ACRS-2460
PDR 3/25/87

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317TH ACRS MEETING
SEPTEMBER 11-13, 1986
WASHINGTON, D.C.

DESIGNATED ORIGINAL

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Patent Application 806,572: Aminophenoxycyclotriphosphazene Cured Epoxy Resins and the Composites Laminates, Adhesives and structures thereof; filed November 21, 1985.
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Patent Application 838,648: Floating Emitter Solar Cell Junction Transistor; filed March 11, 1986.
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Patent Application 831,377: Adjustable Mount for Electro-Optic Transducers in an Evacuted Cryogenic System; filed February 20, 1986.
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Patent Application 831,193: Method and Apparatus for Measuring Distance; filed February 20, 1986.
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Patent Application 855,983: Lightning Discharge Protection Rod; filed April 24, 1986.
Patent Application 832,296: Heat Treatment for Superalloy; filed February 24, 1986.
Patent Application 855,879: Polyether-Polyester Graft Copolymer; filed April 24, 1986.
Patent Application 838,649: Active Control of Boundary Layer Transistor and Turbulence; filed March 11, 1986.
Patent Application 765,991: Planar Oscillatory Stirring Apparatus; filed August 15, 1985.

Dated: August 15, 1986.

Edward A. Frankle,

Deputy General Counsel.

[FR Doc. 86–19177 Filed 8–25–86; 8:45 am]

BILLING CODE 7510-01-M

[Notice 86–56]

Intent To Grant an Exclusive Patent License

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of Intent to Grant an Exclusive Patent License.

SUMMARY: NASA hereby gives notice of intent to grant to Ernest W. Millen, Seaford, Virginia, a limited, exclusive, royalty-bearing, revocable license to practice the invention as described in U.S. Patent No. 4,586,140 for a "Aircraft Liftmeter," which issued on April 29, 1986 to the Administrator of the National Aeronautics and Space Administration on behalf of the United States of America. The proposed exclusive license will be for a limited number of years and will contain appropriate terms and conditions to be negotiated in accordance with the NASA Patent Licensing Regulations, 14 CFR Part 1245, Subpart, 2. NASA will negotiate the final terms and conditions and grant the exclusive license unless, within 60 days of the date of the Notice, the Director of Patent Licensing receives written objections to the grant, together with supporting documentations. The Director of Patent Licensing will review all written responses to the Notice and then recommend to the Associate General Counsel (Intellectual Property) whether to grant the exclusive license.

DATE: Comments to this notice must be received by October 27, 1986.

ADDRESS: National Aeronautics and Space Administration, Code GP, Washington, DC 20546.

FOR FURTHER INFORMATION CONTACT: Mr. John G. Mannix, (202) 453–2430.

Dated: August 14, 1986.

Edward A. Frankle,

Deputy General Counsel.

[FR Doc. 86–19176 Filed 8–25–86; 8:45 am]

BILLING CODE 7510-01-M

NUCLEAR REGULATORY COMMISSION

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 September 11–13, 1986, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published in the Federal Register on August 19, 1986.

317th mtg
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[Thu]rsday, September 11, 1986

[8:]30 A.M.–8:45 A.M.: Report of ACRS Chairman (Open)—The ACRS Chairman will report briefly regarding items of current interest to the Committee.

8:45 A.M.–11:45 A.M.: Improved Light-Water Reactors (Open)—The members of the Committee will discuss proposed ACRS comments and recommendations to the NRC regarding proposed characteristics of improved light-water reactors.

11:45 A.M.–12:30 P.M.: Future ACRS Activities (Open)—The members will discuss anticipated ACRS subcommittee meetings and items proposed for full Committee consideration. The schedule for ACRS full Committee meetings for CY 1987 will also be discussed.

1:30 P.M.–1:50 P.M.: Topics for Meeting with NRC Commissioners (Open)—The members will discuss the presentation of its report dated August 12, 1986 (Revised 8/15/86) on the proposed NRC policy statement on standardization of nuclear power plants.

2:00 P.M.–3:30 P.M.: Meeting with NRC Commissioners (Open)—Presentation and discussion of ACRS report dated August 12, 1986 (Revised 8/15/86) on the proposed NRC [Sta]ndardization Policy Statement.

[3:]45 P.M.–6:00 P.M.: Emergency Core [Coo]ling Systems (Open)—The members will hear presentations and discuss proposed changes in NRC regulatory requirements for emergency core cooling systems. Representatives of the NRC Staff will participate in this discussion.

6:00 P.M.–6:30 P.M.: Primary System Integrity (Open)—The members will hear and discuss the report of its subcommittee regarding research activities related to the integrity of the primary coolant systems in nuclear power plants.

Friday, September 12, 1986

8:30 A.M.–9:30 A.M.: Decay Heat Removal (Open)—The members will hear and discuss a Subcommittee report regarding activities related to resolution of Unresolved Generic Issue 124, Auxiliary Feedwater System Reliability. Members of the NRC Staff will participate as appropriate.

9:30 A.M.–10:30 A.M.: International Operating Experience (Open)—Briefing by member of the U.S. Team regarding the sequences which contributed to the Chernobyl Nuclear Power Plant accident.

10:45 A.M.–12:00 Noon and 1:00 P.M.–2:15 P.M.: Babcock and Wilcox Light-[Wa]ter Reactor Safety (Open/Closed)—[The] members will hear and discuss a presentation by representatives of the Babcock and Wilcox Company regarding plans for review of the long-term safety of B&W nuclear plants.

Portions of this session may be closed as necessary to discuss Proprietary Information related to B&W nuclear plants.

2:15 P.M.–5:15 P.M. Long-Range Planning (Open)—The members of the Committee will discuss proposed ACRS comments and recommendations regarding the preparation of a long-range plan for NRC activities.

5:15 A.M.–6:30 P.M.: ACRS Subcommittee Activities (Open)—The members will hear and discuss reports of designated ACRS subcommittees regarding safety-related matters, including the NRC incident investigation program, activities of the NRC Office of Inspection and Enforcement, and evaluation of seismic margins with respect to nuclear power stations.

Saturday, September 13, 1986

8:30 A.M.–12:30 P.M.: Preparation of ACRS Reports (Open/Closed)—The members will discuss proposed ACRS reports and memoranda to the NRC regarding items considered during this meeting. In addition, proposed ACRS comments on seismic qualification of safety-related equipment in nuclear power plants and use of aptitude testing in the selection of nuclear power plant personnel will be discussed.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to the matters being discussed.

1:30 p.m.–2:30 p.m.: ACRS Subcommittee Activities (Open/Closed)—The members will hear and discuss reports of its subcommittees on management and conduct of ACRS activities, including the prioritization and allocation of ACRS resources and the non-ACRS activities of individual members.

Portions of this session will be closed as necessary to discuss information; the release of which would represent a clearly unwarranted invasion of personal privacy.

2:30 p.m.–3:30 P.M.: Miscellaneous (Open/Closed)—The member will complete discussion of matters noted above.

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

Procedures for the conduct of and participation in ACRS meetings were published in the Federal Register on October 2, 1985 (50 FR 191). 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, R.F. Fraley, prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with subsection 10(d) Pub. L. 92-463 that it is necessary to close portions of this meeting as noted above to discuss Proprietary Information [5 U.S.C. 552b(c)(4)] applicable to the facilities being discussed and information the release of which would represent a clearly unwarranted invasion of personal privacy [5 U.S.C. 552b(c)(6)].

Further information regarding topics to be discussed, whether the meeting has been canceled 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 202/634-3265), between 8:15 a.m. and 5:00 p.m.

Dated: August 21, 1986.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 86-19263 Filed 8-25-86; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-410]

Niagara Mohawk Power Corp.; Nine Mile Point Nuclear Station, Unit 2; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of exemptions from certain requirements of 10 CFR Part 50 to the Niagara Mohawk Power Corporation (the applicant) for the Nine Mile Point Nuclear Station, Unit 2

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

Revised: September 4, 1986

SCHEDULE AND OUTLINE FOR DISCUSSION
317TH ACRS MEETING
SEPTEMBER 11-13, 1986
WASHINGTON, D. C.

Thursday, September 11, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

1)	8:30 - 8:45 A.M.	Report of ACRS Chairman (Open) 1.1) Opening Statement (DAW) 1.2) Items of current interest (DAW/RFF)
2)	8:45 - 11:30 A.M. BREAK: 10:00-10:15)	Improved Light-Water Reactors (Open) 2.1) Discuss proposed ACRS comments/recommendations regarding the characteristics of improved LWRs (DO/RKM)
3)	11:30 - 12:00 Noon	Future ACRS Activities (Open) 3.1) Anticipated ACRS Subcommittee activities (MWL) 3.2) Proposed ACRS activities (DAW/RFF) 3.3) Proposed ACRS meeting dates for CY 1987 (DAW, et al/RFF)
	12:00 - 1:00 P.M.	LUNCH
4)	1:00 - 1:30 P.M.	Primary System Integrity (Open) 4.1) Report of ACRS Subcommittee on Metal Components regarding research activities related to nuclear power plant primary system integrity (PGS/EGI)
5)	1:30 - 1:50 P.M.	Discussion regarding Meeting with Commissioners (Open) 5.1) Discuss presentation regarding ACRS report dated August 12, 1986 (Revised, August 15, 1986) on Proposed NRC Standardization Policy Statement (CJW/HA)
6)	2:00 - 3:30 P.M.	Meeting with NRC Commissioners (Room 1130-H) (Open) 6.1) Presentation and discussion regarding ACRS report on Proposed NRC Standardization Policy Statement dated August 12, 1986 (Revised, August 15, 1986)
	3:30 - 3:45 P.M.	BREAK

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7) 3:45 - 5:45 P.M. Emergency Core Cooling Systems (Open)

7.1) Report of ACRS Subcommittee regarding proposed changes in 10 CFR Part 50.46, Acceptance Criteria for ECCS for Light-Water Reactors and 10 CFR Part 50, Appendix K, ECCS Evaluation Models (CYM/PAB)

7.2) Meeting with NRC Staff

8) 5:45 - 6:45 P.M. Seismic Qualification of Safety-Related Equipment in Operating Nuclear Plants (Open)

8.1) Discuss proposed ACRS report to NRC (CJW, et al/RKM)

8.2) Discussion with representatives of the NRC Staff

Friday, September 12, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

9)	8:30 - 9:30 A.M.	Decay Heat Removal (Open) 9.1) Report of ACRS Subcommittee on Decay Heat Removal regarding resolution of USI A-124, Auxiliary Feedwater Systems Reliability (DAW/PAB)
10)	9:30 - 11:45 A.M. (BREAK: 10:30- 10:45)	Foreign Operating Experience (Open) 10.1) Briefing by representative of U.S. Team regarding the Chernobyl Nuclear Power Plant accident
11)	11:45 - 12:45 P.M.	Babcock & Wilcox Nuclear Power Plant Long-Term Safety (Open/Closed) 11.1) Opening remarks by Chairman, ACRS Subcommittee on B&W Reactors (CJW/RKM) 11.2) Presentation by representatives of the B&W Owners Group regarding evaluation of the long-term safety of B&W reactors (Note: Portions of this session will be closed as required to discuss Proprietary Information applicable to this matter.)
	12:45 - 1:45 P.M.	LUNCH
11)	1:45 - 3:00 P.M.	Babcock & Wilcox Nuclear Power Plant Long-Term Safety (Open/Closed) 11.3) Continue discussion noted above (Note: Portions of this session will be closed as required to discuss Proprietary Information applicable to this matter.)
	3:00 - 3:15 P.M.	BREAK
12)	3:15 - 3:30 P.M.	ACRS Subcommittee Activities (Open) 12.1) Report of ACRS Subcommittee regarding NRC incident investigation procedures (HWL/GRQ)
13)	3:30 - 4:15 P.M.	Seismic Margins Program (Open) 13.1) Discuss results of ACRS August 6, 1986 subcommittee meeting on seismic margins in the design of nuclear power plants (DO/RPS)
	4:15 - 4:30 P.M.	BREAK

14) 4:30 - 6:30 P.M. Long-Range Planning (Open)

14.1) Discuss proposed ACRS comments and recommendations regarding preparation of a long-range plan for NRC activities (MWC/RKM)

Saturday, September 13, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

15) 8:30 - 12:15 A.M. (BREAK: 10:00-10:15)

ACRS Reports to NRC (Open/Closed)

15.1) Discuss proposed ACRS reports on:
- 15.1-1) 8:30-9:30: Seismic Qualification of Equipment in Operating Nuclear Plants (CJW/RKM)
- 15.1-2) 9:30-10:00: B&W Long-Term Safety Review (tentative) (CJW/RKM)
- 10:00-10:15: BREAK
- 15.1-3) 10:15-10:45: Auxiliary Feedwater System Reliability (tentative) (DAW/PAB)
- 15.1-4) 10:45-11:15: Primary System Integrity (tentative) (PGS/EGI)
- 15.1-5) 11:15-11:45: ECCS Requirements (CYM/PAB)
- 15.1-6) 11:45-12:15: Long-Range Plan (MWC/RKM)

12:15 - 1:15 P.M. LUNCH

16) 1:15 - 2:00 P.M.

ACRS Subcommittee Activities (Open)

16.1) 1:15-1:45: Report of ACRS Subcommittee regarding Phase I of the NRC Maintenance Program Plan (CYM/HA)

16.2) 1:45-2:00: Report of ACRS Subcommittee regarding activities of the NRC Office of Inspection and Enforcement (CYM/PAB)

17) 2:00 - 3:15 P.M.

ACRS Subcommittee Activities (Open/Closed)

17.1) 2:00-3:00: Report of Subcommittee meetings regarding ACRS Management/Planning (July 9, and September 10, 1986) and ACRS Procedures and Administration (August 6, 1986) (DAW/RFF)
(Note: Portions of this session will be closed as required to discuss information the release of which would represent a clearly unwarranted invasion of personal privacy.)

17.2) 3:00-3:15: Nomination of ACRS Officers - ACRS Chairman appoint Nominating Panel for ACRS officers for CY 1987/88 (DAW/RFF)

18) 3:15 - 3:30 P.M.

Miscellaneous (Open/Closed)

18.1) Complete discussion of items considered during this meeting.

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MINUTES OF THE
317TH ACRS MEETING
SEPTEMBER 11-13, 1986

The 317th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street, N.W., Washington, D.C., was convened by Chairman D. A. Ward at 8:30 a.m., Thursday, September 11, 1986.

[Note: For a list of attendees, see Appendix I. W. Kerr, G. A. Reed, and H. Etherington did not attend the meeting. D. Okrent and F. J. Remick did not attend on Saturday, September 13.]

Chairman D. A. Ward noted the existence of the published agenda for the meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act and the Government in the Sunshine Act, Public Laws 92-463 and 94-409, respectively. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H Street, N.W., Washington, D.C.

[Note: Copies of the Transcript taken at this meeting are also available for purchase from ACE-Federal Reporters, Inc., 444 North Capitol Street, Washington, D.C. 20001.]

I. Chairman's Report (Open)

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

Chairman D. A. Ward indicated that Kenneth M. Carr was sworn in for a five-year term as a member of the Nuclear Regulatory Commission on August 14, 1986. He also mentioned the fact that ACRS member G. A. Reed is now recovering from a heart attack and will be unavailable to the Committee for at least two months.

II. Babcock & Wilcox Nuclear Power Plant Long-Term Safety (Open)

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

C. J. Wylie reminded the Committee that the NRC made a decision in January 1986 to reassess the Babcock & Wilcox (B&W) plant design after several incidents occurred at B&W reactor plants. At the urging of the NRC, the B&W Owners Group (BWOG) assumed a leadership role in that reassessment. In March 1986, the NRC prepared a proposed NRC version of the reassessment program and forwarded that program to the BWOG. In June, the NRC commented on BWOG's plans taking account of the NRC input. The BWOG Trip Reduction and Transient Response Improvements Program was discussed with the B&W Subcommittee during a June 25, 1986 Subcommittee meeting. The Subcommittee raised several substantive concerns which led the BWOG to postpone its meeting with the full Committee scheduled for July to the ACRS meeting in September. During the July ACRS meeting, the

Committee discussed the BWOG plan as it was then structured, and recorded its concerns in a July 16, 1986 report. On August 14, 1986, a Staff response to the ACRS letter agreed that a broader based program was needed and stated that the Staff was working with the BWOG toward that effort.

R. C. Jones, NRC, indicated that the Staff has been in contact with the BWOG at various working level meetings regarding several major topics. M. W. Carbon asked what reassessment of B&W plants means to the NRC Staff today. R. C. Jones indicated that the BWOG is to examine operating experience at B&W plants over the last several years to identify sensitive systems, kinds of problems that have occurred, kinds of behavior exhibited by B&W plants, and look in detail at specific problem systems or sensitive systems. This broad-based review of the systems should cover their performance, the appropriateness of their design requirements, and whether some changes ought to be made. He presented an overview of the BWOG Safety and Performance Improvement Program (see Appendix IV). He indicated that the operating experience review involves NUREG and EPRI reports as well as LERs. The BWOG plans to interview operators and maintenance personnel to further identify concerns with plant behavior. The BWOG has contracted with MPR Associates to study the sensitivity of B&W plants. The BWOG will also conduct an analytical study to examine the response of B&W plants to trips and upsets and compare the behavior to that of other PWR designs. M. W. Carbon asked whether this study will look at normal, abnormal, or severe transients. R. C. Jones indicated that the primary focus of the program is normal transients, such as instrumentation, or instrumentation and control system (ICS) generated transients, failures of the ICS power supply, loss of feedwater transients, and transients such as stuck-open main steam safety valves. Also analyzed will be the demands upon the operators during these events. They will review operating procedures as part of their system reviews to include the ICS/NNI main feedwater system, emergency feedwater system, auxiliary feedwater system, secondary relief system, and the instrument air system. With the exception of the instrument air system, all of these systems have caused complicated transient response in B&W plants. C. Michelson asked if they are doing some systems interactions studies as they relate to the main feedwater system in particular. R. C. Jones indicated that if transients have occurred they will be identified through the LERs.

R. C. Jones indicated that the final part of the BWOG program is a recommendation tracking system and implementation process. Out of their program will come a series of recommendations which they intend to track and monitor as these items are addressed by individual utilities. He indicated that the NRC Staff believes that the BWOG program is generally on target, but the Staff needs further discussion in the area of the main feedwater system review and its scope. The Staff is also closely monitoring the area of human factors issues and the tracking system and its implementation.

R. C. Jones indicated that the primary effort of the NRC Staff is to review the BWOG results. The Staff will examine demands on operating personnel/procedures, see whether existing B&W PRAs reflect operating experience, and also look at the thermal-hydraulic response and sensitivity of B&W plants as they relate to the overall safety of the plant. The Staff plans an initial SER in December 1986 with open items with supplements, as appropriate. Completion of the program is scheduled for June 1987. C. Michelson noted that the thermal-hydraulic response the Staff is interested in probably refers to normal transient responses. He asked if the Staff plans to look at accident responses. R. C. Jones indicated that the Staff may examine the area of steam generator tube rupture. J. C. Ebersole asked if the Staff intends to examine the nonambiguous vessel level gauge at Arkansas Nuclear One, as well as the primary blowdown system being installed at Davis-Besse. R. C. Jones indicated that level instrumentation as well as ICS instrumentation activities are under NUREG-0737 and will not be part of this program. While the Staff is trying to make the program as broad based as possible, it does not intend to cover the Davis-Besse primary blowdown system because the Staff does not intend to look at new alternate decay heat removal concepts under this program. D. Okrent expressed interest in the reliability of a reactor trip if the turbine generator does not trip. R. C. Jones indicated that, to the best of his knowledge, the turbines always trip. One area the BWOG is to examine is an alternate relief system designed for the steam generator. Continued operation of the turbine is one of the ways to try to minimize operation of the main steam isolation valves. This is a way to keep the steam generator pressure down. B&W plants, following a reactor trip, actuate the main steam safety valves unlike other PWRs, and there has been difficulty with those valves. The BWOG is looking at a two-phased program which will improve the reliability and operability of those valves and, at the same time, minimize the challenge to the valves in the first place. He noted one major area of concern is overcooling transients with steam generator overfill. B&W plants appear to be more sensitive than other PWRs in that area. One has to worry about possible pressurized thermal shock consequences.

H. B. Tucker, Duke Power Company (Chairman of BWOG Executive Committee), explained that the BWOG Safety and Performance Improvement Program constitutes an action plan in response to the January 24, 1986 letter from V. Stello expressing concern about B&W-designed plants. G. R. Skillman, GPU Nuclear, admitted that the BWOG failed to establish recognition of the safety orientation of the program when it met with the ACRS Subcommittee. He mentioned three observations and recommendations in the ACRS letter to V. Stello on July 16, 1986. These were that some B&W plants operate better than others, B&W plants respond differently from other PWRs, and apparently little attention is being given to decay heat removal in B&W plants. He identified four basic issues from the ACRS letter: 1) data base lessons, 2) program safety orientation, 3) safety significance of the once-through steam generator sensitivity, and 4) energy production/removal imbalance--decay heat removal (see Appendix V).

R. Skillman indicated that the BWOG Safety and Performance Improvement Program is a full assessment of fundamental safety and operating issues at B&W plants and includes 13 major tasks. The program consists of an independent sensitivity study of basic thermal-hydraulic plant characteristics. It includes a detailed review of selected key systems. He contended that the need for additional decay heat removal and capability is considered unnecessary. The BWOG recognizes that the imbalance between heat production and removal is a key to understanding B&W units.

R. T. Glaviano, BWOG, indicated that the process of zeroing in on operating experience has meant a review of about 220 transient assessment/performance reports. To accommodate that review, the BWOG developed transient classification guidelines to judge the relative complexity of transients. The central definitions of categories A, B, and C are that Category A events are events where the preferred or expected response of the plant is seen; Category B events are those where the preferred or expected range is exceeded (these events are of concern since they may be precursor events; and Category C events are those where abnormal response was clearly indicated. J. C. Ebersole asked if Category B events include challenges to safety systems such as the auxiliary feedwater system. R. T. Glaviano agreed that they did. D. A. Ward expressed concern regarding the BWOG's ability to differentiate between complexity and seriousness. He hypothesized an accident that was extremely serious but had a rather simple cause. R. T. Glaviano played down the sophistication of the classification system as just a convenient tool for putting transients into relative terms. He noted that the data show that the more complex the event the greater the challenge to the operating crew and to the plant systems. D. W. Moeller asked at what point the BWOG ties the safety significance of the transients to the risk to the health and safety of the public. R. T. Glaviano again stressed that the BWOG is using this method just to develop basic conclusions and recommendations to improve plant response.

R. T. Glaviano indicated that there were 10 Category C events defined. There are four distinct patterns as to what contributed to these events. Two of the events occurred as a result of loss of offsite power. Following the loss of offsite power, the emergency feed system filled at an excessive rate to the natural circulation control set point. When the generators reached that set point either the feed system shut down or the pumps kept running but the flow stopped. The B&W plant has a tendency to reheat and repressurize as a result of the onset of high pressure injection. For both of these events a reheat repressurization following rapid refill of the steam generators resulted in the lifting of the PORV. A second pattern was excessive steam flow through secondary plant valves, whether these were bypass or atmospheric dump valves. These events generally tend to exhibit excessive steam flow which causes a requirement for feedwater to maintain inventory. Feed flow is also increased and the result is an excessive cooling event. The third type is due to ICS/NNI power problems. There is a combination of excessive feed flow and steam flow to the steam flow control valves and emergency flow control valves in a partially open position. These events resulted in excessive cooling. A fourth category of events are associated with actuations of safety systems such as the EFIC and steam feed rupture control system.

R. T. Glaviano stated that preliminary conclusions from the review of transients indicate that the events are generally the result of excessive primary to secondary heat transfer with the steam flow through failed open valves, and the feed flow to steam generator inventory. These events require additional operator attention to achieve plant control in balancing energy removal with energy production and controlling reactor coolant inventory and pressure due to excessive feedback from the secondary side of the plant. The key to mitigating these events is to assure that steam flow and feed flow can be controlled from the control room. This allows balancing of the production terms with the removal terms reducing feedback to the reactor coolant system. It requires that plant control be maintained when ICS/NNI power is lost. Corrective actions are planned in the area of design, maintenance, human interface, and operating experience (see Appendix VI). F. J. Remick noted that in the case of Davis-Besse the Steam Feed Rupture Control System actuated and isolated the steam generator, then the valve stuck closed. Would this be a corrective action to assign to plant maintenance? R. T. Glaviano indicated that it would be either design or maintenance, one of the two. The point is the ability to control the main steam flow path and the main feed flow path.

G. R. Skillman spoke about the sensitivity of B&W plants, noting that the tuning of the ICS, as well as some other items, has caused a great increase in the reliability and overall safety of the plants. It is a plant-specific process. C. Michelson commented that any system that is so arranged that it takes a fine tuning operation to make it function effectively does not sound like a well-designed system to begin with, since it can't stand small drifts and operator maladjustments. G. R. Skillman indicated that the ICS is a system that has a design function and needs to stay within its design limits.

D. Okrent indicated that he thought it certainly valuable to review prior experience and to see what corrective actions are strongly suggested. Nevertheless, he wondered if the BWOG is considering the case of a prior transient and one additional failure occurring. He asked if the BWOG is reviewing the B&W plants for their ability to recover from multiple failures other than those that have occurred. G. R. Skillman indicated that one of the objectives of the sensitivity study is to look beyond the data to other kinds of situations as to quantify the relative sensitivity of B&W plants with respect to other PWR designs. D. Okrent insisted that there were certain aspects of the B&W plant which leave it vulnerable to transients which include certain combinations of failures. G. R. Skillman asked D. Okrent to be more specific. D. Okrent noted that if there is less water in the steam generator one would have less time to recover from an event that was not feeding water to the steam generator. The Instrumentation and Control System in a B&W plant has already been shown to cause multiple instances of incorrect control information to appear simultaneously and to create complex transients. D. Okrent indicated that he is looking for a different philosophy driving the BWOG studies, one that considers multiple failure analysis rather than trying to prevent a single failure. He suggested that that effort seems to be absent from the BWOG studies, and he noted that it would not be difficult to at least pick out vulnerable points in the plant. C. Michelson questioned the BWOG effort regarding systems interactions.

E. Swanson, BWOG, indicated that he was associated with specific system reviews in the BWOG program, such as the air system, emergency feedwater, steam system, ICS/NNI, and main feedwater system reviews. He noted the ACRS' concern regarding removal of decay heat and indicated that the BWOG program is concerned with decay heat removal after shutdown. He explained that the transients the BWOG has seen from experience data from power operation have varied. The objective of the BWOG work regarding decay heat removal is to concentrate on those transients and balance decay heat removal. He noted that decay heat is removed from B&W plants in the same way as it is in other PWRs using the steam generators. B&W plants have the same degree of defense-in-depth as other PWRs. Even the core melt frequency for B&W plants does not differ much from that for other PWRs. C. Michelson pointed out that the coupling between the reactor core and the steam generator is rather unique on a B&W plant. From the steam generator out into the secondary, the situation is not much different from other PWRs. There certainly is considerable difference in how decay heat gets from the core to the steam generator. E. Swanson contended that the difference between B&W PWRs and other PWRs is a matter of time. C. Michelson noted that it is also the thermal-hydraulic arrangement that is significantly different. E. Swanson claimed at least three lines of defense, three lines or levels of defense for removal of decay heat within/without the steam generators. The first line of defense is the main feedwater system, the second line is the emergency feedwater, and the third line of defense is bleed-and-feed capability. He contended that bleed-and-feed is very powerful for B&W plants because of the very high-head high-flow pumps that are able to provide flow through the PORV or through the pressurizer safety valves. The Committee discussed the fact that at Davis-Besse there is enough pump net positive suction head to open the safeties but one cannot get flow into that head. As a result, Toledo Edison has to use the PORV. D. A. Ward expressed concern that the thermal hydraulics of the bleed-and-feed process are not adequately understood and satisfactorily represented in the emergency operating procedures that are in place at B&W plants. E. Swanson admitted that the operators at Davis-Besse were a bit reluctant to engage bleed-and-feed. All plants have taken some action to better define procedures so that the operators will definitely engage bleed-and-feed when necessary. C. Michelson wondered whether thermal-hydraulic understanding through calculations and computer codes is adequate. E. Swanson insisted that the calculations that were made show that the process does work.

E. Swanson attempted to address the ACRS concern regarding the main feedwater system. He indicated that the emphasis of the BWOG program is to keep the main feedwater system on line. The BWOG is looking at what degrades system performance and wishes to improve the reliability of the system and, in particular, the main feedwater pumps. This is important to reduce emergency feedwater challenges which occur even when the main feedwater system is on line. The BWOG wants a smooth post-trip response. It wants to control the ability to keep one pump on line if the other pump trips which is very closely tied to the tuning of the ICS and the main feedwater pump turbine and controls. This is an important part of the BWOG program. Another thing being done in the program is to try to minimize unnecessary feedwater pump trips, and this is done by correcting the controls on the main feedwater system, and the ICS. Mention was made of the interaction between the turbine-driven pumps and steam pressure control. If one does not control steam pressure after the trip the head output of the pumps is affected. J. C. Ebersole commented that electric-motor-driven pumps, while costing more to run, are much more reliable than the turbine pumps and would remain steady in this situation. J. H. Taylor, B&W, indicated that an extensive study of feed pump trips has been done by B&W. He offered to provide a copy to the Committee. J. C. Ebersole expressed interest in the report.

E. Swanson indicated that all B&W plants have initiation of emergency feedwater independent of the ICS power supplies. He explained that the turbine-driven emergency feedwater pump reliability suffers because of the short startup time placed as a requirement on the design. If B&W owners can lengthen the startup time to allow the pumps' speed to be increased more gradually from the low-speed stop to the high-speed stop it will improve or reduce the number of trips on overfrequency or over-speed. The BWOG will probably make such a recommendation to the NRC Staff. He also noted that maintenance practices probably account for a reasonable amount of unreliability of the pumps in the system. Another way of increasing the reliability of the system is to reduce the number of challenges to the system.

G. R. Skillman indicated that the BWOG expects, in its Safety and Performance Improvement Program, to develop about 500 recommendations from the 13 main tasks. These will be available by the second quarter of 1987. It is the intent of the BWOG to dispose of all the recommendations formally and to include the NRC Staff in the disposition of the recommendations. He noted that the implementation phase of appropriate changes will have to fit with the various outage schedules and other key activities and commitments of the utilities. He expected that the implementation phase will continue for several years. P. G. Shewmon noted that the Oconee units did not have any Category C events. He asked if the BWOG has identified how things are being done differently at the Oconee plants than at other B&W plants, and how this might be factored into the program. H. B. Tucker indicated that, while it is true that no Category C or abnormal response events have occurred at the Oconee plants, there are several precursor events or Category B events that were of significance at Oconee. The position of the BWOG is to take the collective response of all the B&W plants to learn the lessons that there are available and apply them to all of the B&W plants. C. Michelson asked how the BWOG was addressing the question of steam generator overfill in this review. G. R. Skillman indicated that it will be addressed partially through the main feedwater system review and, to a large degree, during work on the secondary plant relief system, instrument air, and others.

C. J. Wylie asked if the BWOG program, as presented, resolves the Committee's previous concerns as expressed in its letter of July 16, 1986. C. Michelson thought it premature to endorse the BWOG program without caveats. He remained skeptical about a lack of emphasis on systems interactions in the BWOG program. D. W. Moeller thought that the BWOG has taken the ACRS concerns in the July 16 letter seriously and addressed them. He wondered, however, if the BWOG had studied all the PRAs for B&W plants and produced findings. G. R. Skillman indicated that the BWOG has structured its PRA review around the Class 3 PRA at Crystal River and the Class 1 review at Oconee. The BWOG is in the process of taking the findings from those two PRAs and applying them to Davis-Besse, Rancho Seco, TMI-1, and Arkansas Nuclear One. D. W. Moeller thought it useful for the Committee to hear periodic reports as progress is made in the BWOG program. J. C. Ebersole expressed concern regarding the control of the bleed-and-feed process for B&W plants. P. G. Shewmon mentioned cycling tests done at Davis-Besse. C. Michelson noted that they did not cover the full spectrum of fluid conditions. M. W. Carbon expressed unease regarding the spectrum of accident scenarios examined by the BWOG program. He thought that they should consider accidents which haven't happened yet, and they are only looking at historical experience. He thought that a key question is if the BWOG is doing a systems interactions study of the type that will reveal these new possibilities. No one present was aware of any B&W systems interactions studies that have been published. Chairman Ward noted that there did not appear to be any interest on the Committee to write another letter at this time. He did express the interest of the Committee in hearing more about the program in the future.

III. Decay Heat Removal Subcommittee Report (Open)

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

D. A. Ward indicated that the Subcommittee on Decay Heat Removal Systems held a meeting to review the NRR resolution effort for Generic Issue 124, "Auxiliary Feedwater System Reliability." He explained that, prior to 1975, auxiliary feedwater systems were built to conform to good engineering practice. There were no specific AEC or NRC requirements. After 1975 a requirement was developed that new auxiliary feedwater systems be safety grade, appropriately seismic resistant, subject to QA requirements, and tolerant of single failures. After the Three Mile Island accident, considerable reevaluation and rethinking of the importance of auxiliary feedwater systems led to a new requirement after July 1981 for new plants. They must show that their auxiliary feedwater system should have an unavailability upon demand less than 10^{-4}. This requirement was placed upon applicants for operating licenses rather than for construction permits and was placed in the Standard Review Plan. Twenty to twenty-five units have been licensed under this provision, which means that 75 or 80 plants do not necessarily meet this requirement. This concern was focused in Generic Issue 124. The focus is on 7 plants in which there is a particular concern about unreliability of the systems. All are single units which have two-train auxiliary feedwater systems. While a number of other units have two-train auxiliary feedwater systems, they are on multi-unit sites and credit is given to certain cross-connections between the units. The Committee reviewed the Staff program and in December 1985 wrote a letter which pointed out that the ACRS thought the Staff's resolution for this generic issue was not adequate, the schedule not prompt enough, and the plan not well enough focused. The Committee also complained that operating experience with regard to the performance of auxiliary feedwater systems had not been adequately analyzed for useful information. The Subcommittee concluded on September 9, 1986, after hearing presentations by NRR and AEOD, that the work on analyzing data on actual operating experience is proceeding better than in the last year. The Subcommittee does have some concerns with the approach being taken to review in some detail the systems at each of these seven plants and to recommend specific fixes in hardware or procedures. D. A. Ward scored the lack of development of objective guidelines for making judgments as to what fixes are needed. A more objective basis is needed for deciding whether systems are adequate. He noted that NRR will propose plant-specific hardware-oriented fixes for the original seven plants.

J. C. Ebersole posed a scenario in which loss of main and auxiliary feedwater is assumed at the outset. Relief is accomplished through the PORVs. At some point the flow path shuts down to the point where removal of decay heat can only be effected by relief aided by heat transport through the secondary system. But loss of the secondary system is assumed at the beginning of the event and there is now a lack of options for the reduced relief through the primary system. He asked the NRC Staff how they would deal with such a scenario. [Subsequent to the meeting the Staff agreed to discuss this issue at a future Subcommittee meeting on the resolution of USI A-45.]

