PDR 8/30/82
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261ST ACRS MEETING
JANUARY 7-9, 1982
WASHINGTON, DC

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JANUARY 7-9, 1982

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

December 31, 1981
(Revised)

SCHEDULE AND OUTLINE FOR DISCUSSION
261ST ACRS MEETING
JANUARY 7-9, 1982
WASHINGTON, DC

Thursday, January 7, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 9:00 A.M. — Opening Session (Open)
   1.1) Report of ACRS Chairman re. information related to ACRS activities (JJR/RFF)
      . ACRS presentation to the NRC Commissioners regarding the ACRS position re. reactor pressure vessel water level indicators
      . Ltr. to NRC Chairman re: improved SARs and SERs

2) 9:00 A.M. - 9:30 A.M. — Clinch River Breeder Reactor (Open)
   2.1) Briefing re. project status and the plan for regulatory review of the proposed site and plant design

3) 9:30 A.M. - 12:00 Noon — Boiling Water Reactor Standard Plant Design (Open)
   3.1) Briefing re. proposed changes in standard BWR Plant design
   (Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter)

12:00 Noon - 1:00 P.M. — LUNCH

4) 1:00 P.M. - 5:00 P.M. — NRC Safety Research Program Budget (Closed)
   4.1) Discuss proposed ACRS report to the U.S. Congress re. the proposed NRC Safety Research Budget for FY-1983 (CPS et al/SD et al)
   (This session will be closed to discuss matters which relate to the personnel practices of the agency and information the premature release of which would be likely to significantly frustrate proposed agency action)

5) 5:00 P.M. - 6:30 P.M.

Reports/Comments Regarding Regulatory Activities (Open)

5.1) Report of Subcommittee on Class-9 Accidents re. consideration of Class-9 accidents in the licensing process (WK/GRQ)

(Portions of this session will be closed as necessary to discuss information the premature release of which would be likely to significantly frustrate proposed agency action)

5.2) Proposed letter to the EDO regarding consideration of systems interactions in licensing reviews (DO/JMG)

Friday, January 8, 1982, Room 1046, 1717 H Street, NW, Washington, DC

6) 8:30 A.M. - 10:30 A.M. — Pressurized Water Reactor Standard Plant Design (Closed)

6.1) Briefing regarding proposed design changes in standard Westinghouse PWR nuclear steam supply system design

(This session will be closed to discuss Proprietary Information related to this matter)

7) 10:30 A.M. - 12:30 P.M. — NRC Safety Research Program Budget (Closed)

7.1) Discuss proposed ACRS report to the U.S. Congress re. the proposed NRC Safety Research Budget for FY-1982 (CPS et al/SD et al)

(This session will be closed to discuss matters which relate to the personnel practices of the agency and information the premature release of which would be likely to significantly frustrate proposed agency action)

12:30 P.M. - 1:30 P.M. — LUNCH

8) 1:30 P.M. - 3:30 P.M.. — NRC Safety Research Program Budget (Closed)

8.1) Discuss proposed ACRS report to the U.S. Congress re. the proposed NRC Safety Research Budget for FY-1983 (CPS et al/SD et al)

(This session will be closed to discuss matters which relate to the personnel practices of the agency and information the premature release of which would be likely to significantly frustrate proposed agency action)

9) 3:30 P.M. - 5:30 P.M.	Human Factors Consideration in the Design and Operation of Nuclear Power Plants (Open) 9.1) Report of ACRS Subcommittee on Human Factors (DAW/RKM) and ACRS consultants as appropriate 9.2) Presentation by and discussion with NRC Staff representatives as appropriate regarding NUREG-0700, "Guidelines for Control Room Design;" NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Review;" NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display Systems"
10) 5:30 P.M. - 6:00 P.M.	ACRS Future Activities (Open) 10.1) Discuss anticipated ACRS Subcommittee activity 10.2) Discuss proposed ACRS activities

Saturday, January 9, 1982, Room 1046, 1717 H Street, NW, Washington, DC

11) 8:30 A.M. - 10:30 P.M. NRC Safety Research Program Budget (Closed)

11.1) Discuss proposed ACRS report to the U.S. Congress re. the proposed NRC Safety Research Budget for FY-1983 (CPS et al/SD et al)

(This session will be closed to discuss matters which relate to the personnel practices of the agency and information the premature release of which would be likely to significantly frustrate proposed agency action)

12) 10:30 A.M. - 11:15 A.M. Human Factors Considerations (Open)

12.1) Discuss proposed ACRS comments regarding proposed NRC requirements (NUREG-0700, NUREG-0801, and NUREG-0835)

13) 11:15 A.M.- 12:00 NOON General Discussion (Open)

13.1) Discuss comments/views of ACRS members re. written report of Subcommittee on Reliability of AC/DC Power Supplies (See Memo from J.J. Ray to J. Carson Mark dtd. 12/10/81, Subject: Meeting of AC/DC Power Systems Subcommittee, October 30, 1981) (JJR/RS)

13.2) Discuss proposed changes in ACRS procedures re. conduct of committee activities (CM/RFF)

14) 12:00 NOON - 1:00 P.M. Reports of ACRS Members re. Foreign Regulatory Polices and Practices (Closed)

14.1) Meeting with Japanese nuclear regulatory/development agencies and manufacturers/utilities (PGS et al/RFF)

14.2) Meeting with Canadian ACNS (CM et al/RFF)

(This session will be closed to discuss information considered privileged and provided in confidence by a foreign source)

Note: The movies of the NRX (Chalk River) accident and the SL-1 recovery will be shown at 1:30 P.M. for available/interested members and staff.

to the Acting Administrator, Drug Enforcement Administration, United States Department of Justice, 1405 I Street, NW., Washington, D.C. 20537, Attention: DEA Federal Register Representative (Room 1203), and must be filed no later than January 28, 1982.

Dated: December 18, 1981.

Francis M. Mullen, Jr.,

Acting Administrator, Drug Enforcement Administration.

[FR Doc. 81-36809 Filed 12-24-81; 8:45 am]

BILLING CODE 4410-09-M

NATIONAL COMMISSION FOR EMPLOYMENT POLICY

ACTION: Notice of meeting.

SUMMARY: Under the provisions of the Federal Advisory Committee Act (Pub. L. 92-463, as amended) notice is given of the twenty-fourth meeting of the National Commission for Employment Policy at the Capital Hilton Hotel, 16th and K Streets, NW, Washington, D.C.

DATES: January 14, 1982, 1:30 p.m. to 6:00 p.m. and January 15, 1982, 8:30 a.m. to 4:00 p.m.

Status: This meeting will be open to the public.

Matters to be considered: New Commission members will be sworn in. Members will hear reports on current and planned work, as well as orientation sessions on the Commission's history and structure. Staff work on employment and training delivery systems, labor market problems of Hispanic-Americans and the development of a labor market policy for older workers will be presented. Presentations on the 15th will include congressional and administration viewpoints on the current state of employment policy development.

FOR FURTHER INFORMATION CONTACT: Mr. Ralph E. Smith, Deputy Director, National Commission for Employment Policy, 1522 K Street, NW, Suite 300, Washington, D.C. 20005 (202)-724-1553).

SUPPLEMENTARY INFORMATION: The National Commission for Employment Policy was established as title V of the Comprehensive Employment and Training Act Amendments of 1978 (Pub. L. 95-524). The Act gives the Commission the broad responsibility of advising the President and the Congress on national employment issues. Business meetings are open to the public. People wishing to submit written statements to the Commission that are germane to the agenda may do so, provided that such statements are in reproducible form and are submitted to the Director at least two days before the meeting and not more than seven days after the meeting.

In addition, members of the general public may request to make oral presentations to the Commission, time permitting. Such statements must be applicable to the announced agenda and written application must be submitted to the Director at least three days before the meeting. This application should include: name and address of applicant, subject of presentation, relation to agenda, amount of time needed, individual's qualifications to speak on the subject, and a statement justifying the need for an oral rather than written presentation.

The Commission Chairman has the right to decide to what extent public oral presentations may be permitted at the meeting. Oral presentations will be limited to statements of facts and views and shall not include any questioning of Commission members or other participants unless these questions have been specifically approved by the Chairman.

Minutes of the meeting and materials prepared for it will be available for public inspection at the Commission's headquarters, 1522 K Street, NW, Suite 300, Washington, D.C. 20005.

Signed in Washington, D.C. this 17th day of December 1981.

Ralph E. Smith,

Deputy Director, National Commission for Employment Policy.

[FR Doc. 81-36826 Filed 12-24-81; 8:45 am]

BILLING CODE 4510-30-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on January 7-9, 1982, in Room 1046, 1717 H Street, NW., Washington, DC. Notice of this meeting was published in the Federal Register on November 25, 1981.

The agenda for the subject meeting will be as follows:

Thursday, January 7, 1982

8:30 A.M.-8:45 A.M.: Opening Session (Open)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities.

8:45 A.M.-9:15 A.M.: Clinch River Breeder Reactor (Open)—Briefing by NRC Staff regarding project status and plan for regulatory review of the proposed plant site and design.

9:15 A.M.-12:00 Noon: Standard Boiling Water Reactors (Open)—The Committee will hear and discuss a report by representatives of the General Electric Company regarding design changes in the NSSS for their standard boiling water reactors.

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

1:00 P.M.-5:00 P.M.: NRC Safety Research Program Budget (Closed)—The Committee members will discuss the proposed ACRS report to the U.S. Congress on the proposed NRC Safety Research Program Budget for FY 1983.

This session will be closed to discuss matters which relate solely to the internal personnel rules and practices of the agency and information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy and information the premature release of which would be likely to significantly frustrate proposed agency action.

5:00 P.M.-6:00 P.M.: Reports of ACRS Subcommittees (Open)—The Committee members will hear and discuss reports of designated ACRS Subcommittees regarding the status of assigned activities including consideration of Class 9 accidents in the NRC licensing process.

Friday, January 8, 1982

8:30 A.M.-10:30 A.M.: Standard Pressurized Water Reactors (Closed)—The Committee will hear and discuss a report from representatives of the Westinghouse Electric Corporation regarding proposed changes in the NSSS of their standard pressurized water reactors.

This session will be closed to discuss Proprietary Information applicable to this matter.

10:30 A.M.-12:30 P.M. and 1:30 P.M.-3:30 P.M.: NRC Safety Research Program Budget (Closed)—The Committee members will discuss the proposed ACRS report to the U.S. Congress on the proposed NRC Safety Research Program Budget for FY 1983.

This session will be closed to discuss matters which relate solely to the internal personnel rules and practices of the agency and information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy and information the premature release of which would be likely to significantly frustrate proposed agency action.

3:30 P.M.-5:30 P.M.: Human Factors Considerations (Open)—The Committee will hear the report of its Subcommittee on Human Factors and consultants who may be present regarding proposed NRC requirements regarding design of nuclear power plant control rooms and safety parameter display systems.

5:30 P.M.–6:00 P.M.: Future Committee Activities (Open)—The members will discuss the proposed scope of and schedule for anticipated activities of ACRS subcommittees and full Committee activities.

Saturday, January 9, 1982

8:30 A.M.–10:30 A.M.: Safety Research Program Budget (Closed)—The Committee members will discuss the proposed ACRS report to the U.S. Congress on the proposed NRC Safety Research Program Budget for FY 1983.

This session will be closed to discuss matters which relate solely to the internal personnel rules and practices of the agency and information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy and information the premature release of which would be likely to significantly frustrate proposed agency action.

10:30 A.M.–11:15 A.M.: Design of Control Rooms and Safety Parameter Display Systems (Open)—ACRS report/comments regarding proposed NRC requirements for design of control rooms and safety parameter display systems.

11:15 A.M.–12:00 Noon: Reports of ACRS Members Regarding Foreign Regulatory Policies and Requirements (Closed)—Members of the Committee will report on recent activities related to foreign nuclear regulatory policies and practices.

This session will be closed to discuss information provided in confidence and considered privileged by a foreign source.

12:00 Noon–1:30 P.M.: Concluding Session (Open)—The members will complete discussion of items considered during this meeting.

Members will also exchange views regarding the reliability of AC/DC electrical systems in nuclear power plants.

Proposed changes in ACRS procedures related to the conduct of ACRS activities will be discussed.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 7, 1980 (45 FR 66535). In accordance with these procedures, oral or written statements may be presented by members of the public, recording will be permitted only during those portions of the meeting when a transcript is being kept, an questions may be asked only by members of the Committee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a telephone call to the ACRS Executive Director (R. F. Fraley) prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with Subsection 10(d) Pub. L. 92–463 that it is necessary to close portions of this meeting as noted above to discuss matters which relate solely to the internal personnel rules and practices of the agency (5 U.S.C. 552b(c)(2)), Proprietary Information relating to the matters being considered and information considered privileged and provided in confidence by a foreign source (5 U.S.C. 552b(c)(4)), information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)) and information the premature release of which would be likely to significantly frustrate proposed agency action (5 U.S.C. 552b(c)(9)(B)).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr. Raymond F. Fraley (telephone 202/634–3265), between 8:15 A.M. and 5:00 P.M. EST, 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 relating to the matter being considered (5 U.S.C. 552b(c)(4)), information which will be involved in an adjudicatory proceeding (5 U.S.C. 552b(c)(10)), and information considered privileged and provided in confidence by a foreign source (5 U.S.C. 552b(c)(4)).

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

Dated: December 21, 1981.

John C. Hoyle,

Advisory Committee Management.

[FR Doc. 81–36911 Filed 12–24–81; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Subcommittee on Advanced Reactors; Meeting

The ACRS Subcommittee on Advanced Reactors will hold a meeting on January 21 and 22, 1982, at the Argonne National Laboratory, Building 208, Room C–234, Argonne, IL. The Subcommittee will continue discussion regarding possible design considerations, issues, and criteria for future commercial advanced reactors and plans to prepare a report to submit to the ACRS. Experts in the field of risk perception and aversion will address/discuss those matters with the Subcommittee. Notice of this meeting was published November 25.

In accordance with the procedures outlined in the **Federal Register** on September 30, 1981 (46 FR 47903), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The entire meeting will be open to public attendance except for those sessions during which the Subcommittee finds it necessary to discuss proprietary information. One or more closed sessions may be necessary to discuss such information. (SUNSHINE ACT EXEMPTION 4). To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

Issue Date:

July 12, 1982

MINUTES OF THE
261ST ACRS MEETING
JANUARY 7-9, 1982
WASHINGTON, DC

CERTIFIED

The 261st meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Vice-Chairman J. Ray at 8:30 a.m., Thursday, January 7, 1982.

[Note: For a list of attendees, see Appendix I. P. G. Shewmon, M. Bender, M. S. Plesset and H. Etherington were not in attendance for this meeting.]

The Chairman noted the existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been received from members of the public to present either written or oral statements to the Committee. 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 St. N.W., Washington, DC.

[Note: Copies of the transcript taken at this meeting are also available for purchase from the Alderson Reporting Co., Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. Chairman's Report (Open to Public)

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

A. <u>ACRS Presentation to the NRC Commissioners Regarding Reactor Pressure Vessel Water Level Indicators</u>

The Chairman mentioned a December 29, 1981 memorandum to the ACRS (see Appendix IV) regarding participation in a briefing of the Commissioners tentatively set for January 4, 1982 to address reactor vessel water level indicators. The Committee suggested that C. Mark, W. Kerr, H. W. Lewis and J. Ebersole act as lead spokesmen for the ACRS at this meeting which was actually to be held at 1:30 p.m., on Friday, January 8, 1982. C. Mark and W. Kerr were to present the ACRS position; H. W. Lewis was to present his views as reflected in his added comments in the ACRS report of December 15, 1981 on the Paló Verde Nuclear Generating Station Units 1, 2, and 3. J. Ebersole was to present his views regarding the need to proceed with some system. Acting Chairman Ray solicited comments on a summary prepared by M. Libarkin concerning the history of ACRS comments regarding water level indicators (see Appendix V). H. W. Lewis asked that the precise summary of his position which was handed to Acting Chairman Ray be circulated for internal Committee use. His remarks dealt with the following:

. None of these instruments actually reads water level

. The purpose of instrumentation is to deal with future accidents, not to replay TMI

. Costs incurred without commensurate safety benefits are bound to extract a safety toll elsewhere

. Sheer weight of mandated change in the wake of TMI may have a negative component for safety.

H. W. Lewis expressed concern that the meeting might focus too narrowly to adequately address the subject. J. Ebersole suggested that the discussion be extended to inadequate core cooling.

B. Letter to NRC Staff Regarding Improved SARs and SERs

The Chairman referred to a December 31, 1981 letter from C. Mark to the Commissioners with suggestions for improved summaries in safety analysis reports and safety evaluation reports. R. F. Fraley indicated that the document was based on a poll of the Members (see Appendix VI). J. Ebersole endorsed the statements in the letter concerning the identification of issues shown to be difficult to resolve. W. Kerr and M. W. Carbon noted that they did not endorse the proposed changes since they did not believe it warranted the effort involved.

C. Admiral Rickover Retirement Letter

The Chairman circulated the draft of a proposed letter to Admiral Rickover taking note of his accomplishments and contributions to the nuclear program. The Committee endorsed the letter in principle and Members provided specific comments to the ACRS Executive Director to be incorporated (see Appendix XXI).

D. Future ACRS Activities

Acting Chairman Ray solicited comments on an M. Libarkin handout identifying topics of interest/concern to be considered during Operating License Reviews. R. F. Fraley indicated that this handout was a sort of generic standard review plan (see Appendix VII) provided to assist subcommittee chairmen in addressing items of interest/concern to their fellow members. Added topics (e.g. carry-over items from the CP review and items identified by the NRC Staff during its review) will also be brought to the attention of the designated subcommittee chairmen by the assigned project engineer on a case-by-case basis for consideration as topics during the subcommittee review.

E. Severe Accident Rulemaking and Related Matters

Acting Chairman Ray suggested that policy issue, SECY 82-1, Severe Accident Rulemaking and Related Matters, might be discussed during the discussion of ACRS future activities on January 8, 1982 (see Appendix VIII). A joint subcommittee meeting to review 82-1 has been set up with W. Kerr's Class-9 Accident Subcommittee and D. Okrent's Safety, Philosophy, Technology and Criteria Subcommittee.

II. Clinch River Breeder Reactor, Expedited Review (Open to Public)

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

H. Denton presented background on the CRBR project, indicating that it is probably 90% complete in terms of design and had approximately $500 million of equipment already delivered or on order. C. Mark asked whether environmental questions and site suitability would have to be reopened now that the CRBR review is underway again. H. Denton pointed out that the hearing on site suitability and environmental issues was never completed. However, he did not feel that update of the previous information would require a major effort to reach a conclusion.

P. Check, Project Manager and Director of the Special Program Office at NRC set up to handle the licensing of the CRBR, displayed a viewgraph of licensing milestones (see Appendix IX). D. Okrent cited an ambiguous statement in the Site Suitability Report concerning the capability of designing a 50 or 60 psi containment instead of the current 10 psi design. H. Denton suggested that buildings with different containment pressures could be considered at different sites provided that all necessary technical factors such as bearing loads are properly addressed and evaluated at the sites in question.

P. Check displayed an organization chart of the CRBR Program Office, showing the two sections led by C. Thomas in the Licensing Section and B. Morris in the Technical Section. D. Okrent expressed interest in the decision-making process in NRC Staff, set up to handle new technical issues with respect to the CRBR. P. Check explained that his Program Office will develop a complete array of the acceptance and review criteria which will include existing guidance in the Standard Review Plan for those portions of the plant where existing guidance remains applicable. He explained that an operating plan had been developed which is a combination review plan and schedule. P. Check indicated that he will make the decisions for the Staff with respect to technical matters with the guidance of the technical review groups within NRR. J. Ebersole expressed interest in the new set of general design criteria for the CRBR. P. Check indicated that they are tabulated in Appendix A of the Site Suitability Report. Members of the Committee were quite interested in the Dept. of Energy's exemption request. Of particular interest was the distinction between the LWA-1 and LWA-2 and its relationship to safety. P. Check explained to D. Okrent that the LWA-1 exemption would not involve the building of any safety-related structures.

P. Check introduced a chart which compared the reviewing requirements of the Staff of an exemption under 10 CFR 50.12 with respect to the LWA-1 and LWA-2 (see Appendix IX). J. Ebersole expressed some concern about the siting of the CRBR in the vicinity of the K-25 facility. H. Denton explained that that factor was considered by the reviewers at the time the site was selected.

