated:

Dr. Kerr
Mr. Carroll
Dr. Lewis
Mr. Michelson
Mr. Wylie
Mr. Davis
Dr. Lipinski

Extreme External Phenomena, Date to be determined (February/March), Bethesda, MD (Igne). The Subcommittee will review planning documents on external events. Attendance by the following is anticipated:

Dr. Siess
Dr. Kerr
Dr. Lewis
Mr. Michelson
Mr. Wylie

Instrumentation and Control Systems, Date to be determined (March), Bethesda, MD (El-Zeftawy). The Subcommittee will review the ATWS rule implementation status. Attendance by the following is anticipated:

Dr. Kerr
Mr. Carroll
Dr. Lewis
Mr. Michelson
Mr. Wylie
Mr. Davis
Dr. Lipinski

Advanced Pressurized Water Reactors, Date to be determined (April), Bethesda, MD (El-Zeftawy). The Subcommittee will discuss the comparison of WAPWR (RESAR SP/90) design with other modern plants (in U.S. and abroad). Attendance by the following is anticipated:

Mr. Carroll
Dr. Kerr
Mr. Michelson
Dr. Remick
Dr. Shewmon
Mr. Wylie

Plant Operating Procedures, Date to be determined (spring), Bethesda, MD (Igne). The Subcommittee will review the status of the NRC program on Technical Specifications update. Also, it will review an anonymous letter to Ms. E. Weiss (Union of Concerned Scientist), dated Sept. 27, 1988, on Technical Specifications inadequacies. Attendance by the following is anticipated:

Mr. Michelson	Mr. Ward
Mr. Carroll	Mr. Wylie
Mr. Remick	

Materials and Metallurgy, Date to be determined (2nd or 4th week of May), Bethesda, MD (Igne). The Subcommittee will review low upper shelf fracture energy concerns of reactor pressure vessels. Attendance by the following is anticipated:

Dr. Shewmon	Mr. Ward
Dr. Lewis	Mr. Etherington
Mr. Michelson	

Decay Heat Removal Systems, Date to be determined, Bethesda, MD (Boehnert). The Subcommittee will explore the issue of the use of feed and bleed for decay heat removal in PWRs. Attendance by the following is anticipated:

Mr. Ward	Mr. Wylie
Dr. Kerr	Dr. Catton
Mr. Michelson	Mr. Davis

Thermal Hydraulic Phenomena, Date to be determined, Bethesda, MD (Boehnert). The Subcommittee will discuss the status of Industry best-estimate ECCS model submittals for use with the revised ECCS Rule. Attendance by the following is anticipated:

Mr. Ward	Dr. Catton
Dr. Kerr	Dr. Plesset
Mr. Michelson	Mr. Schrock
Mr. Wylie	Dr. Sullivan
	Dr. Tien

Auxiliary and Secondary Systems, Date to be determined, Bethesda, MD (Duraiswamy). The Subcommittee will discuss the: (1) criteria being used by utilities to design Chilled Water Systems, (2) regulatory requirements for Chilled Water Systems design, and (3) criteria being used by the NRC staff to review the Chilled Water Systems design. Attendance by the following is anticipated:

Mr. Michelson	Mr. Wylie
Mr. Carroll	

APPENDIX IV
MINUTES OF THE 344TH ACRS MEETING
DECEMBER 15-16, 1989

MEETING NOTEBOOK

Tab

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Slides used by the speaker during the presentation
Table of Contents, Tentative Agenda
Status Report with Attachments
Att. I: Letter to V. Stello, NRC from Mr. D. F. Bunch, DOE, dated August 15, 1988 re selection of SAFR and PRISM for continued Department [of Engergy] support
Att. II: Letter to T. J. Garrish, DOE from V. Stello, NRC, dated August 17, 1988, re two issues developed during NRC review of three advanced reactor conceptual designs, MHTGR, PRISM, and SAFR
Att. III: Memo for NRC Commissioners from V. Stello, Subj.: COMMISSION ACTION ON THE KEY LICENSING AND STANDARDIZATION ISSUES ASSOCIATED WITH THE DOE ADVANCED REACTOR CONCEPTS (SECY-88-202 and SECY-88-203), dated August 18, 1988

3 CONTAINMENT SYSTEMS - Table of Contents, Tentative Agenda
Slides used by the speaker during the presentation
Status Report with Attachments
Att. I: ACRS report of December 17, 1986
Att. II: draft copy of SECY on Mark I undated (INTERNAL COMMITTEE USE)
Att. III: Draft Generic Letter undated
Att. IV: Selected Slides Used by RES at Dec. 6, 1988 Meeting of Containment Systems Subcommittee (Mark I issues)
Att. V: Selected Slides Used by NRR at December 6, 1988 Meeting of Containment Systems Subcommittee (Overview of EPG Revision 4, Details on Venting)

4 EQUIPMENT QUALIFICATION-RISK SCOPING STUDY - Table of Contents, Tentative Agenda
Status Report: Memo to ACRS Members and ACRS Technical Staff from S. Duraiswamy, ACRS Staff, Subj.: STATUS REPORT - EQUIPMENT QUALIFICATION-RISK SCOPING STUDY - 344TH ACRS MEETING, DECEMBER 15-17, 1988, BETHESDA MARYLAND, dated November 29, 1988

Executive Summary - Equipment Qualificaton-Risk Scoping Study
General Conclusions/Recommendations - Equipment Qualification Risk-Scoping Study
Discussion of Peer Review Comments - Equipment Qualificaton-Risk Scoping Study