S. S. Diab, NRR, presented the status of a modified resolution approach to Generic Issue 124 (see Appendix VII). He indicated that the modified resolution approach basically consists of short-term, concentrated reliability reviews for each of the seven plants followed by a findings report. The review effort will benefit from ongoing tasks like the Rancho Seco restart effort and the BWOG design reassessment effort. The NRR team, the Auxiliary Feedwater System Review Team, will also benefit from the IE program for Safety System Functional Inspections, and licensee auxiliary feedwater systems reliability analyses that are being done, as well as relevant industry efforts.

S. S. Diab discussed the status of each one of the seven plants in the modified resolution approach, as far as their auxiliary feedwater system reliability studies are concerned. [The seven plants are ANO-1 and -2, Rancho Seco, Crystal River, Prairie Island-1 and -2, and Fort Calhoun.] He indicated that the Staff is also reviewing the BWOG design reassessment. This assessment is attempting to improve the reliability of main feedwater systems, to improve the reliability of auxiliary feedwater systems, and to limit the challenges to the auxiliary feedwater system (reduce scrams). D. Okrent asked if the Staff's review for Generic Issue A-124 will be sufficiently detailed to get into dependencies of not only support systems but support systems of support systems. S. Diab discussed the composition of a Staff auxiliary feedwater system review team, as well as the review's scope, which he indicated would cover support systems such as power supplies, compressed air or nitrogen systems, lubrication, and cooling. The Staff will review all of the post-TMI modifications, paying particular attention to finding common-mode vulnerabilities of the design or the arrangement of the equipment. The reviews will cover operator recovery, control room adequacy for indication control and recovery, ease of LOCA recovery, and alternate decay heat removal means. C. Michelson asked if the Staff will include a fire analysis as well as pipe-break analysis in the feedwater area. S. Diab indicated that the Staff will be looking at the functional reliability, as well as the environment. C. Michelson asked if the Staff will look at fire protection around the auxiliary feedwater turbine and the probability of inadvertent actuation as a factor in reliability. S. Diab indicated that the Staff plans to do that. W. Minners indicated that the Staff does not wish to promise that they will do a formal quantitative reliability analysis on each plant. If that analysis is available and uses plant-specific data the Staff will utilize it. But, based upon schedule and resources, the Staff does not plan to do a formal quantitative, detailed reliability analysis for each of the seven plants. C. Michelson expressed interest in the team reports as they are generated. W. Minners indicated that the Staff could provide them. D. A. Ward still remained concerned regarding the continually changing definition of "reasonably assured reliability" or "sufficient reliability."

IV. Emergency Core Cooling Systems (Open)

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

W. Beckner, NRC, indicated that the existing emergency core cooling system (ECCS) Rule 10 CFR 50-46 was based upon knowledge available in 1975. Even at that time, the Staff knew that parts of Appendix K to the Rule were very conservative, particularly the calculation of decay heat. Conservatism was left in the rule to cover the overall uncertainty and the understanding of the overall performance of ECC systems at that time (see Appendix VIII). Additional features not specifically called out in Appendix K to the Rule have been typically treated very conservatively, either by the Staff's conservative requirements or by the licensee, or applicant, proposing simplified or conservative models. The Staff believes that the current overall ECCS evaluation model is very conservative and that calculated temperatures during a LOCA, using Appendix K models, are much higher than would be expected in reality. The Staff now believes that distortions created by the use of these artificial conservatisms in Appendix K may adversely affect the overall safety of plant design and operation.

W. Beckner explained that the existing ECCS rule is specific and prescriptive. All errors must be immediately reported to the NRC and a reanalysis is required if there is a significant error, even when no regulatory or safety threshold is surpassed. Errors of 20 degrees in peak cladding temperature generate a reanalysis even if the temperature does not exceed 2200°F.

W. Beckner indicated that there is broad support within the NRC Staff, the ACRS, and industry with no indication of any outside opposition to revising this rule. There is also broad support for the basic approach which is based upon SECY-83-472 (Realistic Evaluation Model with an Uncertainty Evaluation). This is an approach originally proposed by the General Electric Company and used in SAFER.

W. Beckner described the proposed rule revision. He noted that calculations would still be required of ECCS performance. However, the analytical technique would attempt to describe the behavior of the reactor systems realistically as defined by comparisons to applicable experimental data. Uncertainty of the calculation would be accounted for so that there would be a high probability that the 2200°F criteria would not be exceeded. Existing evaluation models using Appendix K would be grandfathered. The grandfathering is indefinite if desired by the licensee. Error reporting would be modified to make a reevaluation commensurate with the effect of the change or error, whereas the current rule gives the Staff an option to simply shut down a plant because of an error or exceedence of the 2200°F criterion. The proposed rule allows a utility to de-rate the plant rather than totally shut it down if some problem occurs. All errors and all changes made to the evaluation models should be reported at least annually. If the change is significant there would be a report required within 30 days along with a schedule for the reanalysis. D. W. Moeller noted that these requirements give the impression that each individual utility is doing a separate, totally independent analysis of their specific plant as contrasted to vendors doing a generic calculation. W. Beckner indicated that in reality the vendor typically holds the model and makes changes to it, but the licensee is submitting his model with the vendor acting as his agent.

W. Beckner indicated that the Dougall-Rohsenow correlation is removed from Appendix K and another heat transfer correlation reference is updated. The Staff has found that the Dougall-Rohsenow correlation can be nonconservative in certain areas. As a result, the Staff intends to monitor evaluation models as approved and if there is a significant reduction in conservatism, defined as more than 50°F, the Staff would start to look at such things as Dougall-Rohsenow and pronounce it unacceptable in any area where it is not conservative. The Staff has made changes in documentation requirements to remove the 20°F definition of a significant change, as well as putting into Appendix K explicit reporting requirements for errors. C. Michelson noted that the old requirement of 20°F now becomes 50°F even for the old Appendix K users.

W. Beckner indicated that the ECCS rule revision package contains three pieces:

- Revised Rule
- Regulatory guide
- Compendium of ECCS research (Summary Report by the Office of Research of research performed over the past 10 years)

W. Beckner indicated that the rule itself is general and does not reference either the Regulatory Guide or the compendium. The Regulatory Guide expands upon the Rule by giving guidance on what is acceptable to meet the Rule in practice; if the applicant or licensee follows the guidance of the Regulatory Guide, the NRC will not mount a challenge. The applicant, or licensee, can propose another method of meeting the rule. The Rule permits the use of the latest technology, specifically best-estimate calculations, combined with uncertainty evaluations based on data comparisons. The use of such realistic calculations should lead to more understandable regulations and, hopefully, will be of benefit to safety. The Regulatory Guide proposes acceptable models and data related to Appendix K, but allows flexibility for the Staff to accept industry initiatives. It defines requirements for estimating overall code uncertainty at the 95 percent probability limit.

W. Beckner indicated that the proposed Rule is still being studied by the CRGR and the Staff hopes to send the proposed Rule to the EDO and to the Commission in September. The Rule should be issued with a three-month comment period with a final Rule available sometime in November 1987.

W. Beckner mentioned the Staff's regulatory analysis of the effect of the rule change, noting that there is the potential for large cost savings by the industry (see Appendix IX). Westinghouse plants may be upgraded in power by about 5 percent as a result of this rule change. There is a potential cost savings simply by increasing the flexibility of the fuel cycle and the way plants operate. Because of the more realistic analysis, plants can go to higher peaking factors, alleviate overly-tight diesel generator start times, which will increase diesel reliability, and increase the overall safety of the plant. The rule change may also alleviate PTS concerns.

W. Beckner discussed some of the comments made at the ACRS subcommittee meeting in August 1986 (see Appendix IX). He explained that the Staff finds it hard to justify on a safety basis phasing out Appendix K and forcing all licensees to develop a realistic calculation. C. Michelson asked if a five-to-ten year phaseout would be feasible? W. Beckner indicated that the Staff cannot justify that based on a safety benefit. He noted the ACRS' concern regarding more guidance on uncertainty and he indicated that three potential areas of guidance have been considered. These involve the clarification of high probability, the providing of general principles to be used in realistic best-estimate models, and broad general principles on uncertainty methodology. He defined the objectives of the compendium of ECCS research as supporting the rulemaking and consistent with the Regulatory Guide. He acknowledged that the compendium needs better organization to make it more readable. Regarding the impact of the backfit rule on the NRR review of new evaluation models, he indicated that, according to the NRC Office of the General Counsel, as long as the issues are confined to the adequacy of the evaluation model and whether it meets the criteria, there should be no backfit. The backfit rule could be invoked only if the Staff raises a different issue.

L. Shotkin, NRC, discussed the NRC Research (RES) Staff's proposed methodology for measuring thermal-hydraulic code uncertainty (see Appendix XI). The RES Staff proposes to examine the uncertainty of the Staff's own best-estimate codes. The study will not be incorporated directly in the proposed Rule, nor in the Regulatory Guide. It will be addressed in the compendium of the ECCS research and has very little relationship to the Rule. L. Shotkin explained that RES is proposing a comprehensive methodology for looking at the uncertainty in the calculated peak clad temperature. This methodology will consist of code versus data comparisons in a systematic examination of code models and correlations. The following four factors will be addresssed by the NRC/RES uncertainty methodology, and taken together these four factors constitute the code applicability to analyze a given scenario in a given full-scale vendor geometry: 1) code modeling capability, 2) quantitative measure of code uncertainty, 3) detection of compensating errors, and 4) scalability of the calculated peak cladding temperature. The uncertainty methodology will first provide a quantitative estimate to go to full scale and make an estimate of the uncertainty. At the same time there will be a backup methodology that looks at all of the reasons why the results that the Staff is getting might be wrong. He indicated that the methodology will be reviewed in the beginning of October 1986.

C. Michelson indicated that the ACRS is obliged to provide the Staff with an endorsement of the Rule as far as stipulating that it is ready to issue for public comment. He indicated that the Subcommittee did not have a problem with this endorsement. F. J. Remick applauded the rule revision, indicating that it was long overdue. D. Okrent asked C. Michelson how ACRS consultants received the Staff presentation at the Subcommittee meeting. C. Michelson indicated that all had minor detailed comments that would be of help to the Staff in polishing their documents. None of the consultants condemned the process. D. A. Ward pointed out one sticking point involving grandfathering the old Appendix K--the Staff really does not have much justification because of a lack of an explicit safety advantage in doing it. Some of the consultants thought there ought to be a limitation put on the grandfathering. One ACRS consultant indicated that he did not have any problem with licensees continuing to use the current Appendix K if it is cost effective for them and does not degrade safety. H. W. Lewis was troubled by the confusion between the terms "realistic" and "best estimate," which he suggested are two entirely different concepts. He pointed out that "realistic" means "without deliberate conservatism" whereas "best estimate" is a very specific statistical technique for aiming at the middle of a statistical distribution. He asked whether the Rule uses the term "best estimate" or "realistic." W. Beckner indicated that the two terms were used interchangeably by the Staff, maybe incorrectly. D. A. Ward asked what H. W. Lewis thought best describes what the Staff is trying to do. H. W. Lewis indicated that he thought they are asking for a realistic calculation. When the Staff talks about using a best-estimate calculation, that prescribes a way of doing the calculation which may not be the most appropriate. D. A. Ward asked whether H. W. Lewis thought this would have an impact on the public health and safety. H. W. Lewis thought it would lead to legal squabbles about what the calculations mean.

V. <u>Foreign Operating Experience</u> (Open)

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

D. A. Ward indicated that the Staff is prepared to brief the Committee on the Chernobyl plant accident, specifically the IAEA Meetings that occurred in Vienna in August 1986. He noted that W. Kerr was a member of the U.S. team that attended those meetings. W. Kerr, while not present at this meeting, will provide a follow-up report and a statement of his views during the October ACRS meeting. H. Denton, Director of NRR, acknowledged that W. Kerr was an effective member of the delegation. He indicated that the delegation was led by U.S. Ambassador Kennedy, and contained representatives, not only from ACRS and NRC, but also from the U.S. Department of Energy, the National Institutes of Health, the U.S. Environmental Protection Agency, the Federal Emergency Management Agency, as well as the Defense Nuclear Agency, in addition to several consultants to these groups. R. Wilson from Harvard University and T. Theofanous from the University of California were among the NRC consultants to the delegation. The objective of the meeting was to improve the level of nuclear safety throughout the world, to increase understanding of the phenomena involved in the Chernobyl accident, and to improve international cooperation on nuclear matters. He indicated that Soviet representatives were very forthcoming, quite open and candid, and desired to resume normal relationships with the West in the reactor safety area. He mentioned meeting privately twice with groups from the Soviet delegation and, as a result, there is a plan for follow-up trips by the NRC to the Soviet Union. He noted their desire to cooperate with the West in looking hard at ways to improve the safety of Soviet plants.

H. Denton indicated that the first Soviet speaker, academician Legasoff, explained that, at the time the decision was made to build the Chernobyl plant, the USSR did not have the capability to construct large, thick-walled pressure vessels or steam generators. The Soviet spokesman mentioned a number of advantages to this design that were seen 25 years ago. The design had a flexible fuel cycle and superheating was possible, but not used. The Soviets could do channel-by-channel control and could replace failed fuel. There was recognition of a positive void coefficient and the fact that the Chernobyl design was very sensitive to phase changes and had a high thermal energy. But on balance, in view of the Soviet Union's energy needs at the time, the positive factors outweighed the negative. H. Denton mentioned a 30-minute videotape taken inside the Chernobyl plant and off site, showing the amount of destruction in pictures taken from a helicopter directly over the core, and inside the plant in various room. He recommended that when this videotape is made available (not yet available to anyone outside the IAEA) the ACRS view it.

H. Denton indicated that the main presentation on the causes and sequences of the event spoke of numerous deliberate violations of procedure, and inadequate written procedures. There was lack of proper authorization of the test performed and the test plan was not reviewed by either the plant management or the plant designers. The attitude of the designers was poor and they had lost all sense of danger. It was also acknowledged that the accident was compounded by specific design features. He suggested that there was a mismatch between the control system and the core characteristics.

H. Denton then discussed the accident itself. He indicated that the Soviets described the event as a reactivity excursion and discussed calculations indicating about a beta and one-half of reactivity was inserted in two seconds when voiding of cooling channels occurred. They had violated their own procedures on how much reactivity to have in the control rods. The intent was to keep a beta per second in the control rods at all times. When the accident occurred they had high reactivity input from voids in the coolant. The control rods were slow and ineffective and did not affect the force of the accident. A graphite fire started subsequent to the accident and about 10 percent of the graphite moderator burned (about 250 tons).

H. Denton indicated that a detailed presentation was made on firefighting during the accident and the major problems the Soviets had with the fires that occurred. A number of ideas to better protect firefighters in the future were discussed. The Soviets suggested future international meetings which would discuss the interrelationship of radiation protection and firefighting. He discussed the nature of the releases, indicating that a total of about 50 megacuries, not including noble gases, were released. On the tenth day into the accident the Soviets realized that the core was beginning to heat up (a second peak in activity occurred at about day 10) and they took steps to try to cool the core, including the insertion of liquid nitrogen and water. Instrumented packages were dropped into the core to measure the temperature and radiation levels and considerable time was spent determining exactly how to entomb the plant.

H. Denton indicated that various concepts for a sealed, airtight entombment of Unit No. 4 or a filtered release were discussed. He mentioned the efforts by the Soviets to relocate about 130,000 people. Over 10,000 cattle, as well as other farm animals, were also moved. Efforts to decontaminate Units 1, 2, and 3, as well as houses, streets, farms, and forests, were also discussed. The Soviets pointed out the need for a single, coordinating authority in the event of an evacuation to make the evacuation go smoothly. H. Denton mentioned Soviet attempts to make close-in measurements of meteorological data, using helicopters and airplanes, and a massive effort, still ongoing, to protect water supplies. He indicated that another part of the meeting focused on possible exposures from the accident including the distribution of the release and dosimetry, the characteristics of the release regarding individual and collective doses, and aspects of medical treatment. In the case of exposed individuals, surgical intervention was not very effective and it was found that skin doses were very important in determining the ultimate outcome (recovery) of the patient. No acute fatalities have occurred or are expected to occur to anyone off site. Large amounts of potassium iodide were administered without side effects.

H. Denton indicated that future conferences will be necessary to explore the myriad of questions that were given to the Soviets. About a dozen topics were considered by the IAEA to be fertile topics for future discussion. They ranged from analyses of severe accidents to biological measures of radiation. The International Nuclear Safety Advisory Group (INSAG) is preparing a report of the conference and will recommend what IAEA considers doing as a follow-up to this conference. At the general conference which Chairman Zech will be attending, as well as others, it is intended that the two agreements on prompt notification of accidents involving potential transboundary releases and the agreement providing mutual assistance to countries will be available. H. Denton thought that a report by the U.S. delegation to the Commission summarizing the factual situation at both the public as well as private meetings would be available in about 60 days. At the same time, NRR will prepare a report on the potential implications of Chernobyl. One issue to be scrutinized will be reactivity-excursion accidents in U.S. plants. While much of the accident at Chernobyl has been attibuted to loss of procedural control, much has been done in the U.S. in those areas in training since the Three Mile Island accident. Fruitful areas for NRR consideration are containment designs in a severe accident and the inadequacy of emergency planning. F. J. Remick noted that a statement had been made in the trade press that some research should be done on pumps. H. Denton indicated that there was some discussion at the meeting regarding the contribution of pump cavitation to voiding in the Chernobyl reactor coolant channels. The Soviets had performed tests using cold water but had no tests with hot water. With the assistance of General Electric, pump test data at near-saturated conditions was located and provided to the Soviets. P. G. Shewmon requested information on firefighting. H. Denton indicated that the Soviets would like to see more attention given to protecting firefighters from radiation. C. Michelson asked if the fire that occurred at Chernobyl about a month subsequent to the accident was discussed at the meeting. H. Denton did not think that it was. He noted one of the questions that did get asked and answered was whether there were precursors; although the term precursors may be defined differently by the Soviets, they indicated negatively. They portrayed the accident as an unprecedented violation of administrative and operational rules. C. P. Siess wondered if it was an accident that the Soviets dumped about 5,000 tons of various materials on the damaged Chernobyl core. H. Denton indicated that these were ad hoc solutions and they had good reasons which they detailed. They dumped boron for criticality control and lead for its heat absorption capabilities.

Brian Sheron, NRC, discussed the chronology of the Chernobyl accident (see Appendix XII). He indicated that the objective of the test being performed just prior to the accident was to simulate the load that would be experienced by an emergency feedwater pump following a turbine trip. He explained that Soviet diesels take about a minute longer than U.S. diesels to get up to speed. In their safety analyses they rely on the inertia of the turbine generator to continue to put out enough electricity at the bus bars to power the emergency feedwater pumps for about 40-45 seconds. This would be until the diesels can pick up the load. Four reactor coolant pumps were being used to simulate the load from the emergency feedwater pump.

Brian Sheron discussed the power history of the accident (see Appendix XIII). The plant was running at about 3200 MWt and the plan was to go into a maintenance shutdown at 1:00 a.m. on the morning of April 25, about 24 hours before the accident actually occurred. The operators began their power descent and the rods were withdrawn beyond where they were supposed to be to overcome an expected xenon transient. At 1600 MWt the operators were told to stop their power descent. At this time, the operators disconnected the ECCS according to the test procedure which was developed by an electrical engineer who apparently did not have much of a working knowledge of the physics of the plant. This was a violation of Soviet safety regulations, but probably did not have any real bearing on the actual accident itself. Twelve hours prior to the accident the operators resumed the power descent and switched off the local automatic control system. This caused a mismatch in the power demand setpoint. The control system was being told to drive the power down. As the rods were going in this produced less steam voiding in the core. Since the core has a positive void coefficient reducing the amount of voids resulted in negative reactivity feedback. The reactor actually reached as low as about 30 MWt before the operators tried to bring the power back up to between 700 and 1000 MWt (a stable power level). He noted that these plants are not supposed to run in the Soviet Union below about 50 percent power, or around 1000 MWt, because they are unstable. The operators started pulling control rods in order to increase power and overcome xenon transient. The operator was able to stabilize power at about 200 MWt. There were six reactor coolant pumps running and the operator was having difficulty in stabilizing the reactor from a thermal-hydraulic standpoint. The operator was adjusting feedwater in spurts because there was such a high flow through the core and relatively low power. The system was probably much closer to saturation than they wanted it to be. In order to prevent a reactor trip on fluctuations in the steam drum level and pressure, the operators blocked the trip signals. At about this time, because the test procedure called for tripping four reactor coolant pumps and having four pumps remain running if the tests were to be run at 700-1000 MWt, the operators started two additional reactor coolant pumps. With eight pumps in the system, flow limits were violated and vibrations in the feedwater piping led to some pump cavitation. The control rods were pulled out beyond where they could effectively get negative reactivity into the core very quickly.

B. Sheron explained that the test was initiated by tripping the turbine generator. If both generators are off line there is a reactor trip. Since one of the turbine generators attached to the plant was off line, the operator disconnected the reactor trip in anticipation of the second turbine generator being tripped. One turbine generator was still producing electricity from steam, providing power to four reactor coolant pumps. The other four reactor coolant pumps were being powered by offsite power since the second turbine generator was off line. D. Okrent asked if there are any trips on individual or sets of individual conditions within fuel channels, such as measurements of conditions at the exit of a fuel channel. B. Sheron did not know if there were any trips that were specifically tied to off conditions in a particular fuel channel. D. Okrent asked if the trip was on period or on total power level. B. Sheron didn't recall but noted that they have a graduated set of safety features, a runback system.

B. Sheron explained that about 30 seconds before they started the test the operators cut back the feedwater flow which had just previously been increased to about four times (about 75 percent) of nominal. When they started, the test inlet temperature was beginning to increase because of the cutback of feedwater flow. The system which essentially had no voids in it, was very close to saturation and now started to boil. Voids started occurring in the core, a positive reactivity feedback coefficient resulted from the voids, on top of the fact that the control rods were completely out of the core, and this caused a very sharp reactivity insertion into the core. The operators manually scrammed the plant and the rods started to insert. The operators heard some banging noises and noticed the rods didn't bottom. The rest was history. J. C. Mark asked what the time period of the sharp rise in reactivity was. B. Sheron indicated that it was calculated at less than one second. F. J. Remick asked if the core went prompt critical. B. Sheron indicated that the insertion was 1½ beta over a 2-second period, or prompt critical.

T. Speis, NRC, explained that the first reactivity insertion peak was about 120 times nominal or 3200 MWt. During this time energy deposited in the fuel at about 300 calories per gram. Using all of the fuel in the core, there were about 200,000 kg of energy deposited in the fuel. It is estimated that a substantial part of the fuel in the core saw energies in excess of 300 calories per gram. Fuel is vaporized in the ranges of 500 calories per gram. It has been estimated that during this reactivity insertion period enough energy was deposited in the fuel such that perhaps one-third of it saw energies of between 400-500 calories per gram. The Soviets postulate that the fuel at that time dispersed into the channels and interacted with whatever water was left in the channels. Assuming that the accident proceeded in the lower one-third of the core at this time, and with the additional interaction with the water, pressures were produced on the order of tens of atmospheres to do the damage which resulted.

J. C. Ebersole speculated that, in essence, the fault was a rod-drop accident. The fuel actually evaporated and spit out into the water. T. Speis agreed that it was dispersed into the water. Experts have not agreed that it was a classical steam explosion which produced the pressures which led to the damage. J. C. Ebersole asked if many of the pipes, including vertical risers, were blown out. T. Speis indicated that all or most of them were blown out. The NRC postulates that there was a reactivity excursion, the reactivity then came down, steam production took place in the core as a result of the fuel core interaction, and the steam led to a second reactivity increase, and then to a power spike. C. P. Siess asked if the second peak was after the fuel had been dispersed. T. Speis indicated that that was the consensus. The accident occurred in the lower one-third of the core. H. W. Lewis disputed the homogeneity in the core as analyzed by the Soviets. He thought this a simplified analysis. T. Speis agreed that one would have to do a much more detailed analysis to settle the argument. M. W. Carbon noted that in the Soviet analysis a steam explosion is assumed. T. Speis agreed and indicated that the power can only come from a second voiding wave that took place following the interaction of the fuel with the water in the channel. M. W. Carbon noted that the fuel vaporized and it does not necessarily follow that you had a steam explosion. D. A. Ward thought the Chernobyl accident best described as an "explosive thermal expansion."

T. Speis noted that the Soviets contend that subsequent to the second power peak a hydrogen explosion took place. C. P. Siess asked if the second peak is necessary to explain the damage to the core. T. Speis indicated that the first peak is sufficient to explain the damage, and NRR did not even bother to do any calculations on the postulated second power peak. C. Michelson asked about further effects of the reactivity excursion. T. Speis indicated that the pressure in the core lifted the upper slab plate. While it takes only about two atmospheres to do this, from this type of event tens of atmospheres were probably generated. The crane was pushed up and punctured the wall and many pieces of graphite left the core itself. All the pressure tubes were severed. The forces destroyed the heterogeneity of the core and the reactor finally shut down as a result of the homogenization that took place. H. W. Lewis discussed whether the coupling during the accident was through the neutronics or the hydraulics.

T. Speis discussed the daily radioactive releases, indicating that the early release associated with the explosion and decay lasted 4-5 days. During this phase, a substantial amount of actinides was released. The Soviets estimated that about 3.5 percent actinides were released, which is only seen in the worst kinds of accidents. Basically 100 percent of all the noble gases were released, coupled with about 20 percent of the iodine-131, and about 13 percent of the cesium. D. Okrent asked why only 20 percent of the iodine showed up. T. Speis indicated that the table included only what fell out in the Soviet Union and also contains 50 percent uncertainty. J. C. Ebersole asked the physical form of the iodine. T. Speis indicated that most of it was gaseous. The Soviets estimate that the Chernobyl source term contained 50 million curies of noble gases, and about 50 million curies of other radionuclides (see Appendix XIII).

T. Speis indicated that about 30 fires broke out, mostly on the roof of the building, the top of the machine hole. The fires were quenched by water. D. W. Moeller asked what the firefighters had in the way of protective equipment. T. Speis noted that most of the individuals who were injured or died were firefighters themselves.

T. Speis discussed the entombment of Unit 4 in a concrete building. He indicated that an interconcrete partition wall in the turbine hall will separate the destroyed Unit 4 from Unit 3. A metal partition will separate Units 2 and 3. J. C. Ebersole asked if the Soviets were able to man the other units during the accident and take them down safely. T. Speis noted that Unit 3 continued to produce power for a few more hours after the accident. He pointed out that Units 3 and 4 had a common ventilation system, and the operators of Unit 3 were substantially affected by the radioactivity from the accident. H. Denton added that there will be additional workshops to discuss the fact that they did shut down Unit 3 safely.

T. Speis discussed proposed RBMK-1000 (Chernobyl-type) modifications. Control rods are to be permanently inserted in the core to a depth of 1.2 meters. The rods will not be able to be withdrawn beyond this point. The operators have specific authority to have a minimum of 50 control rods in the core to reduce the void worth so that it never exceeds one beta. F. J. Remick asked whether the Chernobyl reactor, Unit 4 was being used for plutonium production purposes. T. Speis indicated that the Staff's best understanding was that it does produce some plutonium, like all reactors, but it was not being used as a production reactor.

F. Congel, NRC, discussed the latest information on the radioactive plume. He indicated that the plume did rise to 1200 meters in height, and an exposure rate of 1 R per hour was measured by airplanes 5-10 kilometers from the site during the first four days. The individual doses that were ultimately fatal occurred to the firefighters. The control room operator of Unit 4 was apparently dosed with a lethal amount of radiation. At the end of the accident, 203 people were hospitalized because of radiation sickness. Cesium and plutonium were found in the lungs of lethally-exposed individuals during autopsies. Doses exceeding 400 rems up to a maximum of 1200-1600 rem were received by 35 individuals. The child thyroid doses were substantially higher than all of the other general population doses. As a result, this is the only group that the Soviets have decided to watch closely on a long-term basis. He discussed summaries of general population collective doses, noting that the Soviets estimate that about 75 million people are considered exposed as far as external dose is concerned. The Committee discussed the level of conservatism of the Soviet's calculation of health effects, noting that the estimates might be too high by as much as a factor of 10. D. W. Moeller asked if the calculations assume interdictions such as plowing and turning over the soil. F. Congel indicated that that was one of the items brought up at the meeting. He speculated that, if the Soviets thought there was enough cesium in the environment to warrant further study, they could plow deeper layers or discuss addition of chemicals to the soil to cause leachout of the cesium.

F. Congel discussed emergency response measures taken by the Soviets. He indicated that on the first day of the accident open activities were banned at all kindergartens, and potassium iodide was immediately distributed. Consumption of milk with a concentration of 10^{-7} curies per liter or more of iodine-131 was banned. Each evacuee was examined medically and blood tests taken. Some blood tests revealed individuals receiving as much as 40-50 R whole-body dose. In this case, the blood tests and examinations were repeated. Tests were repeated for those evacuees who tested down to 1-10 R whole-body dose. Note was taken of the fact that the Soviets used an examination of blood samples to determine exposure levels. Long-term programs are being established for medical and biological monitoring of populations and plant personnel. A workshop has been set up to ascertain data necessary to follow a group in the city of Pripyat who received 40-50 R whole-body.

H. Denton indicated that his thoughts are that the Soviets sincerely wish to improve the safety of their plants. They are looking for an opportunity to enter a dialogue, but wish to work mainly through the IAEA. He anticipated that there would be future IAEA conferences. At least a dozen have been suggested by the Soviets themselves. Three questions were raised repeatedly and were repeatedly asked by the media: 1) was this a nuclear explosion? 2) would a western-style containment have been of any use? and 3) what is the validity of the man-rem figures to estimate potential latent cancers in the Soviet Union from the Chernobyl release? H. Denton indicated that it is the NRC Staff's view that a large, dry pressurized-water containment could have withstood the energy that was generated and would have limited the releases perhaps to the levels of Three Mile Island. The estimated man-rem figures are speculative regarding the long-term health effects. F. J. Remick asked what the Soviets had to say about additional training such as qualification training and engineering expertise on shift. H. Denton indicated that the Soviets were interested in accreditation and suggested the possibility of international accreditation. The Soviets also stated that they plan to increase the use of simulators. C. Michelson asked about the contribution of any of the fire effects in terms of dispersal of the radioactive materials into the countryside. F. Congel indicated that the fire principally provided the energy that lofted the plume. While there are several phases not completely understood, there was an explosion and an immediate dispersal around the site. Once the fire started, it contributed to a chimney effect.

VI. Improved Light Water Reactors (Open)

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

C. P. Siess discussed a prepared straw-man proposal for performance and design effects of containments (see Appendix XIV). He presented an historical review of ACRS recommendations on containment performance objectives, noting that the first time the Committee mentioned the subject was in a June 9, 1982 letter on the Proposed Policy Statement on Safety Goals for Nuclear Power Plants (NUREG-0880). He noted that the containment design objective that the ACRS has suggested has always been focused on the implementation phase of the Safety Goal Policy Statement. The Committee's comments in an August 9, 1983 report on the Proposed Safety Goal Evaluation Plan indicates that the core melt design objective is an indication of the emphasis on accident prevention, while containment performance objectives are important as an indication of the need for mitigation. He mentioned the two performance guidelines proposed in the April 15, 1986 ACRS report on the Proposed Safety Goal Policy Statement, which talked about a 10^{-4} objective on core damage and a second guideline related to containment performance. The latter stated that a large release of radioactive materials to the environment should be less than 10^{-6} per reactor year. He indicated that the Commission's July 30, 1986 issuance of the "Final Policy Statement on Safety Goals for the Operation of Nuclear Power Plants" talks about defense-in-depth and says that the overall mean frequency for a large release of radioactive material into the environment from a reactor accident should be less than 10^{-6} per reactor year. He noted Commissioner Asselstine's citing of accident prevention, mitigation, and defense-in-depth and his proposed containment performance criterion of the mean frequency of containment failure in the event of a severe core damage accident of less than 1 in 100. A large release is defined as one that would result in a whole body dose of 5 rem to an individual located at the site boundary. This corresponds to the EPA Protective Action Guidelines and would not require an evacuation. He mentioned Commissioner Bernthal's separate view proposing a defense-in-depth philosophy that severe core damage accidents should not be expected on average to occur in the U.S. more than once in 100 years. Since this refers to the 100 or so plants operating in the U.S., it can be derived that a goal of 10^{-4} per reactor year is implied. Containment performance should be such that severe accidents with substantial offsite damage would not occur on the average more than once in 1000 years, which for the 100 plants implies 10^{-5}. Since Commissioner Bernthal talks about a 90 percent confidence that the offsite release goal is met, this brings the number much closer to 10^{-6}.

C. P. Siess suggested some ground rules for the proposed containment objective. He indicated that his Containment Performance Design Objective (CPDO) is limited to future standard plant designs for LWRs. This eliminates from consideration BWR Mark I and Mark II containments and the PWR ice-condenser containments. There would not be anything to preclude applying the CPDO to future HTGR or LMR designs.

C. P. Siess proposed as a CPDO that the overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 10^{-6} per reactor year, where a large release is one that results in a whole body dose of 5 rem to an individual located at the site boundary. This objective is the same as that proposed by the ACRS in its letter of April 15, 1986 and is probably not inconsistent with Commissioner Bernthal's proposal requiring 10^{-3} per reactor year for all plants in the U.S. with 90 percent confidence. He suggested that, taken alone, this statement is more a general performance guideline rather than a containment performance objective. He noted that this objective was coupled to a limit on frequency of severe core damage in the ACRS recommendation; and, although the Safety Goal Policy Statement does not mention a meaningful limit on frequency of severe core damage, it does relate this performance guideline to defense-in-depth and the reliable performance of containment systems. In any case, he suggested that it is highly unlikely that the 10^{-6} probability can be met either without a containment or by the containment alone.

D. A. Ward thought that the Committee ought to be looking for an objective to parallel the core melt objective. This approach does not do that. C. P. Siess noted that his CPDO clearly involves containment but is not exclusively containment. Severe core damage must be considered. as well as the definition of a large release. No credit is taken for evacuation and meteorology. This is another important factor.

J. C. Ebersole spoke in favor of a filtered, vented containment. D. Okrent cautioned that one needs to be a bit careful since some postulated accident sequences develop so quickly that containment venting is very difficult from an engineering sense. The Committee discussed the impact on containment of a steam explosion and pressurized thermal shock.

M. W. Carbon posed several questions regarding the ability to shut down a nuclear power plant safely. C. J. Wylie suggested that the B&W control rod system, which is an adaptation of the drive system used by the U.S. Navy, was an inferior drive system when compared to the Westinghouse design. The magnetic jack design of Combustion Engineering, however, is similar to the Westinghouse design and a good performer. M. W. Carbon spoke in favor of a backup passive shutdown system. Other Committee members thought that was a good idea. J. C. Ebersole spoke briefly about his problems with the hydraulic system associated with the control rod drive system for General Electric BWRs. His particular concern has been the control rod drive dump volume as a common-point vulnerability. C. Michelson noted that European BWRs do not use a dump volume.