P. Check showed a viewgraph entitled <u>Discussion Topics for the February ACRS Subcommittee Meeting</u>. He suggested that the meeting be dominated by safety issues concerned with the question of design criteria and review criteria. M. W. Carbon indicated that the Subcommittee Meeting was scheduled to be held in Washington, DC on February 2, beginning at 1:00 p.m. and continuing through February 3.

D. Okrent brought up a recommendation of some members of the NRC Staff made shortly before the previous review was terminated back in 1977. This recommendation concerned maintaining containment integrity for at least 24 hours given an accident involving a core melt. He inquired whether this was still an appropriate approach. H. Denton indicated that the core design had changed such that the previous values that the Staff used for evaluating a core destructive accident might not be applicable for the new design. But, he stated that the general idea that the plant be designed to sustain a core meltdown and the containment integrity maintained for a period of 24 hours is certainly unchanged. M. W. Carbon questioned whether the current review had proceeded far enough to identify the key or crucial technical issues. P. Check felt that the question of core disruptive accidents would certainly dominate, but the procedural question of acceptance criteria would also be important. H. Denton added that the resultant Los Alamos contract to look at DOE proposed changes since the first review was terminated should be available for the February subcommittee meeting.

III. Boiling Water Reactor Standard Plant Design (Open to Public)

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

A. Design Features of BWR/6-Mark III

D. R. Wilkins discussed with the Committee the current status and future course of the General Electric BWR design (see Appendix X). He listed four major principles in GE's approach to BWR design: continuous simplification of the design; standardization of an entire "nuclear island"; evolutionary rather than revolutionary change; and thorough "test before use" of new features. A series of slides were then shown which described advantages of direct cycle power production which included safety and economic advantages through simplicity, lower pressure, inherent reactivity control, and direct communication between water sources and the reactor vessel, the evolutionary design of commercial BWRs, and descriptions of several representative plants - Dresden 1, KBR, Oyster Creek, and Dresden 2. The Pressure Suppression Containment concept was described and its advantages in seismic, spacial and constructibility considerations explained. D. W. Moeller requested an explanation of an entry on the slide entitled, Increasing Regulatory Emphasis on Nuclear Island, which pointed out 1973 regulatory emphasis on occupational exposure. D. R. Wilkins explained that General Electric had deliberately laid out its STRIDE or Reactor Island design to respond to regulatory requirements as promulgated with particular attention to shielding, lay down space for equipment, and planned maintenance activities which will greatly reduce plant personnel occupational exposures. In answer to a question by Acting Chairman Ray, D. R. Wilkins indicated that the STRIDE design permits better electrical separation for routing cables from power sources to equipment in the plant to meet the increasing regulatory requirements for divisional separation.

The definition of the nuclear island design package being offered by General Electric was discussed. D. R. Wilkins explained that General Electric was not offering a turnkey design but did indicate that the architect/engineer would work for General Electric and GE would provide technical direction to the architect/engineer and assume responsibility for system integration of the entire nuclear island. D. R. Wilkins explained, in addition, that the owner has only to furnish cooling towers, the water supply for the towers, switch yards, the turbine generator and turbine building, and the service building. Within the service building GE provides the offgas system.

D. R. Wilkins discussed the principal features of the current BWR/6-Mark III. J. Ebersole questioned whether GE had developed evaporative cooling for the suppression pool which becomes hot in long-term transients. D. R. Wilkins indicated that that was a potential feature under study. D. R. Wilkins explained, in answer to a question by W. Kerr, that the containment verification testing program is directed at quantifying the containment loads in the suppression pool during loss of coolant accident conditions and during safety relief valve blowdowns to provide a design basis for the structural design.

In answer to an inquiry by Acting Chairman Ray, D. R. Wilkins indicated that this reactor system is designed to handle the ATWS events within the limits of the suppression pool and within the limits of the ATWS events. J. Ebersole expressed concern at GE's ability to depressurize the BWR/6 plant under all circumstances. He pointed out that the rapid depressurization D. R. Wilkins just described is done by opening relief valves by energizing solenoid valves. He suggested that GE should have handled the reliability of that depressurization process by getting rid of the solenoid valves; these were used on reactor designs about 20 years ago. D. R. Wilkins commented on environmental qualification of the solenoid valves.

D. R. Wilkins explained that if reactor vessel water level is maintained, decay heat removal in the BWR/6 is passive. Strong natural circulation internal to the reactor vessel and steam release to either the main condenser or to the suppression pool heat sinks combine to provide this passive decay heat. When the subject of reactor water level instrumentation was brought up, J. Ebersole concluded from the GE presentation that the loss of coolant accident was itself being allowed to destroy the redundant characteristics of water level instrumentation. M. W. Hodges, NRC Staff, indicated that his impression was that the BWR/6 had three sets of taps around the vessel and three trains of level instrumentation such that if a failure of one train caused the inoperability of a second train during an accident, one train of plots would still be available. The discussion that ensued was concluded by a request from Acting Chairman Ray that GE reply in writing concerning the redundancy of core level instrumentation.

D. R. Wilkins described symptom-oriented emergency guidelines for the BWR which would help to minimize the chance of operator error. D. A. Ward questioned whether GE saw any potential for automating the logic in the guidelines with a process computer. D. R. Wilkins presented a detailed description of simple effective operator interface displays but indicated that GE has no plans for automating emergency guidelines. J. Ebersole inquired concerning improvements in the scram discharge volume design such as redundant drain vent valves and diverse redundant water level indicators. D. R. Wilkins volunteered to provide a written explanation of its design of control rod scram discharge volumes. Committee members questioned G. Sherwood about inservice inspection of stub tube welds. W. Kerr was concerned that GE welds met the ASME code which includes inservice inspection. G. Sherwood indicated that the welds do meet the ASME code which requires inservice inspection but some stub tubes are not normally inspected because it is very difficult to do. J. Ebersole endorsed the concept of rapid pressure release embraced by GE of dumping steam to the suppression pool.

D. R. Wilkins discussed accident mitigation. He indicated that the function of containment at a nuclear plant in the case of the BWR pressure suppression containments is accomplished in two ways. The first way is through containment barriers, primary and secondary barriers which are designed to maintain their integrity for all design basis events and have sufficient margin to maintain their integrity for most events beyond the design basis. The second way concerns filtered containment venting or scrubbing of potential releases from the containment. Containment venting and scrubbing of potential releases are an inherent safety features of the Mark III pressure suppression containment. D. Okrent expressed concern that GE was not bringing before the Committee potential pathways or scenarios in which the fission product scrubbing would be ineffective and the decontamination factor of a thousand that D. Wilkins mentioned would be inoperative. D. R. Wilkins explained that GE is focusing its attention on probabilistic risk analyses to show that the probability of the existence of bypass pathways is acceptably small and can be neglected.

D. Okrent questioned whether GE could strengthen its containment in order to be able to handle a larger range of hydrogen events. D. R. Wilkins noted that strengthening the containment from a risk assessment or consequence assessment point of view would be hard to defend from a benefit cost point of view. He stressed the defense in-depth concept that GE uses. J. Ebersole expressed concern that a badly degraded core would segregate the internal dry well in the BWR/6 containment from the outboard side of the drywell and limit mixing such that at the end of an accident, if you were producing hydrogen or if you failed to cool the core, you would have a relatively concentrated source of hydrogen in the drywell compared to a mixed case. D. Wilkins explained that a mixer system did exist which circulates air from the wetwell region back into the drywell as part of the existing hydrogen control.

G. Sherwood of GE described the nuclear island design and suggested that GE is dealing with problems with which the ACRS is concerned such as systems interfaces, standardization and optimization, and in-depth system interaction studies.

D. Okrent questioned whether GE had looked at design features which would be useful in helping to reduce the likelihood of successful serious sabotage by an insider. D. R. Wilkins did not respond in detail because of privileged information that is involved. D. Okrent suggested that a closed meeting be scheduled in the future to discuss this subject. G. Sherwood described the nuclear island licensing program, the GESSAR program status, outstanding regulatory issues, and other generic licensing issues with respect to the BWR/6 Mark III (see Appendix XI).

It was explained that station blackout capability is extended by the containment overpressure relief function. Nuclear island failure modes and effects analyses will identify any needed corrections with regard to systems interactions. When G. Sherwood mentioned that GE would be ready to report on a full risk assessment in the near future, D. Okrent expressed the hope that GE would identify all of the areas where judgment was used in selecting either the parameter or the methods for determining inputs to the analysis including specifying the range of inputs that are possible. D. R. Wilkins indicated that GE planned to include an uncertainty analysis or error budget in its risk assessment which is aimed at addressing this issue.

J. Ebersole was concerned about providing reactor operators with Class 1E type separation and quality levels for operator indications, input on recorders, and indicators and enunciators. He questioned whether GE has taken steps in its STRIDE design to qualify and upgrade the information flow to the operator to 1E caliber. D. R. Wilkins indicated that GE is addressing the issue of reliability of information to the operator but did not plan to upgrade the equipment to Class 1E.

IV. Westinghouse Pressurized Water Reactor Standard Plant Design (Closed to Public)

[R. Savio was the Designated Federal Employee for this portion of the meeting.]

A. Introduction

Press Rahe, Manager of Nuclear Safety for Westinghouse, explained Westinghouse's strategy for seeking NRC approval of a new generation advance PWR design:

- utilize one-step licensing if available
- otherwise, use final design approval to freeze reactor design
- exploit expected market growth due to global oil situation.

He listed certain design objectives which include plant features to preclude core uncovery from small break LOCAs, passive secondary heat removal, a more sophisticated control room, lower power density, moderator control and greater thermal efficiency (see Appendix XII). P. Rahe explained that the Westinghouse design is based upon a nuclear power block concept which includes 70% of the plant and a design approach stressing the importance of safety. This design process allows a single source control of the design, engineering and construction as well as quality assurance of the plant. P. Rahe informed the Committee that Bechtel had been engaged to do the balance of plant for the nuclear power block concept as well as overseas turnkey plants. The Japanese firm Mitsubishi has been engaged as a joint vendor in the licensing and design of plants in Japan.

P. Rahe indicated that in meetings with the NRC Staff, Westinghouse had concluded that the NRC felt that the single source concept would enhance the licensing of this design. He indicated that a full probabilistic risk assessment (PRA) would be available at the time of the docketing of the standard plant design.

B. The Advanced PWR

R. Sero, Manager of Fluid Systems for Westinghouse, explained the Westinghouse approach to plant optimization at the conceptual design stage (see Chart in Appendix XII). He explained that, through the use of integration, systems which affect the safety of the plant have been designated as Westinghouse responsibility. He stressed that the two most important design objectives or considerations for Westinghouse are (1) public risk and, (2) the financial risk to utility customers. W. Kerr expressed concern about product quality, particularly concerning innovative approaches being described by Westinghouse, and quality assurance aspects. He felt that it would be extremely difficult to factor the effects of human error into a PRA analysis if Westinghouse were selling this system to various applicants. R. Sero explained that Westinghouse would model a typical operator in simulation. In concept, they would try to model the instrumentation and typical human actions. Use would be made of the Owners Group experience in training as well as information from the shift technical advisors.

DELETION -3

DELETION -3

DELETION -3

R. Sero discussed the plant protection system which had been subject to a verification and validation program submitted to the NRC with the British as a third party reviewer. He described features of the plant control system which is just an initial design.

DELETION -3

C. Licensing Schedule

D. Call, Manager of Plant Protection for Westinghouse, discussed the power block concept as a key element of licensing with Westinghouse responsibility for specification of safety portions of the plant but not necessarily fabrication or construction. R. C. Axtmann was informed that although the Japanese are using different design bases than those used in the U.S., fundamental criteria of design are still always met. D. Okrent pointed out that there were certain pitfalls in factoring in degradation, sabotage and aging into PRA analyses. He suggested that Westinghouse use PRA as a tool of analysis to determine ATWS design changes.

J. Ebersole expressed concern in regard to the mechanical protection of electrical equipment inside the containment. D. Call addressed this question by indicating that certain vulnerable electrical equipment could be moved outside the containment and assigned a new classification to obviate the safety/nonsafety dichotomy. It would be used on a case-by-case basis in assessing each system individually. C. Mark expressed concern as to whether any factors would cause Westinghouse to lessen its commitment to minimizing occupational exposure. D. Call indicated there would be a certain cost effectiveness economics factor involved in the analysis.

In response to an interest by Westinghouse in an ACRS report on this concept, Acting Chairman Ray suggested that it would be against ACRS policy to write a letter of endorsement to Westinghouse even though the Committee does encourage the advances Westinghouse is making. It was suggested that Westinghouse apply to the ACRS Executive Director to schedule appearances before the ACRS Westinghouse Subcommittee when significant milestones are reached in the design stage. The Committee agreed to discuss on Saturday the matter of how to handle the acknowledgment but did indicate that it was likely that the Committee could include a special paragraph in the Summary Letter commending the progress Westinghouse is making.

V. Report of Subcommittee on Class 9 Accidents (Open to Public)

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

W. Kerr summarized a proposed guide prepared as a result of ACRS subcommittee activities, entitled Outline of Rule for Severe Core Damage (see Appendix XIII). He indicated that the draft document resulted from the Class 9 Accident Subcommittee's consideration of the proposed NRC rule for Class 9 accidents and the licensing process. Members of the Committee disagreed on the use of the word "significant" used with regard to definition of the magnitude of a core melt. The Committee discussed the probability of 100% zirconium in the core being involved in metal/water reaction as proposed. It was suggested that metal/water reaction be considered a two stage process with the second stage containing longer term aspects which would be considered probabilistically. W. Kerr suggested that the rule deal with hydrogen deterministically as a sort of design basis and deal with the core melt as a probability. C. P. Siess and J. Ebersole commented on the use of containment inerting or post-accident inerting as means which preclude the collection of a detonable mixture of hydrogen in the containment. There was some question about the best core melt probability to use. Question was also cast on the wisdom of using an order of magnitude greater probability for the reliability of the decay heat removal system than a core melt. The section of the rule pertaining to the probability of release of more than 10% of the radioactive material in the core at shutdown was particularly confusing in that Mem ers could not reconcile the "96 hours after shutdown" with this probabilit, of release. Although the 96 hour figure was questioned, the Members did suggest that one could define containment performance probabilistically.

C. P. Siess questioned whether this rule would be consistent with the soon to be released Commission safety goals. Acting Chairman Ray summarized the discussion characterizing the draft as a satisfactory approach to the problem. He felt that W. Kerr should proceed with preparation of the guide taking into account the comments noted above and the assistance of one of the AEC fellows to develop related probabilities on a sound basis.

VI. Human Factors Consideration in the Design and Operation of Nuclear Power Plants (Open to Public)

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

A. Subcommittee Report

D. Ward explained that the purpose of the subcommittee meeting was to continue discussions of three NUREG documents prepared by the Division of Human Factors Safety (see Appendix XIV). The first document NUREG-0700, Guidelines for Control Room Design Reviews, was essentially a handbook of good engineering principles for control room design. D. Ward indicated that although the subcommittee was restrained in its acceptance of this NUREG, there was general agreement that it would be useful. The second document, NUREG-0801, Evaluation Criteria for Detailed Control Room Design Review, was described as an aid to the NRC Staff and licensee, a basis for reviewing existing control rooms and assessing human engineering discrepancies in control room designs. The third document, NUREG-0835, Acceptance Criteria for the Safety Parameter Display System, which gives functional criteria for the several different emergency response facilities which will be required, was generally endorsed by the subcommittee with two comments. One comment expressed reservations regarding the NUREG requirement that the SPDS incorporate a computer driven CRT display. The second comment indicated that the SPDS would be more effective if it included some automatic diagnostic capability.

D. Ward explained that the drafted proposal from the Committee to Review Generic Requirements (CRGR) for implementing the guidelines and requirements within these NUREGs favors issuing these documents as guidelines for review but not as strict requirements emphasizing negotiated changes between NRR and the Licensees.

B. Presentation by Division of Human Factors Safety

V. Moore defined the scope of the presentation as dealing particularly and specifically with NUREG-0700, NUREG-0801 and NUREG-0835. He pointed out that the Human Factors Program is also concerned with emergency operating procedures, for upgrading training and qualifications for operators, and for upgrading utility management of operations although these topics will not be discussed at this session. V. Moore presented a short historical perspective regarding the Kemeny Commission and Rogovin Committee which did present some human factors recommendations shortly after the Three Mile Island accident (see Appendix XV). Two specific recommendations of the Lessons Learned Task Force of the NRC Staff were cited: (1) that control rooms be evaluated and improved from the human factors standpoint; and (2) licensees install a safety parameter display panel (SPDP). V. Moore then discussed the Task I.D control room design in NUREG-0660, NRC Action Plan. Task I.D contained Task I.D.1, Control Room Design Reviews and Task I.D.2, Plant Safety Display Console.

V. Moore described NUREG-0700 as containing recommended procedures for planning, reviewing, assessing, implementing and reporting control room design reviews, plus human engineering guidelines. These four aspects were discussed in detail (see Appendix XV).

NUREG-0801 was prepared to be an aid to the NRC Staff and the licensee in determining the adequacy of upgrades to control rooms. He indicated that the bulk of the document was an assessment of human engineering discrepancies (HEDs). He asked the Committee to recognize that there is some flexibility and subjectivity in Staff criteria used in assessing correction of HEDs. W. Kerr questioned how much correlation there would be among different human engineers making a list of HEDs. L. Beltracchi of the Human Factors Staff indicated that there is a repetitive pattern in the correlation of many problems in reviews of control rooms that have been done so far. V. Moore added that in doing the near-term operating reviews, the utility does a preliminary design assessment identifying human engineering discrepancies, usually through the use of consultants. In general, the NRC subsequent reviews find all of the discrepancies found by the utility consultants in addition to others overlooked. In response to a question by D. W. Moeller, V. Moore described the composition of a five-man NRC team as composed of systems engineers, reactor engineers, as well as either human factors engineers or human factors psychologists. Also mentioned were Lawrence Livermore Laboratory and Biotec (their subcontractor) as consultants for these five man teams.

V. Moore described NUREG-0835 as the human engineering criteria or acceptance criteria for the SPDS. V. Moore explained that the function of the SPDS is to assist control room operators to assess whether the plant is in normal operation or to detect abnormal operating conditions which may impact safety. The requirement now reads that for the diagnoses and for the mitigation of abnormal operating conditions, the operators will have to use the SPDS. In answer to a question by C. Mark, V. Moore indicated that the SPDS should show between 7 and 15 critical safety parameters on 1 to 2 CRT displays. In answer to an earlier question by D. A. Ward concerning the fact that NUREG-0835 is limited to acceptance criteria for a computer-based, CRT-displayed SPDS, V. Moore indicated that this course of action was the result of conferences with owners groups, vendors, and individual utilities which all proposed the CRT-type system. V. Moore described the general acceptance criteria (see Appendix XV) which provide guidance for enhancing the operators performance and give guidance for developing displays. V. Moore mentioned section 4.0 of NUREG-0835, Specific SPDS Design Review Criteria, which shows specific acceptance criteria and has a one to one correlation between the functional criteria in NUREG-0696 (Emergency Response Facilities) and acceptance criteria in NUREG-0700.

J. Ebersole questioned whether the NRC is requiring applicants to upgrade the quality of the signals that are incoming to the Class 1E control systems. V. Moore indicated that the Staff is not requiring upgrading to Class IE, but is improving the quality of the signal to a direct reading rather than an indirect reading. V. Moore maintained that the Staff position regarding use of non-class 1E signal generators will not degrade the system because the Staff requirements do provide a better mix of information.

W. Kerr expressed concern that the Staff was not taking the lead responsibility in specifying some sort of reliability for the display systems. L. Beltracchi indicated that SPDS signals would be validated wherever possible with the indication of nonvalidation signified to the operator. He also noted the use of redundant sensors for operator comparison. J. Ebersole expressed concern that redundant sensors might contradict each other.