5 QUANTITATIVE SAFETY GOALS - Table of Contents, Tentative Agenda
Status Report
Slides used by the speaker during the presentation
Certified Minutes of Safety Philosophy, Technology, and Criteria (SPT&C) Subcommittee Meeting on September 1, 1988

ACRS Report of 4/12/88
Draft Plan, "Implementation of Safety Goal Policy," received December 7, 1988

7 REVIEW OF NRC RES CODE SCALING APPLICABILITY AND UNCERTAINTY (CSAU) METHODOLOGY -
Slides used by the speaker during his presentation
Table of Contents, Presentation Schedule,
Status Report
ACRS ltr 12/16/87
ACRS ltr, Prop Rev. ECCS Rule, May 10, 1988
Memo to Paul Boehnert from Ivan Catton, ACRS Consultant, Subj.: TECHNICAL PROGRAM GROUP (TPG) MEETING, DEVELOPMENT OF CODE SCALING, APPLICABILITY, AND UNCERTAINTY (CSAU) METHODOLOGY, NICHOLSON LANE BUILDING, SEPTEMBER 27-28, 1988 (INTERNAL COMMITTEE USE)

8.1 FUTURE ACRS ACTIVITIES - Memo for ACRS Members from R. Fraley, Subj: FUTURE ACRS ACTIVITIES - 345TH ACRS MEETING - JANUARY 12-14, 1989, dated December 14, 1989 w/attachment (Future Agenda)

8.2 empty

8.3 Memorandum for ACRS Staff and ACRS Fell[illegible] from R. Fraley, Subj.: ASSIGNMENT OF ACNW/ACRS RESPONSIBILITIES, dated November 23, 1988 with Attachment (Chart "Distribution of Responsibilities," Revision 2: November 23, 1988

10 OPERATOR REQUALIFICATION
Slides used by the speaker during his presentation
Table of Contents, Tentative Agenda
ES-601, "Administration of NRC Requalification Evaluation"

11 BRIEFING ON ACTIONS TAKEN IN RESPONSE TO LASALLE CORE POWER OSCILLATION EVENT
Slides used by the speaker during the presentation
Table of Contents, Agenda, Status Report
Memorandum for W. Kerr from P. Boehnert, Subject: NRC Augmented Inspection Team (AIT) Report: LaSalle Unit 2 Core Power Fluctuations Event of March 9, 1988, dated May 25, 1988
AEOD Special Report: "AEOD Concerns Regarding the Power Oscillation Event at LaSalle 2,"
NRC Bulletin No. 88-07: Power Oscillations in Boiling Water Reactors (BWRs), dated June 15, 1988
Memorandum for V. Stello from Chairman Zech, Subject: Power Oscillations Event of March 9, 1988 at LaSalle 2, dated July 5, 1988
Memorandum for W. Kerr from P. Boehnert, Subject: NRC Meeting with BWR Owners Group Representatives: LaSalle Core Power Oscillation Event - June 24, 1988, Rockville, MD, dated July 15, 1988
Memo from P. Boehnert to W. Kerr/D. Ward- Report on NRC/BWROG Meeting of November 14, 1988 on T/H Stability (PROPRIETARY)

14.1 REPORT ON PLANNING SESSION FOR INTERNATIONAL CONFERENCE ON QUALITY IN THE NUCLEAR POWER INDUSTRY
Table of Contents
Memorandum for C. P. Siess from H. Alderman, Subject: Planning Session for the International Conference on Quality in the Nuclear Power Industry

14.2 REPORT OF VISIT TO FERMI-2 PLANT
Table of Contents, Status Report

15.1 ACRS Activities
Memorandum for ACRS Members from M. Libarkin, Subject: Information on Topics Discussed and Time Spent by ACRS, dated December 8, 1988

MEETING HANDOUTS

No.	Agenda Item	
1	6.1	Schedule Memorandum for ACRS Members and Staff from S. Duraiswamy, dated December 1, 1988, Status Report - Meeting with the Director of the Office of Nuclear Regulatory Researarch (RES) - 344th ACRS Meeting, December 15-17, 1988, Bethesda, Maryland with the Attachment: NRC Announcement No. 118, dated July 6, 1988, SUBJECT: REORGANIZATION OF THE OFFICE OF NUCLEAR REGULATORY RESEARCH
2		December 13, 1988 Trip Report from Dr. Kerr on Visit to Korea
3	7.0	I. Catton Report on December 7, 1988 T/H Phenomena Subcommittee Meeting (INTERNAL COMMITTEE USE) Working Minutes of December 7, 1988 T/H Phenomena Subcommittee Meeting (INTERNAL COMMITTEE USE)
4	3	I. Catton, Consultant's Comments on Mark I Subcommittee Meeting of December 6, 1988, dated December 11, 1988 (INTERNAL COMMITTEE USE) Latest version of Staff's proposed generic letter on BWR Mark Is, obtained on December 14, 1988 (INTERNAL COMMITTEE USE)
5	15.1	Memorandum for Commissioners from V. Stella, EDO, Subject: Implementation of Severe Accident Policy for Evolutionary LWR Designs, dated December 1, 1988

6	2	Memorandum for D. Ward from Stewart W. Long, ACRS Fellow Subject: Comments on SAFR and PRISM Design Features in Support of Upcoming Subcommittee Meeting, dated 9 December 1988
7	7.0	Memorandum for D. Ward, from P. Boehnert, Subject: Response of T/H Phenomena Consultants to your Question Concerning Usefulness of T/H Codes, dated December 14, 1988
8	8.2	Memorandum for ACRS Members from R. Fraley, dated December 14, 1988, Subject: Future ACRS Activities - 345th ACRS Meeting - January 12-14, 1989