The Committee briefly discussed the issue of the failure of core support structures, noting that major structural failures represent one of the key uncertainties in liquid metal fast reactor safety. He wondered whether greater inspectability is needed for PWR pressure vessels, as well as the core support structures.

M. W. Carbon introduced several other subjects to complement the defense-in-depth concept. He wondered whether an improved decay heat removal system is needed as well as PORV capability in all plants. He noted that G. A. Reed is quite concerned about minimum water levels for all steam generators. He wondered whether a totally independent three-train emergency core cooling system, such as those in West Germany, ought to be a requirement. Should there be minimum requirements for both electrically-driven and steam-driven feedwater pumps. He wondered whether there ought to be a greater emphasis on natural convection cooling and the capability to withstand total station blackout. Should there be greater emphasis on the ability to withstand ATWS events.

M. W. Carbon brought up the subject of potentially severe common-mode failures as a result of seismic events. He speculated on higher seismic requirements for collections of plants near a major city. He brought up the issue of core catchers as a delay mechanism. J. C. Ebersole agreed with M. W. Carbon that a future BWR design would be greatly enhanced with the ultimate plant protection system (UPPS) presented at the CESSAR II review and with an improved scram system.

J. C. Mark expressed alarm that there is no reference to sabotage in the NRC Staff's plan concerning future LWR designs. He thought it quite important that sabotage be considered in the design complex of a plant. He briefly mentioned wire fences and armed guards to take account of the external threat, and also a separate bunkered decay heat removal system that could run when necessary without interference. D. Okrent suggested a systems approach to sabotage to determine the potential effectiveness of a dedicated shutdown heat removal system. This would incorporate plant-specific layout features. He speculated on the possibility of being able to relax certain kinds of restrictions on existing plants. The existence of geographical separation and the existence of certain backup systems might allow one to relax certain restrictions. D. Okrent thought the Committee ought to look to the West Germans and the Swiss regarding heavier and more specific external guard features. He thought it interesting that they use only about one-third the number of guards as are used at U.S. plants. C. J. Wylie suggested one reason for this was the use of guards with dogs and machine guns and the willingness to fire on intruders. The situation is somewhat different for U.S. plants. He noted a document from the International Task Force on Prevention of Nuclear Terrorism by the Nuclear Control Institute. F. J. Remick mentioned seeing that document recently. D. Okrent requested a copy.

F. J. Remick made several suggestions regarding the topic of revisions to NRC regulations. He noted that there are a myriad of Federal acts and regulations that affect utilities; however, he indicated that he would narrow his focus, basically, to the Atomic Energy Act and Title 10 of the Code of Federal Regulations. He raised various topics in the form of questions:

1. Are the NRC regulations too prescriptive or "cookbookish" in contrast to utilizing performance standards to a greater extent? He cited an example of a quite general regulation in the general design criteria of Appendix A to Part 50. He noted that there are other areas that are quite prescriptive, such as Appendix B, Appendix K, and Appendix R to Part 50 as well as Appendix A to Part 55 and Appendix A to Part 100. He wondered whether the NRC should promulgate regulations on the management of nuclear activities.

2. What may be done to reduce or eliminate the adversarial relationship between the NRC and licensees? How can one instill greater mutual respect and cooperation in the process? He cited industry initiatives over the last year or so, such as the readiness review pilot program and the training and requalification of nuclear power plant personnel. Another case would be the Commission Policy Statement on Fitness for Duty which has been recognized as an industry initiative.

3. Does the imposition of enforcement fines contribute to safety? He suggested, as an alternative, personal communication of plant shortcomings to the utility chief executive officer, vice presidents, and plant managers.

4. Should the NRC be restructured to be headed by a single administrator accountable to the President and Congress?

5. Is there still a need for an ACRS?

6. Once training programs at nuclear utilities are accredited, should the licensing and relicensing, or certification and recertification, of ROs and SROs be the responsibility of the utilities or of the Institute of Nuclear Power Operations (INPO)?

7. How can the NRC, the EPA, FEMA, and other agencies be brought to better coordinate their various regulations?

8. Should there be a backfit rule that has some teeth to keep the process under control?

9. Should 10 CFR Appendix R be eliminated as a codified requirement? Should it be designated as a Regulatory Guide or should one start over from scratch?

10. Should the entire NRC license amendment process be revised by defining categories of routine amendments that could be granted with minimum delay?

11. Should the emergency planning requirements of 10 CFR 50.47 and Appendix E to Part 50 be revisited?

12. Should environmental qualifications of electrical equipment requirements be reexamined?

13. Should emergency core cooling requirements of Appendix K to Part 50 be reexamined?

14. Should the physical security requirements of Part 73 be revisited?

15. Do reactor-vessel-level-indication systems, in accord with NUREG-0737, make technical sense?

16. Should a de minimis level for radioactive waste be established? He noted that this issue is probably part of the revision of 10 CFR Part 20. D. W. Moeller noted that the Commission has issued for comment a proposed policy on wastes below regulatory significance.

17. Should the NRC establish a policy which would require that practical experience be a requisite qualification for key staff, and that a program should be arranged to accomplish this?

18. Should the NRC have a program that provides for planned rotation or periodic reassignment of senior staff managers to obtain better interoffice communications and coordination?

19. Should the NRC institute an independent quality assurance audit program for NRC regulations? How can the NRC prevent its Staff from making Regulatory Guides and branch technical positions de facto requirements?

D. Okrent suggested that members who have prepared discussion papers single out specific items they think might be possibilities for a Committee report. He asked that they be part of the discussion materials at the next ACRS meeting in October.

VII. Meeting with the NRC Commissioners (Open)

[Commissioners present were L. W. Zech, Jr., Chairman, T. M. Roberts, F. M. Bernthal, and K. M. Carr.]

Chairman Zech indicated receipt of the ACRS letter on Reactor Standardization dated August 15, 1986. He noted that the ACRS letter considered the NRC Staff Proposed Policy Statement to the Commission of May 14, 1986 and the Draft Policy Statement sent by the Secretary to the Staff on April 10, 1986. He indicated finding the ACRS comments constructive and very helpful and personally thought this was an area in which the ACRS could make a strong contribution.

C. J. Wylie indicated that there were nine points which the Committee discussed in detail in its letter which addressed the April 10 and May 14 drafts individually. The first point mentioned was that the title of the Policy Statement should be changed to the "Policy Statement on Certification for Nuclear Power Plant Standard Designs." He thought this best described the substance and focus of the Policy Statement and that that was where the bulk of the benefit would be derived from standardization of designs. The ACRS did not believe it prudent to universally standardize other areas as the benefits derived would be small compared to the problems this could create. Areas related to site-specific construction, such as concrete mixes, would not be prudent to standardize. Some areas could be standardized, such as nondestructive examination and quality assurance. There are certain aspects of quality control that could possibly be standardized. Procurement as well as training are not areas where it would be beneficial to standardize. Training varies from utility to utility and the accreditation programs by INPO seem to have the process well in hand.

The second point made by the Committee referred to the Commission's draft of April 10 in which the Committee redrafted a statement of the meaning of standardization. In a third point, the Committee did not recommend inclusion of a comment in the policy statement that standardized nuclear power plants should be used to satisfy the ultimate goal of certified designs constructed on preapproved sites. The ACRS thought that the Policy Statement should make it clear that it supersedes the Commission's previous policy on standardization in 1978. The Committee offered two suggested opening paragraphs for the Policy Statement. They place an importance on the standardization process and focus on the reference system design concept, certification concept, and state that the goal is an essentially complete design to be referenced in individual licensing applications. The Committee thought that the Policy Statement should include a reference to the Commission's policies on safety goals, severe accidents, and advanced reactors, as well as other Commission policies. Implementation should be covered in the accompanying NUREG. Regarding comments by former Chairman Palladino on empirical information and prototypical testing, the Committee elected not to comment until it has had further discussion with the NRC Staff and has seen the details of the NUREG which will accompany the Policy Statement. D. A. Ward added that the Committee is concerned that the standardization approach will lack the benefits that come from prototype testing and the gradual evolvement of an improved design. C. J. Wylie indicated that the Committee recommends that the proposed NUREG be submitted for public comment. As a last point, he noted that the Committee thought it absolutely necessary that "essentially complete design" be thoroughly and clearly defined, so that the industry and NRC will know from the outset what information needs to be generated, and the degree to which it needs to be defined in the application for design certification. D. Okrent expressed his concern that there may be a proliferation of designs. He suggested that the regulators limit the current authority to a few very good designs. He expressed concern regarding General Electric's use of a PRA to make design changes to GESSAR II. He thought that if this were meant to be a plant for the future this was not a satisfactory approach. He thought it would be much better if the Commission, for a range of areas, could provide either performance criteria or, in some cases, general design requirements (sabotage would be included in this group) which the Commission expects to see in future standard designs.

C. Michelson presented the comments of G. A. Reed, the ACRS member recovering from a heart ailment and not currently present. He noted G. A. Reed's comment that in the case of PWRs with three vendors for systems, such as the ECCS, three different arrangements have been proposed which essentially have a common purpose, or function. Reed's thought would be to study these functions carefully to see if there is a common way of accomplishing them and then standardizing to one product line. C. Michelson noted that G. A. Reed thought there might be a significant safety gain from this approach, and that this should be one of the thrusts of the Standardization Policy.

Chairman Zech noted that the Committee did not cite the importance of simplicity in future designs. He thought that U.S. plants are often more complicated than they need to be, and the Commission might consider the value of simplicity as it relates to safety. J. C. Ebersole suggested that the Commission consider as a national position whether the U.S. needs both the PWR as well as the BWR. He called attention to the potential of the simpler BWR despite the fact that there are admitted metallurgical problems (stress corrosion cracking) that will have to be solved . He pointed to the use of the suppression pool as a scrubbing system in the event of a severe core-damage condition. He noted that a shutdown BWR core has a power density that is readily cooled by open evaporation with virtually no pressure. Subsequent to the scrubbing operation there would be emission straight to the atmosphere.

C. P. Siess explained why the Committee did not comment on the issue of simplicity. He referred to another Commission Policy Statement on Advanced Reactors which had a considerable list of desirable features for future reactors. While this Policy Statement concentrated on items such as redundancy, diversity, and, possibly, simplicity, the Standardization Policy Statement appears to be procedural. Commissioner Bernthal indicated that the Standardization Policy Statement ought to contain explanations regarding the interdependence of that Policy Statement and the Policy Statements on Advanced Reactors, Safety Goals, as well as Severe Accidents. He expressed concern that the ACRS, and perhaps the Commission, are not quite qualified to comment in any detail on the meaning of a complete plant design. There may be several competing definitions among the ACRS and the Commission. He also noted that from the Commission's earlier deliberations and documents on this subject it appears that one has to be very careful dealing with the legal maze of antitrust considerations. C. J. Wylie suggested that his definition of complete plant design is detail to the level of performance specifications, with everything specified except the nameplate data. D. A. Ward suggested that most of the benefits of standardization would be in assuring that the design is essentially complete and defined. Commissioner Bernthal suggested his definition of "standard plant" would deal with more than merely issuing rather detailed specifications on components, but draw a line that refers to procedural and technical matters that involve site-specific characteristics. Anything not site-specific should be specified. H. W. Lewis mentioned that the FAA has specifications which are precisely performance specifications. It is incumbent upon a manufacturer to produce a working model (aircraft) to get type certification. Eventually there is certification of components based on performance characteristics. He indicated that the process of generating a standardized nuclear plant is quite a different situation since the NRC does not expect generation of a completely new nuclear plant.

Commissioner Carr asked how the Committee thinks standardizing will inhibit or prevent innovation and improvement. C. J. Wylie did not think the concept of designs conceived by vendor-architect/engineer teams would inhibit innovation or advances in design. C. Michelson was not entirely sure that the standardized design that would result from such teams would necessarily be their latest product line. Each team may have two or three standard designs. There could in fact be several, a whole line, of standard plants.

Chairman Zech indicated, in conclusion, that he thought that the NRC ought to concentrate on insuring that the current nuclear power plants in the U.S. continue to operate safely. At the same time, the NRC can assure that the peaceful uses of nuclear materials in other applications is also used to the benefit of the country. The Commission should then focus on assuring the quality of the nuclear plants that are now under construction. It will be very important that the NRC make sure that they will be operated in a safe manner. For the future, standardization is an important issue with many benefits. Part of the problem with regulation, as well as operation and even safety, has been the many different kinds of nuclear plants. Standardization will enhance training, maintenance, as well as other operational matters. He suggested that the direction to go is toward a more disciplined and safe technology. This does not means that it is appropriate that the Commission rigidly standardize. It should allow more than one design and allow the utilities to have a choice. However, he cautioned against proliferation of a multiplicity of designs. He thought that the designs ought to be conservative, safe, reliable and more of an evolutionary character than a revolutionary one. He requested that the Staff comment on the ACRS comments to the Commission. The Commissioners will then review both the ACRS and Staff comments to derive a final consensus on a standardization policy.

Commmisioner Bernthal solicited comment from the ACRS regarding what they thought the worst element in the Policy Statement was. C. J. Wylie thought the statement referring to the standardization of design, as well as construction, will probably give the nuclear industry the most difficulty. M. W. Carbon indicated that he was disappointed in the fact that the Policy was not really a standardized design policy, but a standardized policy for certification of a design. He cited EPRI's attempt to develop standardized requirements from which a proliferation of designs might result. He found this lack of specificity and clarity disturbing. D. A. Ward thought that the Policy Statement did not focus on the desirability of standardizing on a very good design and it could lead to the Commission accepting a 20-year-old design. F. J. Remick and M. W. Carbon thought that the specificity in the NUREG document which will accompany the policy statement will be vital. J. C. Ebersole commented that all should understand that standardization as defined in the Policy Statement does not connote improvement; it is a prelude to continuing to do what has been found permissible. It will tend to lock in current practice and not encourage improvement or enhancement.

VIII. Seismic Qualification of Safety-Related Equipment in Operating Nuclear Plants (Open)

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

C. J. Wylie explained that the present topic is a carryover item from the August 1986 ACRS full Committee meeting, and refers to an August 5, 1986 meeting of the Subcommittee on Reliability Assurance which met with the NRC Staff and Seismic Qualification Utilities Group (SQUG) to review the resolution of USI A-46, Seismic Qualification of Safety-Related Equipment. He indicated that the three main concerns are the adequacy of equipment anchorages, the functional capability of relays, and the subject of outliers. Of import is the maintenance of hot shutdown for 72 hours. The Staff will be looking at equipment in close proximity to the equipment important to hot shutdown. The Subcommittee was concerned that the resolution may be restricted only to equipment essential to maintain hot shutdown and not include in the scope of USI A-46 equipment which may not impact hot shutdown but may be involved in interactions with that equipment. This is considered a separate generic issue by the Staff. The Subcommittee is concerned about the total seismic loading on equipment, as well as the inadvertent actuation of fire protection equipment, seismically-induced fires, and impact on smaller, high-energy pipes. He noted that these items are not being considered by the Staff anywhere else.

C. P. Siess referred to a draft ACRS report on the table noting that it focused on the seismic condition of the nuclear plant and not on the seismic qualification of equipment. C. Michelson noted that the strategy of A-46 is to focus on a minimum set of equipment which will affect and maintain safe shutdown and ignore other equipment. The Staff is looking only at this minimum set of equipment and its protection. He indicated his desire to expand the scope of A-46. C. P. Siess agreed with C. Michelson that the resolution focuses on the seismic qualification of a too-narrowly-defined list of equipment. A member of the NRC Staff pointed out that the resolution does not require redundancy of every component in the minimum set. One needs at least, in the event of failure of a component which does not have a redundant piece of equipment, a parallel path to bypass the failed component (component level single-failure criterion). An alternate path is allowed in the case of a failed nonredundant component, but this component must be seismically qualified. The relay chatter issue is enveloped by the ruggedness success data being collected by EPRI and SQUG. It is, however, only for qualified relays. Systems reviews will focus on a set of relays where chatter will be important. Then those relays that chatter will be tested individually and simultaneously. The effects of relays not included in the minimum set of equipment, and supporting equipment, will not be considered.

C. P. Siess referred to a paragraph in the ACRS draft report prepared by D. Okrent which referred to the omission of indirect interactions, such as flooding from the failure of a nonseismically-qualified tank. He thought this item which refers to Generic Issue 114 ought to be in a separate ACRS letter. D. Okrent agreed to drop the item if the Committee promised to follow up on this issue. The Staff indicated that consideration of the flooding from tanks is outside the scope and is included in USI A-40. A-46 does have limited consideration of tanks with regard to the failure of equipment anchorage. D. Okrent noted that older tanks had other vulnerabilities and less margin than expected. The Staff noted that tank design regarding flexible versus rigid walls is a consideration of USI A-40. D. Okrent suggested flooding of the minimum set of equipment if the refueling water storage tank fails. He expressed concern regarding battery anchorages and noted that the issue of vital water tanks has been a 15-year-old problem. C. J. Wylie cited the ACRS concern regarding interaction of equipment not in proximity to the train of a minimum set of safe shutdown equipment. The Staff indicated that it was not concerned so much by the spilling of the contents of a tank as it is with a flooding issue to be treated generically by walkdowns by SQUG. If problems do occur, they are to be handled separately outside of A-46. C. Michelson thought the Staff's approach was different from the case of Appendix R, or pipe breaks, because of the consideration only of structures that impact and not water effects. The Staff indicated that they have tried to bound the problem.

D. Okrent expressed concern that a sizable number of operating plants may still have undiscovered significant deficiencies involving battery supports and other vital equipment. He pointed out that seismically-induced failure of station batteries can cause loss of function of equipment vital to safe shutdown. He asked the NRC Staff representatives if this issue has been examined for plants being reviewed under the current Standard Review Plan. A formal response was promised.

IX. NRC Incident Investigation Procedures (Open)

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

H. W. Lewis explained that a meeting of the ACRS Subcommittee on Regulatory Policies and Practices was held on June 26, 1986 to discuss the Incident Investigation Team (IIT) review and the regulatory process with regard to the June 9, 1985 event at the Davis-Besse Plant. The Subcommittee reviewed the report, NUREG-1201, dated June 1986. He noted that the IIT set up on Davis-Besse resulted in a letter from the EDO assigning tasks among different groups in the NRC staff. The ACRS had previously written a letter asking if resolution of these 40 tasks would produce closure on the Davis-Besse incident. Chairman Palladino promised that the Staff SER issued in January 1986 would provide closure.

There was an exchange of views from all parties at the Subcommittee meeting regarding noncritical faults in the IIT. H. W. Lewis indicated that the EDO was asked where there was a coherent ending of the IIT investigation. V. Stello, EDO, indicated that the 40 tasks that were assigned were not intended to be a clear response to Davis-Besse, but an event-generic laundry list.

H. W. Lewis thought it not necessary for the ACRS to respond in writing at this time. He thought the Committee ought to let the IIT process, which has been established by the Commission, develop. Weaknesses in the IIT process should show up when a serious incident occurs. C. P. Siess indicated that there was an attempt made at the Subcommittee meeting to determine what the threshold for problems was, and the EDO's response was that it would be left for the regions to discover. C. Michelson thought there was a need for better closure on IIT reports than an SER which deals with other ancillary subjects. He thought that closure on the Davis-Besse incident was not apparent since the SER indicates that it is still under investigation. D. A. Ward thought it apparent that a maintenance problem was the root cause of the Davis-Besse incident.

X. Seismic Margins Program (Open)

[Note: R. P. Savio was the Designated Federal Official for this portion of the meeting.]

C. P. Siess indicated that 12 ACRS members were present at the meeting of the Extreme External Phenomena Subcommittee on August 6, 1986, which was set up as a tutorial on seismic margins. He noted that D. Okrent pointed out the need to stay abreast of new developments and to take note of each seismic PRA that becomes available. He indicated that the discussion confirmed that there is a wide spectrum of opinion on the levels of the seismic hazard in various geographic locations in the U.S. Estimates of seismic hazard vary tremendously from expert to expert. For large earthquakes one is dealing with the tail of the distribution curve. The upper limit which one puts on earthquake magnitude changes that result significantly. He noted that estimates of seismic hazard were further complicated by events such as the discovery of recent activity along the Meers Fault. [It is a large fault east of the Rockies that was previously thought quiescent but has now been found to be active.] He spoke of sand blows discovered in Connecticut that may have been caused by an earthquake as large as the Charleston earthquake. The Staff's invention of the concept of tectonic provinces was discussed. H. W. Lewis thought an historical profile of earthquake activity at actual sites would be more meaningful than for tectonic provinces. C. P. Siess noted that the tectonic-province approach allows the use of data obtained over a region of similar seismic potential. H. W. Lewis thought that seismic PRAs are conservative because they always put the earthquake next to the site, highly overrating the seismic-hazard risk. D. A. Ward asked if there is an NRC research effort on seismic hazard. C. P. Siess noted that research has not decreased the uncertainty regarding seismic hazard and, in some cases, has actually increased the perceived uncertainty. That is why there is the move to a PRA approach.

XI. Executive Sessions (Open)

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

A. Subcommittee Assignments

1. ACRS Workload and Resource Assignments

The members briefly discussed specific ACRS subcommittee assignments for Fiscal Year 1987 made during the September 10, 1986 meeting of the ACRS Planning Subcommittee. M. W. Carbon noted that three subcommittee meetings for review of the DOE advanced reactor concepts appears inadequate. Chairman Ward asked members to review the Summary of the ACRS Planning Subcommittee Meeting September 10, 1986, Bethesda, Md. and transmit any complaints and objections to the Planning Subcommittee for its further consideration. Per the request of the Office of IE, it was agreed at the September 10 meeting to incorporate the meeting of the IE Subcommittee with that of the Regional Subcommittee.

2. Reactor Operations

Discussed during the September 10, 1986 ACRS Planning Subcommittee meeting and briefly by the full Committee during the 317th ACRS meeting, was a proposal to modify the procedure for screening recent significant operating events at nuclear plants. Instead of routine meetings of the Subcommittee on Reactor Operations, J. C. Ebersole, the Subcommittee Chairman, was asked to screen proposed items by making direct contact with the Inspection and Enforcement representative (Jack Rosenthal/Eric Weiss). The objective would be to develop the information he needs to report to the Committee. Other members should be informed of the topics to be discussed at these sessions so that interested members can suggest additional events for consideration. The most significant of these events selected by J. C. Ebersole will be scheduled for a full 2-3 hour presentation and discussion with the full Committee.

3. ACRS Officers for Calendar Year 1986

Chairman D. A. Ward appointed a committee to nominate candidates for Chairman and Vice-chairman of the ACRS for calendar year 1987. The Nominating Panel will be composed of W. Kerr, M. W. Carbon, and D. W. Moeller as Chairman with C. P. Siess as an alternate member should one of the regular members be nominated and not be able to participate. In connection with the charge to the Nominating Panel, proposed changes to the ACRS Bylaws that involve extending the term of office of the Chairman and Vice-chairman to two years as well as a proposal to establish a Planning Committee and define its composition were discussed briefly but no action was taken. C. Michelson proposed three amendments to the Bylaw changes as follows:

1) Retain one year terms for the Chairman and Vice-chairman with unrestricted reelection potential

2) Election of ACRS officers should be by secret ballot without benefit of a Nominating Panel

3) Two members-at-large should be elected to the Planning Subcommittee by the full Committee. The ACRS Vice-chairman should not serve.

Chairman Ward and the Committee members still in attendance on Saturday afternoon agreed that these matters should be taken up with the maximum number of ACRS members present. A prime time session was requested for the 318th ACRS meeting and the designated Nominating Committee was told not to begin deliberations until the Bylaw changes are approved and C. Michelson's proposals are fully considered.

B. Reports, Letters, and Memoranda

1. ACRS Suggestions for an NRC Long Range Plan

The Committee prepared a draft report to the Commissioners consisting of a series of suggestions for development of an NRC Long Range Plan. Time did not permit completion of the report and it will be carried over for discussion during the 318th ACRS meeting.

2. ACRS Comments on the Resolution of USI A-46 (Seismic Qualification of Equipment in Operating Plants)

The Committee prepared a report to the Commissioners of its review of the proposed resolution of USI A-46 (Seismic Qualification of Equipment in Operating Plants). The final resolution of USI A-46 has not yet been presented to the Committee to Review Generic Requirements (CRGR). The ACRS wishes to be kept informed of the outcome of the CRGR review. Additional comments by D. Okrent were appended.

3. ACRS Comments on the Proposed Revision to the ECCS Rule - 10 CFR 50.46 and Appendix K

The Committee prepared a report to the Commissioners of its review of the NRC Staff proposal to issue for public comment a revision to the ECCS Rule - 10 CFR 50.46 and Appendix K which provides for use of "best-estimate" ECCS evaluation models. An extensive Compendium of ECCS Research will be issued in support of the Rule. The ACRS wishes to review the final document.

The ACRS Executive Director was asked to inform OGC of questions related to backfitting requirements for the new "best-estimate" ECCS evaluation models so this can be included in the October meeting with OGC regarding backfitting requirements.

4. Proposed Resolution of Generic Issue 124, "Auxiliary Feedwater System (AFWS) Reliability"

The Committee prepared a letter to the EDO regarding a proposed resolution approach for Generic Issue 124: "Auxiliary Feedwater System Reliability."

5. ACRS Comments on Degraded Piping Research

The Committee prepared a letter to the EDO regarding the NRC research program on degraded piping.

C. Future Agenda

1. Future Agenda

The Committee agreed on tentative agenda items for the 318th ACRS Meeting, October 9-11, 1986 (see Appendix II).

2. Future Subcommittee Activities

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

D. Seabrook Nuclear Power Plant

The Committee agreed that the issues of proposed reduction of source term and distances for emergency planning at the Seabrook Nuclear Power Plant are the "test case" for implementation of generic requirements regarding implementation of the NRC Class 9 Policy, and the September 25, 1986 meeting of the Seabrook Subcommittee was to be rescheduled in favor of a joint meeting of the Severe (Class 9) Accidents and Occupational and Environmental Protection Systems Subcommittees on the above subjects.

E. ACRS Meeting Dates for CY-1987

ACRS meeting dates for CY-1987 have been scheduled as noted:

Meeting	Date
321st	January 8-10, 1987
322nd	February 5-7, 1987
323rd	March 5-7, 1987
324th	April 9-11, 1987
325th	May 7-9, 1987
326th	June 4-6, 1987
327th	July 9-11, 1987
328th	August 6-8, 1987
329th	September 10-12, 1987
330th	October 8-10, 1987
331st	November 5-7, 1987
332nd	December 3-5, 1987

The 317th ACRS Meeting was adjourned at 2:40 p.m. Saturday, September 13, 1986.

APPENDICES
317TH ACRS MEETING MINUTES
SEPTEMBER 11-13, 1986

ACRS-2460

Appendix I

APPENDIX I

ATTENDEES

NRC ATTENDEES
317TH ACRS MEETING

Thursday, September 11, 1986

R. W. Hernan, NRR
M. Mayfield, RES
D. Moran, NRR
M. Vagins, RES
R. Burk, NRR
D. Scaletti, NRR

PUBLIC ATTENDEES
317TH ACRS MEETING

Thursday, September 11, 1986

H. A. Glovier, NDS CORP.
K. Arn, Bechtel
L. Cuoco, Fried, Frank
Lori DiCesare, Fried, Frank
G. Meuzel, CE
H. F. Reis, N&H
L. Guntis, TEPCO
N. P. Smith, CE Co.
W. R. Schmidt, MPR Associates
R. E. Schaffstall, KMC, Inc.
P. G. Starck, MPR Associates

NRC ATTENDEES
317TH ACRS MEETING

Friday, September 12, 1986

R. W. Hernan, NRR
S. Donovan, NRR
R. Pettis, IE
V. Zeoli, RM
A. Szukiewicz, DSRO/EIB
H. Denton, NRR
D. Speis, SODO
B. Shearon, SODO

INVITED ATTENDEES
317TH ACRS MEETING

Friday, September 12, 1986

BABCOCK & WILCOX

R. Borsum
J. Bohart
J. H. Taylor
R. W. Durman
D.A. Downtain
E. Swanson

DUKE POWER COMPANY

H. B. Tucker
L. A. Reed

TOLEDO EDISON

W. T. O'Connor

GENERAL PUBLIC UTILITIES

R. J. McGoey
G. R. Skillman
R. T. Glaviano

TVA

J. J. Ritts

PUBLIC ATTENDEES
317TH ACRS MEETING

Friday, September 12, 1986

J. Trotter, NUS Corp.
W. H. Layman, EPRI
S. Letourneau, Serch Lic.
E. Waxman, Newman & Holtzinger, P.C.
J. Ahearne, Resources for Freedom
D. Nalepka, Wisconsin Public Serv.
E. Dlwiniewski, GRS
T. E. McSpadden, DOE
R. E. Taylor, Wall Street Journal

APPENDIX A
FUTURE AGENDA

OCTOBER ACRS MEETING

NRC Long-Range Plan -- Complete ACRS report	2 hours
Improved LWRs -- ACRS report to NRC regarding characteristics of improved LWRs	3 hours
Removal of Radioactive Iodine from the Containment Atmosphere -- Proposed changes in the NRC Standard Review Plan - ACRS comments	1½ hours
Seabrook Nuclear Power Plant -- Proposed reduction of source term/distances for emergency planning	1½ hours
International Operating Experience -- Report on Russian presentation regarding the Chernobyl Nuclear Power Plant accident	1 hour
Proposed Revisions of NRC Regulatory Guide 1.35, Revision 3, Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments and Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments -- ACRS comments	½ hour
Activities of NMSS - Briefing by the Director, NMSS, regarding items of mutual interest	1½ hours
Backfitting of Systems Interactions Requirements -- Briefing regarding NRC/OGC findings regarding the applicability of backfitting requirements to licensee consideration of systems interactions and ECCS Evaluation Models	½ hour
Status Report regarding USI A-45, Shutdown Decay Heat Removal -- Briefing by NRC/Subcommittee report	1 hour
Recent Events at Operating Nuclear Plants -- Discuss recent significant operating events at nuclear plants	2½ hours
Clinton Nuclear Power Station -- Briefing regarding the following items:	1 hour

1) Restructuring of the construction and operational quality assurance and quality control organizations in response to NRC concerns

2) Seismic Capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines that are part of the decay heat removal system

Subcommittee Reports

°	Standardization of Nuclear Facilities -- NUREG-1225 implementation of NRC Policy on Nuclear Power Plant Standardization	½ hour
°	Westinghouse -- WAPWR plant and NRC Staff review of the Paluel Nuclear Power Plant design changes	3/4 hour
°	Scram System Reliability -- Implementation of ATWS requirements	½ hour
°	Containment Performance Design Objectives -- Proposed generic letter regarding Mark I containment and performance requirements for all containment types	½ hour
°	IE Programs -- Functional Inspection of Safety Systems and Status of the SALP Program	½ hour
°	Maintenance Practices and Procedures -- Completion of Phase I of the NRC Maintenance and Surveillance Program Plan	½ hour
°	Implementation of Severe Accident Policy regarding:	
-	PRA Evaluation: IDCOR Methodology for evaluation of specific plants, NRC Rebaselining NUREG	½ hour
-	NRC Plan for Implementation of the Severe Accident Policy Statement	½ hour
-	Revised Source Term - Introduction of Realistic Source Term Estimates into Licensing	½ hour

REVISED SEP 12 1986

ACRS SUBCOMMITTEE MEETINGS

APPENDIX III

ACRS SUBCOMMITTEE MEETINGS

Containment Requirements, September 23, 1986, 1717 H Street, NW, Washington, DC (Houston), 8:30 A.M., Room 1046. The Subcommittee will review a draft position paper on containment performance design objectives as an addition to the Safety Goal Policy, and a draft of a proposed generic letter on Mark I containment requirements for severe accidents. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of September 22:

Dr. Mark	LOMBARDY	Dr. Okrent	ANTHONY
Mr. Ebersole	DAYS INN	Dr. Siess	ANTHONY
Dr. Kerr	LOMBARDY (9/23)	Mr. Wylie	DAYS INN
		Dr. Plesset	NONE

Joint Severe Accidents and Nuclear Plant Chemistry, September 24, 1986, 1717 H Street, NW, Washington, DC (Houston), 8:30 A.M., Room 1046. The Subcommittees will review the NRR Implementation Plan for Severe Accidents and the IDCOR Methodology for Individual Plant Evaluation. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of September 23:

Dr. Kerr	LOMBARDY	Dr. Siess	ANTHONY
Dr. Moeller	LOMBARDY	Mr. Ward	NONE
Dr. Carbon	LOMBARDY	Mr. Bender	ANTHONY
Mr. Ebersole	DAYS INN	Dr. Catton	DUPONT PLAZA
Mr. Etherington	NONE	Dr. Corradini	ANTHONY
Dr. Mark	LOMBARDY	Mr. Davis	HOLIDAY INN
Dr. Okrent	ANTHONY	Dr. Plesset	NONE
Dr. Shewmon	NONE		

Westinghouse Reactor Plants (Closed), September 25, 1986, 1717 H Street, NW, Washington, DC (El-Zeftawy), 8:30 A.M., Room 1167. The Subcommittee will begin the PDA review of the Westinghouse Advanced Pressurized Water Reactor (RESAR SP/90). In addition, the NRC Staff will brief the subcommittee on NUREG-1206 regarding the French Paluel plant. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of September 24:

Mr. Ward	NONE	Mr. Michelson	DAYS INN
Mr. Ebersole	DAYS INN	Mr. Wylie	DAYS INN

Joint Seabrook/Occupational and Environmental Protection Systems/Severe Accidents, September 26, 1986, 1717 H Street, NW, Washington, DC (Major/Igne), 8:30 A.M., Room 1046. The Subcommittees will gather and exchange information with the NRC Staff and PSNH. The Subcommittees will review efforts by the applicant to reduce the size of the emergency planning zone (EPZ). This effort uses results of the Seabrook Probabilistic Safety Assessment to justify a smaller EPZ. A primary focus will be the credit taken for the strength and leak tightness of the Seabrook containment. A status report on emergency planning around Seabrook will be heard. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of September 25:

Dr. Moeller	CARLYLE	Dr. Remick	NONE
Dr. Mark	LOMBARDY	Dr. Siess	ANTHONY

Decay Heat Removal Systems (Closed), September 26, 1986, 1717 H Street, NW, Washington, DC (Boehnert), 8:30 A.M., Room 1167. The Subcommittee will continue its review of NRR's proposed resolution position for USI A-45, "Shutdown Decay Heat Removal Systems." Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of September 25:

Mr. Ward	NONE	Mr. Wylie	DAYS INN
Mr. Ebersole	DAYS INN	Dr. Catton	DUPONT PLAZA
Mr. Michelson	DAYS INN	Mr. Davis	HOLIDAY INN

Standardization of Nuclear Facilities, October 7, 1986, 1717 H Street, NW, Washington, DC (Alderman), 8:30 A.M., Room 1046. The Subcommittee will review the NUREG for standarization policy statement. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of October 6:

Mr. Wylie	DAYS INN	Dr. Kerr	LOMBARDY
Dr. Carbon	LOMBARDY	Mr. Michelson	DAYS INN
Mr. Ebersole	DAYS INN	Dr. Siess	ANTHONY

318th ACRS Meeting, October 9-11, 1986, Washington, DC, Room 1046.