C. Presentation by the Committee to Review Generic Requirements (CRGR)

E. Blackwood explained that the CRGR has decided that there is need for greater initiative and improvements in coordination in the emergency response area with respect to facilities and capabilities. The CRGR has adopted initiatives in three broad categories. These include the SPDS. control room improvements (design review implementation of Regulatory Guide 1.97, Rev. 2, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, and Regulatory Guide 1.23, Meteorological Measurement Program for Nuclear Power Plants) and the emergency response facilities (including the technical support center, emergency operations facility, the onsite support center, and the nuclear data link). E. Blackwood explained that the CRGR believes that the hardware aspects of the emergency response should be addressed at this time. He described some broad milestones in the review by program offices which will result in EDO decisions and implementation of recommended requirements for emergency response facilities (see Appendix XVI). Proposed basic requirements for the SPDS were explained. The basic premise for these requirements is that seismic or Class 1E requirements are not necessary. No backfitting is anticipated with respect to these proposed basic requirements. With respect to control room improvements, E. Blackwood explained that CRBR would expect licensees to make a relevant cost benefit analysis.

W. Kerr questioned the cost basis for this proposal and asked whether a quantitative value could be placed on these future benefits to licensees. E. Blackwood suggested that the benefits of a control room design review would enhance safety by reducing the rate of human errors. It would have a positive effect on plant availability by a reduction in the rate of forced outages caused by emergencies. E. Blackwood, in answer to an inquiry by W. Kerr, indicated that parameters in Regulatory Guide 1.97, if essential to emergency response, should be treated as requirements by the licensees. He added that other than requirements for NRC approval under 10 CFR 50.59 which is the regulation on changes, tests and experiments, no prior approval of NRC is required for the licensees to make changes or modifications to improve control room designs.

D. A. Ward was concerned with the difference between NRC "post-review" and "pre-review" of planned changes. H. Thompson, Division of Human Factors Safety, indicated that the facility would identify proposed changes and discuss those with the Staff. Regulatory NUREG documents will be used in a flexible manner to evaluate which changes will or will not be made, and the justification for each. In one option, the licensee would then implement the changes in the control room. Under the alternate proposal, the NRC would not look at proposed changes until the licensee has completed his control room review and modifications. The licensee would document what he had done at the same time he presented documentation of his completed control room review to the Staff. D. A. Ward noted that this clearly shifts the burden of proof concerning the acceptability of modifications to the Staff away from the utility with respect to requiring additional changes.

D. Future Committee Actions

The Committee commended the CRGR and the Staff on its approach and took note of the fact that H. Thompson, Division of Human Factors Safety, had not requested a Committee letter with regard to this subject. In the absense of any formal request by the Staff and the fact that the whole Staff did not have a fixed position, the Committee decided that it was not necessary to write a letter, but to follow closely any final CRGR recommendations.

VII. Executive Sessions (Open to Public)

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

A. Subcommittee Summary of Activities

1. Human Factors Consideration in the Design and Operation of Nuclear Power Plants

The Committee heard an NRC Staff presentation regarding NUREG-0700 Guidelines for Control Room Design, NUREG-0801, Evaluation Criteria for Detailed Control Room Design Review, and NUREG-0835, Human Factors Acceptance Criteria for the Safety Parameter Display Systems. In the absence of any formal request by the NRC, the Committee did not elect at this time to write a letter regarding these NUREGs. The Committee did react favorably, however, to the Staff plan to refer these NUREGs to the CRGR for review. The Human Factors Subcommittee noted its intent to follow-up as appropriate regarding the CRGR comments.

2. ACRS Procedures from the Procedures Subcommittee Meeting on November 11, 1981

During the November 1981 (259th) ACRS meeting time did not permit discussion of Item II - Conduct of ACRS meetings; Use (authority) of ACRS Subcommittees; Use of Consultants; and Preparation (content/ scope) of ACRS Reports. The Committee took note of these items in the report of the Procedures Subcommittee Meeting but did not endorse the recommendations presented.

3. General Design Criteria Development for the CRBR

The Committee directed the CRBR Subcommittee to set up a series of meetings to follow the NRC review of the CRBR including consideration of the application of LWR General Design Criteria to the CRBR. M. W. Carbon agreed to discuss the subject at the February 2-3, 1982 CRBR Subcommittee meeting.

B. ACRS Reports, Letters, and Memoranda

1. ACRS Reports on Systems Interactions

The Committee prepared a memorandum to the EDO concerning systems interactions that might lead to significant degradation of safety. The ACRS requested an NRC Staff response with respect to Staff plans for a systems interaction study of Indian Point Unit 3, as well as for other nuclear plants.

2. ACRS Review and Report of the NRC Safety Research Program Budget

The Committee discussed and redrafted sections (Part I and Part II, Chapters 3, 9, 4, 6 and 7) of the proposed ACRS report to the U.S. Congress regarding the proposed NRC Safety Research Budget for FY-83. Additional discussion is planned for the 262nd ACRS Meeting to complete this report.

3. ACRS Comments on the Retirement of Admiral H. G. Rickover

The Committee endorsed in principle a letter to Admiral H. G. Rickover acknowledging his contributions to nuclear power.

C. Generic Safety Items

1. Postponed Generic items

The Committee endorsed follow-up of the R. Savio memorandum entitled "Generic Issue Letters" of January 8, 1982 (see Appendix XVII) which contained proposed generic items from ACRS reports issued during the 259th and 260th full Committee meetings as follows: A letter should be sent to the NRC Generic Items Branch, which cites the issues of occupational exposures at the Palo Verde Nuclear Units 1, 2, and 3 and other Combustion Engineering System 80 plants and shutdown heat removal from the Palo Verde type plants as generic issues. Followup of additional items by designated Subcommittees was endorsed. The ACRS Executive Director will revise subcommittee assignments to reflect these generic items.

2. Proposed Outline of Rule for Severe Core Damage

The Committee heard a presentation by W. Kerr of a proposed ACRS guide on Class 9 accidents and encouraged him to obtain assistance from an ACRS fellow to continue development of a more precise draft of this document.

3. Review of Nuclear Plant Security

Based on a brief discussion of Members' concerns regarding nuclear power plant security, it was suggested that an ACRS fellow should be assigned to perform a background survey of past practice and full Committee actions regarding plant security and antisabotage provisions requested by the Committee.

[Note: K. Kirby has been assigned to do this task.]

4. Participation in Briefing of NRC Commissioners Regarding Reactor Pressure Vessel Water Level Indicators

The Committee sent four Members, C. Mark, J. Ebersole, W. Kerr, and H. W. Lewis to a January 8, 1982 briefing of the Commissioners on Reactor Pressure Vessel Water Level Indicators (see background information in Appendix XVIII). ACRS views and the additional views of H. W. Lewis and J. Ebersole were expressed at this meeting which was reported as inconclusive. H. W. Lewis felt that the NRC Staff had moved prematurely in its endorsement of level indicators which may provide ambiguous level indication under some transient conditions. J. E. Ebersole noted that he favored the use of instrumentation (e.g., delta p cells) with which we are familiar from experience.

D. Future Schedule

1. Future Agenda

The Committee agreed to a tentative agenda for the 262nd ACRS Meeting, February 4-6, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

E. Vendor Briefings on the General Electric Boiling Water Reactor and Westinghouse Pressurized Water Reactor Standard Plants

The Committee members heard briefings by General Electric Company and Westinghouse Electric Corporation and, while they encouraged the advances made by both vendors in plant standardization, they felt it was contrary to ACRS policy to write letters of endorsement at this early stage of development. The Committee did agree to insert augmented paragraphs into its summary letter for the 261st ACRS Meeting regarding this activity. The General Electric and Westinghouse Subcommittees will follow the evolution and licensing of the Westinghouse nuclear island and General Electric STRIDE standard plant designs. In addition, the Safeguards and Security Subcommittee will evaluate these design concepts with respect to design features to reduce the threat of sabotage. The Chairmen of the General Electric (Ray) and Westinghouse (Shewmon) Subcommittees will participate in this aspect of the review to provide continuity. D. Okrent expressed special interest in a detailed discussion of the General Electric STRIDE provisions on this point.

F. Yarway Liquid Level Indication

J. Ebersole, with the assistance of J. McEvoy, an ACRS fellow, will prepare a paper regarding performance of Yarway Liquid Level Indicators used on boiling water reactors.

G. ACRS Report Recommendations

The Committee discussed briefly with Herzel H. E. Plaine, Legal Consultant, the degree of assumed acceptance implied by an ACRS failure to specifically mention a contested item in its report. Mr. Plaine explained that, since the Unresolved Items List has been abolished, the Staff can and does interpret such omissions as tacit acceptance or endorsement of a proposed NRC/licensee handling of these issues. Several Members took exception to this apparent doctrine of implied approval by default.

H. Conflicting Meeting Schedules

Conflicts in Members' attendance due to multiple proposed meetings of the CRBR, Safety Research, and Safety Philosophy Technology & Criteria/Class 9 Accidents Subcommittees on February 3 and March 3, 1982 were brought to the attention of the Committee. Members responded positively to M. W. Libarkin's suggestions for designated attendance (see memorandum, Appendix XIX). Revisions were made to the "Schedule of ACRS Subcommittee Meetings" report in accord with this January 9, 1982 Libarkin memorandum.

I. J. J. Ray Questions of General Electric With Regard to Their STRIDE Standard Reactor Island Design

J. J. Ray asked General Electric to respond in writing concerning redundancy of core level instrumentation. The question arose from comments by J. Ebersole regarding redundant water level instrumentation in accident situations. J. J. Ray also requested an explanation of the design of control rod scram-discharge volumes.

The 261st meeting of the Advisory Committee on Reactor Safeguards was adjourned on Saturday, January 9, 1982, at 1:00 p.m.

APPENDIXES
TO
MINUTES OF THE 261ST ACRS MEETING
JANUARY 7-9, 1982

ACRS- 1943

ATTENDEES
261ST ACRS MEETING
JANUARY 7-9. 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Jeremiah J. Ray, Acting Chairman
Robert C. Axtmann
Max W. Carbon
Jesse Ebersole
William Kerr
Harold W. Lewis
Carson Mark
William M. Mathis
Dade W. Moeller
David Okrent
Chester P. Siess
David A. Ward

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
William M. Baldewicz
Stuart K. Beal
William M. Bock
Paul A. Boehnert
Joseph Donoghue
Sam Duraiswamy
David C. Fischer
J. Michael Griesmeyer
Elpidio G. Igne
Morton W. Libarkin
John A. MacEvoy
Richard K. Major
Thomas G. McCreless
John C, McKinley
Thomas McKone
Austin Newsome
Gary R. Quittschreiber
Christopher Ryder
Richard P. Savio

NRC ATTENDEES

261st ACRS Meeting

Thursday, January 7, 1982

Nuclear Reactor Regulation

C. I. Grimes
P. S. Check
E. F. Goodwin
C. O. Thomas
R. Stark
B. Morris
S. Treby
P. H. Leech
M. W. Hodges
R. L. Tedesco
R. R. Bottimore
H. J. Faulkner
J. H. Conran
F. Coffman
P. Check
A. Notafrancesco

Friday, January 8, 1982

Nuclear Reactor Regulation

C. I. Grimes
M. Virgillo
D. R. Hoffman
R. L. Tedesco
E. F. Goodwin
R. Caruso
D. Beckham
H. Thompson
L. Beltracchi
E. Blackwood

APPLICANT ATTENDEES

261ST ACRS MEETING

Friday, January 8, 1982

Westinghouse Electric Corporation

D. Call
F. T. Johnson
P. Rahe
C. K. Kim
P. B. Haga
P. J. Morris
T. L. Schulz
R. J. Sero
E. T. Murphy
J. W. Miller

PUBLIC ATTENDEES

261ST ACRS MEETING

Thursday, January 7, 1982

R. Lyese, Electric Power Research Inst.
M. D. Newman, Corwell & Moring
P. Pomeroz, Scandpower
Bergler, Pacific Gas & Electric
G. Sherwood, General Electric
M. Shaw, LeBoeuf, Lamb, Leiby & MacRae
R. Vilia, General Electric
J. Quirk, General Electric
D. Wilkins, General Electric
R. W. Englehard, NUS
C. Ropp, Interdevelopment, Inc.
N. G. Chapman, Bechtel
J.Berga, Electric Power Research Inst.
R. Pennington, General Electric
P. Docherty, Westinghouse
C. Grochmal,Stone & Webster
P. Tremblay, NUS
R. Borsum, Babcock & Wilcox
J. Beach, Alderson

PUBLIC ATTENDEES

251ST ACRS MEETING

Friday, January 8, 1982

L. O. Lund, Lund Consultant, Inc.
A. C. Bivens, Atomic Industrial Forum
R. F. Pain, BioTechnology, Inc.
T. R. Kishbaugh, NUTECH
A. Hyde, Inside NRC
E. M. Howard, KMC
M. D. Newman
D. Browne, UCLA
S. Harclerode, Senator Domenici's Office
R. Leyse, Electric Power Research Inst.
A. Riesland
R. Ross, Doub & Muntzing
R. Liddle, General Physics
L. D. Wechsler, Bechtel Power Corp.
Saul J. Harris, EEI
J. Gagnon, NUS Corporation

APPENDIX II
FUTURE AGENDA

FEBRUARY

Reactor Safety Research Program Budget for FY 1983--ACRS Report to Congress	8 hrs
NRC Long Range Research Program Plan	2 hrs
Proposed NRC Policy Statement Regarding Severe Accident Rulemaking Regarding Standardization and Related Matters	4 hrs
Bolting Failures in Nuclear Plants, BNL Report and Proposed NRC Regulatory Guide	Deferred
Update of NRC Report to Congress Regarding Unresolved Safety Issues	Deferred
Proposed Regulatory Guide 1.23, "Meteorology Programs for Nuclear Power Plants	Deferred
Briefing by the NRC Office of Policy Evaluation of its Proposed Rule on Quantitative Risk Criteria	3 hrs
Briefing by the NRC Office of Policy Evaluation of its Proposed Policy and Program Guidance	1-2 hrs
Meeting/Discussion with Regulatory Reform Task Force	1-2 hrs
Clinton Station Units 1 and 2--OL	Deferred

Meeting with NRC Commissioners

- Discuss a policy statement regarding severe accidents
- Discuss meeting with the Regulatory Reform Task Force
- Discuss quantitative risk criteria and Commission safety goals

Waterford Station Unit 3--OL	Deferred

Subcommittee Reports

. Subcommittee on Zimmer Nuclear Station regarding QA/QC deficiencies and proposed ACRS action regarding its OL report of March 13, 1979 (MB/PAB)	Deferred to March
. Subcommittee on Extreme Environmental Phenomena regarding state of the art knowledge relative to the seismicity of the eastern U.S. and strong ground motion (DO/RS)	If time permits

APPENDIX II (Cont.)

- Subcommittee on Dynamic Effects regarding Mark III containment modifications and Mark I and II USIs (MSP/PAB) — If time

- Human Factors Subcommittee Chairman regarding NRC Staff interpretation of ACRS comments/recommendations regarding the composition of licensees safety review committees (DAW/RKM) — 1/4 hr

Future ACRS Activities

Meeting with RSK with likely topics of interest:

- Use of Probabilistic Assessment in the Design, Operation, and/or Licensing of Nuclear Facilites
- Development of Quantitative Risk Criteria
- Developments in Germany
- Reactor Pressure Vessel Thermal Shock

APPENDIX III

1/9/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JANUARY

18	Waste Management (Alderman/McKinley) - Moeller, Axtmann. Purpose: To review the technical assistance program in waste management research and discuss the NRC Safety Research Program budget for FY 1983.
21 & 22	Advanced Reactors (Argonne, IL) (Igne) - Carbon, Mark. Purpose: To continue discussion concerning LMFBR safety philosophy and issues and to prepare a report to submit to the ACRS. Drs. Slovic and Kassperson and Prof. Marrett will discuss the matter of risk perception and aversion.
22	Fluid Dynamics (Los Angeles) (Boehnert) - Plesset, Ward, Ebersole, Etherington, Mathis. Purpose: To continue review of Mark III Containment modifications and discuss status of USIs on Mark I and II Containments.
28 & 29	Extreme External Phenomena (Reston, VA) (Savio) - Okrent, Bender, Etherington, Mark, Moeller, Siess. Purpose: To review status of NRC's research program on geology and seismology and the status of research being performed outside of the NRC programs.

FEBRUARY

2 (pm) & 3	CRBR (Igne) - Carbon, Bender, Mark, Ray. Purpose: To review CRBR program status.
3 (8:30 am)	Nuclear Safety Research Program (Duraiswamy) - Siess, Kerr, Carbon, Okrent, Plesset, Shewmon, Mathis, Moeller, Ward. Purpose: To discuss the Long-Range Research Program.
3 (1:00 pm)	Safety Philosophy, Technology and Criteria/Class 9 Accidents (Griesmeyer/Beal/Quittschreiber) - Okrent, Ebersole, Kerr, Mathis, Ward, Siess, Axtmann. Purpose: To discuss Severe Accident Rulemaking.
4-6	262nd ACRS Meeting
9 (am)	Simulator Tour (Silver Spring, MD) (Major) - Kerr, Ward. Purpose: Visit Singer-Link Corporation.
10	Qualification Program for Safety Related Equipment (Boehnert) - Ray, Ebersole, Kerr. Purpose: To review the NRC Equipment Qualification Program Plan as outlined in SECY-81-504.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

FEBRUARY (CONT'D)

11	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Shewmon, Axtmann, Ray. Purpose: To discuss occupational radiation exposure in BWRs.
12	Joint Metal Components and Waste Management (Igne/Alderman) - Shewmon, Ray, Axtmann. Purpose: To review contractor technical capability and objectives of request for proposal on long-term performance of materials used for high-level waste packaging.
18	Zimmer Plant (Cincinnati, OH) (Boehnert) - Bender, Kerr, Mathis, Shewmon, Carbon (tent.). Purpose: To review QA problems associated with plant construction which resulted in $200,000 fine by NRC/I&E and to discuss plant operations.
CANCELLED	Safety Philosophy, Technology and Criteria (Griesmeyer/Savio) - Okrent, Bender, Ebersole, Kerr, Mathis, Ray, Ward. Purpose: To review the proposed Systems Interaction Study for the Indian Point Nuclear Power Plant.
22 & 23	Watts Bar (Knoxville, TN) (Griesmeyer/Quittschreiber) - Bender, Ebersole, Ward. Purpose: To review application for an operating license.
24 & 25 (tent.)	Byron Station 1 & 2 (Byron, IL) (Igne) - Shewmon, Bender, Mark. Purpose: Site visit and to review application for an operating license.
25 & 26	Clinton (Champagne, IL) (Savio) - Kerr, Axtmann, Ebersole, Moeller, Siess. Purpose: Site visit and to review application for an operating license.

MARCH

~~3 (am)~~ CANCELLED	Babcock & Wilcox (Major) - Ray, Ebersole, Etherington, Okrent, Plesset. Purpose: To explore with B&W changes that have been made to the ICS since the TMI-2, Crystal River 3, and Rancho Seco transients. Other improvements to the plant and plant operations will also be explored such as ATOG guidelines.
3 (pm)	Waterford (Beal/Quittschreiber) - Ward, Bender, Carbon. Purpose: To review Waterford organization, staffing, and training programs.
3	Regulatory Activities (Duraiswamy) - Siess, Kerr, Ray. Purpose: To discuss Regulatory Guides and Regulations.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MARCH (CONT'D)

4-6	263rd ACRS Meeting
Date to Be Determined (early March)	Joint Electrical Systems and ECCS (location to be determined) (Savio/Boehnert) - Kerr, Ebersole, Mark, Mathis, Okrent, Plesset, Ray, Etherington. Purpose: To continue review of the NRC- and Industry-sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.
Date to Be Determined (early March)	Decay Heat Removal Systems (Savio) - Ward, Bender, Carbon, Ebersole, Etherington, Ray. Purpose: To review the status of Task Action Plan A-45 and PWR Decay Heat Removal Systems with the emphasis on the CESSAR System 80 standard design.
Date to Be Determined (early March)	AC/DC Power System Reliability (Savio) - Ray, Ebersole, Kerr, Mathis, Okrent. Purpose to review the status of Task Action Plan A-44 and implementation of the recommendations of NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants."
Date to Be Determined (March)	Transportation of Radioactive Materials (Duraiswamy) - Siess, Mark, Bender. Purpose: To continue the review of the adequacy of the NRC procedures for certifying packages for the transporting radioactive materials.
Date to Be Determined	Human Factors (Major) - Ward, Bender, Lewis, Mathis, Moeller, Ray. Purpose: To review the various vendor SPDS designs, the Status of Disturbance Analysis Systems; to discuss ACRS concerns related to management, organization, staff's and technical resources for utilities that operate nuclear power plants; and to discuss NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures."
Date to Be Determined	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Bender, Kerr, Siess, Mark. Purpose: To review draft Commission Policy Statement on Safety Goals.
Date to Be Determined (June or July)	Joint CRBR and Site Suitability (Igne/Alderman) - Carbon Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, Ray. Purpose: To begin site suitability review for CRBR.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN. 18	Waste Management	(ALDERMAN) Moeller, Axtmann

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To review the technical assistance program in area of waste management research and to discuss the NRC Safety Research Program Budget for FY 1983.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Jan. 21 & 22	Advanced Reactors	(Igne) Carbon, Mark Consultants: Avery, Golden Lipinski, Hartung, Koch, and Siegel

LOCATION: Argonne, IL

BACKGROUND:

Who proposed action: Subcommittee

Purpose: To continue discussion concerning LMFBR safety philosophy and issues and to prepare a report to submit to the ACRS. Drs. Slovic and Kassperson and Prof. Marrett will discuss with the subcommittee the matter of risk perception and aversion.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-12

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN. 22	Fluid Dynamics	(BOEHNERT) Plesset, Ebersole, Etherington, Mathis, Ward

LOCATION: Los Angeles, CA

BACKGROUND:

Who proposed action: M. Plesset/NRC Staff

Purpose: Continue review of Mark III Containment modifications and discuss status of Unresolved Safety Issues on Mark I and II Containments.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided at a later date.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Jan 28-29	Extreme External Phenomena	(SAVIO) Okrent, Bender, Etherington, Mark, Moeller, Siess Consultants: E. Luco, B. Page S. Philbrick (28th only), P. Pomeroy, W. Maxwell, M. Trifunac, G. Thompson

LOCATION: Reston, VA

BACKGROUND:

Purpose: To review the status of the NRC's research program on geology and seismology and the status of research being carried out outside of the NRC programs. The purpose will be to identify the needs for future research in this area. The most likely format for this meeting is a symposium with participation from representatives of the NRC, USGC, various universities, and other organizations workin in this field.