Wingspread International Conference (Closed), October 19-23, 1986, Racine, WI (McCreless). Representatives from the ACRS, RSK, GPR, and Japan will exchange information on nuclear reactor safety.

Waste Management, October 30 and 31, 1986, 1717 H Street, NW, Washington, DC (Merrill), 8:30 A.M., Room 1046. The Subcommittee will provide oversight on the technical quality and direction of NRC's radioactive waste management program by reviewing several pertinent topics, including concerns about the BWIP (Hanford) site, the Staff's review of DOE's five final EAs, assessing licensee compliance with the EPA Standard, seismo-tectonic GTP, and the waste package corrosion program. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Moeller	Dr. Carter
Dr. Carbon (tent.)	Mr. Till
Dr. Kerr	Dr. Orth
Dr. Remick	Dr. Steindler
Dr. Shewmon	

I&E Programs, November 4, 1986, 1717 H Street, NW, Washington, DC (Boehnert), 8:30 A.M., Room 1046. The Subcommittee will continue its review of the activities of the Office of Inspection and Enforcement. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Ebersole	Mr. Michelson
Dr. Kerr (tent.)	Dr. Moeller (tent.)

Safety Philosophy, Technology, and Criteria, November 5, 1986, 1717 H Street, NW, Washington, DC (Savio), 9:00 A.M., Room 1046. The Subcommittee will: (1) continue its review of USI A-17, "Systems Interaction in Nuclear Power Plants," (2) review the status of the NRC work on steam generator overfill, (3) discuss the status of the NRC Staff's development of a Safety Goal Policy Implementation Plan, and (4) discuss the implications of the Chernobyl Accident. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Okrent	Mr. Ward
Dr. Lewis	Mr. Wylie
Mr. Michelson	

319th ACRS Meeting, November 6-8, 1986, Washington, DC, Room 1046.

Extreme External Phenomena, November 20, 1986, 1717 H Street, NW, Washington, DC (Savio), 8:30 A.M., Room 1046. The Subcommittee will continue its review of the Diablo Canyon long-term seismic program. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Siess	Dr. Moeller
Mr. Etherington	Mr. Wylie
Dr. Lewis	

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Spent Fuel Storage, November 21, 1986, 1717 H Street, NW, Washington, DC (Merrill), 8:30 A.M., Room 1046. The Subcommittee will continue its review of 10 CFR Part 72 and Monitored Retrievable Storage (MRS). Lodging will be announced later. Attendance by the following is anticipated:

Dr. Siess
Dr. Kerr
Dr. Moeller
Dr. Remick
Dr. Shewmon

Regional Operations, December 2, 1986, Chicago, IL (Boehnert). The Subcommittee will begin its review of activities which are under the control of the NRC Regional Offices. This meeting will focus on the activities of the Region III Office. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Remick
Dr. Carbon
Mr. Michelson
Mr. Ward
Mr. Wylie

Metal Components, December 4 and 5, 1986 (tentative), Oak Ridge, TN, (Igne). The Subcommittee will review the HSST program, including dosimetry program by HEDL. Lodging will be announced later. Attendance by the following is anticipated:

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

Safety Research Program (Closed), December 10, 1986, 1717 H Street, NW, Washington,DC (Duraiswamy), 8:30 A.M., Room 1046. The Subcommittee will discuss the following and gather information for use by the ACRS in its preparation of the annual report to the Congress on the NRC Safety Research Program and budget for FY 1988: (1) proposed NRC Safety Research Program and budget for FY 1988, (2) preliminary OMB Mark and the impact of the OMB-proposed reductions on the continuing and proposed research contracts, and (3) RES responses to ACRS recommendations contained in its June 11, 1986 report to the Commission. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Siess
Dr. Carbon
Dr. Kerr
Dr. Mark
Mr. Michelson
Dr. Moeller
Dr. Okrent
Dr. Remick
Dr. Shewmon
Mr. Ward
Mr. Wylie

320th ACRS Meeting, December 11-13, 1986, Washington, DC, Room 1046.

Decay Heat Removal Systems, December 16, 1986, 1717 H Street, NW, Washington, DC (Boehnert), 8:30 A.M., Room 1046. The Subcommittee will continue its review of the NRR Resolution Position for USI A-45. Lodging will be announced later. Attendance by the following is anticipated:

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

Structural Engineering, January 21 and 22, 1987, Albuquerque, NM (Igne). The Subcommittee will review containment integrity and Category I structures, programs, and test facilities. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Siess	Dr. Okrent
Dr. Carbon	Dr. Shewmon
Mr. Ebersole	Mr. Bender
Dr. Kerr	Dr. Pickel

AC/DC Power Systems Reliability, Date to be determined (November), Washington, DC (El-Zeftawy). The Subcommittee will review the proposed Station Blackout rule (SECY-85-163). Attendance by the following is anticipated:

Dr. Kerr	Dr. Lewis
Mr. Ebersole	Mr. Wylie

Reactor Operations, Date to be determined (November), Washington, DC (Alderman). The Subcommittee will review recent operating events at nuclear reactors. Attendance by the following is anticipated:

Mr. Ebersole	Dr. Remick
Mr. Michelson	Mr. Wylie
Dr. Moeller	

Instrumentation and Control Systems, Date to be determined (November/December), Washington, DC (El-Zeftawy). The Subcommittee will discuss the effect of adverse conditions such as high temperature on solid-state components in nuclear power plants. Attendance by the following is anticipated:

Mr. Ebersole	Mr. Michelson
Dr. Kerr	Dr. Moeller
Dr. Lewis	Mr. Wylie

Naval Reactors, Date to be determined (December/January), Charleston, SC or Washington, DC (Boehnert). The Subcommittee will review the Naval Reactor Moored Training Ship Project. Attendance by the following is anticipated:

Dr. Kerr
Dr. Lewis
Dr. Remick
Mr. Ward

Metal Components, Date to be determined (January), Washington, DC (Igne). The Subcommittee will review: (1) hear a status report of the Whipjet program (application of broad scope GDC-4 criteria) as applied to lead plant BVPS-2; and (2) review public comments on NUREG-0313, Rev. 2 (long range fix for BWR-IGSCC problems) per ACRS letter. Attendance by the following is anticipated:

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

Seabrook Unit 1, Date to be determined (fall/winter), Washington, DC (Major). The Subcommittee will review the application for a full power operating license for Seabrook Unit 1. Attendance by the following is anticipated:

Dr. Kerr
Dr. Lewis
Dr. Moeller
Mr. Michelson

APPENDIX IV

NRC PRESENTATION ON
B&W PLANT REASSESSMENT PROGRAM

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: B&W PLANT REASSESSMENT PROGRAM

DATE: SEPTEMBER 12, 1986

PRESENTER: R. C. JONES

PRESENTER'S TITLE/BRANCH/DIVISION: NUCLEAR ENGINEER/RSB/DPWR-B

PRESENTER'S NRC TEL. NO.: 492-8004

NRC REASSESSMENT OF B&W PLANT DESIGNS

- o NRC ENCOURAGED B&W OWNERS GROUP (BWOG) TO TAKE THE LEADERSHIP ROLE IN THIS MATTER.
- o BWOG PROGRAM DOCUMENTED IN MAY 15, 1986 SUBMITTAL. SUBMITTAL UPDATED ON AUGUST 29.
- o MAJOR NRC BWOG WORKING MEETINGS
  - ICS/NNI - MAY 21, 1986/AUGUST 26, 1986
  - SENSITIVITY STUDY - JUNE 19, 1986
  - SYSTEMS REVIEW - AUGUST 19, 1986
  - RECOMMENDATION TRACKING SYSTEM - SEPTEMBER 29, 1986 (TENTATIVE)
- o MET WITH ACRS SUBCOMMITTEE ON BABCOCK AND WILCOX REACTOR PLANTS ON JUNE 25, 1986.

BWOG SAFETY AND PERFORMANCE IMPROVEMENT (SPI) PROGRAM

- OVERVIEW OF PROGRAM
  - OPERATING EXPERIENCE REVIEW
  - OPERATOR/MAINTENANCE INTERVIEWS
  - STUDY OF B&W PLANT SENSITIVITY (MPR ASSOCIATES)
  - RISK ASSESSMENT REVIEW
  - PROCEDURES REVIEW
  - SYSTEMS REVIEW
  - RECOMMENDATION TRACKING SYSTEM/IMPLEMENTATION PROCESS
- NRC ASSESSMENT
  - PROGRAM GENERALLY ON TARGET TO IMPROVE SAFETY
  - FURTHER DISCUSSION NEEDED ON
    - SCOPE OF MAIN FEEDWATER SYSTEM REVIEW
    - HUMAN FACTORS ISSUES
    - RECOMMENDATION TRACKING SYSTEM/IMPLEMENTATION
  - STAFF WILL CONTINUE TO MONITOR PROGRAM AND PROVIDE COMMENTS AS APPROPRIATE

NRC PROGRAM PLAN

- PRIMARY EFFORT IS REVIEW OF BWOG PROGRAM RESULTS
- STAFF WILL:
  - REVIEW TAP REPORTS/PREVIOUS NUREG REPORTS
  - UTILIZE FEEDBACK FROM REGIONAL PERSONNEL
  - REVIEW STATUS OF UTILITY COMPLIANCE TO NRC ACTIONS
  - WILL EXAMINE
    - DEMANDS ON OPERATING PERSONNEL/PROCEDURES
    - EXISTING B&W PRAs
    - THERMAL-HYDRAULIC RESPONSE
- INITIAL SER TO BE ISSUED DECEMBER
- PROGRAM COMPLETION - JUNE 1987

RESPONSE TO ACRS CONCERNS

- ACRS LETTER (JULY 16, 1986)

  - BWOG PROGRAM DIRECTED AT IMPROVING ON-LINE PERFORMANCE RATHER THAN SAFETY

  - PROGRAM CONCENTRATES ON SYSTEM DESIGN. SHOULD EXAMINE EFFECT OF OPERATING ORGANIZATION.

  - STAFF AND BWOG SHOULD DETERMINE WHETHER DIFFERENT RESPONSE CHARACTERISTICS OF B&W PLANT ARE SAFETY SIGNIFICANT

  - PROGRAM SHOULD ADDRESS DECAY HEAT REMOVAL RELIABILITY

- STAFF RESPONSE (AUGUST 14, 1986)

  - STAFF ENSURING BROADER PLANT SAFETY ISSUES ARE ADDRESSED VIA WORKING LEVEL MEETINGS WITH BWOG.

  - STAFF BELIEVES MANAGEMENT ISSUES MORE EFFECTIVELY ADDRESSED ON PLANT-SPECIFIC BASIS IN EXISTING PROGRAMS (E.G., SALP, INPO APPRAISAL PROGRAM).

  - UTILIZATION OF MPR STUDY ON SENSITIVITY AND PLANT PRA REVIEW WILL ALLOW STAFF TO JUDGE SAFETY SIGNIFICANCE OF B&W PLANT RESPONSE.

  - COMPOSITE RESULTS OF BWOG ACTIVITIES SHOULD ASSURE RELIABILITY OF DECAY HEAT REMOVAL. NEW ALTERNATE DECAY HEAT REMOVAL CONCEPTS ARE BEING EXAMINED UNDER USI A-45.

- STAFF REQUIREMENTS MEMORANDUM (AUGUST 8, 1986)

  - STAFF IS TO PROVIDE THE COMMISSION WITH RECOMMENDATIONS ON THE GENERIC APPLICATIONS OF TOLEDO EDISON'S "DECAY HEAT REMOVAL RELIABILITY IMPROVEMENT PROGRAM" IMPLEMENTED AT DAVIS-BESSE (SUSPENSE DATE: FEBRUARY 27, 1987).

PRESENTATION BY THE
B&W OWNERS GROUP
ON THE SUBJECT OF THE
SAFETY AND PERFORMANCE IMPROVEMENT PROGRAM
TO THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SEPTEMBER 12, 1986

AGENDA

I	EXECUTIVE OPENING REMARKS	H.B. TUCKER
II	GENERAL INTRODUCTION	G.R. SKILLMAN
III	DATA BASE/TAP INFO & LESSONS	R.T. GLAVIANO
IV	SENSITIVITY OF BWOG PLANTS	G.R. SKILLMAN
V	DECAY HEAT REMOVAL	E.W. SWANSON
VI	SUMMARY	G.R. SKILLMAN
VII	EXECUTIVE CLOSING REMARKS	H.B. TUCKER

PREVIOUS ACRS ACTIVITY

- MET WITH PWR-B SUBCOMMITTEE ON JUNE 25, 1986
- GAVE SEVERAL-HOUR ADMINISTRATIVE PRESENTATION ON THE (THEN) <u>STOP</u>-TRIP PROGRAM (NOW SPIP. THE NAME WAS CHANGED TO REFFLECT ACRS SUBCOMMITTEE GUIDANCE ON THE PROGRAM)
- FAILED TO ESTABLISH RECOGNITION OF THE SAFETY-ORIENTATION OF THE PROGRAM
- THE PROGRAM WAS PERCEIVED TO BE PRIMARILY DIRECTED AT IMPROVING B&WOG PLANT ON-LINE PERFORMANCE RATHER THAN ADDRESSING THE SAFETY OBJECTIVES OF THE NRC/B&W PLANT REASSESSMENT INITIATIVE.
- ACRS LETTER TO V. STELLO OF 7/16/86 OFFERED THREE OBSERVATIONS/RECOMMENDATIONS
  - SOME PLANTS OPERATE BETTER THAN OTHERS; IS IT MORE THAN JUST SYSTEM DESIGN? (I.E., IS IT OPERATING ORGANIZATION?)
  - THE B&W PLANTS RESPOND DIFFERENTLY, AND, DUE IN LARGE PART TO THE INHERENT B&W DESIGN, ARE MORE QUICKLY RESPONSIVE TO LOAD CHANGES AND OTHER CHALLENGES THAN OTHER PWRs.
  - APPARENTLY LITTLE ATTENTION IS BEING GIVEN TO DECAY HEAT REMOVAL.

- WE CONSIDER THE ACRS OBSERVATIONS TO BE HELPFUL. WE CATEGORIZE THEM AS FOUR BASIC ISSUES WHICH ARE:

  - DATA BASE LESSONS
  - PROGRAM SAFETY ORIENTATION
  - SAFETY SIGNIFICANCE OF BWOG/OTSG PLANT SENSITIVITY
  - ENERGY PRODUCTION/REMOVAL IMBALANCE -- DECAY HEAT REMOVAL

- WE WILL ADDRESS AND DISCUSS EACH IN DETAIL IN THIS PRESENTATION -- AND USE THEM AS THE FRAMEWORK FOR OUR PRESENTATION.

PROGRAM SAFETY ORIENTATION

- THE SPI PROGRAM CONSISTS OF A FULL REASSESSMENT OF FUNDAMENTAL SAFETY AND OPERATING ISSUES AT B&W PLANTS. IT INCLUDES THIRTEEN MAJOR TASKS WHICH COVER THE FOLLOWING:

  - RE-REVIEW OF SIX YEARS (1980 - 1985 INCLUSIVE) OF B&W PLANT OWNERS' OPERATING EXPERIENCE OF OVER 200 TRIPS AND TRANSIENTS PLUS SELECTED PRE-1980 TRANSIENTS. THE TMI-2 EXPERIENCE WAS FACTORED IN OBLIQUELY DUE TO ITS INCLUSION IN ESSENTIALLY ALL POST-79 RULEMAKING.

  - AN INDEPENDENT SENSITIVITY STUDY OF BASIC THERMAL-HYDRAULIC PLANT CHARACTERISTICS COMPARING B&W PLANTS WITH OTHER PWRs.

  - DETAILED REVIEW OF SELECTED KEY SYSTEMS

  - AN ASSESSMENT OF NEARNESS-TO-CORE-MELT RISKS ASSOCIATED WITH SELECTED OPERATING TRANSIENTS.

PROGRAM SAFETY ORIENTATION (CONTINUED)

- THE PRIMARY GOAL OF THE SAFETY AND PERFORMANCE IMPROVEMENT PROGRAM (SPI PROGRAM) IS TO MAKE THE B&WOG PLANTS SAFER.

- THE PLURALITY AND OVERLAP OF SAFETY LEVEL INCREASE, RISK REDUCTION AND RELIABILITY INCREASE ARE RECOGNIZED; INCREASE IN THE LEVEL OF SAFETY IS THE DOMINANT ISSUE IN THIS CONTEXT.

- REVIEW AND ACTION TO PREVENT REOCCURRENCE OF INCIDENTS LIKE THE DAVIS-BESSE AND RANCHO SECO INCIDENTS IS INCLUDED AMONG THE MAJOR THEMES OF THE PROGRAM.

- THE PROGRAM IS RESPONSIVE TO MR. V. STELLO'S 1/24/86 LETTER REQUESTING A DESIGN REASSESSMENT. BWOG INTERPRETATION OF THAT LETTER, AS TRANSLATED INTO SPI PROGRAM SCOPE, IS APPROPRIATE PER NRC STAFF INFORMATION. BWOG AND NRC STAFF ARE COMMUNICATING WITH EACH OTHER FREQUENTLY ON THE COMPOSITION AND PROGRESS ON THE REASSESSMENT.

- THE PROGRAM IS BEING EXECUTED BY BWOG (UTILITY AND B&W) PERSONNEL. B&W IS PART OF THE BWOG.

PROGRAM SAFETY ORIENTATION (CONTINUED)

- THE SPI PROGRAM HAS SOUGHT INDEPENDENT COUNSEL REGARDING PROGRAM DIRECTION AND DEPTH. GUIDANCE HAS BEEN ACCEPTED FROM:

  MPR ASSOCIATES
  INDEPENDENT ADVISORY BOARD
  OTHERS

- THE SPI PROGRAM SCOPE HAS EXPANDED CONSIDERABLY SINCE ITS INITIATION; IT MAY CONTINUE TO EXPAND AS THE ACTIVITIES IN THE PROGRAM DEMONSTRATE THE NEED FOR FURTHER REVIEW AND GREATER THOROUGHNESS. EXAMPLES OF CHANGE SINCE THE PROGRAM WAS INITIATED INCLUDE:

  - REORIENTATION OF SOME TASKS (I.E., FOCUS OF INSTRUMENT AIR SYSTEM REVIEW TOWARDS DECAY HEAT REMOVAL)

  - TRIP INITIATORS REVIEW

- THE ISSUE OF HEAT PRODUCTION/REMOVAL IMBALANCE AND RELIABLE DECAY HEAT REMOVAL CAPABILITY IS RECOGNIZED AND ACCEPTED AS A DOMINANT ISSUE. A NEED FOR ADDITIONAL DECAY HEAT REMOVAL CAPABILITY IS CONSIDERED UNNECESSARY.

<u>DATA BASE LESSONS</u>
ZEROING IN ON THE OPERATING EXPERIENCE REVIEW

- WE HAVE AVAILABLE $\approx$220 TRANSIENT ASSESSMENT/ PERFORMANCE REPORTS.

- TRANSIENTS HAVE BEEN REVIEWED IN DETAIL BY OUR TRANSIENT ASSESSMENT COMMITTEE.

- THE CRITERIA BY WHICH ONE DETERMINES TRANSIENT CLASSIFICATION IS A <u>TOOL</u> DEVELOPED BY THE BWOG, BASED PRINCIPALLY ON ATOG GUIDELINES FOR USE IN MONITORING BEHAVIOR. IT HAS BECOME A HANDY TOOL FOR CATEGORIZING SEVERITY. IT IS <u>NOT</u> AN ABSOLUTE "SAFETY" INDICATOR.

- THE CATEGORY C'S (10) PROVIDE SEVERAL VERY IMPORTANT CONCLUSIONS RELEVANT TO PLANT SAFETY. THE CONCLUSIONS WILL BE USED IN THIS PRESENTATION TO PROVIDE A LOGICAL ORDER FOR DESCRIBING THE SPIP TASKS.

B&W OWNERS GROUP
SAFETY AND PERFORMANCE IMPROVEMENT PROGRAM
REVIEW OF CATEGORY 'C' EVENTS
(PRELIMINARY RESULTS)

SEPTEMBER 12, 1986

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PRESENTATION

- EVENT REVIEW PROCESS
- SUMMARY OF FINDINGS
- CONCLUSIONS
- RECOMMENDATIONS

EVENT REVIEW PROCESS

- DEVELOP CLASSIFICATION METHOD TO FACILITATE REVIEW OF COMPLEX TRANSIENTS

  METHODOLOGY IS BASED ON THE ABNORMAL TRANSIENT OPERATING GUIDELINES (ATOG) STABILITY FUNCTIONS. THREE LEVELS (A,B,C) ARE DEFINED FOR EACH FUNCTION TO JUDGE THE RELATIVE COMPLEXITY OF TRANSIENTS AT B&WOG PLANTS

- SEARCH TAP DATA BASE AND CATEGORIZE EACH EVENT

- REVIEW SIMILAR EVENTS, WITH EMPAHSIS ON COMPLEX TRANSIENTS

EVENT REVIEW PROCESS

TRANSIENT CLASSIFICATION

TRANSIENT RESPONSE CATEGORIES DEFINED

CATEGORY	DEFINITION
A	PLANT RESPONSE WITHIN PREFERRED BOUND
B	PREFERRED OR EXPECTED RANGE EXCEEDED. THESE ARE OF OF CONCERN SINCE THESE MAY BE PRECURSOR EVENTS.
C.	ABNORMAL RESPONSE INDICATIVE OF A COMPLEX TRANSIENT. PLANT CONDITIONS REACH LIMITS WHICH MAY REQUIRE EXTENSIVE SAFETY SYSTEM & OPERATOR RESPONSE TO MITIGATE.

NOTE:

- o CLASSIFICATION METHOD IS VALID FOR JUDGING THE RELATIVE COMPLEXITY OF TRANSIENTS AT B&WOG PLANTS, BUT IS NOT AN ABSOLUTE MEASURE OF SAFETY
- o THIS CLASSIFICATION METHOD IS UNIQUE TO THE B&W PLANTS. TO OUR KNOWLEDGE NONE OF THE OTHER LIGHT WATER REACTOR MANUFACTURERS, OR THEIR OWNERS, HAVE A TRANSIENT CLASSIFICATION SYSTEMS

TRANSIENT EVENT PATH

THERMAL & FLUID EFFECTS

INADEQUATE PRI → SEC HEAT TRANSFER → RCS HEATUP → RCS FLUID EXPANSION → PZR INSURGE → PZR STM. BUBBLE COMPRESSION

EXCESSIVE PRI → SEC HEAT TRANSFER → RCS COOLDOWN → RCS FLUID CONTRACTION → PZR OUTSURGE → PZR STM. BUBBLE EXPANSION & LIQUID FLASHING

ATOG STABILITY PARAMETER

STM. GEN. PRESS & INVENTORY → RCS TEMP → RCS INVENTORY → RCS PRESS

CONTRIBUTING OR MITIGATING FACTORS

STM. GEN. PRESS & INVENTORY	RCS TEMP	RCS INVENTORY	RCS PRESS
$\dot{M}_{STEAM}$ $\dot{M}_{FEED}$	$\dot{Q}_{D.H.}$ $\dot{Q}_{RCPS}$ $\dot{Q}_{HPI}$	$\dot{M}_{MU/LD}$ $\dot{M}_{HPI}$	$\dot{Q}_{PORV/SV}$ $\dot{Q}_{SPRAY}$ $\dot{Q}_{HEATERS}$

SUMMARY OF FINDINGS

- 0 CATEGORIZATION OF 1980 - 1985 EVENTS
- 0 TRANSIENT DISTRIBUTION
- 0 CAUSES & CONTRIBUTING FACTORS
- 0 DATA SUMMARY

SUMMARY OF FINDINGS

CATEGORIZATION OF 1980 - 1985 EVENTS

CATEGORY	NUMBER	PERCENTAGE
A	78	35%
B	134	60%
C	10	5%

TRANSIENT DISTRIBUTION

PLANT	CAT. C	SIG CAT. B	TOTAL
ANO-1	1	6	7
CR-3	3	3	6
DB	2	8	10
OC-1	0	8	8
OC-2	0	3	3
OC-3	0	2	2
RS	4	5	9
	10	35	45

*TMI-1 SHUTDOWN DURING THIS PERIOD

CATEGORY 'C' EVENTS
AT THE B&WOG OPERATING UNITS

UNIT	1980	1981	1982	1983	1984	1985
ANO-1	LOOP	---	---	---	---	---
CR-3	ICS/NNI PWR	LOOP	---	---	---	EFIC ACT POWER
D-B	---	---	---	---	SFRCS ACT [MSSV]	FWP TRIP SFRCS ACT
OC.1	---	---	---	---	---	---
OC.2	---	---	---	---	---	---
OC.3	---	---	---	---	---	---
RS	---	TBV/ADV	---	---	NNI POWER	(1) FW HTR RV (2) ICS/NNI PWR
TMI-1	---	---	---	---	---	---
TOTAL	2	2	0	0	2	4
# TRIPS	35	47	42	44	20	34
FREQUENCY % Category C trips	6%	4%	0%	0%	10%	12%

SUMMARY OF FINDINGS

COMPLEX TRANSIENT DATA SUMMARY

- O FREQUENCY OF OCCURRENCE IS 5%
- O TREND SHOWS 3 FOLD INCREASE FOR '84-'85 OVER '80-'83 EXPERIENCE
- O CAUSES
  - o EXCESSIVE HEAT REMOVAL - OVERSTEAM & OVERFEED BY EFW FOLLOWING
    - LOSS OF OFFSITE POWER
    - ICS/NNI POWER LOSS
    - TBV/ADV/RV FAILED OPEN
    - SPURIOUS EFIC/SFRCS ACTUATION
  - o INADEQUATE HEAT REMOVAL - LOSS OF STEAM GENERATOR(s) AS HEAT SINK
    - SPURIOUS/IMPROPER SFRCS/EFIC ACTUATION
- O SECONDARY PLANT RESPONSE IS REFLECTED INTO THE PRIMARY PLANT

PRELIMINARY CATEGORY 'C' REVIEW CONCLUSIONS

BASED ON A REVIEW OF CATEGORY 'C' TRANSIENTS

1. EVENTS ARE GENERALLY THE RESULT OF EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

   O STEAM FLOW THROUGH FAILED OPEN VALVES

   O FEED FLOW TO MAINTAIN SG LEVEL

2. CATEGORY 'C' TRANSIENTS REQUIRE ADDITIONAL OPERATOR ATTENTION TO ACHIEVE PLANT CONTROL

   O BALANCE ENERGY PRODUCTION AND REMOVAL

   O CONTROL OF RCS INVENTORY & PRESSURE DUE TO FEEDBACK FROM THE SECONDARY PLANT

3. KEY TO MITIGATION IS ENSURING CONTROL OF STEAM FLOW & FEED FLOW FROM THE CONTROL ROOM

   O BALANCING ENERGY TERMS REDUCES FEEDBACK INTO THE RCS

   O MAINTAIN PLANT CONTROL WHEN ICS/NNI POWER IS LOST

4. CORRECTIVE ACTIONS REQUIRED IN THE FOLLOWING AREAS

   O PLANT DESIGN
   O PLANT MAINTENANCE
   O HUMAN INTERFACE
   O LEARN FROM OPERATING EXPERIENCE

PRELIMINARY CORRECTIVE ACTIONS AREAS

BASED ON A REVIEW OF CATEGORY 'C' TRANSIENTS

1. PLANT DESIGN

   - O ENHANCE CONTROL OF STEAM FLOW & FEED FLOW FROM THE CONTROL ROOM UNDER POST-TRIP CONDITIONS

   - O REDUCE SFRCS & EFIC ACTUATIONS AND IMPROVE RESPONSE ONCE ACTUATED

2. PLANT MAINTENANCE

   - O IMPROVE MAINTENANCE PROGRAM WITH SPECIAL EMPHASIS ON STEAM FLOW & FEED FLOW COMPONENTS, CONTROLS AND THEIR MOTIVE POWER

3. HUMAN INTERFACE

   - O IDENTIFICATION OF OPERABLE CONTROLS & INSTRUMENTATION ON LOSS OF ICS/NNI POWER

   - O VERIFICATION OF ATOG STABILITY PARAMETERS

   - O CONTROL OF PLANT PER ATOG GUIDANCE

4. OPERATING EXPERIENCE

   - O IMPROVE PROCESS TO IDENTIFY & ELIMINATE RECURRING PROBLEMS AT EACH PLANT

   - O IMPROVE PROCESS TO LEARN FROM B&WOG COLLECTIVE EXPERIENCE

CATEGORY 'C' EVENT REVIEW

CATEGORY 'C' EVENT REVIEW IS ONE ELEMENT OF THE SPI PROGRAM

DATA ENCOMPASSES 6 FULL YEARS OF B&WOG OPERATING EXPERIENCE

REVIEW CONCLUDES 4 MAIN ITEMS:

- O EVENTS ARE GENERALLY THE RESULT OF EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER
- O CATEGORY 'C' TRANSIENTS REQUIRE ADDITIONAL OPERATOR ATTENTION TO ACHIEVE PLANT CONTROL
- O KEY TO MITIGATION IS ENSURING CONTROL OF FEED FLOW & STEAM FLOW FROM THE CONTROL ROOM
- O CORRECTIVE ACTION IS REQUIRED IN THE FOLLOWING AREAS:
  - PLANT DESIGN
  - PLANT MAINTENANCE
  - HUMAN INTERFACE
  - LEARN FROM OPERATING EXPERIENCE

CATEGORY C CONCLUSIONS AND THEIR RELATIONSHIP TO THE SPI PROGRAM TASKS

THE SPIP PROGRAM BUILDS ON KNOWN AND UNDERSTOOD OPERATING EXPERIENCE. FOCUS ON THE CATEGORY C CONCLUSION:

- O EVENTS ARE GENERALLY THE RESULT OF EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER
  - 1154 EFFORT - MAIN FEEDWATER SYSTEM REVIEW
  - EFW/AFW REVIEW
  - ICS/NNI SYSTEM REVIEW
  - INSTRUMENT AIR REVIEW
  - SECONDARY PLANT RELAY SYSTEM REVIEW

- O CATEGORY C TRANSIENTS REQUIRE ADDITIONAL OPERATOR ATTENTION TO ACHIEVE PLANT CONTROL
  - SYSTEMS REVIEW (ABOVE)
  - OPERATIONS/MAINTENANCE PERSONNEL INTERVIEWS
  - RANCHO SECO - NUREG 1195 SYSTEMS

- O KEY TO MITIGATION IS ENSURING CONTROL OF FEED FLOW AND STEAM FLOW FROM THE CONTROL ROOM
  - SAME AS ABOVE

- O CORRECTIVE ACTION IS REQUIRED IN PLANT DESIGN, PLANT MAINTENANCE, HUMAN INTERFACE, LESSONS LEARNED
  - OPERATIONS/MAINTENANCE PERSONNEL INTERVIEWS
  - OPERATING EXPERIENCE REVIEW
  - PAST RECOMMENDATIONS/INDUSTRY REVIEW

B&W OWNERS GROUP

SAFETY & PERFORMANCE IMPROVEMENT PROGRAM

1. SENSITIVITY STUDY
2. ICS/NNI EVALUATION
3. OPERATING EXPERIENCE REVIEW (PAST 6 YRS)
4. EFW SYSTEM REVIEW
5. MFW SYSTEM REVIEW
6. SEC. PLANT RELIEF SYSTEM REVIEW
7. INSTRUMENT AIR SYSTEM REVIEW
8. RISK ASSESSMENT
9. EMERGENCY OPERATING PROCEDURES REVIEW
10. OPS/MAINT PERSONNEL INTERVIEWS
11. REVIEW OF PAST RECOMMENDATIONS
12. DAVIS BESSE TASK FORCE (NUREG-1154)

    E.G. - NEW ROOT CAUSE PROCESS
    - MOV WORKSHOPS
    - MFW RELIABILITY
    - STEAM TRAP REVIEW

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13. RANCHO SECO RRG ACTIONS (NUREG-1195)

B&WOG/OTSG PLANT
SENSITIVITY STUDY - OVERVIEW

- THE STUDY'S OBJECTIVE IS TO QUANTIFY THE RELATIVE SENSITIVITY OF B&W NSSS PLANTS WITH RESPECT TO OTHER PWR DESIGNS

  - PROVIDES A LOGICAL, SYSTEMATIC BASIS FOR COMPARISON
  - QUANTIFIES "SENSITIVITY" IN MEASURABLE TERMS
  - CORRELATES WITH ESTABLISHED SAFETY LIMITS AND PERFORMANCE REQUIREMENTS
  - IS BROAD IN SCOPE TO FULLY ASSESS DIFFERENCES

- MPR IS PERFORMING CALCULATIONS AND ANALYSES OF THE THERMODYNAMIC RESPONSE OF THE THREE NSSS PWR DESIGNS TO UPSETS IN FW FLOW, STEAM FLOW, AND REACTIVITY

  - USES SPECIFIC PLANTS, CHOSEN TO REFLECT DIFFERENCES WITHIN VENDOR LINES
  - IMPORTANT B&W PLANT DIFFERENCES ARE BOUNDED
  - ASSESS THE RESPONSE OF CONTROL AND PROTECTIVE SYSTEMS TO UPSETS, FAILURES
  - ADDRESSES FAILURES REFLECTED IN OPERATING EXPERIENCE
  - LOOKS AT THE NEED FOR OPERATOR ACTION AND CONSEQUENCES OF FAILURE TO PERFORM
  - LOOKS AT PLANTS AT POWER

QUANTIFICATION OF SENSITIVITY

INDICES	SAFETY PARAMETERS	OPERATIONAL LIMITS/CRITERIA
	o SECONDARY DESIGN PRESSURE	o STEAM LINE FLOODING
		o SG OVERFILL
o MARGIN	o PRIMARY DESIGN PRESSURE, TEMPERATURE	o SG DRYOUT
		o SAFETY VALVE CHALLENGE
	o SATURATION MARGIN	o PORV CHALLENGES
o TIME		
	o Kw/FT LIMIT	o STEAM/FEED ISOLATION
		o LOSS OF PRESSURIZER LEVEL
o FREQUENCY	o MINIMUM DNBR	o RPS TRIP LIMITS
	o PTS LIMITS	o SAFETY INJECTION LIMITS
		o HEATUP/COOLDOWN RATE LIMITS

ANALYSIS MATRIX (PARTIAL LISTING)
(SPECIFIC ANALYTICAL ACTIVITIES)
SIGNIFICANT DISTURBANCES, WITH CORRECTIVE ACTION

A. TURBINE TRIP
- WITH REACTOR RUNBACK
- WITH REACTOR TRIP
- WITH STEAM FLOW UPSET

B. LOSS OF ONE FEEDWATER PUMP
- WITH SLOW RUNBACK
- WITH FAST RUNBACK

C. LOSS OF ALL FEEDWATER PUMPS
- NORMAL EFW
- DELAY EFW
- NO EFW
- EXCESSIVE EFW

D. CONTROL SYSTEM UPSETS
- VARYING FEED SYSTEM CONFIGURATIONS AND INITIAL CONDITIONS

E. LOSS OF COOLANT FLOW
- LOSS OF ONE PUMP
- LOSS OF ALL PUMPS

- THESE CALCULATIONS WILL BE REVIEWED TO DETERMINE IF, WHEN, AND HOW OFTEN THESE OPERATIONAL LIMITS ARE REACHED.

- AS A RESULT, THE RELATIVE SENSITIVITY OF THE THREE DESIGNS WILL BE QUANTIFIED.

B&WOG SENSITIVITY STUDY - SUMMARY

- THE STUDY WILL CLEARLY IDENTIFY WHERE THE B&W PLANT RESPONSE IS DIFFERENT, BOTH POSITIVELY AND NEGATIVELY, FROM THE OTHER DESIGN.

  - WHERE THERE IS DIFFERENCE IN MARGIN OR DIFFERENCE IN TIME TO RESPOND.

- THE STUDY WILL PROVIDE AN ENGINEERING BASIS FROM WHICH TO ASSESS THE SAFETY SIGNIFICANCE OF THESE DIFFERENCES.

  - SAFETY SIGNIFICANCE IN TERMS OF ESTABLISHED LIMITS

- THE STUDY WILL PRODUCE RECOMMENDATIONS FOR CHANGES TO CORRECT AREAS WITH ADVERSE DIFFERENCES, TO BE EVALUATED IN LIGHT OF THEIR SIGNIFICANCE.

DECAY HEAT REMOVAL
B&W PLANTS
AN OVERVIEW

A-46

PURPOSE OF PRESENTATION

- PROVIDE A GENERAL OVERVIEW OF DHR ON B&W PLANTS

- SHOW THE SPIP RELATIONSHIP TO DHR

DECAY HEAT REMOVAL REQUIRED

- o AFTER SHUTDOWN FROM POWER
  - o NORMAL
  - o ABNORMAL

- o DURING REFUELING

IN A BROAD SENSE:

- o DECAY HEAT REMOVAL IS ACCOMPLISHED IN B&W PLANTS AS IN ANY PWR

- o DEFENSE IN DEPTH IS PROVIDED IN B&W PLANTS TO THE SAME DEGREE OR MORE THAN IN OTHER PWRs

- o PRA RISK EVALUATIONS INDICATE THAT THE CORE DAMAGE FREQUENCY FOR INTERNAL EVENTS IS THE SAME FOR B&W AS FOR OTHER PWRs

DECAY HEAT REMOVAL

USING STEAM GENERATOR		WITHOUT STEAM GENERATOR
FIRST LINE OF DEFENSE	SECOND LINE OF DEFENSE	THIRD LINE OF DEFENSE
MAIN FEEDWATER	EMERGENCY FEEDWATER	FEED AND BLEED B&W PLANTS - GENERAL: HPI AND PORV OR PRESSURIZER SAFETY DAVIS-BESSE: MAKEUP AND PORV
CONDENSATE		
STEAM RELIEF • MSSV • TBV/ADV	STEAM RELIEF • MSSV	
ICS/NNI CONTROLS	EFIC, SFRCS OR EQUIVALENT CONTROLS	
RC PUMPS (FORCED FLOW)	• RC PUMPS OR • NATURAL CIRCULATION	

BOXES INDICATE ITEMS INCLUDED IN SPIP

SPIP REVIEW OF SYSTEMS IMPORTANT TO DECAY HEAT REMOVAL

COVERS ALL ASPECTS

- o DESIGN
- o OPERATIONS
- o TESTING/INSPECTIONS
- o MAINTENANCE

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OBJECTIVE - INCREASE LEVEL OF OVERALL PLANT SAFETY BY:

- o BETTER SYSTEM AND COMPONENT RELIABILITY
- o BETTER OPERATION FOR NORMAL AND ABNORMAL CIRCUMSTANCES
- o REDUCED PLANT/SYSTEM TRIP FREQUENCY

DECAY HEAT REMOVAL - FIRST LINE OF DEFENSE

MAIN FEEDWATER/ICS CONTROLS

BACKGROUND INFO OF INTEREST

TRANSIENT (1983 & PRIOR DATA)

o	LOFW	0.9/YR
o	EXCESS F/W	0.2/YR
o	ICS POWER	5×10^{-2}/YR

EQUIPMENT PERFORMANCE (APPROXIMATE)

o	OTHER PWR MFP TURBINE	3.3 PUMP TRIPS/YR
o	B&W MFP TURBINE	2.3 PUMP TRIPS/YR
o	NNI SENSOR	0.6 Rx TRIPS/YR

SPIP REVIEW OBJECTIVES

- IMPROVE SYSTEM AND MFW PUMP RELIABILITY
- KEEP MFW ON LINE
- REDUCE EFW CHALLENGES
- SMOOTH POST TRIP RESPONSE

OBSERVATIONS TO DATE

- MFW/ICS REQUIRE DILIGENCE TO MAINTAIN TUNING AND CONTROL

  o SPEED CONTROL AND RESPONSE

- UNNEEDED MFW PUMP PROTECTION TRIPS

  o LOW SUCTION PRESSURE

- ICS SENSOR INPUT FAILURES

- NNI POWER SUPPLY MIDSCALE FAILURES (AFFECTS CONTROL COMPONENT POSITIONS)

DECAY HEAT REMOVAL - SECOND LINE OF DEFENSE

EMERGENCY FEEDWATER/CONTROLS

BACKGROUND INFO OF INTEREST

- o ALL B&W PLANTS - INITIATION AND CONTROL TO BE INDEPENDENT OF ICS AND ICS POWER SUPPLIES

- o SPECIFIC PLANT ACTIONS UNDERWAY, E.G.
  - o RANCHO SECO - EFIC INSTALLATION
  - o DAVIS-BESSE - SFRCS MODS
    - MOTOR DRIVEN EFW PUMP
    - TURBINE DRIVEN PUMP STEAM
  - o RESPONSE TO SSFI REVIEWS OF EFW
  - o MOV PROGRAMS

SPIP - REVIEW GOALS

- o REDUCE SFRCS AND EFIC INITIATIONS AND IMPROVE PERFORMANCE ONCE INITIATED.
- o IMPROVE RELIABILITY FOR DECAY HEAT REMOVAL

OBSERVATIONS TO DATE

- o TURBINE DRIVEN EFW PUMP RELIABILITY
  - INCREASE START TIME
- o MORE CONSISTENT TESTING MAY BE DESIRABLE
- o MAINTENANCE PRACTICES ARE IMPORTANT
- o UNNECESSARY INITIATIONS

DECAY HEAT REMOVAL - FIRST/SECOND LINE OF DEFENSE

STEAM PRESSURE AND RELEASE CONTROL

BACKGROUND INFORMATION OF INTEREST

- o SAFETY VALVE FAILURES - STUCK OPEN
  ~2×10^{-3} (95% CONFIDENCE)

- o OPERATOR ACTIONS TO SOLIDLY RESEAT MSSVs
  ARE NEEDED

- o FAILURES OF ICS/NNI POWER CAUSING OPEN BYPASS
  CONTROL VALVE FAILURES
  - o ONE PLANT - BEING CORRECTED

SPIP PROGRAM REVIEW GOALS

- o REDUCE CHALLENGES TO SAFETIES

- o SMOOTHER PRESSURE CONTROL
  - PRESSURIZER LEVEL
  - MFP/EFWP RESPONSE

- o REDUCE OPERATOR ACTIONS

OBSERVATIONS TO DATE

- o SAFETY VALVE PERFORMANCE IS GOOD BUT NOT
  OPTIMAL
  - EARLY LIFTS
  - BLOWDOWN GREATER THAN ORIGINAL DESIGN

- o STEAM PRESSURE CONTROL UNEVENESS
  - INTERACTION BETWEEN TBVs AND MSSVs
  - EFW

SUMMARY

- o THREE LEVELS OF DEFENSE FOR DECAY HEAT REMOVAL ARE PROVIDED FOR THE SPECTRUM OF NORMAL AND ABNORMAL CONDITIONS
- o THE OTSG CHARACTERISTICS DEMAND CAREFUL CONTROL OF NORMAL SYSTEMS
- o NORMAL SYSTEMS PROVIDE THE PREFERRED METHOD FOR HEAT REMOVAL (FIRST LINE OF DEFENSE) AND ARE BEING ADDRESSED
  - -- MAIN FEEDWATER
  - -- STEAM
  - -- CONTROLS
- o IMPROVEMENTS IN EMERGENCY FEEDWATER AND CONTROLS ARE BEING ADDRESSED (THE SECOND LINE OF DEFENSE)
- o FEED AND BLEED COOLING (THIRD LINE OF DEFENSE) IS NOW MORE THAN ADEQUATE AND REQUIRES NO FURTHER ACTION*

*DAVIS-BESSE MAY ELECT ADDITIONAL CHANGES

CLOSING COMMENTS

- EXPECT TO DEVELOP APPROXIMATELY 500 SAFETY AND PERFORMANCE IMPROVEMENT RECOMMENDATIONS FROM 13 MAIN TASK EFFORTS BY 2Q 87.
- INTEND TO DISPOSITION ALL RECOMMENDATIONS FORMALLY
- IMPLEMENTATION OF SEVERAL RECOMMENDATIONS HAS ALREADY RESULTED IN BENEFIT (TRIP AVOIDANCE).
- EXPECT THAT ISSUE/CONCERN DISTILLATION AND PRIORITIZATION ACTIVITIES WILL IDENTIFY AREAS MOST SAFETY-SIGNIFICANT-APPROPRIATE FOR CHANGE, AND FOR RECOMMENDATIONS AFFECTING THOSE AREAS TO BE ACTED UPON FIRST.
- EXPECT THE IMPLEMENTATION PHASE TO BE CONTINUING FOR SEVERAL YEARS.

ACRS OBSERVATIONS
AND
RECOMMENDATIONS

1. THE PROGRAM APPEARS TO CONCENTRATE ENTIRELY ON DESIGN; IT SHOULD INCLUDE THE EFFECT OF THE OPERATING ORGANIZATION ON SYSTEM PERFORMANCE.

2. THE B&W SYSTEMS WERE DESIGNED TO RESPOND DIFFERENTLY. THE PROGRAM SHOULD DETERMINE WHETHER THESE CHARACTERISTICS ARE GOOD, BAD, OR INDIFFERENT FROM A SAFETY PERSPECTIVE.

3. TOO LITTLE ATTENTION APPEARS TO BE GIVEN TO DHR. IF DHR IS ADEQUATE PUBLIC SAFETY IS ENSURED.

APPENDIX VII

NRC STAFF PRESENTATION ON
GENERIC ISSUE NO. 124

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: AUXILIARY FEEDWATER SYSTEM RELIABILITY, GENERIC ISSUE NO. 124

DATE: SEPTEMBER 12, 1986

PRESENTER: SAMMY S. DIAB

PRESENTER'S TITLE/BRANCH/DIV: TASK MANAGER, REACTOR SAFETY ISSUES BRANCH/DIVISION OF SAFETY REVIEW & OVERSIGHT

PRESENTER'S NRC TEL. NO.: 492-4083

FULL COMMITTEE:

SEPTEMBER 12, 1986

PRESENTATION TO THE ACRS

AUXILIARY FEEDWATER SYSTEM (AFWS) RELIABILITY

GENERIC ISSUE NO. 124

- THIS IS A STATUS PROGRESS REPORT
- MODIFIED RESOLUTION APPROACH
- MODIFIED RESOLUTION PLAN
- PROGRAM
- SCHEDULE FOR RESOLUTION

THE MODIFIED RESOLUTION APPROACH

- SHORT TERM CONCENTRATED RELIABILITY REVIEW FOLLOWED BY AN AFWS REVIEW FINDINGS REPORT FOR EACH OF THE SEVEN PLANTS
- REVIEW CARRIED OUT BY ONRR MULTIDESCIPLINE AFWS REVIEW TEAM
- REVIEW EFFORT WILL BENEFIT FROM
  - RANCHO SECO RESTART EFFORT
  - B&W OG DESIGN REASSESSMENT EFFORT
  - THE OIE SSFI PROGRAM
  - THE LICENSEE'S AFWS RELIABILITY ANALYSES
  - RELEVANT INDUSTRY EFFORTS

WHY NOT STICK WITH "OLD" PROPOSED RESOLUTION?

- NEGATIVE COMMENTS FROM NRR REVIEW OF BACKFIT ANALYSIS

- LENGTHY PROCESS (ISSUANCE OF GL, LICENSEES TO CONDUCT AFWS RAs, STAFF REVIEW OF RAs FOLLOWED BY ISSUANCE OF SERs)

MODIFIED RESOLUTION PLAN

- TWO TIER APPROACH

  - THE SEVEN PLANTS:

    LOW ESTIMATED PRE-TMI AFWS RELIABILITIES.

    TWO PUMP AFWSs.

  - THE REST OF PWRs:

    ACCEPTABLE AFWS RELIABILITIES.

    MOSTLY THREE PUMP AFWSs.

- OUTCOME OF THE SEVEN PLANT REVIEW WILL DICTATE ACTION REGARDING REST OF PWRs.

THE SEVEN PLANTS

ANO-1

* SSFI CONDUCTED BY OIE
* B&W OG FOR DESIGN REASSESSEMENT
* AFWS REVIEW TEAM

RANCHO SECO

* RESTART EFFORT:

  EXTENSIVE MULTIDECIPLINE STAFF REVIEW OF AFWS

  - DESIGN MODS
  - PROCEDURES & TRAINING
  - SUPPORT SYSTEMS
  - INDICATION & CONTROL
* B&W OG FOR DESIGN REASSESSEMENT
* AWFS REVIEW TEAM

CRYSTAL RIVER

- B&W OG FOR DESIGN REASSESSMENT
- LICENSEE'S AFWS RELIABILITY ANALYSIS
- AFWS REVIEW TEAM

PRAIRIE ISLAND 1&2

- LICENSEE'S AFWS RELIABILITY ANALYSIS
- AFWS REVIEW TEAM

ANO-2

- LICENSEE'S AFWS RELIABILITY ANALYSIS
- DEDICATED BLEED & FEED ARRANGEMENT
- AFWS REVIEW TEAM

FT. CAHOUN

- AFWS REVIEW TEAM

B&W OG (INCLUDES R. SECO, C. RIVER & ANO-1)

- B&W DESIGN REASSESSMENT
  - IMPROVE RELIABILITY OF MFWS
  - IMPROVE RELIABILITY OF AFWS
  - LIMIT CHALLENGES TO AFWS
- CURRENTLY UNDERWAY
- WILL BE PRESENTED TO THE FULL COMMITTEE TODAY

REVIEW PROGRAM

AFWS REVIEW TEAM

- A FIVE PERSON TEAM PLUS A TEAM LEADER
- REVIEW MATERIAL REQUESTED AND ENROUTE
- TEAM STARTS OPERATION MONDAY, SEPTEMBER 15

REVIEW SCOPE

- AREAS TO BE COVERED ARE SHOWN ON TABLE 1
- AFW AND SUPPORT SYSTEMS
- AUDIT SELECTED PREVENTIVE AND CORRECTIVE MAINTENANCE ACTIVITIES DURING LAST 12 MOS.
- AUDIT SELECTED SURVEILLANCE TESTING PROCEDURES AND POST MAINTENANCE TESTING
- POST TMI MODS
- COMMON MODE VULNERABILITIES
- OPERATOR RECOVERY & WALK-THROUGHS
- CONTROL ROOM ADEQUACY FOR INDICATION, CONTROL & RECOVERY
- EASE OF LOCAL RECOVERY, INDICATION & CONTROL
- ALTERNATE DECAY HEAT REMOVAL MEANS

TABLE 1. AREAS COVERED BY STAFF REVIEW

1. P&IDs
2. FSAR
3. SYSTEM DESCRIPTION
4. TECHNICAL SPECIFICATION
5. I&C LOGIC DIAGRAMS
6. SERs SINCE 1980

AFWS, ADVs, PORVs (FEED & BLEED) SUPPORT SYSTEMS (E.G., POWER SUPPLIES, COMPRESSED AIR OR NITROGEN SYSTEMS, LUBRICATION, COOLING)

7. MAINTENANCE (PREVENTIVE PROGRAMS, CORRECTIVE ACTIVITIES) LAST 12 MOS
8. SURVEILLANCE TESTING (FEW PROCEDURES AND RESULTS), POST MAINTENANCE TESTING, LERs, NPRDS, SOEs, ROs, AOs DURING THE LAST 12 MOS
9. EMERGENCY OPERATING PROCEDURES (LOSS OF HEAT SINK)
10. F&BLEED PROCEDURES (SEE 9 ABOVE), TRAINING (YES/NO)

POSSIBLE REVIEW OUTCOMES

- THE REVIEW TEAM IS REASONABLY ASSURED THAT THE AFWS IS ADEQUATE AND SUFFICIENTLY RELIABLE
- PROVIDED CERTAIN PLANT-SPECIFIC MEASURES WERE TAKEN THE AFWS IS ADEQUATE AND SUFFICIENTLY RELIABLE.
- WITHOUT SUBSTANTIAL CHANGES TO THE AFWS' DESIGN MAINTENANCE, TESTING AND/OR OPERATION, THE REVIEW TEAM CAN NOT BE REASONABLY ASSURED THAT THIS SAFETY SYSTEM IS SUFFICIENTLY RELIABLE

IMPLEMENTATION OF FINDINGS

- ANY REVIEW TEAM FINDINGS WILL BE DISCUSSED WITH LICENSEE
- ANY PLANT BACKFITS WILL BE HANDLED BY NORMAL PROCEDURES.

SCHEDULE

	REVIEW TEAM VISIT	REPORT
P. ISLAND 1&2	SEP. 29, 1986	Nov. 3, 1986
ANO-2	NOV. 17, 1986	DEC. 22, 1986
FT. CALHOUN	JAN. 5, 1987	FEB. 9, 1987
ANO-1	NOTE (1)	NOTE (2)
R. SECO	NOTE (1)	NOTE (2)
C. RIVER	NOTE (1)	NOTE (2)

NOTE (1) THE NEED FOR AND SCHEDULE OF VISITS TO THESE PLANTS WILL BE DETERMINED IN LIGHT OF THE ONGOING STAFF WORK ON THESE PLANTS.

NOTE (2) REVIEW TEAM REPORTS FOR THESE PLANTS WILL BE COORDINATED WITH ONGOING STAFF EFFORTS.

APPENDIX VIII
REVISION OF THE ECCS RULE
10 CFR 50.46 AND APPENDIX K

ACRS-REVIEW OF PROPOSED

REVISION OF THE ECCS RULE

10 CFR 50.46 AND APPENDIX K

317TH ACRS MEETING

SEPTEMBER 11, 1986

WILLIAM BECKNER

BACKGROUND-CURRENT RULE

- 50.46(A) REQUIRES THAT EMERGENCY CORE COOLING SYSTEMS "...BE DESIGNED SUCH THAT ITS CALCULATED COOLING PERFORMANCE CONFORMS TO THE CRITERIA SET FORTH IN PARAGRAPH (B) ...APPENDIX K, ECCS EVALUATION MODELS, SETS FORTH CERTAIN REQUIRED AND ACCEPTABLE FEATURES OF EVALUATION MODELS."

- 50.46(B) CRITERIA INCLUDE: (1) CALCULATED PEAK CLADDING TEMPERATURE LIMIT OF 2200°F (2) MAXIMUM CLADDING OXIDATION OF 17% OF THICKNESS, (3) MAXIMUM HYDROGEN GENERATION LESS THAN 1% OF HYPOTHETICAL AMOUNT FROM TOTAL METAL REACTION, (4) COOLABLE GEOMETRY MAINTAINED, AND (5) LONG TERM COOLING PROVIDED.

- APPENDIX K CONTAINS SPECIFIC FEATURES THAT EVALUATION MODELS (EMS) MUST CONTAIN (E.G., SOURCES OF HEAT TO BE INCLUDED IN THE CALCULATIONS), SPECIFIC FEATURES EMS MUST EXCLUDE (E.G., STEAM COOLING ONLY FOR LOW REFLOOD RATES), SPECIFICALLY REQUIRED MODELS (E.G., MOODY BREAK FLOW), AND MODELS WHICH ARE ACCEPTABLE, BUT NOT REQUIRED (E.G., SPECIFIC HEAT TRANSFER CORRELATIONS).

A-71

EXISTING RULE HAS PROVIDED SUCCESSFUL REGULATION AND WILL BE "GRANDFATHERED" UNTIL INDUSTRY PHASES IN NEW RULE: HOWEVER, IT IS UNIQUE IN TWO ASPECTS:

1. PRESCRIPTIVE MODELS
2. RESTRICTIVE REPORTING OF ERRORS, EVEN WHEN NO REGULATORY OR SAFETY THRESHOLD IS SURPASSED

IT IS APPROPRIATE, AFTER A DECADE OF ECCS RESEARCH, TO REVISE THE RULE TO MAKE IT MORE CONSISTENT WITH OTHER NRC REGULATIONS.

BACKGROUND-APPENDIX K

- APPENDIX K AND 50.46 WERE ISSUED IN 1975 BASED ON THE KNOWLEDGE OF ECCS PERFORMANCE AT THAT TIME AFTER EXTENSIVE HEARINGS. CERTAIN AREAS OF APPENDIX K WERE KNOWN AT THAT TIME TO BE VERY CONSERVATIVE (E.G., DECAY HEAT), BUT WERE USED TO COVER UNCERTAINTY IN THE OVERALL UNDERSTANDING OF ECCS PERFORMANCE.

- MANY FEATURES OF EMS ARE NOT SPECIFIED BY APPENDIX K. IN GENERAL, THESE AREAS HAVE ALSO BEEN TREATED CONSERVATIVELY DUE TO STAFF REQUIREMENTS OR DUE TO APPLICANTS PROPOSING SIMPLIFIED AND/OR CONSERVATIVE MODELS.

- RESEARCH PERFORMED SINCE 1975 HAS SHOWN THAT THE NET EFFECT OF THESE CONSERVATISMS IS UNREALISTIC, YET HIGHLY CONSERVATIVE, EVALUATION MODELS.

- DISTORTIONS CREATED BY THE USE OF ARTIFICIAL CONSERVATISMS IN APPENDIX K CAN ADVERSELY AFFECT THE OVERALL SAFETY OF PLANT DESIGN AND OPERATION

  - EXAMPLES INCLUDE OVERLY TIGHT TRIP SETPOINTS CAUSING NEEDLESS SCRAMS; OVERLY TIGHT DIESEL GENERATOR START TIME REQUIREMENTS (REDUCING DIESEL RELIABILITY); FLAT RADIAL POWER PROFILES LEADING TO HIGHER FLUX ON VESSEL (PTS CONCERN); ARTIFICIAL DESIGN OF ECCS ACCUMULATORS

A-72

- NRC, DOE, INDUSTRY AND FOREIGN RESEARCH PERFORMED SINCE 1975 PROVIDES A TECHNICAL BASIS WHICH PERMITS MORE REALISTIC TREATMENT OF ECCS ANALYSES.

- 1973 COMMISSION POSITION STATEMENT:

  "THE COMMISSION EXPECTS THAT BOTH GOVERNMENTAL AND PRIVATE PROGRAMS WILL BE PURSUED DILIGENTLY, AND EXPECTS TO CONSIDER PROMPTLY THE NEW KNOWLEDGE AS IT BECOMES AVAILABLE, AND TO CONSIDER SUCH CHANGES IN THESE REGULATIONS AS THEY APPEAR APPROPRIATE IN THE LIGHT OF ALL INFORMATION THEN AVAILABLE".

- BROAD SUPPORT FOR REVISING THE RULE BY NRC STAFF, ACRS, AND INDUSTRY; NO INDICATION OF OPPOSITION FROM PUBLIC OR INTERVENORS AT THIS TIME.

- BROAD SUPPORT FOR CURRENT RULE REVISION APPROACH BASED ON SECY-83-472 (I. E., REALISTIC EM WITH UNCERTAINTY EVALUATION).

- PRESENT APPENDIX K REQUIREMENTS RESULT IN CONSIDERABLE DIVERSION OF INDUSTRY <u>(W)</u> AND REGULATORY RESOURCES TO SAFETY ANALYSIS THAT ARE IN FACT IRRELEVANT, TAKING ATTENTION AWAY FROM MORE IMPORTANT ISSUES

A-73

PROPOSED RULE REVISION

50.46(A)(1)

(I) "...ANALYTICAL TECHNIQUE REALISTICALLY DESCRIBES THE BEHAVIOR OF THE REACTOR SYSTEM..."
"COMPARISONS TO APPLICABLE EXPERIMENTAL DATA..."
"UNCERTAINTY MUST BE ACCOUNTED FOR SO THAT...THERE IS A HIGH LEVEL OF PROBABILITY THAT THE CRITERIA [PARAGRAPH B] WOULD NOT BE EXCEEDED."

(II) "ALTERNATIVELY, AN ECCS EVALUATION MODEL MAY BE DEVELOPED IN CONFORMANCE WITH...APPENDIX K..."

A-74

50.46(A)(2)

"RESTRICTIONS ON REACTOR OPERATION WILL BE IMPOSED...IF THE EVALUATIONS OF ECCS COOLING PERFORMANCE SUBMITTED ARE NOT CONSISTENT WITH PARAGRAPHS (A)(1)(I) AND (II)...AND REQUIRED TO PROTECT PUBLIC HEALTH AND SAFETY."

50.46(A)(3)

"...ESTIMATE THE EFFECT OF ANY CHANGE TO OR ERROR IN AN ACCEPTABLE EVALUATION MODEL..."

"...SIGNIFICANT CHANGE OR ERROR IS ONE WHICH RESULTS IN...TEMPERATURE DIFFERENT BY MORE THAN 50°F..."

"FOR EACH CHANGE TO, OR ANY ERROR...REPORT THE NATURE OF THE CHANGE OR ERROR AND ITS ESTIMATED EFFECT ON THE LIMITING ECCS ANALYSIS...AT LEAST ANNUALLY..."

"IF THE CHANGE OR ERROR IS SIGNIFICANT...REPORT WITHIN 30 DAYS..."

"...REPORT A PROPOSED SCHEDULE...TO SHOW COMPLIANCE..."

"...FACILITIES NOT HAVING NRC APPROVED INTEGRATED SCHEDULING SYSTEM, A SCHEDULE FOR ACHIEVING COMPLIANCE WILL BE ESTABLISHED BY THE NRC STAFF WITHIN 60 DAYS..."

"ANY CHANGE OR ERROR...DOES NOT CONFORM TO CRITERIA SET FORTH IN PARAGRAPH (B)...IS A REPORTABLE EVENT AS DESCRIBED IN 50.55E, 50.72 AND 50.73."

A-75

APPENDIX K

I.C.5.B - DOUGALL-ROHSENOW CORRELATION REMOVED AND ANOTHER HEAT TRANSFER CORRELATION REFERENCE UPDATED.

I.C.5.C

"EVALUATION MODELS APPROVED AFTER [EFFECTIVE DATE OF RULE]...SHALL NOT USE...[DOUGALL-ROHSENOW]..."

"EVALUATION MODELS...APPROVED PRIOR TO [EFFECTIVE DATE OF RULE] CONTINUE TO BE ACCEPTABLE UNTIL...CHANGE...RESULTS IN A SIGNIFICANT REDUCTION IN OVERALL CONSERVATISM IN THE EVALUATION MODEL."

"...SIGINIFICANT REDUCTION IN THE OVERALL CONSERVATISM...WOULD BE A REDUCTION IN THE CALCULATED PEAK FUEL CLADDING TEMPERATURE OF AT LEAST 50°F FROM THAT WHICH WOULD HAVE BEEN CALCULATED ON [EFFECTIVE DATE OF RULE]..."

II. - CHANGES IN DOCUMENTATION REQUIREMENTS TO:

1. REMOVE 20°F DEFINITION OF SIGNIFICANT CHANGE
2. PROVIDE A COMPLETE LISTING OF COMPUTER PROGRAM ONLY IF REQUESTED BY STAFF.
3. MAKE CONSISTENT WITH CHANGES IN 50.46(A).

A-76

° ECCS RULE REVISION PACKAGE HAS 3 PARTS:

1. REVISED RULE
2. REGULATORY GUIDE
3. COMPENDIUM OF ECCS RESEARCH

1. RULE DOES NOT MENTION EITHER THE REG GUIDE OR THE COMPENDIUM; THESE ARE ONLY REFERENCED IN THE SUPPLEMENTARY INFORMATION AND THUS HAVE LITTLE LEGAL STATUS.

2. REG GUIDE PROVIDES "GUIDANCE" ON "ACCEPTABLE" MODELS/DATA AND UNCERTAINTY PROCEDURES WHICH, IN PRACTICE, NRC WILL NOT CHALLENGE IF USED.

3. COMPENDIUM IS REFERENCED IN REG GUIDE AS REPRESENTATIVE OF "A LARGE BODY OF DATA GENERALLY APPLICABLE TO B.E. MODELS"; BUT THERE IS NO GUARANTEE THAT ANYTHING MENTIONED IN THE COMPENDIUM IS "ACCEPTABLE" IN THE REG GUIDE SENSE OF THE TERM "ACCEPTABLE".

A-77

1. REVISED RULE

- NO SPECIFICITY ON MODELING REQUIREMENTS (OTHER THAN DOUGALL-ROHSENOW)
- MORE FLEXIBILITY ON REPORTING REQUIREMENTS
- PERMITS USE OF LATEST TECHNOLOGY
  - BEST-ESTIMATE CODES
  - CODE UNCERTAINTY BASED ON DATA COMPARISON
- USE OF B. E. CODES SHOULD LEAD TO MORE UNDERSTANDABLE REGULATION
- CONSISTENT WITH INDUSTRY COMMITMENT TO USE OF SECY-83-472 (B.E. CODE + UNCERTAINTY + APPENDIX K CORRECTIONS)

2. REGULATORY GUIDE

- PROPOSES, FOR PUBLIC COMMENTS, ACCEPTABLE MODELS/DATA RELATED TO APPENDIX K.
- DEFINES REQUIREMENTS FOR ESTIMATING OVERALL CODE UNCERTAINTY AT THE 95% PROBABILITY LIMIT.
- FLEXIBILITY TO ACCEPT INDUSTRY INITIATIVES.

3. COMPENDIUM OF ECCS RESEARCH

- ROAD-MAP TO DECADE OF RESEARCH ON ECCS SINCE AMERICAN PHYSICAL SOCIETY REPORT.
- SHOWS NRC HAS PERFORMED SUFFICIENT RESEARCH TO BETTER UNDERSTAND MARGIN OF SAFETY IN CALCULATED ECCS OPERATION AND TO JUSTIFY THE RULE REVISION.

SCHEDULE FOR REVISION OF ECCS RULE

° ACRS MEETING TO UPDATE ECCS RULE ACTIVITIES	APRIL, AUGUST, SEPTEMBER 1986
° NRR, DRR, ELD, RES CONCUR WITH PROPOSED RULE	JUNE 1986
° CRGR MEETING	JULY, AUGUST 1986
° EDO	SEPTEMBER 1986
° COMMISSION	SEPTEMBER 1986
° COMPENDIUM OF ECCS RESEARCH ISSUED FOR PUBLIC COMMENT	OCTOBER 1986
° NOTICE OF PROPOSED RULEMAKING ISSUED FOR PUBLIC COMMENT	NOVEMBER 1986
° REGULATORY GUIDE ISSUED FOR PUBLIC COMMENTS	NOVEMBER 198~~7~~6
° COMMENTS PERIOD ENDS	FEBRUARY 1987
° FINAL RULE PUBLISHED	NOVEMBER 1987

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APPENDIX IX
REGULATORY ANALYSIS
REVISION OF ECCS RULE

REGULATORY ANALYSIS

REVISION OF ECCS RULE

317TH ACRS MEETING

SUPPLEMENTARY INFORMATION

A-81

REGULATORY ANALYSIS-EFFECT OF RULE CHANGE

* CALCULATED PEAK CLADDING TEMPERATURES (PCT) DURING LARGE BREAK LOCA (INCLUDING UNCERTAINTY BOUNDS) WOULD BE REDUCED. AMOUNT OF REDUCTION WOULD BE PLANT SPECIFIC AND ALSO DEPEND ON THE ACCURACY OF THE CALCULATION. HOWEVER, THE REDUCTION IN CALCULATED PCT WOULD LIKELY BE LARGE ENOUGH SO THAT LARGE LOCA CONSIDERATIONS WOULD NO LONGER BE LIMITING. OTHER CONSIDERATIONS (E.G., DNB, SBLOCA) WOULD LIMIT PLANT OPERATION.

* SMALL BREAK LOCA (SBLOCA) MODELS ARE GENERALLY MORE REALISTIC THAN LARGE BREAK MODELS AND WOULD BE LESS AFFECTED BY THE PROPOSED RULE CHANGE, SBLOCA MAY BECOME LIMITING.

* REDUCED CALCULATED LARGE BREAK LOCA PCT COULD RESULT IN:

  - INCREASED ALLOWED PEAK LOCAL POWER
  - INCREASED TOTAL POWER
  - CHANGES IN EQUIPMENT, SURVEILLANCE OR LCO

* ALL THE ABOVE CHANGES WOULD LIKELY NOT BE POSSIBLE AT THE SAME TIME AND MANY OTHER FACTORS ALSO WOULD HAVE TO BE CONSIDERED SUCH AS:

  - DNB LIMITS
  - PLANT HARDWARE LIMITS
  - OTHER CHAPTER 15 EVENTS

REGULATORY ANALYSIS-PLANTS EXPECTED TO BENEFIT

* GE AND WESTINGHOUSE PLANTS ARE GENERALLY LIMITED IN OPERATING FLEXIBILITY AND/OR TOTAL POWER BY THE ECCS RULE AS EVIDENCED BY:

  - WRITTEN RESPONSES FROM GE AND WESTINGHOUSE
  - SIGNIFICANT RESOURCES INVESTED BY GE AND W IN IMPROVED ECCS CALCULATIONS
  - RECENT BWR APPLICATIONS OF NEW GE SAFER MODEL

* THE GE SAFER MODEL APPROVED UNDER SECY-83-472 PROVIDES BWRS WITH THE CAPABILITY TO OBTAIN SIGNIFICANT REDUCTION IN OPERATING LIMITATIONS. IT MAY BE MORE DIFFICULT TO REDUCE OPERATING LIMITATIONS ON W PLANTS THOUGH THE USE OF SECY-83-472 W/O A RULE CHANGE.

* B&W AND CE CLAIM NO BENEFIT FOR THEIR PLANTS.

A-83

REGULATORY ANALYSIS-POTENTIAL COST SAVINGS

* WESTINGHOUSE CLAIMS MOST PLANTS ARE LOCA LIMITED AND COULD BE UPGRADED IN TOTAL POWER BY ABOUT 5% IF LOCA LIMITS REMOVED. PRESENT VALUE (10% DISCOUNT RATE) OF SUCH A POWER INCREASE FOR A W PLANT ASSUMING A 30 YEAR LIFE RANGES FROM \$13M TO \$147M, WITH AN AVERAGE FOR THE 47 CURRENTLY OPERATING W PLANTS OF \$68M. LOWER ASSUMED DISCOUNT RATES OR LONGER PLANT LIVES WOULD INCREASE THESE ESTIMATES.