PERTINENT PUBLICATIONS:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 2 (p.m.) & FEB. 3	Clinch River Breeder Reactor	(IGNE) Carbon, Bender, Mark, Ray Consultants:

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To review CRBR program status.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 3 8:30 am	Nuclear Safety Research Program	(DURAISWAMY), Siess, Kerr, Okrent, Plesset, Shewmon, Mathis, Moeller, Ward.

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action:

Purpose: To discuss the NRC Long-Range Research Program Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 3 1:00 pm Wash, DC	Safety Philosophy/Class 9	(GRIESMEYER/QUITTSCHREIBER/BEAL) Okrent, Ebersole, Kerr, Mathis, Ward, Siess, Axtmann.

BACKGROUND, ETC.

Who proposed action: D. Okrent

Purpose: To discuss Severe Accident Rulemaking.

Pertinent Publications

Memo from EDO to Commissioners regarding: Severe Accident Rulemaking and Related Matters, January 5, 1981.

Pertinent publications:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb. 9 (p.m.)	Simulator Tour	(Major) Kerr, Ward

LOCATION: Singer-Link Corporation, Silver Spring, Md.

BACKGROUND:

Who proposed action: W. Kerr

Purpose: To visit Singer-Link Corporation

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

ADDITIONAL DETAILS:

This will be an afternoon trip to Singer-Link Corporation located in Silver Spring, Maryland to observe several Nuclear Power Plant Simulators under construction, possibly witness a demonstration of one, and discuss the engineering behind the simulator with employees of Singer-Link. The tour will start and end at the ACRS Offices at 1717 H Street.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 10	Qualification Program for Safety Related Equipment	(BOEHNERT) Ray, Ebersole, Kerr

LOCATION: Washington, D.C.
(Federal Home Loan Bank Board Conference, Room: 1700 G St, N.W.; Fifth Floor)*

BACKGROUND:

Who proposed action: J. Ray

Purpose: To review the NRC Equipment Qualification Program Plan as outlined in SECY-81-504

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SECY-81-504 plus additional material to be provided later.

*To test integrated communications/recording system.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 11	Reactor Radiological Effects	(ALDERMAN) Moeller, Shewmon Axtmann, Ray Cons: R. Dillon

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller/P. Shewmon

Purpose: To discuss occupational radiation exposure in BWRs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. P. Shewmon memo to D. Moeller

2. SEC-81-517

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 12	Metal Components and Waste Management	(IGNE/ALDERMAN) Shewmon, Ray, Axtmann. Cons: Steindler, Orr, Rodabaugh, Readey, Dillon, Kassner

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Commission

Purpose: To review contractor technical capability and objectives of request for proposal on long-term performance of materials used for high-level waste packaging.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Request for Proposed RS-RES-81-173, "Long Term Performance of Materials Used for High-Level Waste Packaging."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 18	Zimmer Plant	(BOEHNERT) Bender, Kerr, Carbon (tent), Mathis, Shewmon

LOCATION: Cincinnati, OH

BACKGROUND:

Who proposed action: M. Bender/ACRS

Purpose: To review QA problems associated with plant construction which resulted in a $200,000 fine by NRC/I&E and to discuss plant operations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. I&E Investigation Report (to be distributed to Committee).
2. I&E Notification of Violations and Appraisal of Fines (distributed to Committee)
3. Other pertinent documentation as it becomes available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 22-23	Watts Bar	(GRIESMEYER, QUITTSCHREIBER) Bender, Ebersole, Ward

LOCATION: Knoxville, TN

PROPOSED BY: NRR

BACKGROUND:

Purpose: To review the Watts Bar for an OL.

PERTINENT PUBLICATIONS:

Watts Bar SER and Supplement (not yet published)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 24-25 (TENTATIVE)	Byron Station 1 & 2	(IGNE) Shewmon, Bender, Mark Cons: Kassner

LOCATION: Site Visit at Byron (24th). Subcommittee meeting nearby

BACKGROUND:

Who proposed action: NRC Staff & P. Shewmon

Purpose: OL review.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Safety Evaluation Report due 2/07/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 25-26	Clinton	(SAVIO) Kerr, Axtmann, Ebersole, Moeller, Siess

LOCATION: Champagne, IL
Site Visit at the Clinton site with a Subcommittee meeting near the site.

BACKGROUND:

Who proposed action:

Purpose: To review application for OL.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Safety Evaluation Report expected to be available by February 5, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3 (a.m.)	Babcock & Wilcox	(MAJOR) Ray, Ebersole, Etheringt Okrent, Plesset Cons. Catton, Ditto, Epler, Lipinski, Ybarrondo, Zuda

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: J. Ray

Purpose: The purpose of this meeting is to explore with B&W changes that have been made to the ICS since the TMI-2, Crystal River 3, and Rancho Seco transients. Other improvements to the plant and plant operations will be explored such as ATOG guidelines during this meeting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Cancelled

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3	Waterford	(Beal/Quittschreiber) - Ward, Bender, Carbon. Cons: Pearson, Binford

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: To review Waterford organization, staffing, and training programs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SER Supplement scheduled to be issued in January 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 3	Regulatory Activities	(DURAISWAMY) Siess, Kerr, Ray.

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To discuss Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Early March	Combined ECCS/Electrical Systems Subcommittee	(SAVIO/BOEHNERT) Kerr, Ebersole, Mark, Mathis, Plesset, Okrent, Ray, Etherington

LOCATION: To be determined.

BACKGROUND:

Purpose: To continue the review of the NRC and Industry sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.

PERTINENT PUBLICATIONS:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Early March	Decay Heat Removal Systems	(Savio), Ward, Bender, Carbon, Ebersole, Etherington, Ray

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: ACRS

Purpose: To review the status of Task Action Plan A-45 and PWR Decay Heat Removal Systems with the emphasis on the CE system 80 standard design.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Early March	AC/DC Power Systems Reliability	(Savio), Ray, Ebersole, Kerr, Mathis, Okrent

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the Status of the NRC work on Task Action Plan A-44 and the NRR Impementation of the reccommendation of NUREG-0666.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-31

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To Be Determined (March)	Transportation of Radioactive Materials	(DURAISWAMY) Siess, Mark, Bender Cons: Zudans, Langhaar, Shappert

LOCATION:

BACKGROUND:

Who proposed action:

Purpose: To continue the review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Date to Be Determined (March)	Human Factors	(MAJOR) Ward, Bender, Lewis, Mathis, Moeller, Ray. Cons: Arnold, Buck, Debons, Keyserling, Pearson, Salvendy

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: Topics to include reviewing the various vendor SPDS designs, the Status of Disturbance Analysis Systems, a discussion with representatives from industry on ACRS concerns related to management, organization, staff's and technical resources for utilities that operate nuclear pwoer plants, and a discussion of NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
TO BE DETERMINED	Reliability and Probabilistic Assessment	(Griesmeyer/Quittschreiber) Okrent, Bender, Kerr, Siess, Mark

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Office of Policy Evaluation

Purpose: To review draft Commission Policy Statement on Safety Goals.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June or July	Joint CRBR and Site Suitability	(Igne/Alderman), Carbon, Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, and Ray. Consultants (to be determined).

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To begin site suitability review for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Please delete the following pages A-36 thru __________ as deletion:

DELETION 1

The following pages has been deleted as deletion 6 - Predecisional information.

Pages A-37 thru ________

Please delete the following pages A.38 thru A.48 as deletion:

DELETION 1

Please delete the following pages A-49 thru A-51 as deletion:

DELETION 1

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

December 31, 1981

APPENDIX VI
SUGGESTIONS FOR IMPROVED SUMMARIES IN SAFETY ANALYSIS REPORTS AND SAFETY EVALUATION REPORTS

MEMORANDUM FOR: Chairman Palladino

FROM: J. Carson Mark, Chairman, ACRS

SUBJECT: SUGGESTIONS FOR IMPROVED SUMMARIES IN SAFETY ANALYSIS REPORTS AND SAFETY EVALUATION REPORTS

During the meeting with the Commissioners on December 11, 1981, the ACRS Members commented on various concerns regarding the presentation of information in Safety Analysis Reports (SARs) and Safety Evaluation Reports (SERs). Key among the concerns discussed was the need for a concise presentation of information in a manner that describes key features and emphasizes unique features or significant safety issues. To put some of the ideas discussed into more specific terms, the following suggestions for revisions to SARs and SERs are provided for your consideration:

1. Safety Analysis Reports - It is suggested that the SAR include a concise summary document describing the plant and related licensing information. This document should be a compact digest of SAR information that contains sufficient detail to be of value as an overview or reference document independent of the remainder of the SAR, while also serving as and improving upon Chapter 1 of present SARs. Little or no new information would be required for the document which should essentially provide a condensation of the SAR information into a compact, easy-to-use form by utilizing summary tabulations and illustrations to emphasize quantitative information and unique plant features. The following types of information should be included:

    a. diagrams of structures and systems important to safety (pictorial representations as well as engineering drawings should be included for perspective);

    b. a comparison of the major characteristics with those of similar facilities;

    c. a tabulation of the principal characteristics of such structures and systems, including plant features which differ significantly from established designs;

    d. a summary of site characteristics;

    e. a summary of the accident sequences analyzed and the results of such analyses;

f. a summary description of system performance requirements and design features; and

g. a list of national industrial standards, NRC branch technical positions, NUREGs and other similar requirements in addition to the Code of Federal Regulations and NRC Regulatory Guides used as the basis for design of substantive safety features.

A narrative text may be needed to provide a brief description of the information with appropriate amplification of non-quantitative aspects. The level of detail of the document should be such that it is of general value for licensing review purposes and reference by utility engineers, perhaps with cross-reference to more detailed sections of the SAR. Hopefully, this document would be smaller than a typical, single SAR volume.

2. Safety Evaluation Reports - Suggestions for improvement in the SER include providing some additional information and reducing the amount of standardized text ("boiler-plate"). Since these suggestions, specifically noted below, may impact on the concept of the SER as a "stand-alone" document integral to the hearing process, alternative methods of addressing these suggestions may be required.

   a. Include in the SER an identification of those issues with major safety significance where divergence in technical viewpoint between staff members or between licensee representatives and staff members required "out-of-the-ordinary" resolution actions. When appropriate, discuss the nature of the issue, the judgment considerations, and the resolution basis. It may be appropriate for the ACRS to comment on such issues while they are still in a "predecisional" status.

   b. If a compact summary description of the plant were to be provided in the SAR and were readily available to the ACRS, ASLB, and other users, the SER could exclude routine descriptive material and focus its discussion on the substantive safety issues. Also, in many cases a simple checklist of review matters that could be signed by responsible reviewers would provide a suitable record of review actions for legal purposes in connection with license hearings and eliminate repetitive narrative. If the checklist and sign-off were organized to cover major features such as containment, ECCS, emergency power, emergency heat sinks and site phenomenological characteristics, the reviews could be made much more systematic. Such lists could note exceptions or unusual aspects of the review to be covered in the SER narrative.

c. The SERs now provide some comparisons with other plants in tabular form and these help in establishing a reference basis for the review. A categorized tabulation of special features, peculiar to the plant being reviewed, would help to show the areas in which safety matters were given unusual attention. For example, if the plant design is based on a particular steam generator feedwater pump arrangement and the NRC reviewers prefer a different scheme, then a comparison would make the issue clear to the SER reader without the need to refer back to bulky reference material.

Since SARs and SERs, the primary reference documents used in the licensing process are bulky, awkwardly organized, and difficult to scan, improvements in the presentation and clarity of their contents could be of significant value. In addition, a more comprehensive discussion of the safey logic concerning technical issues could help in improving ACRS effectiveness. ACRS representatives would be pleased to meet with representatives of the NRC Staff to pursue this matter further, including consideration of a pilot demonstration.

cc:
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
S. Chilk, SECY
W. J. Dircks, EDO

Please delete the following pages A-55 thru ____________ as deletion:

DELETION 1

POSSIBLE SUBCOMMITTEE REVIEW ITEMS

. Status of the Applicant's Compliance with the Requirements of the TMI Action Plan - Should address completed items, schedule commitments and proposed resolutions.

. Seismic Reevaluations Since CP Letter - The OL plants which received a CP before the implementation of 10 CFR 100, Appendix A are being reevaluated against Appendix A and Regulatory Guide 1.60. The approach which is being used to generate site-specific spectra and then to reevaluate the seismic integrity of a representative sample of plant structures/components should be discussed. In cases examined by the ACRS to date, the site-specific spectra have been accepted, but the reevaluation of the seismic integrity of the plant on an "accident" basis had not begun. The subcommittee should probe the plans for reevaluation of all plant components and structures necessary for safe shutdown and decay heat removal. In the cases of North Anna, Davis Besse 2, Sequoyah, and Summer, such recommendations were made.

. Implementation of NSSS Vendor-Generated Degraded Core Cooling Emergency Procedures (including operator training) - The NSSS vendors are writing guidelines for such procedures and are supplying them to the utilities for implementation on a plant-by-plant basis. Individual utility implementation and interfacing with the balance-of-plant design would be of interest. The Subcommittee should ask about:

- the mechanisms which will be used to incorporate such guidelines into plant-specific procedures;

- the mechanisms to be used to tie vendor-generated guidelines into the balance of plant;

- the schedule for writing the actual emergency procedures;

- how the procedures are to be incorporated into the training program and on what schedule (Similarly, for that part of the training program to be conducted on a simulator);

- the internal or external review process to be used in finalizing such procedures (i.e., is the NSSS vendor/AE to review the final, detailed, plant-specific procedures?) On what schedule will any of the above be carried out?

- will any event-based (as opposed to symptom-based) procedures continue to be used? How is it decided which type will be used in any given circumstance?

. Use of an Unfiltered Vent[1]- GE is proposing the use of an unfiltered vent on Mark III containment to mitigate the consequences of accident

[1]Applicable to BWRs with Mark III Containments, only.

sequences which result in the overpressurization of the containment before the release of significant quantitites of radioactive material. In view of the fact that this has been identified by GE as a worthwhile safety improvement, the subcommittee should ask the applicant to address any plans to install such a system, as well as guidelines which have been (will be?) established for its use. Any investigations made by the applicant into the risk-reduction potential of this system should also be examined.

- Reactor Vessel Level Indication[1]

  - What device has been chosen for a vessel water level indicator? What was the basis for this decision?

  - What role will the information obtained from this system play in assisting the operator to deal with degraded core cooling scenarios? What uncertainties are associated with the use of this information?

  - To what extent has the use of this instrument system been incorporated into operator training and emergency procedures?

- Capability of Decay Heat Removal Systems to Cope With Plant Transients or Degraded Conditions

  - What is the applicant's assessment of the plant's decay heat removal system to cope with the above, and what is the basis for this assessment? Has the applicant accepted the NSSS/AE design or has he imposed his own acceptance criteria? On what basis has the applicant concluded that the plant's system is acceptable?

  - The NSSS vendors are preparing guidelines for the removal of decay heat under degraded plant conditions. What is the status of the NRC Staff's review of this material? How has the applicant incorporated these guidelines into plant emergency procedures and considered the balance of the plant?

- AC/DC Power System Reliability

  - The NRC Staff has issued a report (NUREG-0666) on the reliability of DC power system in which a 2-train DC system found to meet minimum NRC requirements was evaluated. As a result, the DC power system was identified as a potentially high contributor to core melt. The applicant could be asked with his assessment of his DC system is and what consideration he has given to the recommendations of NUREG-0666.

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[1] For PWRs only.

- What is the applicant's assessment of grid reliability and what procedures exist for restoring offsite power to the plant in the event of this loss.

- What are the results of the applicant's station blackout analyses? Has the applicant made a best-estimate analysis of the accident sequence and evaluated what might be done to improve the plant or has a conservative analysis been made with a core melt assumed after some specified degradation of the battery?

- What is the applicant's assessment of his diesel generator system? To what extent has LER and operating experiences been used to improve the design?

- Has the applicant performed low power testing and a simulated loss of offsite power test? If so, what are the results and what has the applicant learned?

. Hydrogen Control - The Regulatory Staff should be asked the current status of the proposed rule, as it will apply to large dry containment and GE Mark III Containments.

Results of any analyses made in compliance with either the Rule adopted for other than Mark III pressure suppression containment or with the proposed Rule should be examined. Were any modifications required/ made as a result?

The Subcommittee should look into procedures which will be established for the use of any equipment installed to deal with large hydrogen releases. If a "burn" is proposed, the survivability of vital equipment must be demonstrated, or a plan and schedule for such a demonstration should have been developed.

. Anticipated Transients Without Scram - The proposed ATWS Rule was recently issued (Dec. 1981) for public comment. The Rule contains two alternatives (the "Hendrie Rule" version which relies on PRA-based resolution of ATWS and the NRC Staff recommendations which are hardware-based fixes). In addition to the above, there is a third Rule proposal from the "Utility Group on ATWS" (a group of 20 utilities) that was published in the Federal Register.

Questions/topics to explore include:

- Discussion of the proposed Rule alternatives with the applicant asked to express his preferred alternative(s) and defend his preference.

- Has the applicant committed to implement whatever requirements are specified in the final ATWS Rule?

- (For BWRs)
  Has the applicant explored "unusual" ATWS mitigation methods such as enriching the concentration of B_{10} isotope in SLCS?

- (For CE plants)
  Does your ATWS analysis indicate the possibility of vessel "head lift" due to high pressure? If so, how will you cope with this event and safely shut the plant down?

- (For W plants)
  Have you explored the feasibility of modifying the scram breaker configuration to increase scram system reliability?

- Explore with applicant his opinion of the reliability assurance program and PRA techniques advocated in the Hendrie Rule. If he is not in favor of them (most likely), ask why he cannot dovetail an ATWS-PRA in with the overall PRA now being done by most OL plants.

- (For B&W plants)
  What is the calculated peak pressure for an ATWS? Is there a vessel "head lift" problem similar to CE's? Can the plant be safely shut down given the above peak pressure and/or head lift?

• Suppression Pool Dynamics - This topic is applicable to Mark I, II, and III plants. (Only 1 Mark I plant, Hope Creek, remains for OL review).

Topics/Questions for Mark II and III plants include:

Mark II

- Applicant should discuss compliance to NUREG-0808 (Mark II Containment Program Load Evaluation and Acceptance Criteria), and NUREG-0783 (Suppression Pool Temperature Limits for BWR Containments).

- Discuss the status of results of the BNL technical assistance contract, led by NRR, to study the consequences of a failure of the SRV line in the wetwell airspace. The study will analyze the consequences based on a PRA approach.