* INCREASING ALLOWED LOCAL PEAK POWER (SAME TOTAL POWER) RESULTS IN MORE FLEXIBLE FUEL MANAGEMENT AND MANEUVERING CAPABILITIES. THESE ARE COMPLICATED SUBJECTS AND LOCA LIMITS ARE ONLY ONE OF SEVERAL LIMITING FACTORS. HOWEVER, SAVINGS OF \$3-6M PER PLANT PER YEAR MAY BE POSSIBLE.

* GENERIC RELOAD ANALYSES AND FEWER REANALYSES OFFER POTENTIAL SAVINGS.

* POTENTIAL HARDWARE CHANGES, LCO, ETC. HAVE NOT BEEN EVALUATED

A-84

LOCA LIMITS ON WESTINGHOUSE PLANTS

* CURRENT TECHNICAL SPECIFICATIONS AND UPDATED FSAR'S INDICATED THAT:

  - 15% OF W PLANTS ARE NOT RESTRICTED BY APPENDIX K LOCA CRITERIA (F_Q = 2.32; LIMITING LBLOCA PCT < 2000°F)

  - 41% HAVE MODERATE OPERATIONAL RESTRICTIONS

    - LIMITATIONS ON LOAD FOLLOWING
    - LIMITATIONS ON STEAM GENERATOR TUBE PLUGGING (F_Q = 2.32; LIMITING LBLOCA PCT > 2000°F)

  - 44% HAVE STRONG OPERATIONAL RESTRICTIONS

    - LIMITATIONS AS ABOVE
    - INCREASED CORE MONITORING REQUIREMENTS
    - POSSIBLE DIFFICULTIES IN ACHIEVING FULL POWER
    - (E.G., D. C. COOK 2; F_Q = 1.97; LIMITING LBLOCA PCT = 2187°F)
    - (GENERALLY F_Q 2.2; LIMITING LBLOCA PCT > 2100°F)

REGULATORY ANALYSIS-SAFETY IMPACT

* CHANGES WHICH MAY RESULT FROM PROPOSED RULE COULD RESULT IN POSSIBLE NEGATIVE AND POSITIVE SAFETY IMPACTS.

* DO NOT RECOMMEND ATTEMPTING TO QUANTIFY NET IMPACT BECAUSE:

  - NEGATIVE ASPECTS EXAMINED WERE FOUND TO BE SMALL COMPARED TO UNCERTAINTY IN OVERALL RISK
  - MANY OF THE POSITIVE IMPACTS, WHICH WE BELIEVE TO BE REAL, ARE HIGHLY SUBJECTIVE.
  - MAJOR RISK IMPACT BELIEVED TO RESULT FROM POTENTIAL CHANGES TO PLANT EQUIPMENT WHICH, WHILE POSSIBLE UNDER THE PROPOSED RULE, IS NOT THOUGHT TO BE A LIKELY RESULT.

* THE PROPOSED RULE MAY:

  - ALLEVIATE OVERLY TIGHT SETPOINTS; REDUCING NEEDLESS SCRAMS
  - ALLEVIATE OVERLY TIGHT DIESEL GENERATOR START TIMES; INCREASING DIESEL RELIABILITY
  - PERMIT NEUTRON FLUX PROFILES WHICH REDUCE FLUENCE ON VESSEL AND CORRESPONDING PTS RISK
  - POSSIBLY PERMIT POWER INCREASE

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SUMMARY

* A RULE REVISION HAS BEEN PROPOSED BASED ON INCREASED KNOWLEDGE OF ECCS PERFORMANCE GAINED SINCE THE RULE WAS WRITTEN

* THE EFFECT OF THE RULE IS TO REDUCE UNNECESSARY PLANT RESTRICTIONS WITH A POTENTIAL FOR ECONOMIC BENEFIT WITH NEGLIGIBLE EFFECTS ON SAFETY.

* RULE INCORPORATES THE EXISTING LICENSING METHODS SO AS TO NOT PLACE ANY ADDITIONAL BURDEN ON PLANTS NOT NEEDING OR DESIRING TO MAKE USE OF NEW RULE PROVISIONS.

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APPENDIX X
RESPONSE TO ACRS COMMENTS ON
REVISION OF ECCS RULE

RESPONSE TO ACRS COMMENTS ON REVISION OF ECCS RULE

317TH ACRS MEETING

SEPTEMBER 11, 1986

WILLIAM BECKNER

A-88

ACRS COMMENT:

SHOULD APPENDIX K BE AVAILABLE AS AN OPTION FOREVER OR SHOULD IT EVENTUALLY BE PHASED OUT?

RESPONSE:

THERE APPEARS TO BE POTENTIAL BENEFITS FROM A REALISTIC CALCULATION AS OPPOSED TO CONSERVATIVE, BUT UNREALISTIC APPENDIX K CALCULATIONS. THUS, IT WOULD BE "NICE" TO EVENTUALLY HAVE ALL PLANTS LICENSED UNDER THE NEW METHOD. HOWEVER, IT IS DIFFICULT TO JUSTIFY FORCING ALL LICENSEES TO SUBMIT NEW EVALUATION MODELS, PARTICULARLY IN LIGHT OF THE BACKFIT RULE. A CASE FOR SAFETY IS HARD TO MAKE BECAUSE OF THE LARGE CONSERVATISM IN APPENDIX K AND THE COST WOULD BE HIGH. SUCH AN ACTION WOULD ALSO BE DIFFICULT TO JUSTIFY EVEN FOR NEW PLANTS.

ACRS COMMENT:

WHAT GUIDANCE IS THE REGULATORY GUIDE EXPECTED TO PROVIDE? WHAT IS THE OBJECTIVE OF THE GUIDE? MORE GUIDANCE SHOULD BE PROVIDED RELATIVE TO THE UNCERTAINTY EVALUATION SINCE THIS IS A KEY PART OF THE NEW METHODOLOGY.

RESPONSE:

THREE POTENTIAL AREAS OF GUIDANCE HAVE BEEN CONSIDERED:

1. CLARIFICATION OF "HIGH PROBABILITY"-

   A MAJOR OBJECTIVE OF THE REGULATORY GUIDE IS TO CLARIFY THE RULE WORDING "...HIGH LEVEL OF PROBABILITY THAT THE CRITERIA WOULD NOT BE EXCEEDED." SPECIFIC REQUIREMENTS WERE INTENTIONALLY LEFT OUT OF THE RULE SO AS TO AVOID THE POTENTIAL FOR LITIGATION ABOUT MATHEMATICS AND STATISTICS. THE GUIDE PROVIDES SPECIFIC CRITERIA AND GUIDANCE AS TO WHAT THE STAFF REQUIRES TO MEET THE RULE INTENT.

2. BEST ESTIMATE MODELS-

   THE GUIDE PROVIDES GENERAL PRINCIPLES TO BE USED IN REALISTIC MODELS. IN ADDITION, THE STAFF IS REVIEWING SPECIFIC MODELS THAT MAY BE LISTED AS "ACCEPTABLE" FOR USE IN REALISTIC CALCULATIONS. THE DEGREE OF SPECIFIC GUIDANCE ON MODELS THAT WILL BE INCLUDED IN THE GUIDE HAS NOT YET BEEN DETERMINED AND PUBLIC COMMENT IN THIS AREA WILL BE SOLICITED.

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3. UNCERTAINTY METHODOLOGY-

THIS SECTION ONLY PROVIDES BROAD PRINCIPLES W/O DETAILED METHODOLOGIES. THE STAFF HAS DECIDED NOT TO PROVIDE DETAILED GUIDANCE IN THIS AREA BECAUSE:

A. THE INDUSTRY IS ALREADY WORKING ON THEIR OWN METHODS AND IS IN MANY AREAS AHEAD OF THE NRC STAFF IN THIS AREA.

B. SINCE THIS IS A NEW AREA, THE STAFF WANTS TO PROVIDE MAXIMUM INCENTIVE AND FLEXIBILITY FOR THE INDUSTRY TO DEVELOP AND PROPOSED SPECIFIC METHODS.

ACRS COMMENT:

WHAT IS THE OBJECTIVE OF THE COMPENDIUM OF ECCS RESEARCH? WHAT IS THE RELATIONSHIP OF THE COMPENDIUM OF RESEARCH TO THE REGULATORY GUIDE AND WHAT IS THE STATUS OF THE COMPENDIUM ONCE REFERENCED BY THE GUIDE? THE TWO DOCUMENTS SHOULD BE CONSISTENT.

RESPONSE:

THE OBJECTIVE OF THE COMPENDIUM OF ECCS RESEARCH IS TO PROVIDE A BROAD "ROAD MAP" TO THE EXTENSIVE ECCS RESEARCH THAT HAS BEEN PERFORMED SINCE THE ECCS RULE WAS WRITTEN. IT SUPPORTS THE RULEMAKING IN THE SENSE THAT IT PROVIDES EVIDENCE THAT THE TECHNOLOGY NOW EXISTS TO PERFORM REALISTIC CALCULATIONS OF ECCS PERFORMANCE. THE REGULATORY GUIDE REFERENCES THE COMPENDIUM AS A SOURCE OF INFORMATION ABOUT THIS TECHNOLOGY. HOWEVER, THE REGULATORY GUIDE LANGUAGE HAS BEEN CAREFULLY WRITTEN SO THAT MODELS OR DATA CONTAINED IN THE COMPENDIUM ARE NOT, BY VIRTUE OF REFERENCE ALONE, TO BE CONSIDERED "ACCEPTABLE" IN THE REGULATORY SENSE AND THUS NOT SUBJECT TO REVIEW. THIS WAS DONE BECAUSE THE COMPENDIUM IS A VERY BROAD AND EXTENSIVE SUMMARY. AS SUCH, A DETAILED REVIEW OF WHAT RESEARCH IS ACCEPTABLE UNDER WHAT CONDITIONS WOULD NOT BE PRACTICAL.

THE STAFF AGREES THAT THE TWO DOCUMENTS SHOULD BE CONSISTENT. THIS HAS BEEN AN OBJECTIVE FROM THE START AND WE WILL ATTEMPT TO MEET THIS OBJECTIVE.

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ACRS COMMENT:

THE COMPENDIUM OF ECCS RESEARCH NEEDS WORK IN SEVERAL AREAS. THE ORGANIZATION COULD BE CHANGED TO MAKE IT MORE READABLE, THE DEFINITION OF ISSUES AND THE RELATIONSHIP BETWEEN ISSUES (CHAPTER 4) AND RESULTS (CHAPTER 7) IMPROVED, AND MAJOR IMPROVEMENTS MADE TO THE UNCERTAINTY DISCUSSION. PARTS OF THE DOCUMENT ARE SO NEGATIVE SO AS TO APPEAR NOT TO SUPPORT THE RULE REVISION.

RESPONSE:

THE LARGE SIZE OF THIS REPORT HAS MADE THIS A MONUMENTAL TASK AND THUS DELAYED THE COMPLETION OF THE REPORT. THIS DOCUMENT IS A ROUGH DRAFT BEING REVIEWED BY THE ACRS IN PARALLEL WITH THE RES STAFF REVIEW. WE GENERALLY AGREE WITH THE ACRS COMMENTS AND WELCOME SPECIFIC INPUT TO ASSIST US IN THE TASK OF SMOOTHING SOME VERY ROUGH EDGES.

WE ALSO RECOGNIZE THE IMPORTANCE OF A GOOD DOCUMENT TO ENSURE A SMOOTH RULEMAKING AND ARE GIVING THIS EFFORT A HIGH PRIORITY.

ACRS COMMENT:

WHAT IS THE IMPACT OF THE BACKFIT RULE ON NRR REVIEW OF NEW EM's? IF A LICENSEE APPLIES FOR AN AMENDMENT TO HIS LICENSE RELATIVE TO A NEW EM AND BELIEVES THAT THE STAFF IS IMPOSING UNDUE CONSERVATISM OR CHANGE TO A PROPOSED MODEL, MAY THE LICENSEE INVOKE THE BACKFIT RULE?

RESPONSE:

ACCORDING TO OGC STAFF, AS LONG AS THE ISSUES ARE CONFINED TO THE ADEQUACY OF THE EM AND WHETHER IT MEETS THE CRITERIA, THERE SHOULD BE NO BACKFIT. THE BACKFIT RULE COULD BE INVOKED ONLY IF THE STAFF RAISED A DIFFERENT ISSUE.

THIS CONCERN DIFFERS FROM THE "INSULATION ISSUE" PROMPTING CONCERN OVER THE ADEQUACY OF SUMP PUMP DESIGN. IN CASE OF THE ECCS RULE, THE LICENSEE ALWAYS HAS THE FALLBACK OF USING APPENDIX K AND HIS EXISTING EM.

APPENDIX XI
NRC/RES PROPOSED METHODOLOGY FOR
MEASURING THERMAL-HYDRAULIC CODE UNCERTAINTY

NRC/RES PROPOSED METHODOLOGY FOR MEASURING THERMAL-HYDRAULIC CODE UNCERTAINTY

FUAT ODAR, LOUIS SHOTKIN & NOVAK ZUBER

ACRS

9/11/86

STATUS

- REVISED ECCS RULE REQUIRES QUANTITATIVE MEASURE OF UNCERTAINTY (95% PROBABILITY) IN CALCULATED PEAK CLAD TEMPERATURE (PCT)
- NRC/RES IS PROPOSING COMPREHENSIVE METHODOLOGY, FOR INDEPENDENT PEER REVIEW IN OCTOBER, 1986, COMPOSED OF BOTH
  - CODE VS. DATA COMPARISONS AND
  - SYSTEMATIC EXAMINATION OF CODE MODELS AND CORRELATIONS
- METHODOLOGY ORIGINALLY DEVELOPED FOR
  - COHERENT INTEGRATION OF ICAP ASSESSMENT RESULTS
  - BASIS FOR ESTABLISHING CODE APPLICABILITY TO ANALYZE B&W PLANTS (PROGRAM AT T/H TECHNICAL INTEGRATION CENTER)
- WHEN APPLIED TO ECCS RULE, BOTH ASPECTS COMPLEMENT EACH OTHER IN A SINGLE COHERENT METHODOLOGY

PROPOSED NRC/RES UNCERTAINTY METHODOLOGY

- REQUIRES, AT START
  - CODE DOCUMENTATION (MANUAL, USER GUIDE, Q/A DOCUMENT, ASSESSMENT REPORTS)
  - ASSESSMENT AGAINST DATA
  - IDENTIFICATION OF KEY PROCESSES AND PARAMETERS FOR GIVEN SCENARIO
  - UNCERTAINTY IN INPUT AND BOUNDARY CONDITIONS
  - UNCERTAINTY IN FUEL PARAMETERS
  - UNCERTAINTY IN EXPERIMENTAL DATA

- FOUR FACTORS ADDRESSED BY NRC/RES UNCERTAINTY METHODOLOGY

  1. CAPABILITY OF CODE MODELLING TO ANALYZE AND SCALE UP MAJOR PHENOMENA EXPECTED IN THE ACCIDENT SCENARIO

  2. QUANTITATIVE MEASURE OF CODE BIAS AND STANDARD DEVIATION TO WITHIN 95% PROBABILITY, INCLUDING NODALIZATION SENSITIVITY WITHIN USER GUIDELINES

  3. DETECTION OF COMPENSATING ERRORS WHICH WOULD MASK THE PERCEIVED ACCURACY OF THE CALCULATED PCT

  4. SCALABILITY OF THE CALCULATED PCT

- TAKEN TOGETHER THESE FOUR FACTORS CONSTITUTE THE CODE APPLICABILITY TO ANALYZE A GIVEN SCENARIO IN A GIVEN FULL-SCALE VENDOR GEOMETRY

PROPOSED NRC/RES UNCERTAINTY METHODOLOGY:
DETAILS OF EACH FACTOR

1. CODE MODELING CAPABILITY

   A. IDENTIFY AND DEFINE MAJOR PHENOMENA EXPECTED, FOR A SCENARIO OR SET OF SCENARIOS

   B. ASSESS CODE MODELS TO CALCULATE AND SCALE THESE PHENOMENA

   C. IF MODELLING DEFICIENT, ESTIMATE EFFECT ON ACCURACY OF CALCULATED RESULT

   D. IDENTIFY SCENARIOS WHICH CANNOT BE ADDRESSED BY CODE

2. QUANTITATIVE MEASURE OF CODE UNCERTAINTY (BIAS AND STANDARD DEVIATION)

   A. CALCULATE Δ_{PCT} = MEASURED - CALCULATED, FOR SEVERAL TESTS IN DIFFERENT SCALED FACILITIES AND SIMILAR (LOCA) SCENARIOS

   B. COMBINE ALL Δ'S TO CALCULATE CODE BIAS

   C. ESTIMATE PROBABILITY DENSITY FUNCTION WITH HISTOGRAM OF FREQUENCY OF Δ VS Δ; DERIVE STANDARD DEVIATION FROM CURVE MATCHING THIS HISTOGRAM

   D. WITHOUT FURTHER ANALYSIS, WOULD ASSUME THIS BIAS AND STANDARD DEVIATION CAN BE EXTRAPOLATED TO FULL-SCALE

   E. HOWEVER, WE STILL NEED TO EVALUATE EFFECTS OF COMPENSATING ERRORS AND SCALABILITY TO CONFIRM EXTRAPOLABILITY

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3. COMPENSATING ERRORS

A. REPEAT (2) FOR OTHER KEY PARAMETERS

B. MAKE ENGINEERING JUDGEMENT ON ACCURACY REQUIRED FOR THESE KEY PARAMETERS; THEN

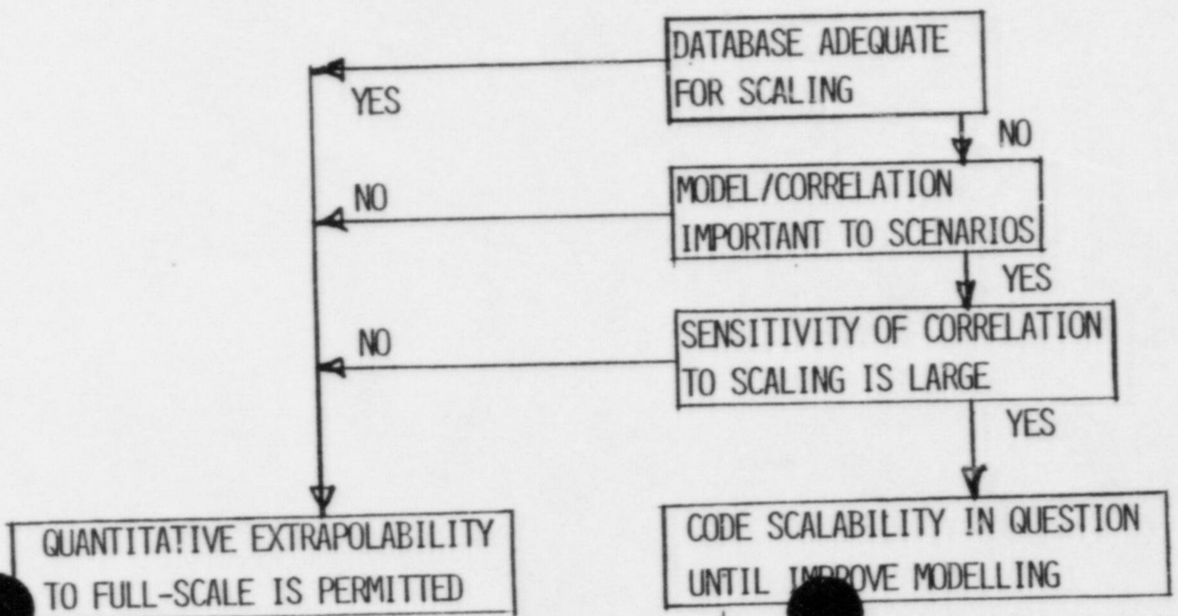
C. IDENTIFY FROM (1), MODELS/CORRELATIONS LEADING TO COMPENSATING ERRORS AND DETERMINE SENSITIVITY AND IMPORTANCE

D. EITHER IMPROVE MODELLING OR ADD EFFECT TO UNCERTAINTY

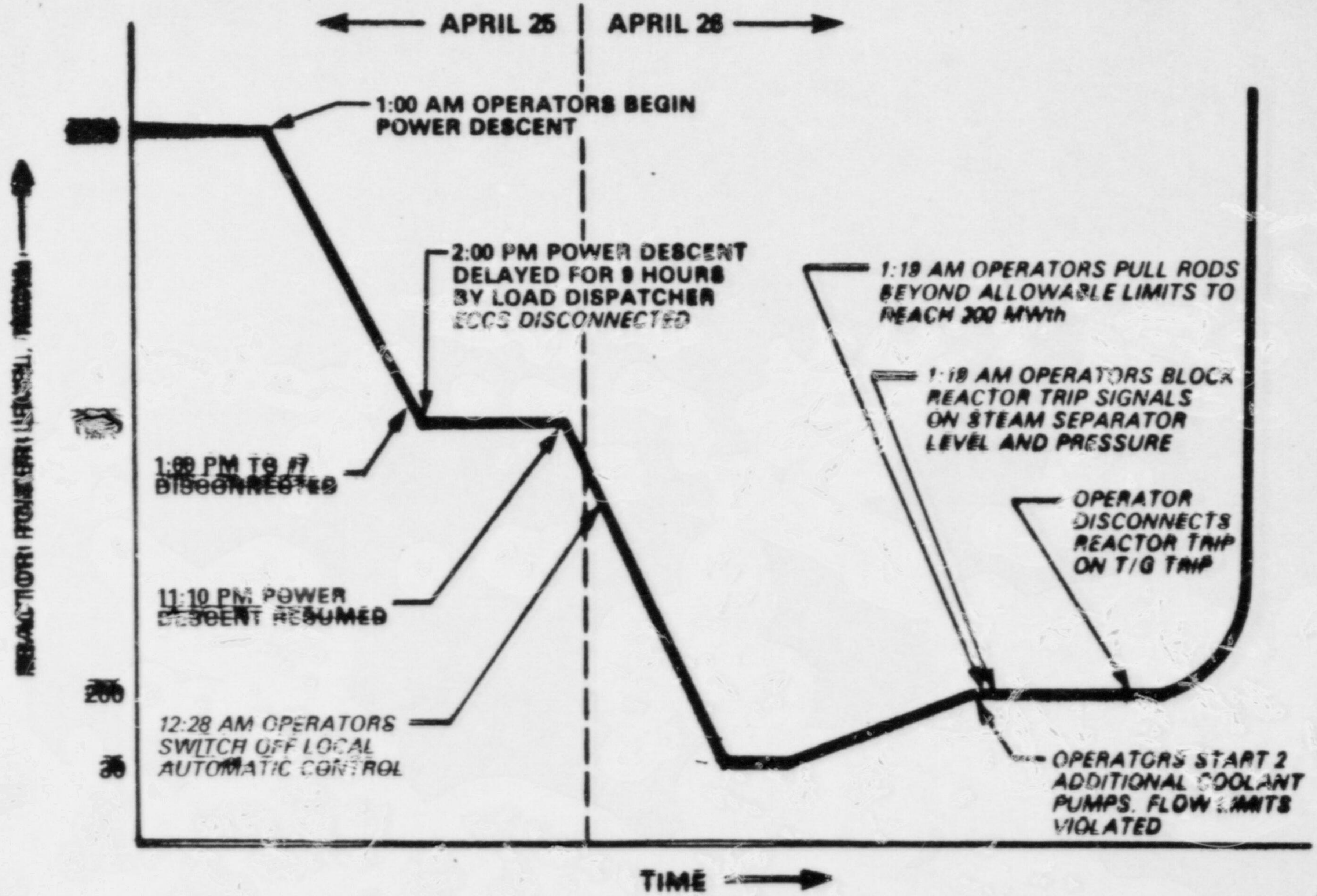
4. SCALABILITY

A. PLOT BIAS OF EACH FACILITY VS. SCALE AND LOOK FOR SIGNIFICANT SLOPE; AT SAME TIME

B. EXAMINE CODE MODELLING/CORRELATIONS:



CHRONOLOGY OF THE CHERNOBYL ACCIDENT



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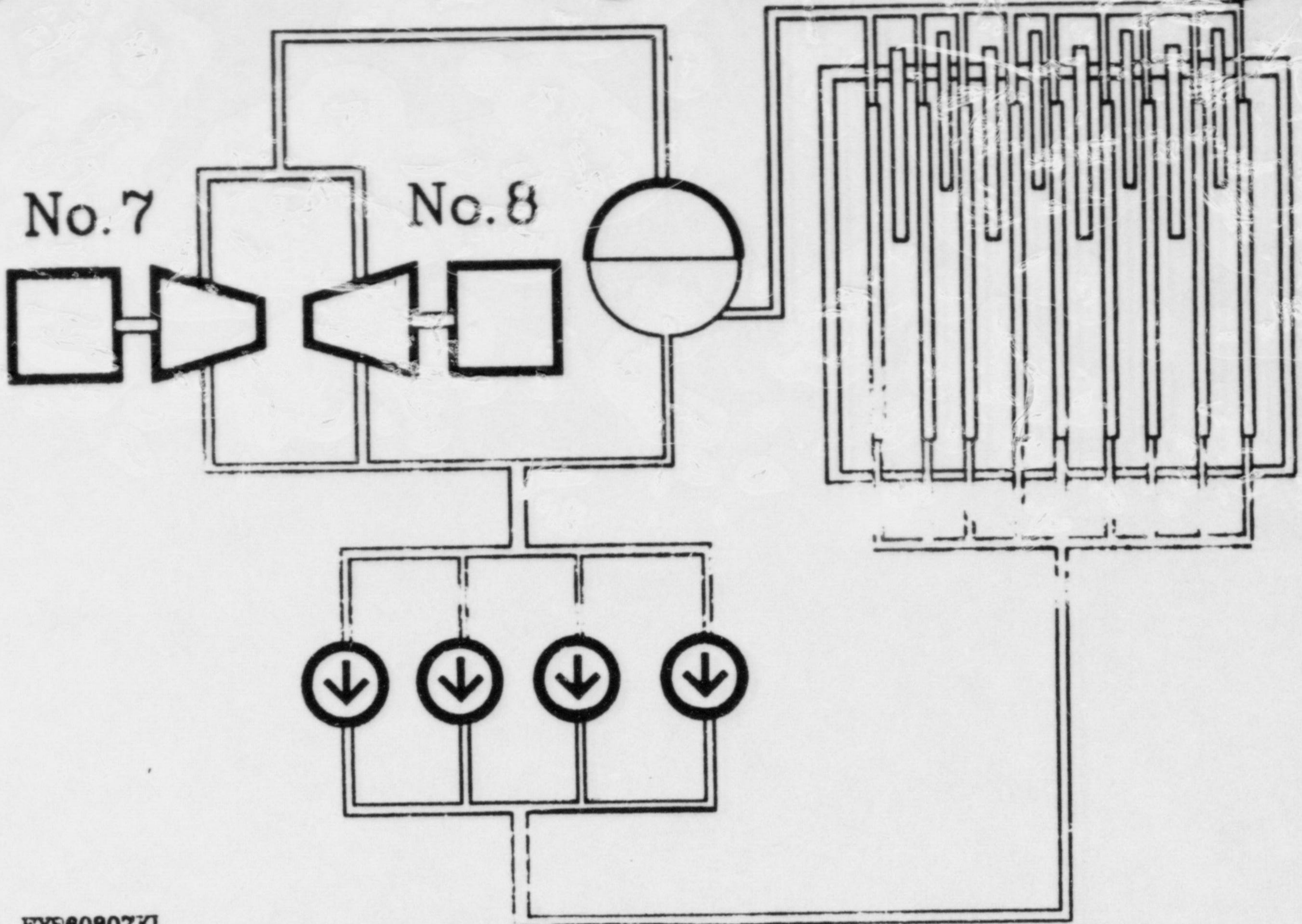
APPENDIX XII
CHRONOLOGY OF THE CHERNOBYL ACCIDENT



EYE60807XL

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REACTOR SPACE
1.8 Kgf/cm^2 (abs)
= 25.6 psia

0.08 MPa (12 psi')

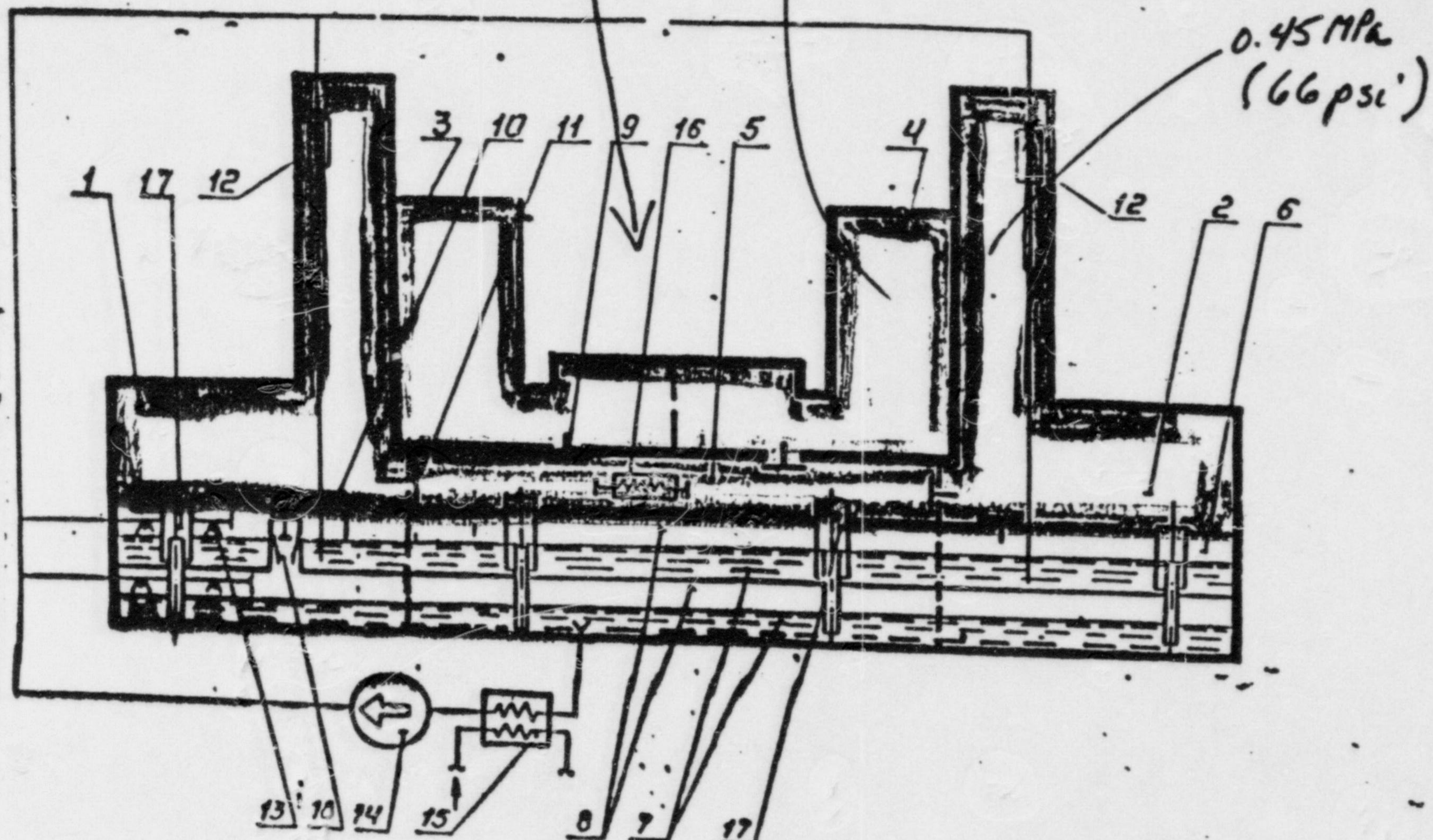
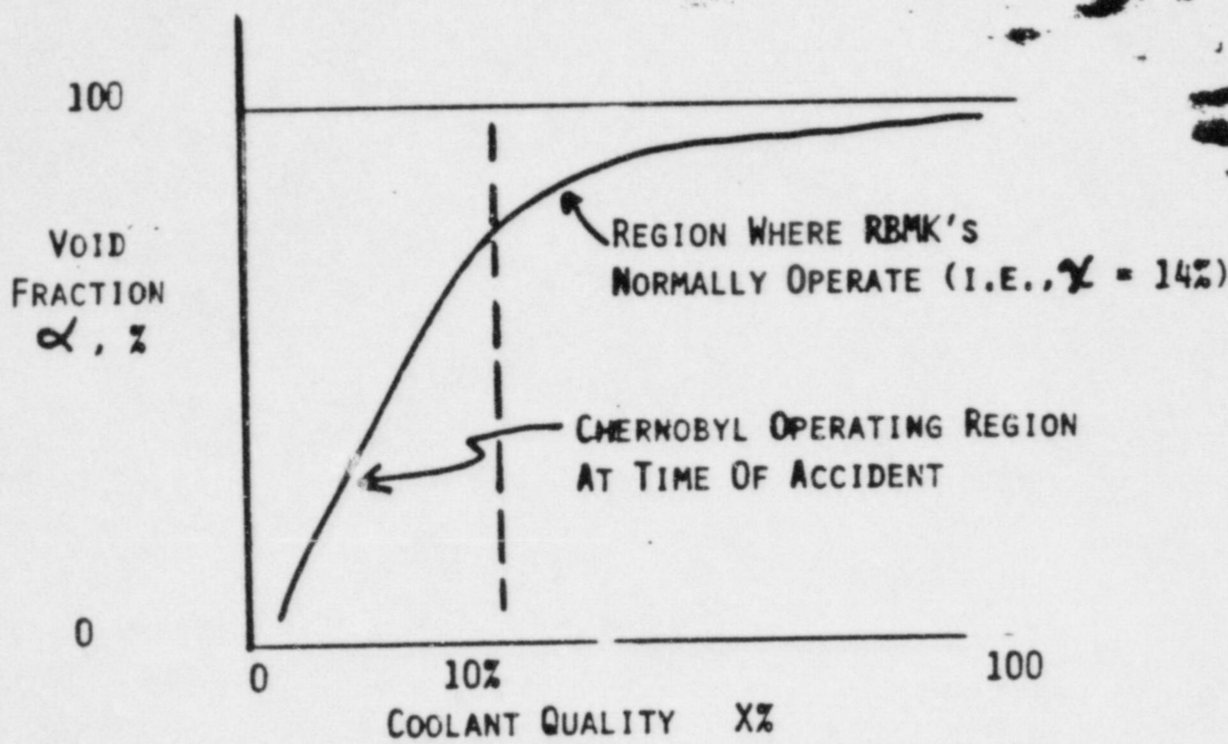


Fig 2.47 Schematic diagram of the confinement system

FIGURE (1)



$$X = \frac{\text{MASS OF STEAM}}{\text{MASS OF STEAM \& LIQUID}} = \text{FLOW QUALITY}$$

$$\alpha = \frac{\text{VOLUME OF STEAM}}{\text{VOLUME OF STEAM \& LIQUID}} = \text{VOID FRACTION}$$

<u>RELATIONSHIP</u>: $\alpha = \left\{ \frac{x}{1-x} \right\} \left\{ \frac{1}{(\rho_v/\rho_l)\, S} \right\}$

WHERE ρ_v/ρ_l = DENSITY RATIO OF VAPOR TO LIQUID AT PREVAILING PRESSURE

S = RATIO OF STEAM VELOCITY TO LIQUID VELOCITY (USUALLY EMPIRICALLY DETERMINED, BUT APPROXIMATELY UNITY AT LOW QUALITY)

9/12/86

REPORT ON THE IAEA MEETING ON THE CHERNOBYL ACCIDENT

HAROLD R. DENTON
THEMIS P. SPEIS
BRIAN W. SHERON
FRANK J. CONGEL

PRESENTED TO THE NRC COMMISSIONERS
SEPTEMBER 3, 1986

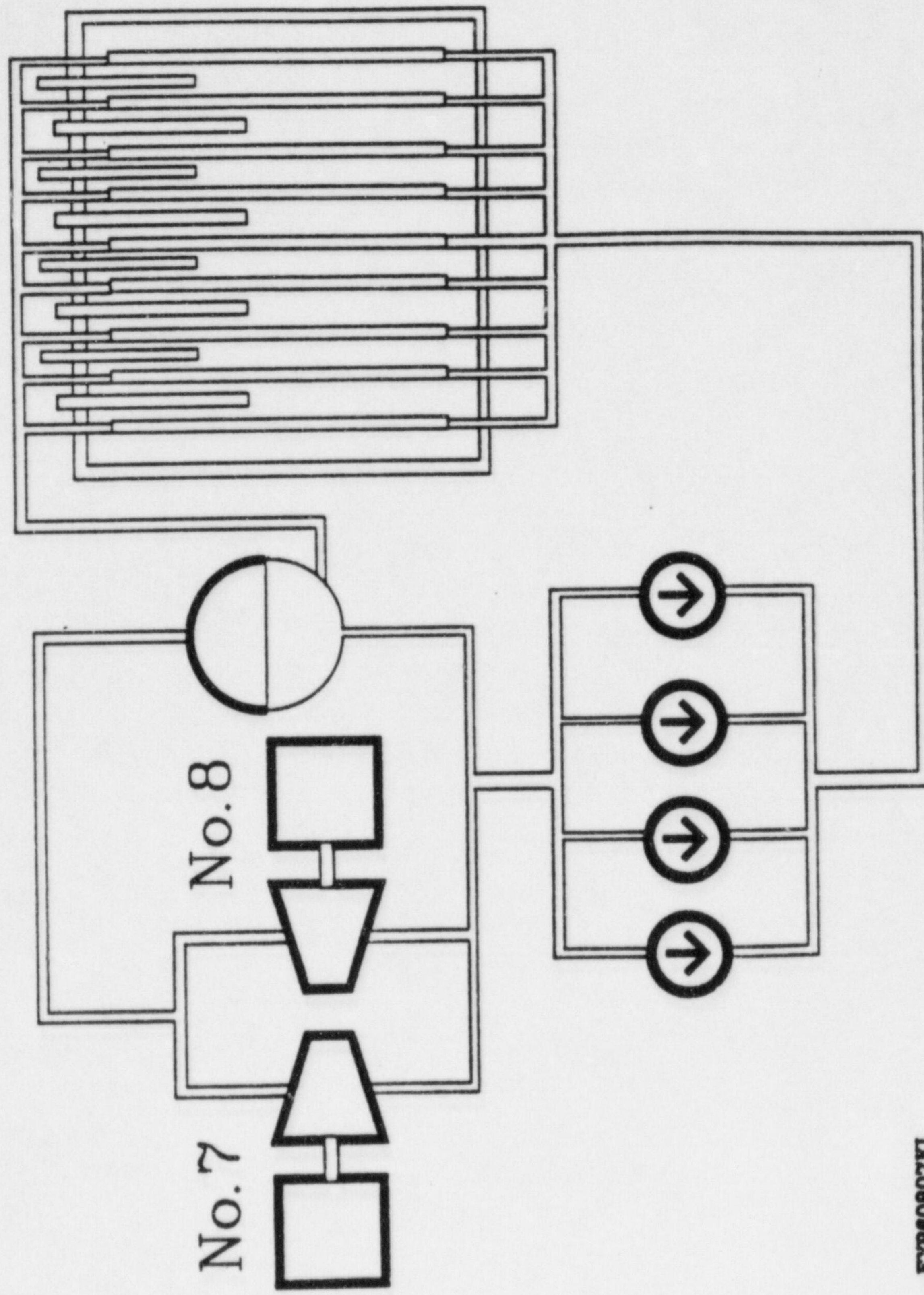
KEY DESIGN AND HUMAN FACTOR ASPECTS WHICH CONTRIBUTED TO ACCIDENT SEVERITY

HUMAN FACTOR ASPECTS

- TEST PROCEDURES VIOLATED CERTAIN SOVIET SAFETY REGULATIONS.
- OPERATORS VIOLATED CERTAIN PARTS OF TEST PROCEDURE.
- SOVIETS BELIEVE OPERATORS LOST SENSE OF VIGILANCE TOWARDS SAFETY.

DESIGN ASPECTS

- APPARENT SIMPLICITY WITH WHICH SAFETY AND PROTECTION SYSTEMS COULD BE OVERRIDDEN.
- SLOW CONTROL ROD INSERTION RATE.
- POSITIVE VOID REACTIVITY FEEDBACK.



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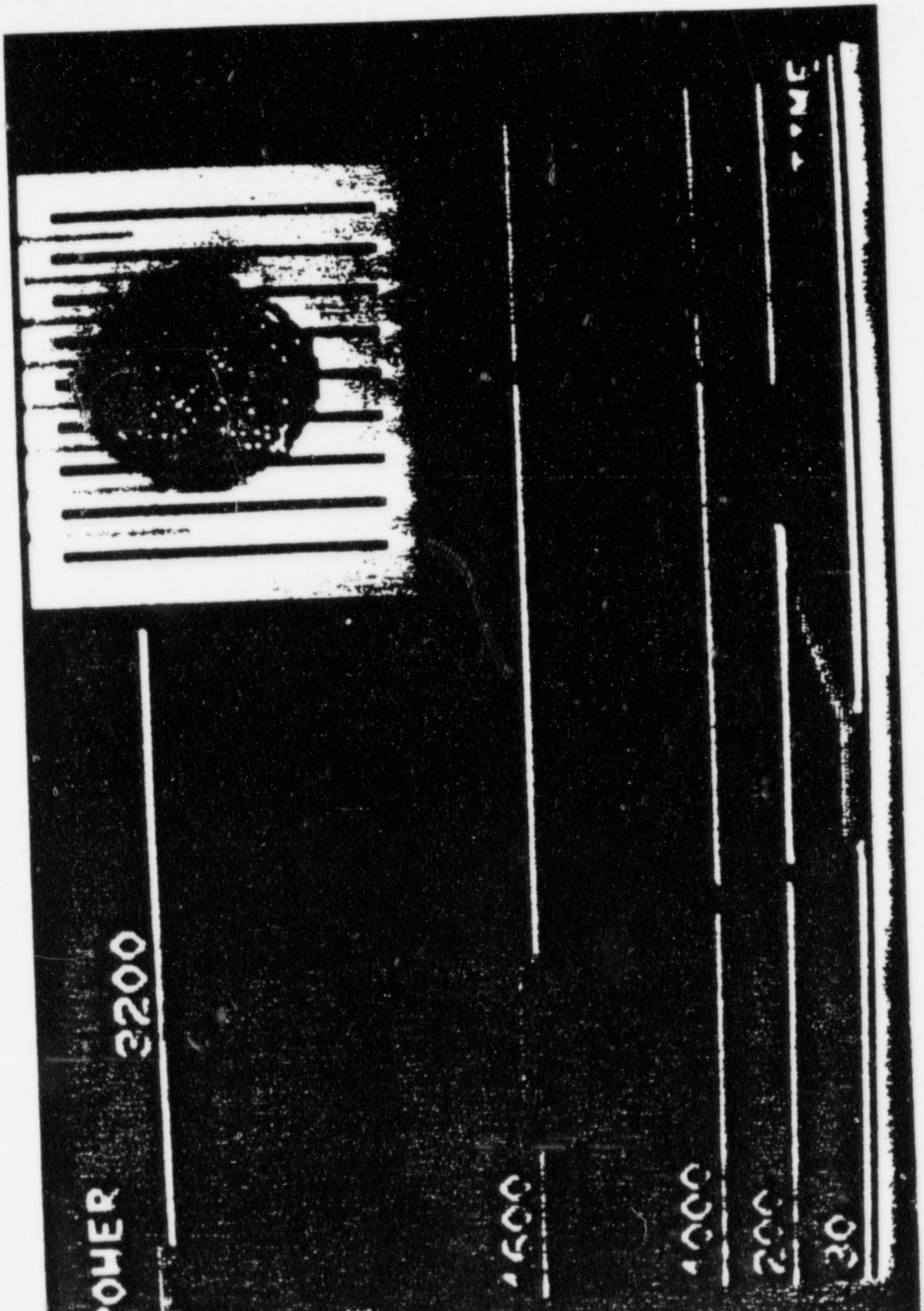

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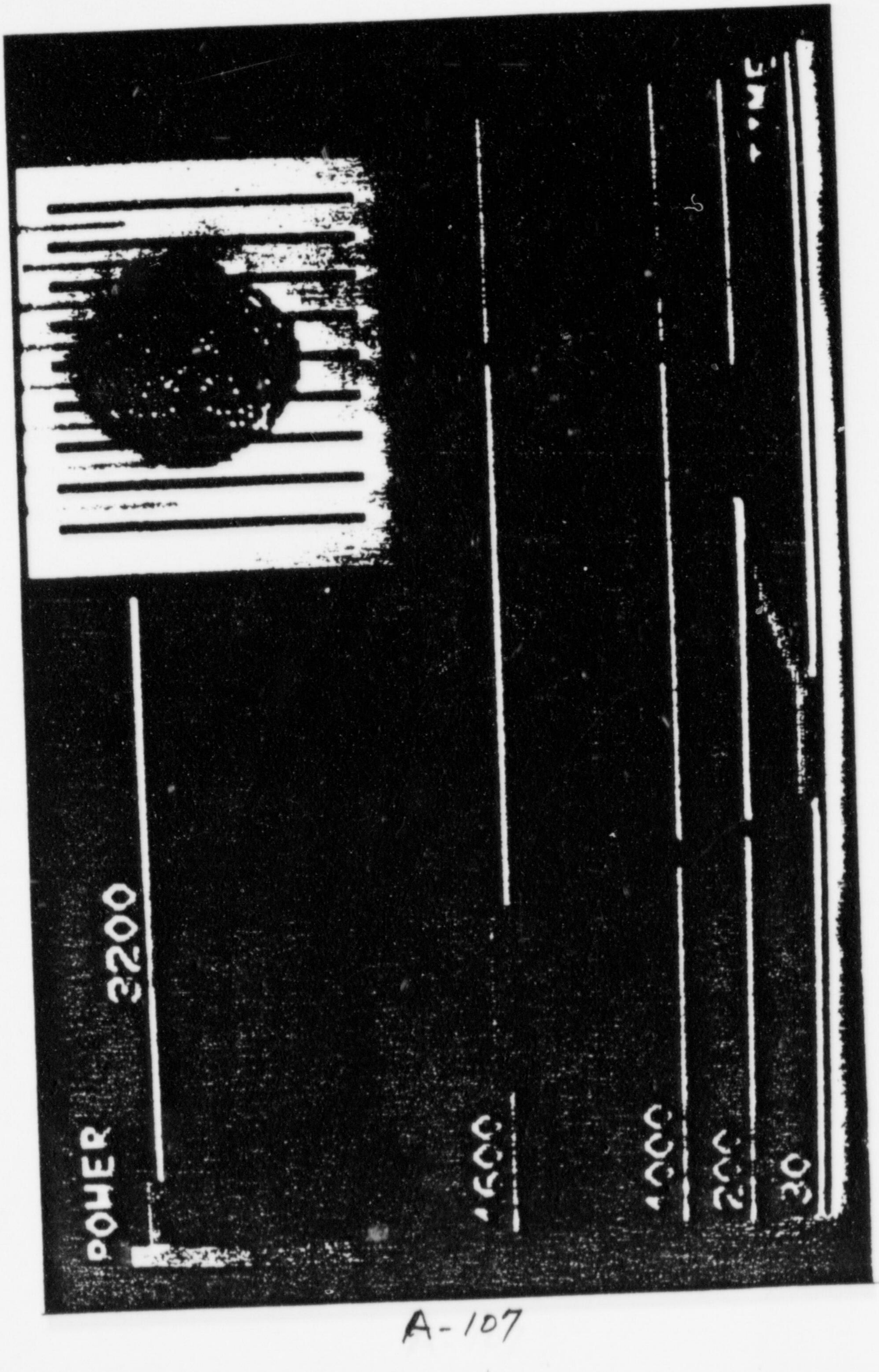

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A - power
B - total reactivity
C - pressure in the drum separator
D - power
E - rod AC-1 (automatic control rod)

G - rod AC-2
H - rod AC-3
K - flow of the main circulation pump
L - feed water flow

M - steam flow of the drum separator
N - T_{top} (T-temperature)
O -
P -
S - water level in the drum separator

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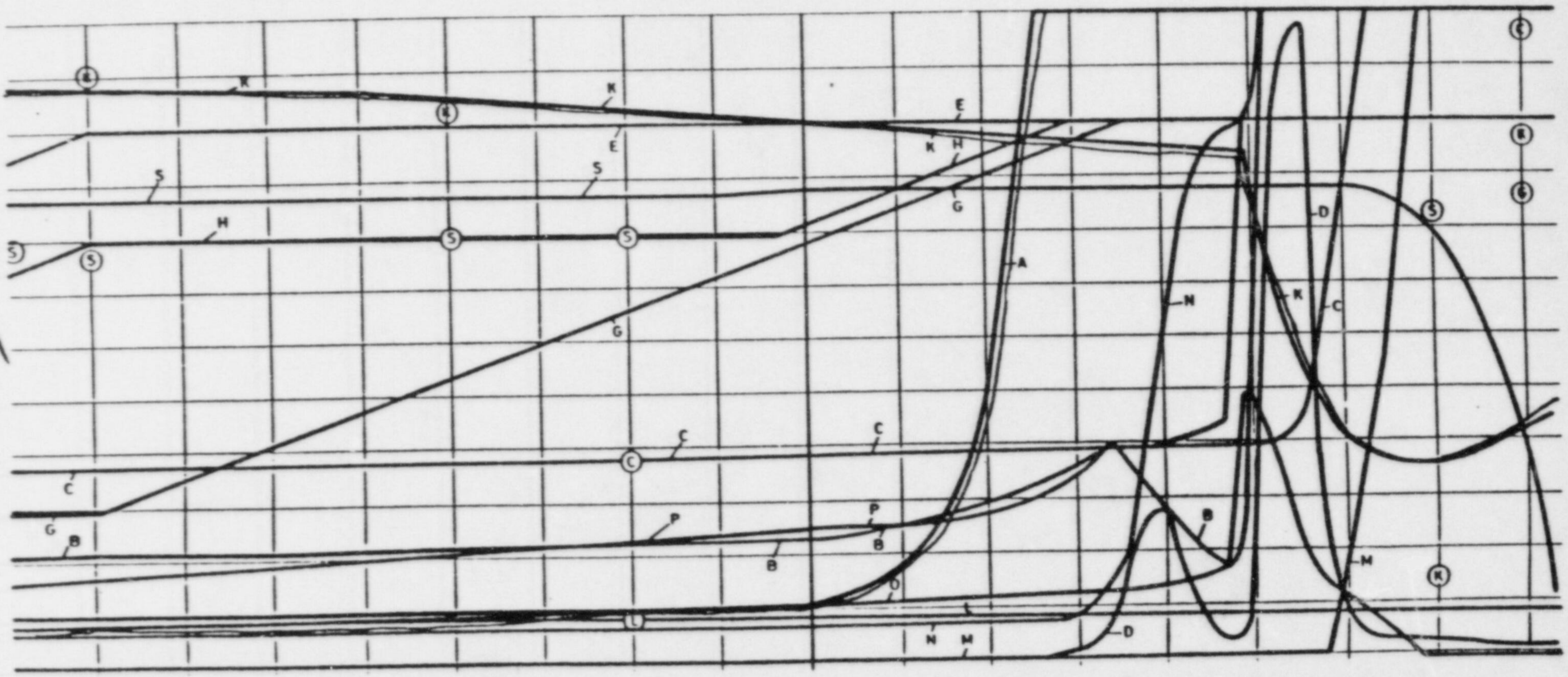

ind of
operation
PK up

1.23'40"
AZ-5

Fault in measurement section of automatic regulators AR1, AR2
Over pressure in drum separator
Triggering of fast-acting steam-dump system

ENVIRONMENTAL RELEASE TERMS
IN THE CHERNOBYL ACCIDENT

OVERVIEW

THE SOVIETS HAVE DEVELOPED ESTIMATES FOR ENVIRONMENTAL RELEASES OF RADIONUCLIDES AS A FUNCTION OF TIME.

THE ESTIMATED RELEASE FRACTIONS APPEAR TO BE:

1) CONSISTENT WITH THE SOVIET CONCEPTION OF THE ACCIDENT SCENARIO.
2) CONSISTENT WITH MEASUREMENTS MADE IN OTHER COUNTRIES.

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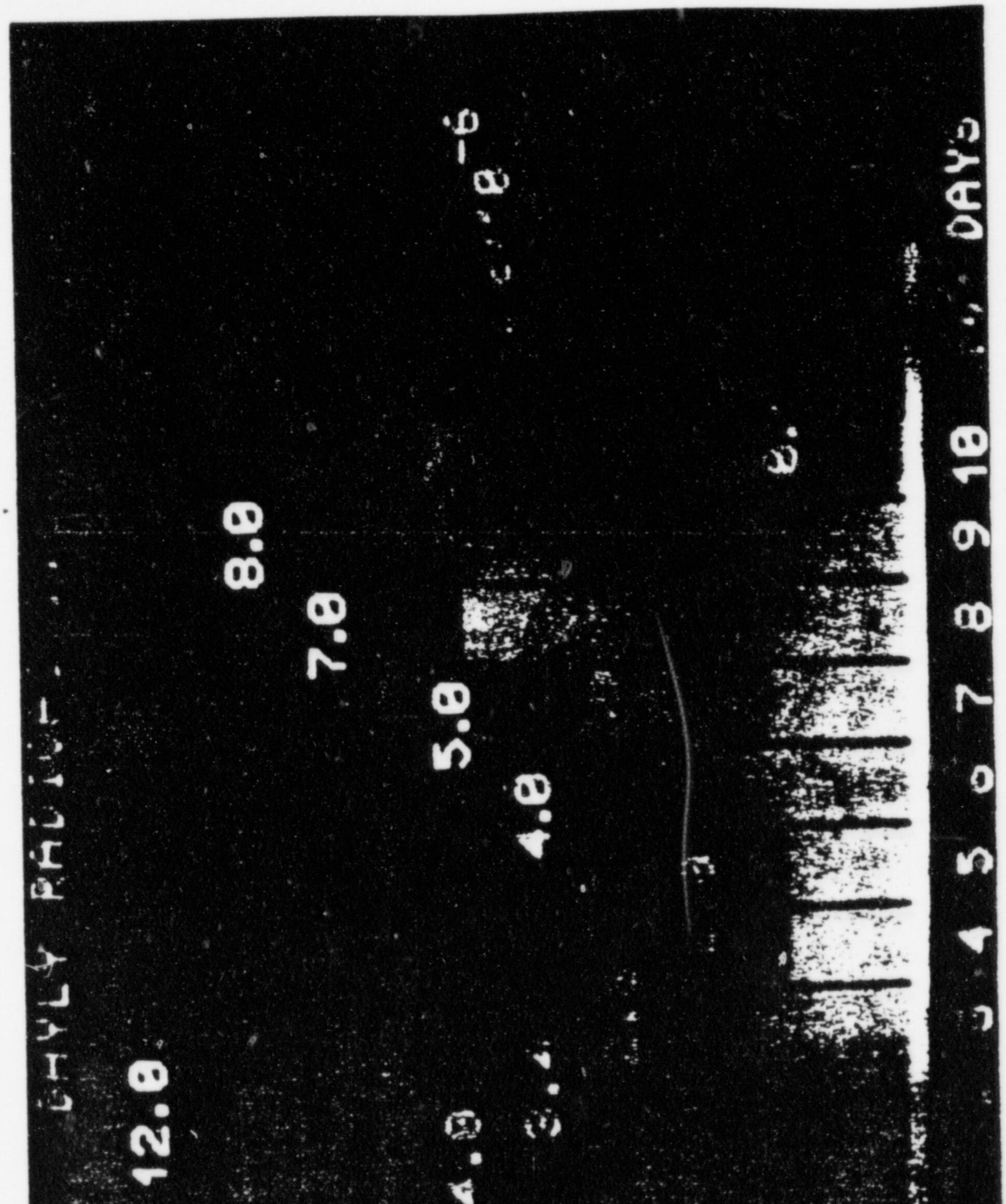

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ESTIMATED RELEASES OF RADIONUCLIDES FROM THE ACCIDENT UNIT OF CHERNOBYL NUCLEAR POWER PLANT*

Nuclide	Released activity, MCi		Release percentage by
	26.04.86	06.05.86**	06.05.86
Xe-123	5	45	may be up to 100
Kr-85m	0.15	-	-"-
Kr-85	-	0.9	-"-
I-131	4.5	7.3	20
Te-132	4	1.3	15
Cs-134	0.15	0.5	10
Cs-137	0.3	1	13
Mo-99	0.45	3	2.3
Zr-95	0.45	3.8	3.2
Ru-103	0.6	3.2	2.9
Ru-106	0.2	1.6	2.9
Ba-140	0.5	4.3	5.6
Ce-141	0.4	2.8	2.3
Ce-144	0.45	2.4	2.8
Pu-238	0.1E-3	0.8E-3	3.0
Pu-239	0.1E-3	0.7E-3	3.0
Pu-240	0.2E-3	1E-3	3.0
Pu-241	0.02	0.14	3.0
Pu-242	0.3E-6	2E-6	3.0
Cm-242	0.3E-2	2.1E-2	3.0
Sr-89	0.25	2.2	4.0
Sr-90	0.015	0.22	4.0
Np-239	2.7	1.2	3.2

* Estimated error ± 50%.

** Total release by May 6, 1986.

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CHERNOB6.LV

SOVIET ESTIMATES OF THE CHERNOBYL SOURCE TERM

ESTIMATED RELEASE TO ENVIRONMENT

NOBLE GASES	50 MCi	eg. Argon Krypton neon Xenon Radon
OTHER RADIONUCLIDES	50 MCi	
	100 MCi	

AVERAGE RELEASE FRACTION EXCLUDING NOBLE GASES 3.5%

ESTIMATED RELEASE FRACTIONS FOR KEY ELEMENTAL GROUPS

NOBLE GASES	UP TO 100%
IODINE	20%
TELLURIUM	10%
CESIUM	13%
INVOLATILE GROUPS	3 - 4%

PHASES OF RELEASE

PHASE 1. APRIL 26

RELEASE ASSOCIATED WITH VERY HIGH FUEL TEMERATURES IN EXCURSION

HIGH RELEASE OF VOLATILE FISSION PRODUCTS

PHASE 2. APRIL 26 - MAY 2

REDUCED RATE OF RELEASE

RELEASE PROPORTIONAL TO INITIAL INVENTORY OF FUEL

(IMPLYING TRANSPORT AS FUEL PARTICLES)

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SOVIET ESTIMATES OF THE CHERNOBYL SOURCE TERM (CONT.)

PHASE 3. MAY 2 - MAY 9

INCREASED RELEASE OF RADIONUCLIDES, PARTICULARLY VOLATILE SPECIES

IMPLIED HEATUP OF CORE

PHASE 4. MAY 10 -

RAPID DECREASE IN RELEASE

OXIDATION OF UO_2 TO U_3O_8 OBSERVED (MECHANISM FOR PRODUCING FULE AEROSOLS)

FIRE FIGHTING

- 3 TEAMS WENT TO SITE IMMEDIATELY
- FIRES LOCALIZED TO ROOFS BY 2:30 A.M.
- FIRES QUENCHED BY 5:00 A.M.
- OBJECTIVE WAS TO PREVENT SPREAD TO UNIT #3
    - PROTECT CABLE ROOMS
    - OIL TANK ROOMS
- USED PRIMARILY WATER TO EXTINGUISH FIRES. FIRES WERE MAINLY ON SURFACE
- SOVIETS IDENTIFIED "FIRE-FIGHTING LESSONS LEARNED"
    - LIST OF PROPOSALS GIVEN TO IAEA FOR CONSIDERATION

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ENTOMBMENT OF UNIT 4

A CONCRETE-WALLED BUILDING WILL BE CONSTRUCTED AROUND UNIT 4 TO ENSURE CONFINEMENT OF RADIOACTIVITY.

AN INNER CONCRETE PARTITION WALL IN THE TURBINE HALL WILL SEPARATE THE THIRD AND FORTH UNITS.

A METAL PARTITION WALL WILL SEPARATE UNITS 2 AND 3.

A PROTECTIVE ROOF WILL COVER THE TURBINE HALL.

THE CENTRAL HALL AND OTHER REACTOR ROOMS WILL BE SEALED OFF.

CONCRETE WILL BE POURED OVER DEBRIS IN SOME AREAS.

ENTOMBMENT OF UNIT 4 (CONT)

CLOSED LOOP AND OPEN LOOP SYSTEMS WERE CONSIDERED.

CLOSED LOOP SYSTEMS ARE MORE DESIREABLE FROM THE VIEWPOINT OF PUBLIC PERCEPTION.

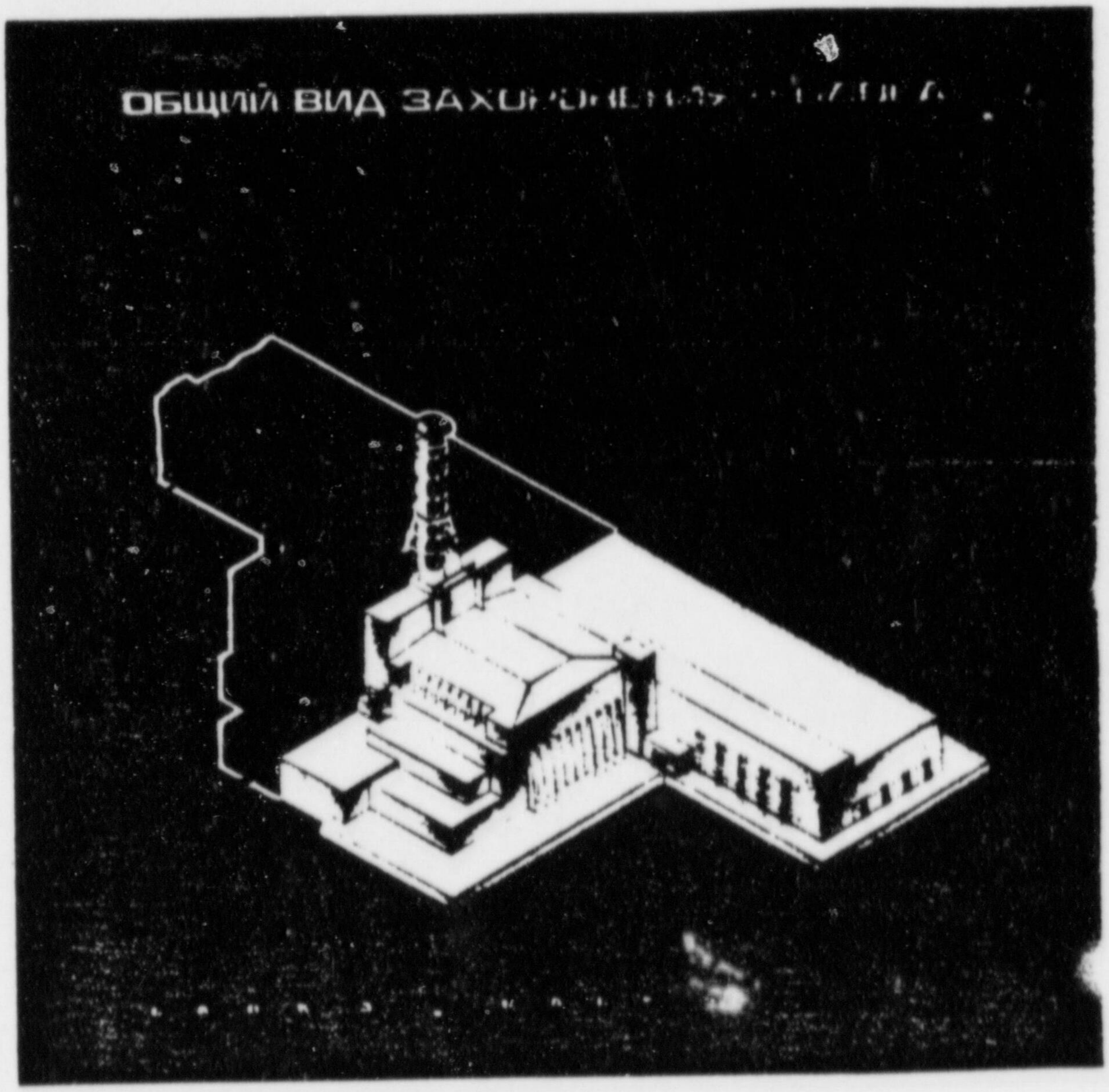
AN OPEN LOOP SYSTEM WAS SELECTED BECAUSE:

- PERMITS SIMPLIFIED CONTROL OF HYDROGEN (DILUTION).
- EASIER MONITORING AND MAINTENANCE.

A NEGATIVE PRESSURE DIFFERENTIAL WILL BE MAINTAINED BETWEEN THE BUILDING AND THE ENVIRONMENT.

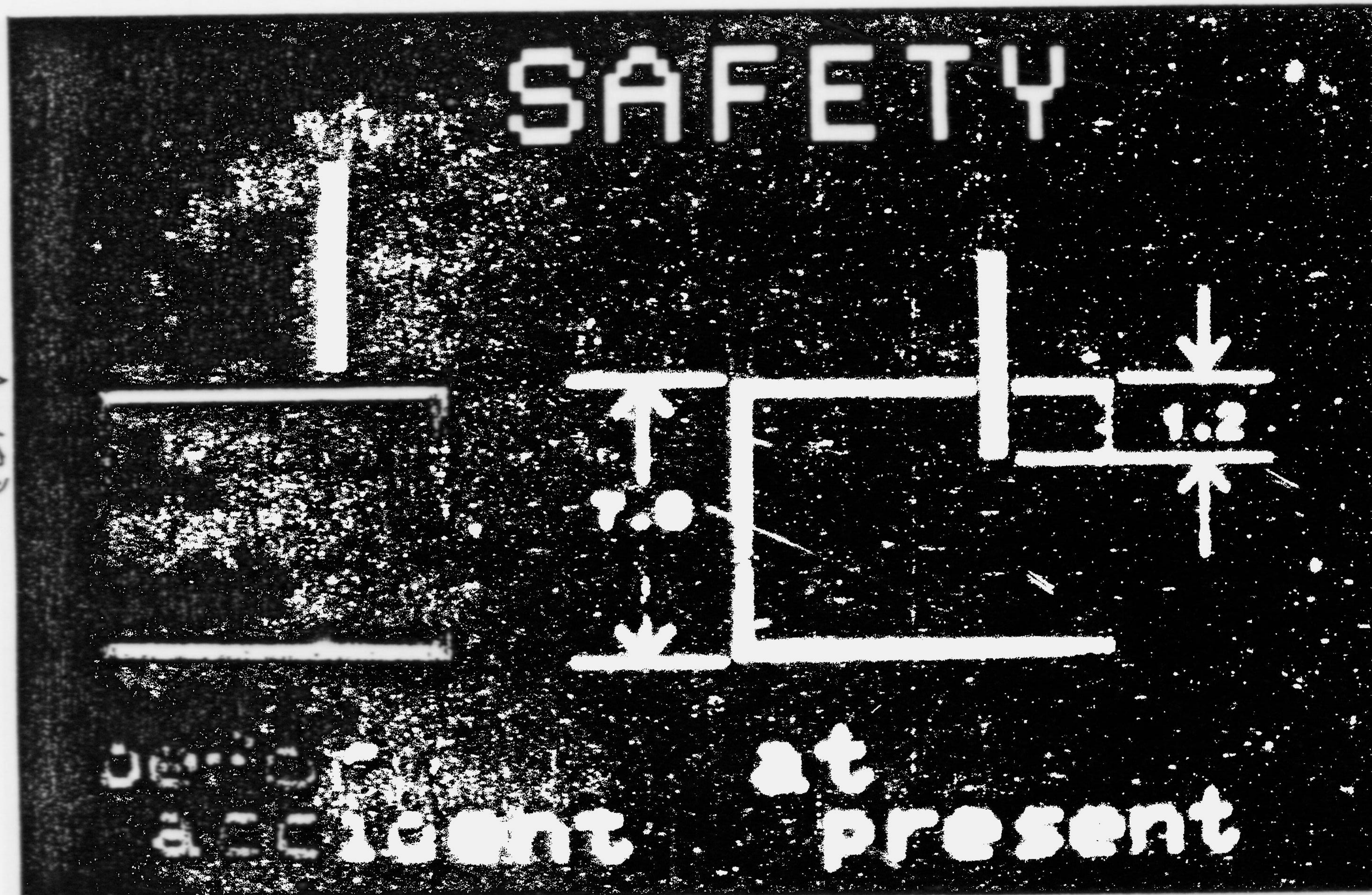
UNITS 1 AND 2 ARE EXPECTED TO RESUME OPERATION IN 1986.