- Inquire whether or not the plant under review has had to modify the downcomer vacuum breaker valves due to potential failure during "chugging". (Applicable to Mark II plants with vacuum breakers attached to downcomer pipes.) If modification is necessary, discuss details of fix.

- The applicant may have elected to make "plant specific" modifications that are exceptions to the criteria specified in NUREG-0808. The applicant should detail any exceptions he plans to take. NRC should discuss the acceptability of these exceptions and the applicant should discuss their analyses to assure the exceptions are not deleterious to plant safety.

Mark III

The Mark III Program is still under review. This has not been handled as a Unresolved Safety Issue because there was deemed to be sufficient time to develop suitable load acceptance criteria prior to operation of the first Mark III plant. The program is near resolution with the exceptions noted below:

- Items of disagreement between the NRC and GE (as of two months ago) include: pool swell velocity used in load definition, froth drag on HCU floor gratings, and pool swell impact loads. The Subcommittee should explore the resolution of these specific areas with the Mark III plant applicant.

- The applicant should be questioned on the consequences of loss of HCU floor vis-a-vis safely shutting the plant down (vital equipment survivability).

- Ask what are the consequences of loss of the "pool dump" capability for long-term, post-LOCA cooling and explore the steps taken to prevent loss of dump function.

- Ask if the applicant will perform in-plant tests of SRVs to assure no unusual loads are generated on containment (plant under review may reference tests conducted at another plant termed "identical" to plant under review).

- Inquire as to what "plant specific" modifications/exceptions are being taken to generic load acceptance criteria. Assure that these exceptions have been carefully analyzed.

. Instrumentation and Control System - The Subcommittee should ask whether the Rancho Seco event has been investigated in detail, with an eye toward revealing similar vulnerabilities. Is it clear that this plant is not similarly vulnerable? Why?

Has a failure modes and effects analysis been done? With what result? Were any changes made as a result?

Can the plant be brought to (hot standby) (cold shutdown) with equipment independent of the control room? If so, the system(s) to be used and their reliability and independence from the normal control system should be investigated. Were considerations of sabotage vulnerability factored in?

How would the control room operators recognize an instrument failure?

- Implementation of Regulatory Guide 1.97, "Instrumentation to Follow the Course of a Serious Accident." The applicant should be asked to discuss any areas of non-compliance and the bases therefor.

- Performance of PRA - The Subcommittee should ask about:

  - any PRAs performed by the applicant. Who actually performed the analysis? Does the applicant have any PRA experience/ expertise on its own staff? Were any modifications of plant or procedures made or are any planned as a result of PRAs?

  The Subcommittee should also probe the extent of the operational and management staffs' knowledge of the PRA (or WASH-1400) sequences and vulnerabilities.

- Performance of Systems Interaction Studies - Have any Systems Interaction studies been done or are any planned? Were they (will they be) analytical or "walkthrough", or both? Were any plant modifications made as a result? Will the outline for such studies which was provided in the Committee's letter on the Indian Point 3 SIS, 10/12/79, be followed? (copy attached).

- Plant Security - What are the security-related plant design features; are security features "add-ons" or were they considered in the original design? Has the interaction between plant security and other safety considerations been considered? In particular, conflicting administrative requirements (e.g., controlled access areas) between safety and security should be examined.

  Have any plant features been incorporated to reduce sabotage vulnerability other than those related to access control?

  What is the local legal situation with regard to the use of "deadly force?"

- Emergency Planning - Both the NRC Staff and Applicant should be asked to address:

  - Support facilities and organization of support personnel.

  - Coordination with FEMA and State and Local authorities.

  - Experience from emergency drills.

  - Impact of projected popluation growth.

  Does the emergency plan contemplate the use of KI or other blocking agents? How are such decisions to be made? (Both NRC and the applicant should be asked to discuss this subject in general, as well as answer the specific questions posed)

Have local hospitals agreed, and are they adequately equipped, to handle contaminated accident victims?

- Organization and Management - A review could be structured around the following issues:
  - Construction management and construction QA/QC.
  - Interfaces with outside support organizations (NSSS Vendors, AE, NRC, and Industry consultants).
  - Plant Management and utility qualifications.
  - Training programs.
- Operations Staffing - To be considered are:
  - Operations staff and planning program.
  - Operators experience with similar type BWRs and/or additional support for start-up programs and first year of operation.
  - Organization capability for developing and retaining an adequate operations staff; current, on-board operations staff.

Specific questions to be raised are:

- The chain of command within the plant management organization with a clear delineation of final decision authority in operating, engineering, and safety matters.
- The degree to which full-time utility personnel are skilled in the fields of material properties, nuclear system controls, waste management, fluid mechanics, energy transfer and transport, electrical distribution, rotating machinery, and other important areas. If some part of these capabilities are to be provided by other than full-time utility personnel, how will they be factored into operational planning for the plant?
- What is the staff development plan, including the time prior to plant operation at which recruitment and integration of the operating staff with contractor personnel will take place?
- What will be the communication arrangement for normal operations and for maintenance, refueling, testing and emergencies?

- What arrangements have been made for the services of review committees, outside consultants, and supporting contractors? How will such committees be constituted? In what circumstances will they be used? What mechanisms will exist for utilizing or considering their advice?

- The applicant should be requested to provide information (as in the material which David Fischer has prepared in recent OL reviews) on the organization's structure. A presentation on the advantages and disadvantages of the particular structure chosen should be requested.

- What criterion will be used to determine that the operating organization is capable of taking over from the "startup" crew?

- To the extent that contactor personnel are used to provide operational capabilities, how is their competence established by management?

- Will the training program be a contracted or in-house operation? What qualifications must the Director of Training (or equivalent) have?

- To what extent does management view the development of a competent operating organization as different from simply passing the NRC RO and SRO exams? Have other criteria been set for the judgement of the competence of operators and other in-plant personnel? What are they?

- What fraction of candidates survive the training program(s)? Is there any pre-training screening process?

- Is simulator training used? What fractions of such training are devoted to normal operating evolutions, to transients, and to situations approximating DBAs and beyond? What confidence does the utility have in the simulation of situations beyond the DBAs?

- At multi-unit sites, is a given operator qualified to operate all units? Is this expected?

- What interfaces/feedback exists between the preparation of operating procedures and the training programs?

- What is the status within the organization of the Training Coordinator or equivalent?

. Review of Construction Experience - The Subcommittee should review, with the applicant and I&E, the construction experience. A list of construction deficiencies should be requested.

Has an independent audit been performed of the conformance of the plant as constructed, with design?

In view of the numerous lapses in the Diablo Canyon case, why does the applicant feel confident of his QA/QC program?

. Equipment Qualification:

- NUREG-0588, Environmental Qualification of Electrical Equipment is currently under review by a Subcommittee established for this purpose.

- Qualification of equipment for a hydrogen burn environment has also been a subject of recent interest to the Committee.

Other items which should be questioned include:

- a description of the valve qualification program;

- the specific flood, fire, and other conditions for which equipment and cabling have been qualified;

- will DC-operated equipment actually function on loss of AC power, including such conditions as loss of ventilating air?

- the QC program used to assure that contractor-provided equipment is acceptable.

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

October 12, 1979

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

SUBJECT: SYSTEMS INTERACTIONS STUDY FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Dear Mr. Gossick:

In a report dated July 13, 1978 concerning operation of the Indian Point Unit No. 3 at its full power level of 3025 MWt, the ACRS made several recommendations, including one that requested, "Review of the Station for systems interactions that might lead to significant degradation of safety."

In its earlier report of June 9, 1976 concerning full power operation of Zion Units 1 and 2, the ACRS had made a similar recommendation for that plant. In response to the recommendation for Zion, Commonwealth Edison arranged to have a study performed of Licensee Event Reports (LERs) covering the period between 1969 and 1977 to determine which indicated a potential systems interaction question. The results of this study were then applied to the Zion station to see if the potential for any of the same systems interactions were present and needed correction.

The ACRS has recently been asked by Consolidated Edison and the NRC Staff whether an LER systems interactions study similar to that performed for Zion would be an adequate response to its recommendation for a systems interactions study for Indian Point Unit No. 3, which, like Zion, was designed and constructed prior to ACRS identification of the generic need to examine the matter of systems interactions (letter to L. M. Muntzing dated November 8, 1974).

The ACRS believes that some types of systems interactions can be identified by an LER study such as that performed for Zion. However, the Committee believes that such an effort can only be considered to represent a treatment of part of the problem and does not recommend that type of study for Indian Point Unit No. 3.

As the Committee has stated in NUREG-0572 (September 1979), "Review of Licensee Event Reports (1976-1978)," a detailed review of LERs cannot be expected to identify all systems interactions. By far, the bulk of the LERs deal with failure of individual components and equipment, with relatively few cascades of failures resulting from an initiating event. It is not to be expected that LERs will include a relatively comprehensive set of examples of low probability events involving the coupled failures of systems where the initiating event itself is unlikely.

A-65

Thus, there will be important aspects of systems interactions which are unlikely to be exposed by a study of LERs. The important question is how to uncover vulnerabilities which may have potentially serious effects the first time they occur. In its letter of November 8, 1974 to Mr. Muntzing, the ACRS gave several examples of possible systems interactions to illustrate the matter. Since a question has arisen concerning what constitutes a reasonably appropriate study of systems interactions at Indian Point Unit No. 3, the ACRS has the following additional comments.

There are at least two general areas of investigation of systems interactions which are unlikely to be covered by a review of LERs.

1. There is a possibility of systems interactions within an interconnected electrical or mechanical complex. In such a study, it is necessary to consider failures which may be outside the usual context of failure analysis. For example, a component may run away or it may partly fail and hang up somewhere between its normal and its "failed" state, in either case leading to some excess in whatever service (voltage, frequency, flow, pressure, temperature, etc.) is provided or controlled by the system complex under consideration. This kind of failure, which usually is less likely than total functional failure of a sub-system, is unlikely to be revealed by LERs. Investigation of such failures generally will require an appropriate application of failure modes and effects analysis with the use of the systems diagrams.

2. There is a possibility of interactions between nonconnected systems due to the physical arrangement or disposition of equipment and to possibilities of transporting damaging influences, such as heat or water, within a given plant or site. Such interactions are likely to be unique to each plant and are unlikely to be revealed by LERs since the probability for such interaction to occur may be modest. There are exceptions to this, of course, and many reductions in the potential for systems interactions resulted from evaluation of the Quad Cities event of June 9, 1972 in which a rupture in the circulating water system flooded the turbine building basement and some safety-related equipment. Generally speaking, however, neither LERs nor a study of plant diagrams and other drawings will consistently reveal the potential for such interactions between nonconnected systems, because such drawings generally show single features or systems; composite drawings which include all systems are difficult to make without their becoming unmanageably complicated. Thus, uncovering the potential for interaction of nonconnected systems will usually require careful, in-situ examination of the physical plant. This examination must consider all features having the potential to damage safety systems, including the safety systems themselves.

   The physical inspection of the plant could be approached by dividing the plant into "compartments" following discernable structures — such as walls, ceilings, and floors with appraisable strengths and weaknesses. Doors, stairs, ventilation ducts, piping, and other penetrations would be evaluated for potential influence transport (fire, steam, hot air, etc.). Structures, which act as barriers to the flow of a damaging influence, would be assessed for the adequacy of their resistance to such influences.

In each compartment the elements of the safety systems, including such extensions as instrument lines and power or control wiring should be identified on a "train" basis. The physical vulnerability of the safety system elements to nonstandard conditions (temperature, pressure, water, spray, etc.) should be identified. The characteristics of such systems as influence generators under faulted conditions would have to be assessed if such system elements exist as redundant elements within the identified "compartment" boundaries.

The influence potential of all non-safety elements including such items as sewer and drain lines, combustible gas transport and storage, compressors, and heavy-power-circuits and transformers, within the given compartment should be assessed with respect to potential for damaging or disrupting (as with induced electrical noise) critical system(s) within the "compartment" and the "compartment" boundary itself.

The invasion of damaging influences through the barriers or boundaries into the identified compartment would also have to be assessed. This would include consideration of entry of personnel carrying influence generators such as welding equipment.

Special consideration would have to be given to the identification of convergence of safety functions into single compartments and the degree of convergence within the given space. The study of interactions between nonconnected systems would also have to include the possibility of non-visible interactions, such as the possibly adverse effect of failure of one buried pipe on a neighbor due to scouring. A study of plant drawings would be required in connection with this aspect.

The ACRS believes that one practical method to pursue such a systems interactions investigation is by formation of a small but competent interdisciplinary team, perhaps four to six individuals, who would pursue the two areas of investigation described above. The report of the team should identify the detailed approach employed and tabulate the results in a reviewable form.

The Committee believes that the two areas of investigation described above can be used in defining a suitable approach to a systems interactions study for Indian Point Nuclear Generating Unit No. 3 and are generally applicable to such studies on other LWRs.

Sincerely,

Max W. Carbon
Chairman

January 4, 1982

SECY-82-1

POLICY ISSUE
(Notation Vote)

For: The Commissioners

From: Executive Director for Operations

Subject: SEVERE ACCIDENT RULEMAKING AND RELATED MATTERS

Purpose: To recommend a change in approach on the severe accident rulemaking.

Category: This paper covers a major policy issue requiring Commission approval.

Issue: Whether to substitute rulemakings on specific standard plant designs for the generic severe accident rulemaking. If approved, this approach would lead to establishment of design requirements for severe accidents for the next generation of nuclear power plants. It would not, at this time, require any more than the Interim Rule for operating plants nor any more than the CP Rule for near term CPs.

Background: The TMI Action Plan (NUREG-0660, May 1980) contained task II.B.8, "Rulemaking Proceeding on Degraded Core Accidents." It envisioned a long-term rulemaking extending beyond 1982 to establish policy, goals, and requirements related to accidents involving core damage greater than the present design basis. The task also included the interim steps of an Advanced Notice of Rulemaking and an Interim Rule. The Advanced Notice of Rulemaking was issued on December 2, 1980 (45FR65474). The Interim Rule is in two parts; the first was issued in effective form on December 2, 1981 (46FR58484), and the second was issued as a proposed rule

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on December 23, 1981 (46FR62281).

In the Action Plan, it was stated that the long-term rulemaking would consider several significant matters not addressed in the Interim Rule, namely:

- use of filtered, vented containment;
- hydrogen control measures;
- core retention devices;
- reexamination of design criteria for decay heat removal, and other systems;
- post-accident recovery plans;
- criteria for locating highly radioactive systems;
- effects of accidents at multi-unit sites; and,
- comprehensive review and evaluation of related guides and regulations.

Since the Action Plan was issued, there have been several significant changes in the Commission's severe accident policies which deal with issues intended to be included in the severe accident rulemaking. First, the Commission required more protection for severe accidents in some near term licensing actions than was envisioned in the Action Plan. Thus, in the Sequoyah operating license review, TVA was required to provide a severe accident hydrogen control system for the ice condenser containment and to assure the survivability of critical safety equipment in the case of a hydrogen burn. This led to similar requirements for other small containments and for large dry containments in Part 2 of the Interim Rule. When issued in effective form, this part of the Interim Rule will apply to all operating plants.

A second significant change was the development of a rule to specify additional requirements for pending construction permit (CP) and manufacturing license (ML) applications. This rule requires more protection for severe accidents than was envisioned in the Action Plan (100% metal-water reaction, post-CP probabilistic risk assessment and allowance for possible backfit of a containment venting system).

Three other things of significance to severe accident concerns have happened since the Action Plan was issued. First, new probabilistic risk assessments for large dry containments, when coupled with new thinking on accident source terms, are leading to a considerable scaling down of risk estimates for this design type. This may eventually be shown to be true of other design types. Second, the changes in reactor design and operation that are being implemented in accord with the guidance in NUREG-0737 are substantially improving safety, although the efforts and time required are larger than anticipated. Third, the industry has initiated a substantial program of study of costs and benefits of design features to deal with severe accidents (IDCOR).

After its issuance, there was coordination of the various Action Plan rulemaking efforts involving severe accidents, siting and emergency preparedness. To that end, the EDO created a degraded core cooling steering group, which functioned during the period October 1980 through April 1981. Its report contained a plan recommending, among other things, several years of extensive research. That research program is now underway. The responsibility for severe accident policy development was assigned to the Office of Research in April 1981. The Office of Research is preparing and will shortly transmit to the Commission its plans for producing the research information needed to confirm regulatory decisions in the severe accident area. It is coordinated with this paper.

Based on the experience of the past two years and because of the progress summarized above, we feel it is timely that the Commission reconsider its approach to severe accident rulemaking. As we will discuss below, our current thinking is also influenced by the Commission's progress toward establishment of a reactor safety goal.

Discussion: Because of improvements in our understanding of severe accidents and because of progress on other regulatory actions that improve protection (reduce risk) from severe accidents, the staff has been reconsidering the purpose of the long-term severe accident rulemaking. This reconsideration was in part prompted by public comments on the Advanced Notice of Rulemaking and by the ACRS. The comments varied widely, but a recurring theme was the call for greater clarity in our description of what we are trying to achieve and the steps by which we are intending to proceed.

These questions about "where to" and "how" are crucial. The first goes to the question of a safety goal. The second has not been addressed in any detail until now. Recently, several initiatives have come together which offer an opportunity to address these questions of where we are going with severe accidents and how we intend to get there.

The NRC has developed a safety goal that is about to be issued for comment, and perhaps for trial use. The ACRS has asked for NRC's outlook on future designs, and several vendors are considering submitting new designs for review. The AIF and several reactor vendors are seeking some statement of Commission policy on referencing of approved standard designs (i.e., final design approvals) in future CP applications. per WJ Dircks 1/6/82

These initiatives could be brought together to ~~replace~~ redirect the severe accident rulemaking. The safety goal would provide a measure of how far we need to go with severe accidents. The offering of new designs is an opportunity to review real plants. At the same time, these are plants still on the drawing board where designs for severe accidents can be optimized. New designs also provide an incentive for constructive industry participation. The use of standardization of design offers NRC and industry an opportunity to maximize the effectiveness of limited resources. The current standardization policy of NRC also describes, in Appendix O of 10CFR Part 50, the steps for resolving design questions, such as those involving severe accidents, for standardized designs.

The staff believes that it is possible to begin reviews of specific standard plant design applications with an expectation of fully resolving the severe accident questions in the course of the review. This belief is predicated on the availability of results from ongoing NRC, IDCOR and vendor research; confirmatory conclusions from the Zion and Indian Point risk assessments; and the availability of a reactor safety goal. Such an approach would provide incentive to industry to address severe accident phenomena. It would replace the current unfocused long-term generic rulemaking and concentrate our efforts on real plants and real choices in design and operations.

Such an approach would not address the question of whether design or operating features for severe accidents, beyond those of the Interim Rule, need to be backfit to operating plants. But this question can probably be safely deferred for several years based on the results of our ongoing risk assessments and reviews of risk assessments for operating plants when viewed in light of the Commission's draft safety goal. In any event, the development of Phase III for the systematic evaluation program (SEP) to include a plant specific, probabilistic risk assessment (PRA) is perhaps a more effective way to address operating plants. The impracticalities of further backfits to operating plants in the next few years also lead us to believe that the question of backfits for severe accidents, beyond those already mandated, should be deferred for now.

The Commission could, at this time, issue a policy statement as a logical successor to the series of changes in requirements for design and operation of nuclear power plants which have already been made as a result of the accident at Three Mile Island. We have drafted such a statement (see Enclosure A) as though it were to be issued after or concurrent with the Safety Goal. It discusses a group of interrelated standards that would make the next generation of reactors safer than the best of those presently being licensed. The Commission has already taken a step in this direction with the recent Construction Permit/ Manufacturing License Rule which would accept current designs with some changes in principal features, to be determined during OL review, that can be incorporated without major redesigns.

per WJ Dircks 1/6/82

for standardized plants.

Issuance of a policy statement such as Enclosure A would not in itself accept the designs. Rather, it would serve to clarify that current designs could be submitted for license review without the likelihood of fundamental change, and applicants could, after about 2 or 3 years of evaluation against specified standards, expect to receive design approvals. This is probably faster than the case of a generic rule followed by standard design submittals. No new CP could be expected to be issued before about 1984 or 1985 under the standard design approval approach, but realistic plans and cost estimates could be drawn up by utilities on the basis of the policy statement. This approach would give clear preference to standardized designs in licensing by emphasizing rulemaking (see 10CFR Part 50, Appendix O, Section 7) as the method of design approval. This would also encourage movement towards one step licensing since most standard designs are now nearly final.

In the interim, i.e., between now and 1984-85, before completion of the reviews of standard designs against the full list of criteria in the policy statement, the Commission should consider allowing new applications to reference approved FDAs (under the previous standardization policy), provided the application is amended and reviewed against the new CP/ML rule. Some background on standardization is provided in Enclosure B.

We believe that with the use of PRA and compliance with current regulations (including the CP rule) it can be shown that present generation standard designs, with due care in construction, test and operation, satisfy the safety policy the Commission is considering. We believe that the Commission should substitute rulemakings on specific standard plant designs for the generic severe accident rulemaking. We caution that the enclosed policy statement attaches considerable importance to the use of Probabilistic Risk Assessment (PRA) to measure the plant risk against the quantitative guidelines of the Commission's safety policy. There is some weakness in this approach because of the significant uncertainties in PRA. That is why PRA alone is not a sufficient test. The policy must rest on a balance of conventional review and PRA. That is also why the research program must be closely coupled to the Commission's policies for use of PRA in programs relating to operating reactors (Phase III of the Systematic Evaluation Program and the National Reliability Evaluation Program) and to new CP applications. We recommend this approach on the presumption that the Commission's safety policy will include quantitative guidelines, require cost-benefit analyses, and require applicants to demonstrate that risk is as low as reasonably achievable (ALARA). It should also address the degree to which sabotage and other external factors are to be included in comparisons with the safety goal.

<u>Recommendation</u>: That the Commission:

1. <u>Approve</u> the concept outlined in this paper and in the draft policy statement of Enclosure A.

2. <u>Note</u> that:

   a) Staff will further refine the policy statement in Enclosure A by February, 1982;

   b) Further information will be included on:

      . basis for compliance with the safety goal;

      . escape clause for additional requirements, if needed;

      . legal aspects.

   c) Review will be requested of ACRS;

   d) Industry reaction will be solicited;

e) Public comment could be coordinated with comments on the safety goal.

Scheduling: Commission action on this paper is required by January 15, 1982 in order to allow time for ACRS review, industry comment, necessary revisions, and return to the Commission by February 15.

William J. Dircks
Executive Director for Operations

Enclosures:
"A" - Draft Commission Policy Statement on Licensing Policy for New Power Plant Construction Permit Applications

"B" - Background on Standardization

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. January 15, 1982.

Commission staff office comments, if any, should be submitted to the Commissioners NLT January 8, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for consideration at an open meeting during the week of January 4, 1982. Please refer to the appropriate weekly Commission schedule, when published, for a specific date and time.

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ENCLOSURE A

POSSIBLE COMMISSION STATEMENT

ON

LICENSING POLICY FOR NEW POWER PLANT

CONSTRUCTION PERMIT APPLICATIONS

Introduction: History and Purpose of Policy Statement

The Three Mile Island (TMI) accident prompted the Commission to consider and mandate a series of changes in design and operation of nuclear power plants as a response to deficiencies revealed by that accident. Since many reactors were already operating, the Commission focused first on requirements for operating Babcock and Wilcox reactors and then turned to other operating reactors. Later, the Commission set requirements for plants whose operating license (OL) review was interrupted by the attention paid to operating plants. Still later a separate set of requirements was developed for plants whose construction permit (CP) review was interrupted by the accident. This last set of requirements, embodied in the Construction Permit/Manufacturing License (CP/ML) Rule is expected to be published in effective form by January 1982.

The Commission now intends to outline its policy for new CP applications and reactivated CP applications with respect to design requirements. In this policy statement, we will connect our present requirements for the current classes of reactors to our new safety policy, and to our previous standardization rules, thereby identifying for new or reactivated CP applications the general licensing policy for new light water nuclear power reactors.

As part of the Commission's response to TMI, an Action Plan (NUREG-0660, May 1980) was issued. Section II.B of that Plan deals with the siting of plants and the requirements for coping with severe accidents. The thrust of that section of the Plan is to reconsider the manner in which severe accidents should be factored into the regulatory process. The Commission has already issued interim rules concerning degraded core cooling. However, promulgation of final generic requirements for severe accidents will take additional time and research. The Commission does not believe that, in the interim, this continuing research should be a deterrent to the placement of orders or the initiation of licensing reviews for new CP applications. We thus need to state our current general licensing policy as it relates to policy for coping with severe accidents. For the reasons discussed below, the Commission believes that reactors of modern design, improved to take into account the CP regulation promulgated as a result of the TMI accident and the insights afforded by contemporary probabilistic risk assessments, can be shown to be acceptable for severe accident concerns. This conclusion embodies due consideration of the Commission's proposed statement of safety policy. It permits plants to be sited at locations similar to those required by current siting practice. These reactors of modern design are typified by those represented in vendor-specific and architect/engineer-specific standardized safety analysis reports. As discussed below, the policy contains seven inter-related components: safety policy, use of probabilistic risk analysis in safety review, lessons learned from TMI, the Standard Review Plan, standardization, further research in severe accidents, and implementation of the policy.

Safety Policy

The Commission has recently developed and published for comment a safety policy which presents several qualitative safety goals and quantitative probabilistic guidelines. It also includes numerical guidance to assure that the risks of nuclear power plants are as low as reasonably achievable. Implementation of the safety goal will require extensive use of probabilistic risk assessment (PRA). Although there are limitations on the use of probabilistic risk assessment, the Commission considers it to be a valuable adjunct to the established safety review process. Many probabilistic risk assessments of U.S. reactors have been made since two reactors were analyzed and reported in the Reactor Safety Study (WASH-1400). These include risk assessments done under the NRC's Reactor Safety Study Methodology Application Program (RSSMAP) and Interim Reliability Evaluation Program (IREP), as well as a number of industry studies (Big Rock Point, Limerick, Zion, etc.).

These assessments have varied in scope, depth and quality; but, taken as a whole, they indicate a careful growth in the constructive use of these techniques as a supplement to the safety review process. They improve our understanding of the severe accident sequences to which a plant is most vulnerable, and they provide a tentative measure of the overall risk posed by a plant (as well as the constituents of that risk). The studies conducted to date also provide the basis for a preliminary, but important, generic conclusion: that the risk from some (perhaps all) nuclear power reactors of current modern design could be shown to meet the goals and guidelines of the Commission's safety policy, provided that these reactors are built, tested, and operated with suitable care.

Some of these risk assessments and appraisals of operational experience have identified new equipment and specific plant features that have a high potential for risk reduction. The features so identified typically involve system design and operation details and not fundamental design characteristics. Examples include the interfacing systems' loss-of-coolant accident identified in the assessment of the Surry plant; simple surveillance procedures now undertaken on pressurized water reactors (PWR) reduce this dominant risk contributor. The study of Sequoyah in RSSMAP identified blockage of the drains connecting the ice condenser compartments as risk-significant; here again, simple surveillance procedures provide significant risk reduction for this ice-condenser type of containment. Post-TMI special assessments of PWR auxiliary feedwater systems identified many changes that substantially increased the system reliability. Some of these changes were small system design refinements; the largest change was the addition of a third pump to a two-pump system. At the Peach Bottom facility, anticipated transients without scram (ATWS) were found to be predominant contributors to risk. Therefore, a requirement for prompt trip of the recirculation pumps was imposed for BWRs. While this did not solve the problem, it greatly reduced the early pressure risk (which could make the vessel fail).

In sum, considering the risk assessments thus far made, we note that current generation light water reactors are estimated to be close to or below the risk levels we believe acceptable, and that cost-effective means of further risk reduction are often readily apparent through PRA. We are thus persuaded that most recent U.S.-designed standardized reactors, with suitable balance-of-plant equipment, can be shown to meet our safety goals and that PRA can be used to decide what design measures are required to be included to reduce risks to as low as reasonably achievable levels consistent with the guidance in our safety policy.

Use of PRA in License Review

The Commission recognizes that probabilistic risk assessment (PRA), if used effectively in the design process, can improve both the safety and the cost-effectiveness of the plant. The performance of a PRA, as a supplement to current regulations and the current licensing process, is considered sufficiently valuable that it should be completed by an applicant and reviewed by NRC before issuance of a CP for any new application or for any reactivation of a previously-deferred CP application. We believe that such studies can help to identify design features that would either lessen the likelihood of degraded core cooling events (prevention), arrest the extent of damage by successful interdiction of a degraded core cooling event (management), or lessen the ensuing consequences of a core meltdown (mitigation). We do not mean to imply that our policy is to require zero risk. We expect that PRA can help to expose those design variations which are practical and which make a significant contribution to risk reduction commensurate with their cost, consistent with the cost-effectiveness criterion of our safety policy. We have already proposed for near-term CPs that PRA be factored into the design process shortly after CP issuance. Thus, our policy for new CP applications is to simply move the PRA and associated reliability engineering programs forward in the design process and the regulatory process.

Lessons Learned from TMI

The lessons learned from TMI have been applied to operating reactors, plants in OL review, and plants now undergoing CP review. The Commission's policy for the near-term CP applicants has been that each such plant should comply with the appropriate subset of the Action Plan, now embodied in the CP/ML Rule and discussed in NUREG-0718. It is our policy that future CP applications or reactivations will also be required to comply with this regulation.

Standard Review Plan

The Commission staff issued an updated version of the Standard Review Plan (NUREG-0800) in July 1981. It is our expectation that new CP applications and reactivations would include an evaluation of adherence to this SRP. It is Commission policy to ensure that every new CP application or reactivation meet the current SRP, unless differences are described and justified.

Standardization

The Commission reiterates its support for standardization. To this end, it expects holders of Preliminary Design Authorizations (PDA) and applicants for Final Design Authorizations (FDA) to modify their applications to conform with this policy statement, if the design approvals are to be used in any future CP applications. When reviewed by the staff and approved in rulemaking, the Commission intends that acceptance of the standardized designs would be binding, as appropriate, for a period of ten years, except where new information showed that the design did not meet our safety goal. Both the staff and applicants would be bound by the approval. Ten years appears a reasonable choice on two grounds: first, it permits the completion of relevant research and due consideration of the results before changes are made; and, second, it is a time span consistent with practical use of standardized designs. The Commission intends that approval of standard designs be accomplished by rulemaking in accordance with 10CFR Part 50, Appendix O, Section 7. Applicants seeking this course will be given priority, over custom plant applications, in the assignment of staff review resources.

The Commission acknowledges the importance of reviewing final design information in the performance of probabilistic risk assessments. This will probably require that the FDA-level of design detail for the nuclear steam supply system (NSSS) and for a substantial portion of the balance-of-plant (BOP) equipment be available before successful completion of the licensing review for a new plant. Because of this, and in view of our limited resources for initiating new CP reviews, the Commission will give preference, at the time of docketing, to standard plant applications for which a substantial portion of the NSSS and

BOP design has been completed.*

Further Research on Severe Accidents

The Commission is conducting a research program on the phenomenology of severe accidents. This program includes studies in probabilistic risk assessment, analyses of systems transients (including severe accidents where core damage is hypothesized), fuel behavior, fission product behavior, and containment response to severe loadings. There is substantial uncertainty in the risk calculations which would be used in testing adherence to our safety policy, and this plays a role in shaping future research. The uncertainty in the "front end," or accident initiation likelihood, is generally believed to be optimistically biased, due to possible lack of completeness in describing scenarios and to difficulties in identifying and modeling common cause failures. On the other hand, the "back end," or consequence estimation of risk assessment, is generally believed to be conservatively biased, since even the partial failure of core cooling is usually taken as core melt--and recent research (see NUREG-0772) indicates that radioactive releases, at least in some major accident sequences where containment failure is delayed, are likely to be substantially lower than previous predictions. These biases tend to cancel one another in the overall risk estimate; however, they limit the usefulness of PRA for weighing the relative merits of different design or operating features.

Continued research is warranted to provide confirmatory information on the level of safety of plants. The research will also enable a more precise appraisal of specific design and operational refinements, especially from the

*See previous Commission policy statement (e.g., 43FR38954, August 31, 1978) with respect to antitrust aspects involved in standardization approach.

value-impact view, that may be indicated for further reduction of risk. In addition, better understanding of the dominant severe accident sequences and of the magnitude of radioactive releases can lead to substantial improvements in emergency preparedness. The principal areas of research are:

- common cause accident contributors;
- system interactions;
- accident management, including guidelines for recovery from a core damage event;
- human factors;
- phenomenological research on behavior of damaged cores;
- applications research on behavior of existing systems and components given the accident environment;
- fission product release and transport; and,
- value impact analysis of potential add-on systems.

We expect these research programs to be essentially complete by 1985. The Commission also notes a substantial commitment of industry resources through 1983 for severe accident evaluation under the auspices of the IDCOR program. It is important to our promulgation of this policy statement that the IDCOR program continue on its present course and schedule.

Our present view is that we do not expect our policy to change as a result of ongoing research with respect to the fundamentals of the present designs and their general adherence to our safety policy. We expect research results to permit further risk reduction by identifying worthwhile refinements in present design or operating practices rather than indicating major redesign needs. The research will also help to develop more accurate probabilistic risk assessment methods for use in regulatory decisionmaking and to provide greater assurance of safety.

Implementation

The Commission sets the following guidelines for new CP applicants and the staff for review of these applications or any CP reactivations that might be submitted:

(i) demonstration of compliance with current Commission regulations, including the TMI requirements set out in 10CFR 50.34;

(ii) completion of a PRA before CP-issuance; demonstration of adherence to the draft safety goal and guidelines; and commitment to install cost-effective design features for prevention, management, or mitigation of severe accidents (within the guidelines of the safety goal policy);

(iii) adherence to the current version of the Standard Review Plan (NUREG-0800); and,

(iv) consideration of all applicable Unresolved Safety Issues.

For th.e CP applications and reactivations meeting the guidelines above, it is the Commission's expectation that no additional fundamental design requirements relating to severe accidents would be issued, unless new safety information shows nonconformance to our safety goal.

ENCLOSURE B

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BACKGROUND ON STANDARDIZATION

In August 1978, the Commission issued a policy statement, "Statement on Standardization of Nuclear Power Plants," which expanded on the standardization concept for nuclear plants and described specific policy changes being made to improve the usefulness of the Commission's standardization program. That policy statement, among other things, defines the effective time periods for design approvals under each of the four standardization concepts. For example, under the replicate plan concept, an applicant for a new construction permit (CP) can reference a base plant up to 3 years after the issuance of the CP Safety Evaluation Report for the base plant. Under the reference system concept, the Final Design Approval (FDA-1) expires 3 years after the expiration of the related Preliminary Design Approval.

In a letter dated July 23, 1981, the Atomic Industrial Forum (AIF) recommended that the Commission consider (1) extending the reference period for replicate plant applications and (2) allowing a plant with an operating license SER to serve as a base-plant for a replicate plant application for a CP.

In a letter dated September 15, 1981, Combustion Engineering (CE) requested that the Commission either (1) replace the allowed time period under the reference system concept with a limit on the number of plants which may reference the design (CE suggests that 50 units be the limit); or (2) change the reference limit to 50 units or a minimum of 10 years from the issuance of the FDA-1, whichever comes first.

The two letters raise narrow policy issues regarding the Standardization Policy. However, there is a broader policy issue imbedded in the industry letters. The current Standardization Policy was enunciated before the TMI-2 accident. Following the accident, the notion of a demarcation between the current generation of plants (those operating and those under O! or CP review) and a future generation of plants was raised. The distinction was that approval of future plants would be designed based on a reformulation of the siting criteria and design requirements to reflect all that had been learned over the years, including broader lessons from TMI-2. Thus, the TMI Action Plan (including the Near-Term Construction Permit and Manufacturing License requirements) was developed with the current generation of plants in mind, leaving open the question of possible broader changes for a future generation of plants. Since the Commission has already devoted considerable time and effort on ORs, OLs and pending CPs, it now is logical to turn to questions involving future CP applications. This subject cannot be separated from the severe accident rulemaking and associated research.

ACRS BRIEFING OUTLINE

- o BACKGROUND
- o ORGANIZATION
- o REVIEW PLAN & SCHEDULE
- o RECENT EVENTS
- o UPCOMING EVENTS

BACKGROUND

APPLICATION DOCKETED	APR. 75
ACRS REPORT ON CDA core disruptive accident	AUG. 76
FES ISSUED	FEB. 77
SSR ISSUED	MAR. 77
ASLB PROCEEDINGS SUSPENDED	APR. 77
GAMMILL - CAFFEY LETTER	NOV. 78
DAVIS - PALLADINO LETTER	AUG. 81
CRBRPO ESTABLISHED	SEPT. 81

DIRECTOR, NRR

CRBRPO

DIRECTOR
PROGRAM COORDINATOR

LICENSING SECTION C. Thomas
LEADER
2 PMs R. Stark
P. Leech
LIC ASST

TECHNICAL REVIEW SECTION W. Morris
LEADER
3-6 TECH REVIEWERS

PRESENT STRENGTH: 12
AUTHORIZED: 15

REVIEW PLAN & SCHEDULE

3 PRINCIPAL PRODUCTS (DOCUMENTS):	FES*, SSR*, SER
ISSUE DATES:	9/82 6/82 3/83
LICENSING ACTION (DATE):	LWA (9/83) CP(6/84)

* SUPPLEMENTS

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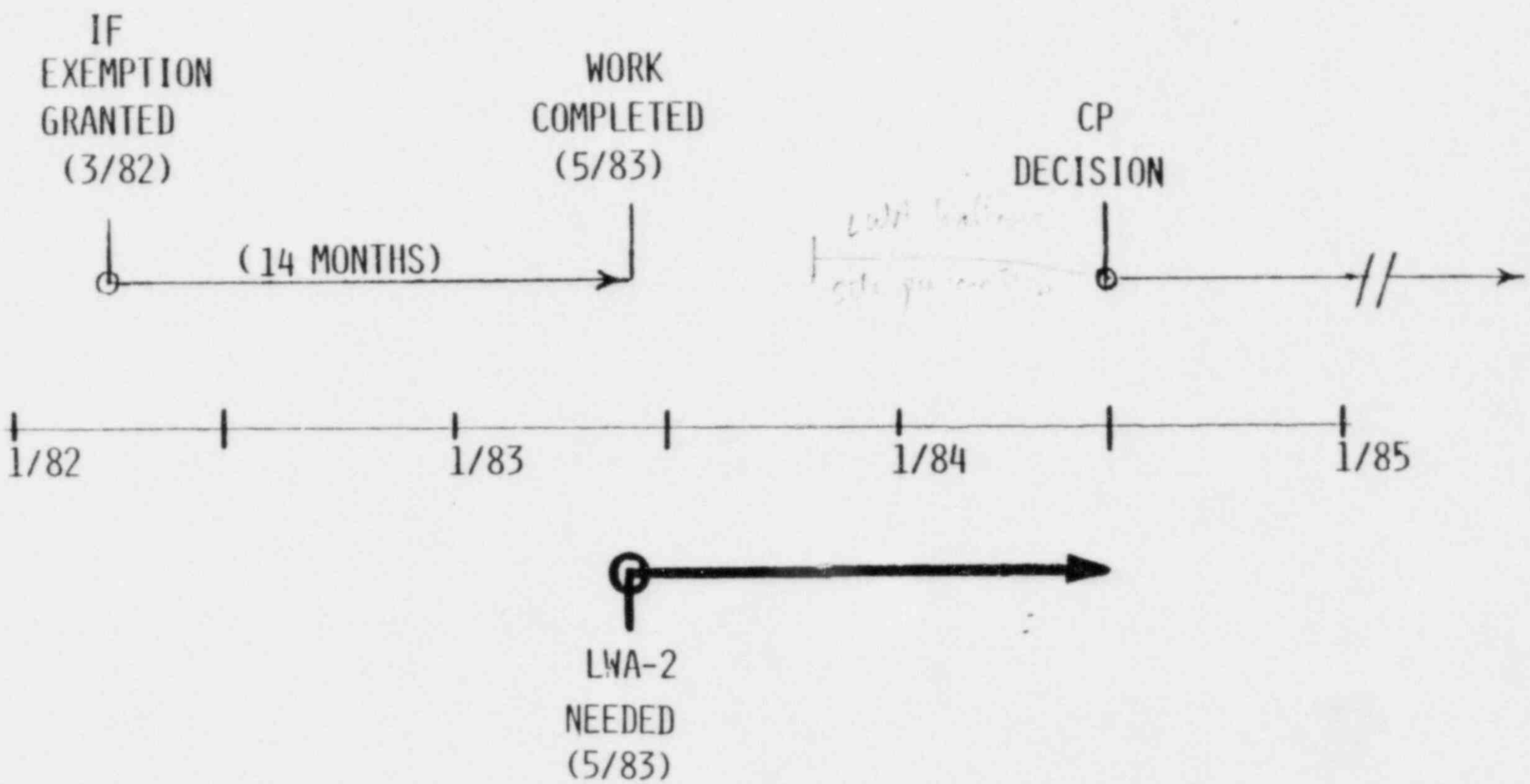
RECENT EVENTS