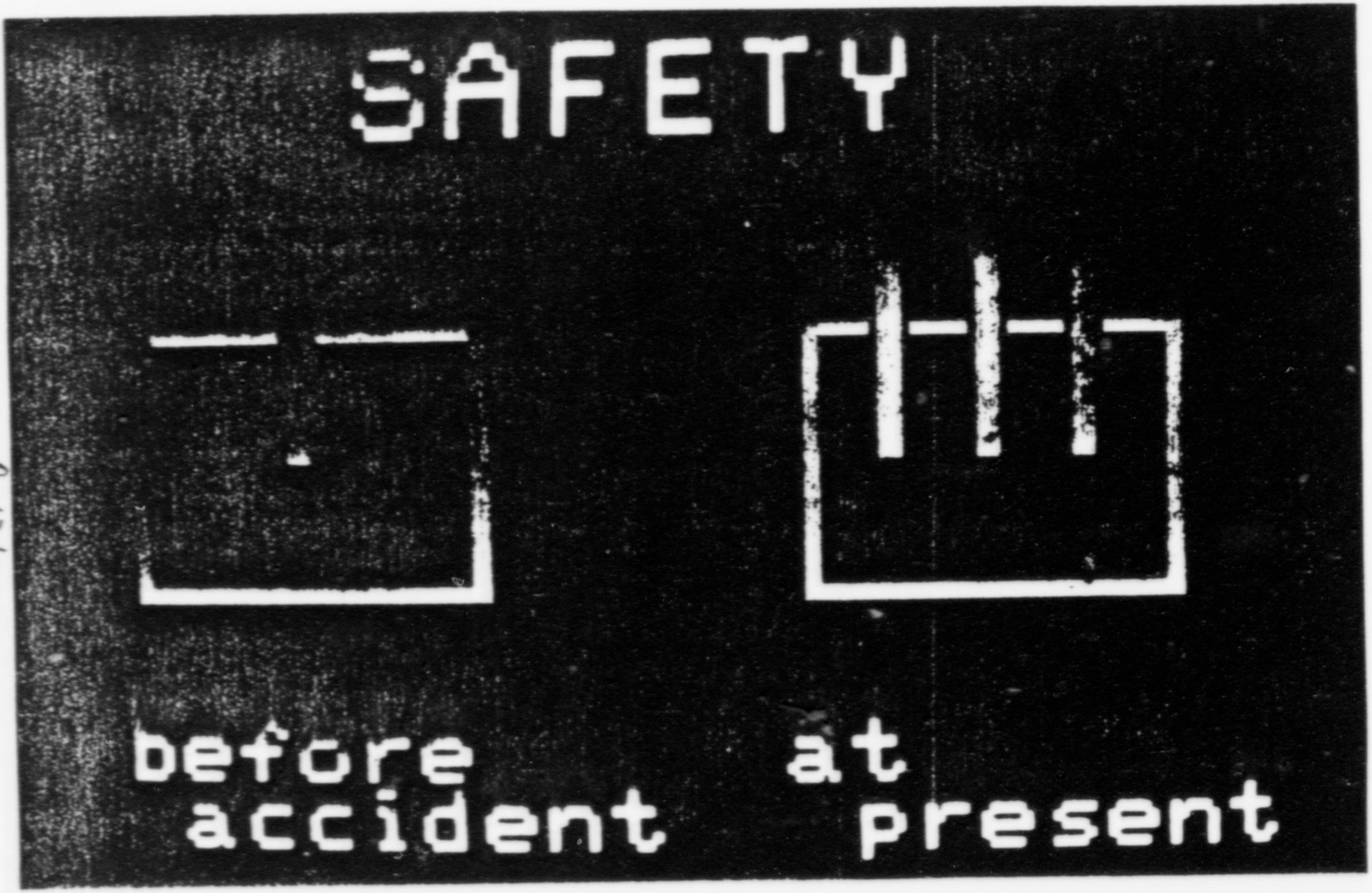
UNIT 3 WILL UNDERGO A THOROUGH SAFETY REVIEW BEFORE RESUMING OPERATION.


ОБЩИЙ ВИД

PROPOSED RBMK-1000 MODIFICATIONS (SHORT AND LONGER TERM)

- CONTROL RODS PERMANENTLY INSERTED IN THE CORE TO A DEPTH OF 1.2M.
- ABSORBER-TYPE CONTROL RODS ALWAYS PRESENT IN THE CORE WILL BE INCREASED TO 80, TO FURTHER REDUCE THE POSITIVE VOID COEFFICIENT (BY A FACTOR OF 2); TEMPORARY MEASURE UNTIL FUEL ENRICHMENT IS INCREASED TO 2.4% FROM THE PRESENT 2.0%.
- ADDITIONAL INDICATORS OF THE CAVITATION OF THE MCPs WILL BE INSTALLED.
- AUTOMATIC CALCULATION OF REACTIVITY WITH EMERGENCY SHUTDOWN SIGNAL WHEN EXCESS REACTIVITY MARGIN < SPECIFIED LEVEL.
- ORGANIZATIONAL STEPS TO REINFORCE TECHNOLOGICAL DISCIPLINE AND TO IMPROVE QUALITY OF OPERATIONS.
- EVALUATE ADDITIONAL DIVERSE AND FAST ACTING ABSORBERS SUCH AS LIQUID, GAS, OR SOLID FOR FUTURE USE.


SAFETY
1.0
1.2
accident
at
present



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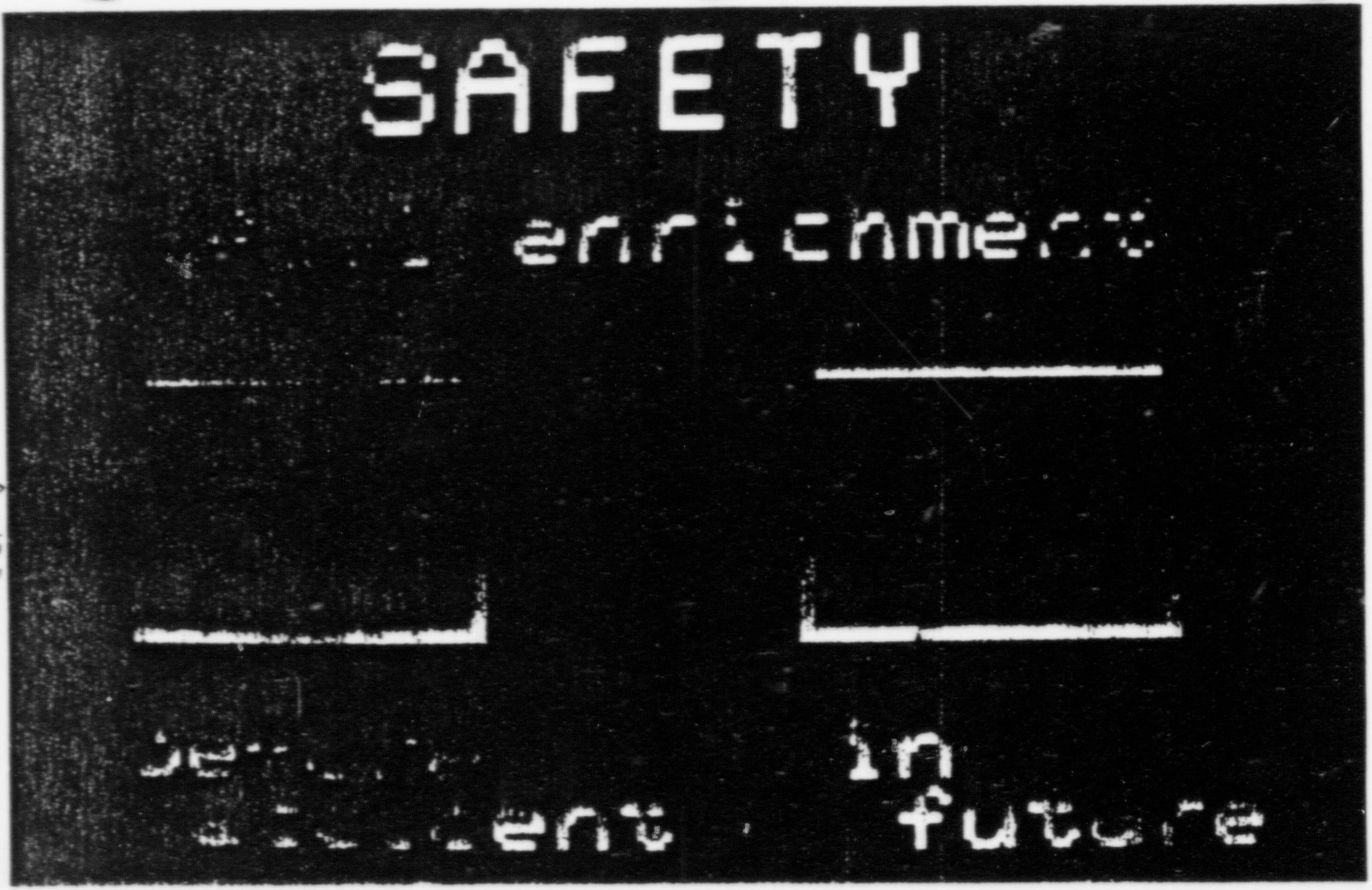

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PLUME MOVEMENT

1st Day	-	Toward the west and north (around Pripyat)
2nd-3rd days	-	Toward the north (through Pripyat)
4th day	-	Toward the south (toward Kiev)

Plume height exceeded 1200m on April 27

Dose rate in plume $\sim$ 1 R/hr as measured by airplanes 5-10 km from site

Plume contained fission and activation products (Cs-134 & Np-239)

INDIVIDUAL DOSES—WORKERS

- WORKERS - PRIMARILY FIREFIGHTERS

  1ST VICTIMS - ONE DIED AT 6:00 AM FROM BURNS
  ANOTHER WAS APPARENTLY BURIED
  IN COLLAPSED SECTIONS OF BUILDING

- 12 HOURS INTO ACCIDENT - 350 WORKERS CLOSELY EXAMINED FOR BLOOD CHANGES AND OTHER PHYSICAL SYMPTOMS OF HIGH RADIATION EXPOSURE

- 129 PEOPLE HOSPITALIZED IN SPECIAL CENTER IN MOSCOW

- 84 DIAGNOSED AS EXPERIENCING ACUTE RADIATION ILLNESS - IMMEDIATE SYMPTOMS

- 45 HAD LOWER EXPOSURES AND WERE OBSERVED FOR 1 TO 1.5 MONTHS

INDIVIDUAL DOSES - WORKERS (CONT.)

- TOTAL OF 203 PERSONS HOSPITALIZED WITH ACUTE RADIATION SICKNESS - NO MEMBERS OF THE GENERAL PUBLIC WERE INCLUDED IN THIS GROUP
- NO EVIDENCE OF NEUTRON EXPOSURE BASED ON ANALYSIS FOR NA-22
- SUBSTANTIAL AMOUNTS OF CS AND PU FOUND IN VICTIMS
- DOSE DISTRIBUTION - ALL 203 HOSPITALIZED VICTIMS RECEIVED > 100 REM. OF THAT GROUP, 35 PEOPLE EXCEEDED 400 REM UP TO A MAXIMUM OF 1200-1600 REM

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Table 7.2.1

Estimated doses of public exposure in some populated areas in the 30 km – zone arround ChNPS

Settlement	Distance from ChNPS km	Dose rate on "D" + 15 mR/h	Dose from discharge cloud R	Dose on child's thyroid rem	Dose from Radioactive fallout in 7 days R	
					estimate	actual
Chistowka	5.5	12	10	120	8.4	3.2
Leviev	9	25	7	250	17	10
Chernobyl	16	8	1.2	80	5.6	3.0
Rudki	22	8	[illegible]	80	5.6	2.2
Crevichi	29	2.5	0.2	25	1.8	4.4

NRC NOTE: Several other communities could have experienced doses (individual) of 30-40 rem.

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Table 7.2.3

ESTIMATED COLLECTIVE DOSES OF EXTERNAL IRRADIATION OF THE EVACUATED POPULATION

AREA AROUND CH NPS	NUMBER OF POPULATION THOUSAND/MEN	COLLECTIVE DOSE MLN.MEN.REM
T. PRIPYAT	45	0,15
3 - 7 km	7,0	0,38
7 - 10 km	9,0	0,41
10 - 15 km	8,2	0,29
15 - 20 km	11,6	0,06
20 - 25 km	14.8	0,09
25 - 30 km	39.2	0,18
TOTAL	135	1,6

DOSE TO AN INDI[illegible] USE OF CONTAMINATED FOOD FOR ONE MONTH

Food Type	Dose
Leafy Ve[illegible]le[1]	40 millirem
Meat	20 - 200 millirem
Milk	2 rem (adult thyroid) 5 rem (child thyroid)

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1. Assuming DF of 10 due [illegible]shing

Table 7.2.9

PREDICTED [illegible] [illegible] THE EXTERNAL EXPOSURE POPULATION [illegible] SOME REGIONS OF THE USSR EUROPEAN PART

REGION	POPULATION MILLIONS	DOSE FOR 1986 REM/YEAR		COLLECTIVE DOSE 10^6 MAN*REM	
		RURAL	URBAN	FOR 1986	FOR 50 YRS
UKR.SSR CENTR.	[illegible]	0.270	0.150	2.750	9.31
UKR.SSR WEST	[illegible]	0.067	0.036	0.440	1.47
UKR.SSR EAST	14[illegible]	0.077	0.041	0.750	2.52
UKR.SSR SOUTH	[illegible]	0.045	0.024	0.730	2.47
BSSR SOUTH-EAST	[illegible]	0.980	0.520	2.050	6.94
BSSR NORTH-WEST	7.[illegible]	0.094	0.050	0.470	1.58
MOLDAVIA	4.1	0.084	0.045	0.270	0.92
BRYANSKAYA REG.	1.6	0.500	0.270	0.440	1.49
KALININGR. REG.	0.[illegible]	0.012	0.003	0.006	0.02
KALUJ.,TULSK.					
SMOL. REG.	[illegible]	0.120	0.064	0.320	1.06
LIPETSK. REG.	[illegible]	0.140	0.075	0.350	1.17
TOTAL	74[illegible]	–	–	8.6	29.0

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CHERNOBYL.ACS

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SUMMARY OF GENERAL POPULATION
COLLECTIVE DOSES & ASSOCIATED ESTIMATED
HEALTH EFFECTS

PATHWAY	NO. OF EXPOSED PEOPLE	COLLECTIVE DOSE
EXTERNAL DOSE	135,000	1.6×10^6 PERSON-REM
EXTERNAL DOSE	74,500,000	29×10^6
FOOD PATHWAY - Cs	UNSPECIFIED	210×10^6*
THYROID EXPOSURE	UNSPECIFIED	-

*SOVIETS STATED THIS NUMBER IS CONSERVATIVE BY AS MUCH AS A FACTOR OF 10

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EMERGENCY RESPONSE MEASURES TAKEN

- IMMEDIATELY AFTER ACCIDENT PRIPYAT POPULATION (45,000) ADVISED TO REMAIN INDOOR AND CLOSE WINDOWS

- ON APRIL 26 OPEN-AIR ACTIVITIES BANNED AT ALL KINDERGARDENS, AND SCHOOLS; IODINE PROPHYLATIC TREATMENT GIVEN THERE

- EVACUATION OF PRIPYAT BEGAN AT 2 PM ON APRIL 27 AS DOSE RATE WORSENED; COMPLETED BY 5 PM THE SAME DAY

- REMAINING POPULATION (90,000) FROM 30-KM ZONE EVACUATED IN NEXT FEW DAYS BECAUSE OF CONTINUING CONTAMINATION DUE TO CHANGING PLUME DIRECTION

- CONSUMPTION OF MILK CONTAINING 1×10^{-7} CI/L OR MORE OF I-131 WAS BANNED

- ALL CHILDREN FROM 30-KM ZONE WERE SENT TO SUMMER SANITORIUM IN THE COUNTRY

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EMERGENCY REPONSE MEASURES TAKEN (CONT.)

- STANDARDS FOR PERMISSIBLE LEVELS OF CONTAMINATION IN FOOD PRODUCTS ISSUED BEGINNING EARLY IN MAY 1986
- 1240 DOCTORS, 920 NURSES AND SEVERAL THOUSAND SUPPORTING ASSISTANTS MOBILIZED TO PROVIDE MEDICAL CARE OF EVACUEES
- EACH EVACUEE EXAMINED; BLOOD TESTS CARRIED OUT; IN SOME CASES EXAMINATION AND TESTS REPEATED
- EVACUEES WHO SHOWED IRREGULARITIES WERE HOSPITALIZED IN SPECIAL SECTIONS SET-UP AT CENTRAL REGIONAL HOSPITALS
- LONG-TERM PROGRAMS ARE BEING ESTABLISHED FOR MEDICAL AND BILOGICAL MONITORING OF POPULATION AND PERSONNEL

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DECONTAMINATION--OFFSITE

- BUILDING AND HOUSES ARE BEING DECONTAMINATED BY SPRAYING DECONTAMINATION SOLUTION
- AFTER WASHING, CONTAMINATED SOIL AROUND THE BUILDINGS TURNED OVER OR REMOVED WITH BULL-DOZERS AND TAKEN AWAY
- TRANSPORT VEHICLES DECONTAMINATED USING SOLUTIONS BY SPRAYING AND STEAM JETS.
- LONG TERM DECONTAMINATION PROCEDURES OF THE ENVIRONMENT ARE BEING RESEARCHED

DECONTAMINATION -- PLANTSITE

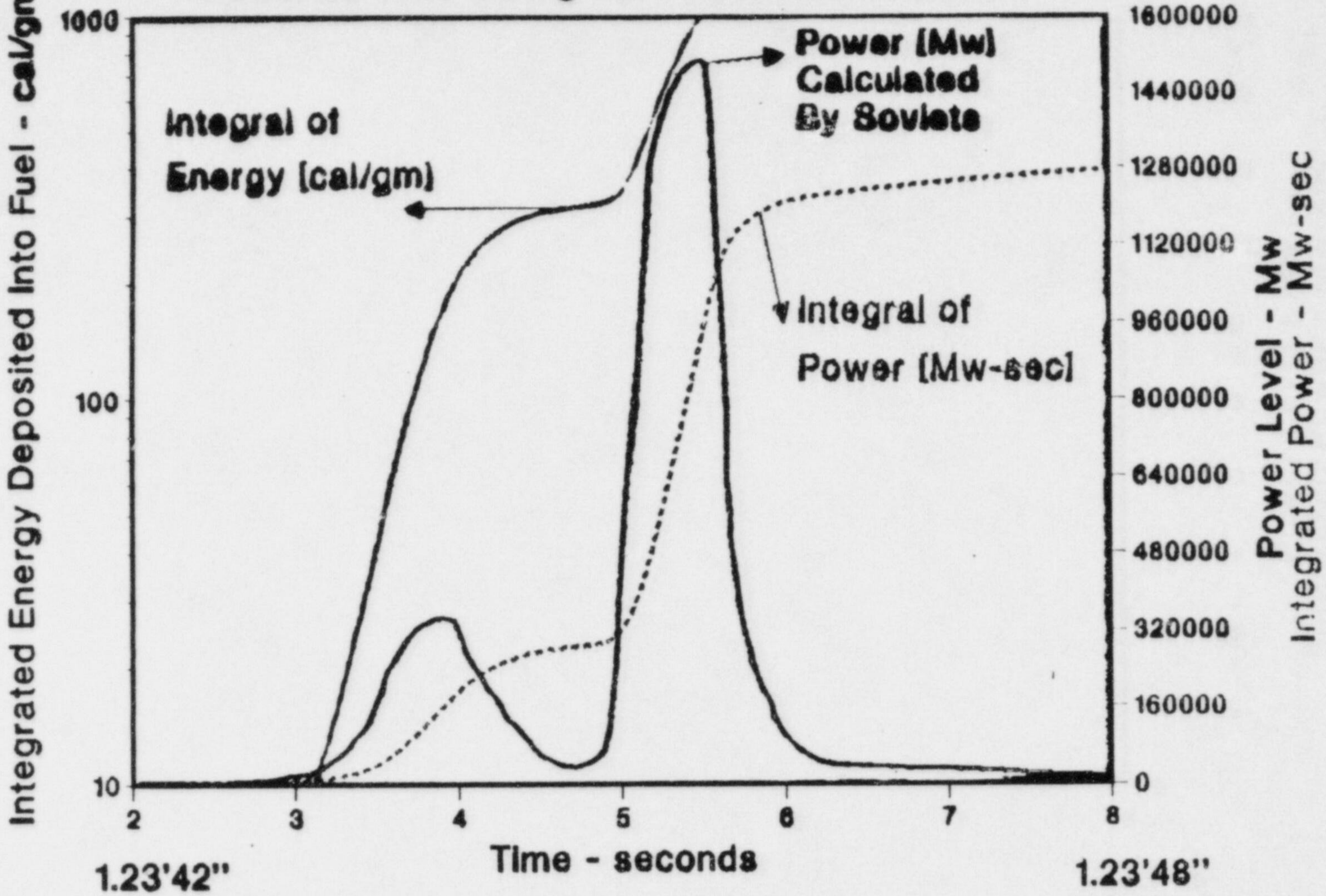
. THE SITE, THE TURBINE BUILDING ROOF AND THE SIDES OF THE ROADS TREATED WITH RAPID POLYMERIZING SOLUTIONS TO REINFORCE UPPER LAYERS OF SOIL AND PREVENT DUST FORMATION.

. THE NUCLEAR POWER PLANT SITE WAS DIVIDED INTO ZONES FOR THE PURPOSES OF DECONTAMINATION;

. DECONTAMINATION IN EACH ZONE CARRIED OUT IN THE FOLLOWING ORDER:

- REMOVAL OF DEBRIS AND CONTAMINATED EQUIPMENT FROM SITE;
- DECONTAMINATION OF ROOFS AND EXTERNAL SURFACES OF BUILDING;
- REMOVAL OF A SOIL LAYER, 5-10 CM THICK, AND TRANSPORTATION TO REPOSITORIES;
- LAYING, WHERE NECESSARY, OF CONCRETE SLABS OR FILLING IN WITH CLEAN SOIL;
- COVERING OF SLABS AND NON-CONCRETED PARTS OF SITE WITH FILM-FORMING MATERIAL;
- ACCESS RESTRICTION TO THE TREATED SITE

Chernobyl Data Evaluation
Power vs Time During Core Destruction Phase



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PROPOSED CONTAINMENT PERFORMANCE DESIGN OBJECTIVE

1. INTRODUCTION

At the 316th meeting, August 1986, I was directed to prepare a "straw man" Containment Performance Design Objective for consideration by the ACRS as part of its recommendations for design criteria for improved reactors.

This has been done, and is offered herewith. It is based on previous recommendations by the ACRS and is more or less consistent with what the Commission has said in its Policy Statement and what individual Commissioners have proposed.

The remainder of this report is arranged as follows:

2. Historical Review. An attempt to summarize previous ACRS recommendations and provisions or comments in the Policy Statement. The first may be incomplete; it begins in 1982.

3. Containment Performance Design Objective. A statement of the ground rules followed, a statement of the proposed CPDO itself, comments relating it to previous recommendations and the Policy Statement, and comments on its application and possible significance.

4. Design to Meet the CPDO. A brief list of some of the design changes that might be made or required to meet the CPDO.

5. Questions. A very brief and incomplete list of what we might need to know before such a CPDO is adopted.

2. HISTORICAL REVIEW

ACRS RECOMMENDATIONS

With no fear of prejudicing what we might say now, I will review briefly below what we have said about Containment Performance Objectives in previous ACRS letters.

(1) Letter of 9 June 1982: Comments on Proposed Policy Statement on Safety Goals for Nuclear Power Plants (NUREG-0880, a Discussion Paper)

> "For plants yet to be designed, it may be practical to set containment performance standards for accidents leading to large-scale-core melt, but not automatically involving a direct loss of containment integrity by

the frequency of core melt accidents that are directly coupled with an early loss of containment integrity is and must be kept very low, recent studies indicate that it is practical to establish stringent performance requirements on containment capability for other core melt accidents. We believe that additional study is needed before numerical guidelines are set for the containment of future plants."

(2) Letter of 15 September 1982: ACRS Comments on the NRC Staff Questions to the Commission Concerning the Policy Statement on Safety Goals for Nuclear Power Plants.

"We believe that, in view of the continuing uncertainties to be expected in the art of PRA and a continuing inability to satisfactorily treat all initiators and other contributors to core melt frequency, and in view of the potentially very large differences in release magnitudes among different core melt accidents, containment performance design objectives are needed and should be developed expeditiously."

(3) Letter of 15 September 1982: ACRS Report on the Draft Action Plan for Implementing the Commission's Proposed Safety Goals for Nuclear Power Plants.

"We believe that priority should be given to developing containment performance criteria for several reasons, including the following:

a. There are major uncertainties in the calculation of statistical health effects from very small doses to large numbers of people.

b. There are large uncertainties in calculation of accident dose. Evacuation models, for example, are fairly arbitrary and do not reflect the potential effects of earthquakes or offsite loss of power on the effectiveness of emergency actions.

c. Assumptions concerning land areas which would require interdiction and problems in large-scale decontamination require futher study.

d. Uncertainties in prediction of core melt frequency would be compensated, at least in part, by a containment having a significant potential to mitigate core melt accidents."

(4) Letter of 9 August 1983: ACRS Comments on Proposed Safety Goal Evaluation Plan.

"We observe that the proposed safety goals contain no design objective for containment performance. It is stated that the evaluation process will include a review of whether containment performance is to be a specific design objective. Discussions with the NRC Staff indicate that they have concluded that uncertainties in containment performance are too great to make a performance objective meaningful at this time. It is strange that the NRC Staff considers the uncertainty in describing the progress of a large scale core melt to be significantly less than the uncertainty in describing containment performance. We continue to believe that containment performance objectives are important as an indication of the need for mitigation, just as the core melt design objective is an indication of the emphasis on accident prevention."

(5) Letter of 17 July 1985: ACRS Comments on Proposed NRC Safety Goal Evaluation Report.

"The NRC Staff has not developed a containment performance guideline, nor has any serious NRC Staff effort to do so been apparent to the Committee. The ACRS continues to believe, as it did in its report of June 9, 1982, that the development of a containment performance guideline warrants high priority, and recommends that the Commission require early NRC Staff attention to this matter as part of maintaining its defense-in-depth principle. Approximate compliance to an appropriate criterion should be an NRC objective."

(6) Letter of 19 March 1986: ACRS Comments on Proposed Safety Goal Policy.

"In a severe accident, it is the releases from the containment which constitute the risk to the health and safety of the public. Thus, risk cannot be assessed without a judgment on containment performance. We reiterate our recommendations to develop a containment performance objective."

(7) Letter of 15 April 1986: Additional ACRS Comments on Proposed NRC Safety Goal Policy Statement.

"We believe the Commission should adopt certain performance guidelines as one satisfactory means to assure conformance with the safety goal objectives. these guidelines should be structured so that the principle of defense-in-depth is maintained....

"We propose that the plant performance guidelines be regarded as fully acceptable surrogates for the safety goal objectives.....

"There should be two performance guidelines and consideration should be given to the development of a third.

"The first guideline should be that the chance of a loss of adequate core cooling with consequent severe core damage should be less that 10E-4 per reactor-year for all but a few small reactors.

"The second guideline should relate to containment performance and should be such that the chance of a very large release of radioactive materials to the environment should be less than 10E-6 per reactor-year."

COMMISSION POLICY STATEMENT

On 30 July 1986, the Commission issued its "Final Policy Statement on Safety Goals for the Operation of Nuclear Power Plants." Portions of that Statement, or of the attached remarks of Commissioners Asselstine and Bernthal, relating to containment performance are cited in the following:

(8) Under the heading V. Guidelines for Regulatory Implementation.

> "....the staff will require specific guidlines to use as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy. .... The guidance would be based on the following general performance guideline which is proposed by the Commission for further staff examination--
>
> > "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

(9) Commissioner Asselstine, in his additional views, citing accident prevention and mitigation, defense-in-depth, and the ACRS, proposes "the following containment performance criterion:

> "In order to assure a proper balance between accident prevention and accident mitigation, the mean frequency of containment failure in the event of a severe core damage accident should be less than 1 in 100 severe core damage accidents."

(10) Commissioner Asselstine further addresses the "general performance guideline" of one in a million probability of a large release, and proposes that this be adopted as a performance objective. He then defines a "large release" as one "that would result in a whole body dose of 5 rem to an individual located at the site boundary". He points out that this is consistent with the EPA's emergency planning Protective Action Guidelines and thus would not require evacuation of the public."

(11) Commissioner Bernthal, in his separate views, proposes the following as being consistent with the Commission's defense-in-depth philosophy:

> "1) Severe core-damage accidents should not be expected, on average, to occur in the U.S. more than once in 100 years;
> "2) Containment performance at nuclear power plants

should be such that severe accidents with substantial offsite damages are not expected, on average, to occur in the U.S. more than once in 1000 years;

"3) The goal for offsite consequences should be expected to be met after conservative consideration of the uncertainties associated with the estimated frequency of severe core-damage and the estimated mitigation thereof by containment. [Footnote omitted]

" The term "substantial offsite damages" would correspond to the Commission's legal definition of "extraordinay nuclear occurrence" [5 rem whole body ?]. "Conservative consideration of associated uncertainties" should offer at least 90 percent confidence (typical good engineering judgment, I would hope) that the offsite release goal is met."

3. CONTAINMENT PERFORMANCE DESIGN OBJECTIVE

GROUND RULES

This "straw man" Containment Peformance Design Objective (CPDO) is intended to be used in connection with the ACRS recommendations for future plants. Earlier discussion suggests that this would be limited to future standard plant designs for LWRs. With this limitation, BWR Mark I and II containments and PWR ice-condenser containments would not be included. Some previous discussion indicated also that the proposed ACRS recommendations would not necessarily apply to existing standard plant designs, such as GESSAR II and CESSAR, or to possible reactivated CPs or replications of existing plant designs. However, there is nothing in the proposed CPDO that would not apply to such plants if they do not involve Mark I or II or ice-condenser containments. Nor is there anything that would preclude applying the CPDO to future HTGR or LMR designs.

PROPOSED CPDO

The following is proposed as a CPDO:

> The overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 10E-6 per reactor-year.
>
> A "large release" is one that results in a whole body dose of 5 rem to an individual located at the site boundary.

This objective is the same as that proposed by the ACRS in its letter of 15 April 1986 [1.(7) above] and included in the Policy Statement for consideration by the NRC Staff [1.(8) above]. It probably is not inconsistent with Commissioner Bernthal's proposal [1.(11) above], requiring 10E-3 per year for all plants in the U.S. with 90 percent confidence.
The definition of a large release corresponds to that offered by

Commissioner Asselstine [1.(10) above], and is consistent with Commissioner Bernthal's definition of "substantial offsite damage" [1.(11) above].

COMMENTS

Taken alone, this can be considered a "general performance guideline" [1.(8) above] rather than a containment performance design objective. However, in the ACRS recommendation [1.(7) above], this objective was coupled to a limit on frequency of severe core damage. Although the Policy Statement does not mention a meaningful limit on frequency of severe core damage, it does relate this performance guideline to defense-in-depth and "the accident mitigation philosophy requiring reliable performance of containment systems. In any case, it is highly unlikely that the 10E-6 probability can be met either without a containment or by the containment alone.

The definition of a large release allows no credit for evacuation. In fact, it is intended to define an accident that does not require evacuation. To this extent, it is intended to both protect and assure the public. If a severe core damage accident should occur--but no evacuation is required--some of the psychological trauma expected from a severe core damage accident might be avoided. This may be idealistic or optimistic in view of the possible bases upon which a decision to evacuate may be made, but it is clearly one reason behind the Commission's proposal.

An important, and perhaps controversial feature of this CPDO is the calculation of dose to a hypothetical individual at the site boundary. This, however, is essential if the CPDO is to be used to evaluate standard designs to be approved for use at unknown sites. Some assumption will have to be made about the meteorology, but this should be tractable since the release point is a function of the scenario and only local meteorology is needed.

I have no idea as to whether this limit on releases will satisfy the health effects safety goals. Most likely, it will be well below the quantitative goals of the Policy Statement. Such conservatism may be considered desirable or even necessary to offset the uncertainties in estimating the frequency of severe core damage accidents, their course, and the performance of containment systems.

Compliance with this CPDO will of course require a PRA. In fact it is the ultimate use of the "bottom line". Since a PRA will be required for all standard plant designs, and probably for any other future designs, this should introduce no more than the usual problems, including what to do about external events such as earthquakes and floods. In this respect, should some thought be given to external events of such magnitude that the event itself, absent the presence of the nuclear power plant, would require evacuation or result in catastrophic consequences?

For the most likely severe-core-damage accidents (SCDAs), the challenge to the containment is slow overpressure. If unmitigated, this scenario will lead eventually to failure of the containment, perhaps by gross rupture but most likely by large leakage. For this case, the CPDO will require that the containment either

(a) be able to resist the overpressure for a long enough time that the eventual release is not "large", or

(b) be vented prior to failure in such a manner that the release will not be "large."

For SCDA scenarios leading to very large pressures on the containment at early stages of the accident, it may not be possible to prevent containment failure. Such scenarios include steam explosion producing a missile, hydrogen detonation on a large scale, or direct heating. In such cases the CPDO must be met by insuring that the frequency of all such accidents is below 10E-6 per reactor-year.

Similarly, accidents involving containment bypass (Event V), pre-existing leakage, or failure to isolate cannot be mitigated by containment or containment system designs. Again, the CPDO must be met by keeping the frequency of such accidents, in total, below the 10E-6 criterion.

4. DESIGN TO MEET THE CPDO

Application of the proposed CPDO could lead to design changes of the following kinds:

1. Increased pressure capacity of the containment to prevent or further delay failure by slow overpressure.

2. Provisions for venting through a filtering medium, and procedures for doing so.

3. Provisions to reduce the probability of pre-existing leakage, such as continuous leak monitoring to detect gross openings.

4. Provisions to prevent containment bypass as a result of Event V sequences, if the frequency is not low enough.

5. Provisions to reduce the probability of hydrogen detonation, if it is not low enough.

It is likely, or at least we can hope, that provisions such as these will be either passive or procedural, and simple enough that complex analyses of their effectiveness will not be required.

5. QUESTIONS

1. Most of the discussion above relates chiefly to large dry containments for PWRs. What is not applicable to BWRs with Mark III containments? What should be added? How do Bernero's recommendations for BWRs fit in? Would GESSAR II meet the proposed criterion.

2. Can the proposed CPDO be achieved?

3. If so, how much margin will be provided against the quantitative safety goals?

4. For existing designs, will the doses for late containment failure under slow overpressure, or for venting, meet the definition of large releases?

5. Could the performance required by the proposed CPDO be met by requiring certain design features, as Bernero proposes, rather than by the proposed performance criterion?

APPENDIX XV

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, H. Etherington to ACRS members, Graphite-Steam Reaction at Chernobyl, September 9, 1986

2. Announcement No. 135, NRC Chairman L. W. Zech, Jr. to all NRC employees, Director, Office of Nuclear Regulatory Research, September 9, 1986

3. Report, USSR State Committee on the Utilization of Atomic Energy, The Accident at the Chernobyl' Nuclear Power Plant and Its Consequences, Part I. General Material (Information compiled for the IAEA Experts' Meeting, 25-29 August 1986, Vienna), August 1986

4. Report, USSR State Committee on the Utilization of Atomic Energy, The Accident at the Chernobyl' Nuclear Power Plant and Its Consequences, Part II. Annexes 1, 3, 4, 5, 6 (Information compiled for the IAEA Experts' Meeting, 25-29 August 1986, Vienna), August 1986

5. Report, USSR State Committee on the Utilization of Atomic Energy. The Accident at the Chernobyl' Nuclear Power Plant and Its Consequences, Part II. Annexes 2, 7 (Information compiled for the IAEA Experts' Meeting, 25-29 August 1986, Vienna), August 1986

6. Report of the International Task Force on Prevention of Nuclear Terrorism, a project of the Nuclear Control Institute, June 25, 1986