- TECHNICAL MEETINGS WITH APPLICANT
- COMMISSION BRIEFING
- EXEMPTION REQUEST

IMPACT OF EXEMPTION, WERE IT AUTHORIZED, ON LICENSING REVIEW



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UPCOMING EVENTS

- o SUBCOMMITTEE MEETING
- o EXEMPTION DECISION
- o REVISE PLAN FOR REVIEW OF CRBR

DISCUSSION TOPICS FOR FEBRUARY ACRS SUBCOMMITTEE MEETING

- CRBRPO ORGANIZATION
- REVIEW STATUS IN MID-1977
- DESIGN CHANGES SINCE MID-1977
- REVIEW PLANS AND SCHEDULES INCLUDING EXEMPTION
- DESIGN CRITERIA
- RESEARCH AND TECHNICAL ASSISTANCE NEEDED TO SUPPORT LICENSING

ADDITIONAL INFORMATION

PROJECT LICENSING PROFILE

- o DESIGN AND R&D NEARLY COMPLETE
- o $500 MILLION IN HARDWARE
- o FES & SSR ISSUED
- o PREVIOUS SAFETY REVIEW, NOT ENTIRELY RECOVERABLE
- o HEARING STATUS
  - 7 PARTIES
  - 18 CONTENTIONS
  - EXTENSIVE DISCOVERY
- o '76 COMMISSION DECISION ON NEPA RESPONSIBILITY
- o HEAVY RELIANCE ON DOE LABS

FY'82 NRR PROGRAM SUPPORT BY DOE LABORATORIES

LAB	PROGRAM SUPPORT ($ IN THOUSANDS)	PRINCIPAL TASKS
LOS ALAMOS	1,125	- REACTOR CORE DESIGN - CONTAINMENT - CORE DISRUPTIVE ACCIDENTS - REACTIVITY TRANSIENTS
BNL	750	- HEAT TRANSPORT SYSTEMS - ENVIRONMENTS FROM SODIUM LEAKS - UNDER-COOLING ACCIDENTS - PRA REVIEW
INEL	600	- INSTRUMENTATION AND CONTROL - HIGH-TEMPERATURE MECH. DESIGN - SODIUM SYSTEMS - SODIUM FIRE PROTECTION
PNL	25	- ENVIRONMENTAL ISSUES
TOTAL	2,500	

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NRC RESEARCH PLAN

- o JOINT EFFORT OF NRR AND RES TO ESTABLISH NEEDS FOR CRBR LICENSING
- o WORKING GROUP ACTIVITIES
  - MEETINGS WITH DOE
  - REVIEW STATUS OF MAJOR ISSUES
  - IDENTIFY SPECIFIC INFO NEEDED FOR LICENSING
  - PARTICIPATION BY BNL, LANL, SNL
- o ISSUES DOMINATED BY CDA
- o COMPLETE INITIAL PLAN BY END OF YEAR
- o APPLICANT PARTICIPATION EXPECTED

STATUS OF APPLICATION

- o UPDATE OF CRBR ER
  - LMFBR PROGRAM EIS
  - ANSWERS TO RECENT QUESTIONS
- o PSAR REFLECTS CURRENT DESIGN
- o AMEND PSAR TO SHOW SYSTEMATIC EVALUATION OF NEW NRC REQUIREMENTS
  - REGULATIONS
  - REG. GUIDES
  - TMI
  - USIs

HIGHLIGHTS

CRBR DESIGN CHANGES

- HETEROGENEOUS CORE
- ADDED DIVERSE DC POWER SUPPLY
- NEW ENGINEERED SAFETY FEATURES
- ENHANCED SEISMIC DESIGN

LMFBR (VS LWR)

- o PRIMARY COOLANT SYSTEM
  - LOW PRESSURE
  - HIGH TEMPERATURE
- o ENERGETIC CHEMICAL REACTIONS OF SODIUM WITH
  - OXYGEN
  - WATER
  - CONCRETE
- o POSITIVE REACTIVITY FEEDBACK FROM
  - VOIDS IN SODIUM
  - FUEL RELOCATION

SPECIAL REVIEW PROBLEMS

- o MANY LWR STANDARDS AND REG GUIDES NOT DIRECTLY APPLICABLE
- o CR LICENSING CRITERIA
- o CORE DISRUPTIVE ACCIDENTS
  - HISTORICALLY MAJOR LMFBR ACCIDENT
  - LOW PROBABILITY BUT POTENTIALLY SEVERE CONSEQUENCES
  - EVALUATION REQUIRES MODELING OF CORE BEHAVIOR AFTER DISASSEMBLY AND MELTING

PROPOSED ACTIVITIES UNDER 50.12

- SITE CLEARING, GRADING, AND EXCAVATION FOR MAIN BUILDINGS, TEMPORARY SUPPORT, AND CERTAIN SERVICE FACILITIES.

- CONSTRUCTION OF TEMPORARY SUPPORT AND SERVICE FACILITIES INCLUDING:
  - ROADS AND RAILROADS SPURS
  - ON-SITE QUARRY AND CRUSHING FACILITY
  - CONCRETE BATCHING AND MIXING PLANT
  - BARGE UNLOADING FACILITY
  - SERVICE FACILITIES (CONSTRUCTION PARKING, WORK AND STORAGE AREAS)
  - FIRE PROTECTION SYSTEM
  - STORM DRAINAGE SYSTEM

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PROJECT PARTICIPANTS

CO-APPLICANTS:	DOE, PMC AND TVA
REACTOR MANUFACTURERS:	WESTINGHOUSE (LEAD) GENERAL ELECTRIC ATOMICS INTERNATIONAL
ARCHITECT ENGINEER:	BURNS AND ROE
CONSTRUCTOR:	STONE & WEBSTER

	EXEMPTION	LWA-1	LWA-2
GOVERNED BY:	10 CFR 50.12	10 CFR 50.10	10 CFR 50.10
ISSUED BY	COMMISSION	DIRECTOR, NRR	DIRECTOR, NRR
REQUIRES:	CONSIDERATION OF (1) ENVIRONMENTAL IMPACT, (2) REDRESSABILITY, (3) FORECLOSURE OF ALTERNATIVES, (4) PUBLIC INTEREST. NO FES, SSR OR PUBLIC HEARING REQ'D.	COMPLETION OF ENVIRONMENTAL REVIEW AND PUBLIC HEARING ON NEPA AND SITE SUITABILITY AND BOARD ISSUES FINDING.	SAME AS LWA-1, PLUS FINDING OF NO UNRESOLVED SAFETY ISSUES RELATED TO PROPOSED ACTIVITIES.

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CRBRP SAFETY REVIEW SCHEDULE

Reinitiate Safety Review.	10/01/81
Identify remaining safety information requirements.	01/01/82
Receive remaining safety information from applicant.	03/01/82
SER inputs provided to Project Manager.	05/15/82
Issue draft SER	07/01/82
Meeting with applicant to discuss resolution of SER open items.	07/15/82
Meetings with applicant to resolve SER open items.	As necessary
SER inputs on open items provided to Project Manager.	01/15/83
Issue SER.	03/01/83
ACRS Meeting.	05/07/83
Receive ACRS letter.	05/15/83
Issue SER Supplement.	07/01/83
Begin safety hearing.	10/01/83
Complete safety hearing.	02/01/84
ASLB issues partial initial decision on safety matters.	05/01/84
Issue CP	06/01/84

BWR FUTURE DIRECTIONS

MANUSCRIPT

PRESENTATION BY
D. R. WILKINS
TO
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C.

JANUARY 7, 1982

BWR FUTURE DIRECTIONS

SLIDE 1 - TITLE

MY NAME IS DAN WILKINS. I AM THE GENERAL MANAGER OF THE NUCLEAR POWER SYSTEMS ENGINEERING DEPARTMENT AT THE GENERAL ELECTRIC COMPANY -- RESPONSIBLE FOR THE DESIGN OF OUR BOILING WATER REACTOR PRODUCT LINES.

I AM PLEASED TO HAVE THE OPPORTUNITY TO DISCUSS WITH THE COMMITTEE THE CURRENT STATUS AND FUTURE OF BWR DESIGN. FUTURE LWR DESIGNS ARE AN IMPORTANT SUBJECT AND IT'S ENCOURAGING TO SEE THE ACRS THINKING ABOUT THE FUTURE OF NUCLEAR TECHNOLOGY IN TODAY'S DIFFICULT CLIMATE.

SLIDE 2 - OUTLINE

BEFORE DISCUSSING THE FUTURE, I WOULD FIRST LIKE TO VERY BRIEFLY REVIEW PAST BWR DESIGN EVOLUTION AND THE PRESENT BWR PRODUCT. THE REASON FOR THIS APPROACH IS THAT OUR OWN VIEWS OF THE BWR FUTURE ARE STRONGLY INFLUENCED BY OUR LEARNING EXPERIENCES OF THE PAST. WITH THIS BACKGROUND, I WILL THEN DISCUSS SOME OF OUR MORE RECENT POST-TMI EVALUATIONS OF THE BWR -- THE CONCLUSIONS WE HAVE REACHED -- AND THE PLANT PROTECTION IMPROVEMENTS WE HAVE IDENTIFIED AND EITHER WILL BE IMPLEMENTING OR ARE CONSIDERING FOR FUTURE PLANTS.

From my remarks this morning you will see that our approach to BWR design has rested on four major principles:

- o Continuous simplification of the design
- o Standardization of the entire Nuclear Island
- o Evolutionary -- rather than revolutionary -- change
- o Thorough "test before use" of new features.

We expect these same principles to guide BWR design evolution in the future.

Slide 3 - BWR Evolution

Let's look briefly at the evolution of the BWR.

Slide 4 - Direct Cycle Advantages

Twenty-five years ago, indirect cycle Pressurized Water Reactor technology was being developed for the Navy. GE was heavily involved in that program, and others were selecting PWR technology for central power station application. GE, however, departed from this course and chose the direct cycle BWR. We made that decision because we felt that the direct cycle offered safety and economic advantages through simplicity, lower pressure, inherent reactivity control, and direct communication between water sources and the reactor vessel

We recognized that the BWR represented a more developmental product -- and would require greater investments in supporting technology -- but we were convinced that the benefits of the direct cycle justified the investment.

Slide 5 - Dresden 1 BWR

Dresden 1 was characteristic of GE's first generation of commercial BWR's. It exhibits many features similar to a PWR -- reactor vessel, four steam generators, and primary and secondary loops. The main difference between the Dresden 1 BWR and early PWR designs is that Dresden 1 had a steam drum instead of a pressurizer. It operated as a "dual cycle" providing steam to the turbine from both the steam drum and the steam generators. This dual cycle approach allowed us to demonstrate the feasibility of the direct cycle while retaining the proven capability of the indirect cycle.

Slide 6 - KRB

Our first major move toward BWR simplification came with KRB in Germany. In that plant, the steam drum was eliminated and the steam separation and drying functions were moved inside the reactor vessel -- a change which would become permanent to the BWR design. KRB still retained the steam generators -- operating as a three-loop dual cycle plant.

Slide 7 - Oyster Creek

Our next step in simplifying the BWR occurred at Oyster Creek. By this time, we had accumulated sufficient direct cycle experience that we could justify eliminating the steam generators. The increased power rating of Oyster Creek required the use of five external recirculation loops.

Slide 8 - Dresden 2

With Dresden 2, we made our most recent major step in simplifying the BWR. Jet pumps were added inside the reactor vessel to reduce the number of external recirculation loops to two. This two recirculation loop configuration has remained the basic GE-BWR design to this day. It is incorporated in our BWR/3,4,5,6 product lines.

Slide 9 - BWR Evolution Summary

So as you can see, the past 20 years of BWR evolution have seen the BWR design evolve from the more complex dual cycle at Dresden 1 to day's simpler direct cycle. Our design path has been one of evolutionary simplification with each change being thoroughly tested and proven before being implemented.

SLIDE 10 - ABWR

WE EXPECT TO CONTINUE TO EVOLVE AND SIMPLIFY THE BWR IN THE FUTURE. WE ARE ALREADY DOING PRELIMINARY EVALUATION ON AN ADVANCED BWR CONCEPT. A POTENTIAL NEXT STEP IN FURTHER SIMPLIFYING THE DIRECT CYCLE MAY BE TO ELIMINATE THE EXTERNAL RECIRCULATION LOOPS ALTOGETHER BY EMPLOYING INTERNAL RECIRCULATION PUMPS. WE ARE CURRENTLY EVALUATING THE FEASIBILITY OF SUCH AN ADVANCED BWR WITH OUR BWR PARTNERS IN JAPAN.

SLIDE 11 - PRESSURE SUPPRESSION ADVANTAGES

OUR PIONEERING OF THE BWR DIRECT CYCLE WAS ACCOMPANIED BY A PARALLEL DEVELOPMENT OF PRESSURE SUPPRESSION CONTAINMENT. WHILE EARLY GE BWR'S WERE HOUSED IN DRY CONTAINMENTS, WE SAW ADVANTAGES IN PRESSURE SUPPRESSION THROUGH ITS ABILITY TO STORE LARGE QUANTITIES OF HEAT INSIDE THE CONTAINMENT, OPERATE AT A LOWER PRESSURE, PROVIDE THE CAPABILITY TO BLOW DOWN THE PRIMARY SYSTEM AND TO "SCRUB" ANY FISSION PRODUCTS WHICH MIGHT BE RELEASED FROM THE PRIMARY SYSTEM.

SLIDE 12 - PRESSURE SUPPRESSION CONTAINMENT

OVER THE YEARS, THE BWR REFERENCE PRESSURE SUPPRESSION CONTAINMENT HAS EVOLVED FROM THE MARK I CONFIGURATION TO THE MARK II, AND THEN TO THE CURRENT MARK III. THIS EVOLUTION WAS MOTIVATED PRIMARILY BY SEISMIC, SPACIAL AND CONSTRUCTABILITY CONSIDERATIONS.

WHILE PHYSICAL CHARACTERISTICS OF THE BWR REFERENCE CONTAINMENT HAVE CHANGED THROUGH THIS EVOLUTIONARY PROCESS, THE PRESSURE SUPPRESSION CONCEPT HAS REMAINED THE SAME AND REPRESENTS A SIGNIFICANT DIFFERENCE BETWEEN BWR CONTAINMENT DESIGN AND THOSE OF OTHER LWR'S.

SLIDE 13 - INCREASING REGULATORY EMPHASIS ON NUCLEAR ISLAND

ANOTHER IMPORTANT STEP IN THE GE BWR PRODUCT EVOLUTION OCCURRED WITH THE INTRODUCTION OF THE STANDARD REACTOR ISLAND DESIGN (STRIDE) IN 1972. THIS WAS PARTLY IN REPONSE TO AN INCREASING TREND OF REGULATORY REQUIREMENTS AFFECTING NOT ONLY THE NUCLEAR STEAM SUPPLY SYSTEM, BUT THE ENTIRE NUCLEAR ISLAND. MORE STRINGENT REGULATORY REQUIREMENTS IN THE AREAS OF SEISMIC DESIGN, PIPING FAILURES, OCCUPATIONAL EXPOSURE, ELECTRICAL SEPARATION, EQUIPMENT QUALIFICATION, AND FIRE PROTECTION -- FOR EXAMPLE -- ALL POINTED TO THE NEED FOR STRENGTHENED SYSTEM ENGINEERING AND TECHNICAL INTEGRATION OF THE ENTIRE NUCLEAR ISLAND DESIGN. STRIDE RESPONDED TO THIS NEED.

SLIDE 14 - STRIDE

THE GE STRIDE DESIGN PROVIDES A TECHNICALLY INTEGRATED DESIGN OF ALL SYSTEMS IN THE REACTOR, FUEL, AUXILIARY, DIESEL, RADWASTE AND CONTROL BUILDINGS, AND THE OFFGAS SYSTEM IN THE TURBINE BUILDING, AND ENCOMPASSES ALL RADIOLOGICALLY SIGNIFICANT PARTS OF THE PLANT. THE STRIDE DESIGN IS ENGINEERED FOR AN ENVELOPE OF SITE PARAMETERS WHICH ENCOMPASS AN ESTIMATED 90% OF USA SITES. IT ELIMINATES MANY OF THE SYSTEM INTERFACE PROBLEMS INHERENT IN THE MORE TRADITIONAL SCOPE SPLITS BETWEEN THE NSSS MANUFACTURER AND THE ARCHITECT ENGINEER.

WE BELIEVE THAT STANDARDIZATION OF THE NUCLEAR ISLAND DESIGN OFFERS BENEFITS NOT ONLY IN SAFETY AND LICENSING, BUT ALSO IN LOWER ENGINEERING COSTS, SHORTER CONSTRUCTION SCHEDULES, LOWER INSTALLED COST, IMPROVED PLANT CAPACITY FACTOR, AND IMPROVED OPERATION AND MAINTENANCE. WE EXPECT TO SEE AN INCREASING TREND IN THE FUTURE TOWARD THE UTILIZATION OF STANDARDIZED NUCLEAR ISLAND DESIGNS, NOT ONLY IN THE USA BUT AROUND THE WORLD.

THE GE STRIDE DESIGN IS 95% COMPLETE AND CURRENTLY UNDER CONSTRUCTION AT THE TVA HARTSVILLE AND PHIPPS BEND SITES. IT FORMS THE BASIS FOR THE GESSAR STANDARD SAFETY ANALYSIS REPORT SUBMITTED TO THE NRC.

SLIDE 15 - BWR TODAY

HAVING DISCUSSED THE EVOLUTION OF THE DIRECT CYCLE NUCLEAR SYSTEM, PRESSURE SUPPRESSION CONTAINMENT, AND STRIDE, IT'S APPROPRIATE TO LOOK AT THE CURRENT PRODUCT OF THESE EVOLUTIONS -- THE BWR/6 - MARK III. I WILL GIVE YOU A VERY QUICK DESCRIPTION OF IT WHICH WILL BE USEFUL IN SOME OF THE DISCUSSIONS TO FOLLOW.

SLIDE 16 - BWR/6 REACTOR VESSEL

LET'S START WITH THE REACTOR VESSEL. FEEDWATER ENTERS THE REACTOR VESSEL AT ABOUT THE MID HEIGHT AND IS ROUTED DOWN TO A LOWER PLENUM -- AND COMES UP THROUGH THE CORE REGION. BOILING OCCURS IN THE CORE AND THE STEAM THEN PASSES THROUGH THE STEAM SEPARATORS AND DRYERS -- INTERNAL TO THE REACTOR VESSEL -- AND EXITS OUT THE MAIN STEAM LINE. THE CONTROL ROD DRIVES IN THE BOILING WATER REACTOR ARE BOTTOM ENTRY CONTROL ROD DRIVES.

SLIDE 17 - BWR SCHEMATIC

LET'S LOOK AT THE OVERALL SYSTEM. THE STEAM COMES OUT OF THE VESSEL THROUGH THE MAIN STEAM LINE AND PASSES DIRECTLY TO THE TURBINE. ISOLATION VALVES ARE PROVIDED ON THE MAIN STEAM LINES, BOTH INBOARD AND OUTBOARD OF THE CONTAINMENT. FEEDWATER IS RETURNED FROM THE CONDENSER THROUGH DEMINERALIZERS AND FEEDWATER HEATERS AND DIRECTLY INTO THE REACTOR VESSEL. THE RECIRCULATION SYSTEM CIRCULATES WATER

THROUGH THE REACTOR CORE. SUCTION FLOW IS TAKEN FROM THE VESSEL, PASSES THROUGH A RECIRCULATION PUMP AND FLOW CONTROL VALVE (NOT SHOWN), AND IS ROUTED BACK TO THE VESSEL WHERE IT DRIVES JET PUMPS AND FORCES RECIRCULATION THROUGH THE REACTOR CORE.

THE MARK III PRESSURE SUPPRESSION CONTAINMENT CONSISTS OF A DRYWELL STRUCTURE, A SUPPRESSION POOL, AND A PRIMARY CONTAINMENT SURROUNDED BY A SECONDARY CONTAINMENT. IN CASE OF A PIPE BREAK IN THE PRIMARY SYSTEM, ANY STEAM RELEASED IS ROUTED TO THE SUPPRESSION POOL. IN ADDITION SAFETY RELIEF VALVES ON THE MAIN STEAM LINES ARE PIPED TO THE SUPPRESSION POOL. FOR BOTH TRANSIENT AND ACCIDENT EVENTS ANY RELEASE FROM THE PRIMARY SYSTEM GOES INTO THE SUPPRESSION POOL -- WHERE IT IS QUENCHED AND SCRUBBED -- BEFORE REACHING THE PRIMARY CONTAINMENT. FUTHERMORE, ANY LEAK IN THE MAIN CONTAINMENT IS STILL CONTAINED WITHIN THE SECONDARY CONTAINMENT WHICH IS MAINTAINED AT NEGATIVE PRESSURE RELATIVE TO THE ATMOSPHERE.

THIS COVERS THE MAIN BWR/6 - MARK III FEATURES THAT WE WILL BE TALKING ABOUT FROM TIME TO TIME THROUGHOUT THE REST OF THE PRESENTATION.

SLIDE 18 - TEST PROGRAMS

BEFORE LEAVING TODAY'S BWR/6-MARK III DESIGN, I WANT TO SAY A FEW WORDS ABOUT VERIFICATION TESTING -- OUR "TEST BEFORE USE" PHILOSOPHY AS WE REFER TO IT. THE BWR/6 - MARK III IS A VERY THOROUGHLY TESTED PRODUCT. IT IS BACKED UP BY:

- O SMALL, INTERMEDIATE AND LARGE SCALE MARK III CONTAINMENT TESTING
- O FULL SCALE FUEL BUNDLE HEAT TRANSFER TESTING
- O FULL SCALE FLOW INDUCED VIBRATION TESTING OF REACTOR INTERNALS
- O ACCELERATED, FULL SCALE PIPE TESTING FOR STRESS CORROSION PREVENTION
- O EXPANDED, FULL-SCALE ECCS TESTING, AND
- O EXTENSIVE MECHANICAL COMPONENT VERIFICATION TESTS.

THIS IS IN ADDITION TO NUMEROUS TESTS WHICH HAVE BEEN CONDUCTED TO QUALIFY THE 650 COMPUTER PROGRAMS CURRENTLY APPROVED FOR USE IN DESIGN. ALL-IN-ALL, WE HAVE MORE THAN 50 MAJOR TEST FACILITIES IN SAN JOSE WHICH HAVE CONTRIBUTED TO THE BWR/6-MARK III DESIGN AND SUPPORTING TECHNOLOGY BASE -- AND THESE HAVE BEEN AUGMENTED BY NUMEROUS OFF-SITE TESTS CONDUCTED AT VENDOR FACILITIES AND OPERATING

REACTORS.

I MENTION THIS TESTING TO GIVE YOU SOME APPRECIATION OF BOTH THE DEPTH OF ENGINEERING WHICH SUPPORTS THE BWR/6 - MARK III DESIGN -- AND THE REASONS WE HAVE ADOPTED AN EVOLUTIONARY -- RATHER THAN REVOLUTIONARY -- APPROACH TO DESIGN CHANGES. OUR EXPERIENCE HAS TAUGHT US TO BE SKEPTICAL -- TO ANTICIPATE UNFORSEEN PROBLEMS WITH EVEN THE BEST-INTENTIONED DESIGN "IMPROVEMENTS" -- AND THAT THERE IS NO SHORTCUT FOR "TEST BEFORE USE" TO ELIMINATE DESIGN PROBLEMS BEFORE THEY REACH THE FIELD. OUR COMMITMENT TO THE "TEST BEFORE USE" PHILOSOPHY SOMETIMES CAUSES US TO APPEAR SLOW IN IMPLEMENTING DESIGN IMPROVEMENTS OTHERS BELIEVE TO BE "OBVIOUS", BUT IT HAS PAID OFF MANY TIMES IN HELPING US FIND AND ELIMINATE DESIGN WEAKNESSES BEFORE THEY REACH THE FIELD. WE BELIEVE -- IN THE LONG RUN -- "TEST BEFORE USE" IS IN THE BEST INTEREST OF THE PUBLIC, OUR CUSTOMERS AND OURSELVES -- EVEN IF IT COSTS MORE AND TAKES LONGER IN THE SHORT TERM.

SLIDE 19 - RECENT GE EVALUATIONS

WE HAVE SO FAR DISCUSSED THE EVOLUTION OF THE BWR AND TODAY'S BWR/6 - MARK III - STRIDE DESIGN. WITH THIS BACKGROUND I'D NOW LIKE TO CHANGE COURSE AND PRESENT THE RESULTS OF GENERAL ELECTRIC COMPANY REEVALUATIONS OF THE GE BOILING WATER REACTOR PERFORMED SINCE THE MARCH 1979 ACCIDENT AT THREE MILE ISLAND.

THE OBJECTIVE OF THESE EVALUATIONS WAS TO ASSESS THE CAPABILITY OF THE GE-BWR TO PROTECT BOTH THE PUBLIC AND THE PLANT CAPITAL INVESTMENT AGAINST DEGRADED TRASIENT AND ACCIDENT EVENTS. THEY ENCOMPASSED A BROAD SPECTRUM OF INITIATING EVENTS RANGING FROM TRANSIENTS WITHOUT PIPE BREAK TO LARGE PIPE BREAK ACCIDENTS. THEY INCLUDED PERFORMANCE ANALYSES OF THE PLANT RESPONSE TO DEGRADED EVENTS, RELIABILITY ANALYSES TO DETERMINE THE PROBABILITY OF EACH EVENT DETERIORATING TO THE POINT OF SIGNIFICANT CORE DAMAGE, AND ASSESSMENTS OF POTENTIAL OFF-SITE CONSEQUENCES.

I WILL PRESENT THE RESULTS OF THESE ANALYSES THIS MORNING IN THREE PARTS. FIRST, I WILL DISCUSS CERTAIN FEATURES OF THE GE-BWR WHICH OUR ANALYSES HAVE IDENTIFIED AS PROVIDING HIGHLY EFFECTIVE PLANT PROTECTION FOR MANY OF THE DEGRADED EVENTS WE STUDIED. SECOND, I WILL DISCUSS SOME OF THE GE-BWR IMPROVEMENTS WHICH ARE BEING MADE AND CONSIDERED SINCE THREE MILE ISLAND. FINALLY, I WILL PRESENT SOME QUANTITATIVE RESULTS OF OUR RELIABILITY ANALYSES FOR THE GE-BWR.

FOR CONSISTENCY MY REMARKS WILL BE RESTRICTED TO OUR CURRENT BWR/6 PRODUCT IN A REFERENCE MARK III PRESSURE SUPPRESSION CONTAINMENT, ALTHOUGH MANY OF THE CONCLUSIONS ARE ALSO APPLICABLE TO EARLIER GE-BWR'S.

Slide 20 - Core Protection Functions

The key to plant protection in any nuclear plant is protection of the reactor core. This involves three functions -- scram -- supplying water to the core -- and removing decay heat. The scram function has been extensively discussed before this Committee in connection with planned ATWS rulemaking -- so I will focus today's discussion on the last two functions. Our analyses have identified several features of the BWR/6 which provide high assurance that all of these functions will be performed -- even during highly degraded events.

Slide 21 - Systems To Supply Water To Core

Let's look first at systems to supply water to the core, the BWR/6 has four high pressure systems and three low pressure systems which can provide water to the reactor core. These systems contain 13 pumps -- 11 of which have sufficient capacity to individually provide makeup water for decay heat removal in a non-break event. At both high and low pressures these systems provide the capability to spray the core from above and flood it from below. The top entry core sprays provide core cooling capability even if the core is completely uncovered.

Numerous water supplies are an inherent feature of the BWR direct cycle in which normal pumping systems -- feedwater, condensate, isolation cooling, control rod drive cooling -- routinely provide water to the reactor vessel. These systems are backed up by the emergency cooling systems.

Slide 22 - BWR/6 Plant

The systems to supply water to the BWR core are not only numerous -- they are diverse in terms of physical location, motive power and water supplies. Some perspective of this diversity is provided by an isometric view of the plant. The condensate and feedwater pumps are located in the turbine building, the emergency core cooling pumps are located in the auxiliary building, and the control rod drive cooling pumps are located in the fuel building. Motive power is provided by two off-site power sources, three diesel generators located in two physically separated diesel generator buildings, and -- in the case of the reactor core isolation cooling system -- by a steam driven turbine. Available water supplies include the condensate storage tank, the condenser hot well, and the suppression pool -- which are again physically separated. While not all pumps have access to all sources of motive power or water, the overall network provides a water delivery capability and reliability far in excess of that required to meet regulatory requirements.

THE DIVERSITY OF SYSTEMS TO SUPPLY WATER TO THE BWR CORE PLAYED AN IMPORTANT ROLE IN PREVENTING DAMAGE TO THE REACTOR CORE DURING THE 1975 FIRE AT THE BROWNS FERRY PLANT. AT ONE TIME DURING THE FIRE, WATER LEVEL WAS MAINTAINED ABOVE THE CORE BY NON-SAFETY GRADE PUMPS UNTIL ADDITIONAL WATER MAKEUP CAPABILITY COULD BE RESTORED. THIS EXPERIENCE SERVES TO UNDERSCORE THE VALUE OF THE NUMEROUS DIVERSE SYSTEMS TO PROVIDE WATER TO THE REACTOR VESSEL WHICH ARE INHERENT IN THE DIRECT CYCLE BOILING WATER REACTOR.

SLIDE 23 - DEPRESSURIZATION

THE NUMEROUS BWR/6 WATER SUPPLIES ARE MADE EVEN MORE EFFECTIVE BY THE CAPABILITY TO RAPIDLY DEPRESSURIZE THE REACTOR. THIS CAN BE DONE IN AS LITTLE AS 5 MINUTES BY RELIEVING STEAM TO THE SUPPRESSION POOL. THE SUPPRESSION POOL IS SIZED TO ACCEPT A FULL BLOWDOWN OF THE PRIMARY SYSTEM WITHIN NORMAL OPERATING TEMPERATURE LIMITS. THIS RAPID DEPRESSURIZATION CAPABILITY CAN BE EMPLOYED IF NEEDED TO MAKE ALL OF THE BWR/6 WATER SUPPLIES AVAILABLE FOR ALL EVENTS.

SLIDE 24 - DECAY HEAT REMOVAL

IF REACTOR VESSEL WATER LEVEL IS MAINTAINED, DECAY HEAT REMOVAL IN BWR/6 IS PASSIVE. STRONG NATURAL CIRCULATION INTERNAL TO THE REACTOR VESSEL, AND STEAM RELIEF TO EITHER THE MAIN CONDENSER OR SUPPRESSION POOL HEAT SINKS, COMBINE TO PROVIDE THIS PASSIVE DECAY HEAT REMOVAL. BECAUSE OF PASSIVE DECAY HEAT REMOVAL, BWR/6 CORE

COOLING REQUIRES ONLY THAT ADEQUATE WATER LEVEL BE MAINTAINED IN THE REACTOR VESSEL. THIS ENABLES THE OPERATOR TO CONCENTRATE HIS ATTENTION DURING AN INVENTORY THREATENING EVENT ON THE SINGLE PRIMARY OBJECTIVE OF MAINTAINING REACTOR WATER LEVEL -- ALL OTHER CONSIDERATIONS ARE SECONDARY. THIS GREATLY SIMPLIFIES THE OPERATOR EMERGENCY RESPONSE AS WE WILL SEE SHORTLY.

SLIDE 25 - WATER LEVEL MEASUREMENT

BECAUSE OF ITS CENTRAL IMPORTANCE TO CORE COOLING, WATER LEVEL IS MEASURED DIRECTLY ON THE BWR/6 REACTOR VESSEL. THE MEASUREMENT IS BASED ON DIFFERENTIAL PRESSURE TECHNIQUES, PROVIDES CONTINUOUS MEASUREMENT OF WATER LEVEL ABOVE THE TOP OF THE ACTIVE FUEL, AND IS HIGHLY REDUNDANT. THE MEASURED WATER LEVEL IS USED FOR BOTH AUTOMATIC AND MANUAL INITIATION OF PLANT PROTECTION SYSTEMS.

SLIDE 26 - SUPPRESSION POOL HEAT SINK

I WOULD NOW LIKE TO LEAVE THE REACTOR AND DISCUSS THE ROLE OF THE MARK III PRESSURE SUPPRESSION CONTAINMENT IN PROVIDING PROTECTION AGAINST DEGRADED TRANSIENT AND ACCIDENT EVENTS. LET'S FIRST DISCUSS THE SUPPRESSION POOL HEAT SINK WHICH PROVIDES A VERY LARGE QUENCH TANK FOR TRANSIENT AND ACCIDENT VENTS.

THIS SLIDE COMPARES THE STORED ENERGY IN BWR/6 WITH THE PASSIVE HEAT CAPACITY OF THE MARK III SUPPRESSION POOL HEAT SINK.

THE ENERGY STORED IN THE BWR/6 PRIMARY SYSTEM IS 470 BILLION JOULES. FOR RELIEF VALVE DISCHARGES THE MARK III SUPPRESSION POL IS SIZED TO ACCEPT 520 BILLION JOULES WITHOUT EXCEEDING ITS 66^{0} C NORMAL OPERATING LIMIT. THIS CAPABILITY OF THE CONTAINMENT TO ACCEPT TOTAL BLOWDOWN OF THE PRIMARY SYSTEM IS THE KEY TO THE REACTOR DEPRESSURIZATION CAPABILITY -- WHICH WE HAVE ALREADY SEEN CONTRIBUTES SIGNIFICANTLY TO BWR/6 CORE PROTECTION.

FOR A LOSS OF COOLANT ACCIDENT THE MARK III SUPPRESSION POOL IS SIZED TO ACCEPT 810 BILLION JOULES WITHOUT EXCEEDING ITS 85^{0}C EMERGENCY OPERATING LIMIT. THIS PROVIDES SEVERAL HOURS OF POST-ACCIDENT DECAY HEAT STORAGE CAPACITY BEFORE ACTIVE CONTAINMENT COOLING SYSTEMS ARE NEEDED. THIS PASSIVE CAPABILITY FREES THE OPERATOR TO FOCUS HIS ATTENTION ON THE REACTOR DURING THE EARLY STAGES OF AN ACCIDENT SEQUENCE.

Slide 27 - Plant Protection Improvements

OUR POST-TMI EVALUATIONS OF THE BWR HAVE NOT ONLY REINFORCED OUR CONFIDENCE IN THE EXISTING BWR PLANT PROTECTION FEATURES -- THEY HAVE ALSO IDENTIFIED A NUMBER OF PLANT PROTECTION IMPROVEMENT OPPORTUNITIES WHICH WE ARE EITHER INCORPORATING OR CONSIDERING IN THE DESIGN OF FUTURE BWR/6 PLANTS.

SLIDE 28 - OPERATOR RESPONSE TO INVENTORY THREATENING EVENTS

ONE OF THE MOST SIGNIFICANT BWR PLANT PROTECTION IMPROVEMENTS UNDERWAY IS IN THE OPERATOR INTERFACE. THE DIRECT CYCLE BWR HAS SEVERAL ATTRACTIVE FEATURES WHICH ARE BEING EXPLOITED TO SIMPLIFY AND IMPROVE THE OPERATOR INTERFACE. ONE IS THE INHERENT SIMPLICITY OF THE MACHINE -- ONE VESSEL WITH WATER LEVEL MEASURED DIRECTLY ON IT, AND PASSIVE DECAY HEAT REMOVAL TO THE SUPPRESSION POOL IF WATER LEVEL IS MAINTAINED. THIS SIMPLICITY ENABLES THE OPERATOR TO FOCUS HIS ATTENTION DURING AN EMERGENCY ON THE PRIMARY OBJECTIVE OF RESTORING AND MAINTAINING THE WATER LEVEL IN THE REACTOR. IN ADDITION, THE BWR/6 OPERATES IN THE BOILING MODE FAMILIAR TO PLANT OPERATORS UNDER BOTH NORMAL AND EMERGENCY CONDITIONS.

BECAUSE OF ITS SIMPLICITY, THE BWR/6 PERMITS A COMMON OPERATOR RESPONSE TO ALL INVENTORY THREATENING EVENTS. THIS RESPONSE IS BASED ON SYMPTOMS RATHER THAN EVENT DIAGNOSIS AND CONSISTS OF:

- FIRST - MAINTAIN REACTOR WATER LEVEL. THIS IS ACCOMPLISHED USING HIGH PRESSURE SYSTEMS OR BY DEPRESSURIZING AND USING LOW PRESSURE SYSTEMS. DURING THE TIME THE OPERATOR IS DOING THIS THE PASSIVE DECAY HEAT REMOVAL PROCESSES WILL TRANSFER DECAY HEAT FROM THE REACTOR TO THE SUPPRESSION POOL HEAT SINK WHERE IT CAN BE STORED FOR SEVERAL HOURS IF NECESARY WITHOUT OPERATOR ACTION.
- SECOND - ESTABLISH A LONG TERM HEAT SINK. THIS IS ACCOMPLISHED BY REGAINING THE MAIN CONDENSER, OR BY ESTABLISHING SUPPRESSION POOL COOLING.

Slide 29 - Emergency Procedure Guidelines

The existence of a common operator response to all inventory threatening events has enabled us to develop a simple and yet comprehensive set of emergency guidelines -- based on symptoms rather than event diagnosis -- for BWR operators. The development of these guidelines has been a joint effort of BWR Owners and GE since Three Mile Island. Two emergency guidelines -- one addressing reactor water level control and one addressing containment control -- are sufficient for all inventory threatening events. Each guideline has certain well-defined symptoms for entry, and provides the operator with prioritized actions for accomplishing the overall function. For example the level control guideline gives the operator specific guidance on how to deploy his 13 pumps to restore and maintain water level in the reactor vesel.

We are quite proud of these symptom oriented emergency guidelines for the BWR and believe they contribute significantly to minimizing the chance of operator error. In my view they represent the single most important contribution to BWR safety since the Three Mile Island accident.

SLIDE 30 - LEVEL CONTROL DISPLAY

ANOTHER OF THE OPERATOR INTERFACE IMPROVEMENTS WHICH IS BEING DEVELOPED IS AN EMERGENCY RESPONSE INFORMATION SYSTEM THAT WILL ASSIST THE OPERATOR IN RECOGNIZING AND RESPONDING TO POTENTIALLY UNSAFE CONDITIONS.

FOR THE GE-BWR THE EMERGENCY RESPONSE INFORMATION SYSTEM IS BEING DESIGNED TO, AMONG OTHER THINGS, SUPPORT THE OPERATOR IN FOLLOWING THE PREVIOUSLY DISCUSSED EMERGENCY GUIDELINES. THIS PERMITS THE BASIC SIMPLICITY OF THE BWR SYSTEM AND THE SYMPTOM-BASED EMERGENCY GUIDELINES TO BE REFLECTED IN VERY SIMPLE AND YET VERY EFFECTIVE OPERATOR INTERFACE DISPLAYS.

THE LEVEL CONTROL DISPLAY FOR ASSISTING THE OPERATOR IN HANDLING A REACTOR WATER INVENTORY THREATENING EVENT IS AN EXAMPLE. THIS RELATIVELY SIMPLE DISPLAY -- WHICH BY THE WAY IS STILL BEING REFINED -- PROVIDES THE OPERATOR THE KEY INFORMATION HE NEEDS TO KNOW ABOUT REACTOR WATER LEVEL -- ITS LOCATION RELATIVE TO THE CORE -- WHETHER IT IS INCREASING OR DECREASING -- WHETHER THE REACTOR IS ISOLATED -- WHETHER ANY RELIEF VALVES ARE OPEN -- THE OPERATIONAL AND STANDBY STATUS OF ALL WATER DELIVERY SYSTEMS -- AND THE INVENTORY OF THE SEVERAL AVAILABLE WATER SOURCES.

WE SAW EARLIER THAT THE BWR OPERATOR'S PRIMARY OBJECTIVE DURING AN EMERGENCY IS TO RESTORE AND MAINTAIN CONTROL OF REACTOR WATER LEVEL. HE ONLY HAS TO DO THIS ONE THING RIGHT TO AVOID DAMAGE TO THE REACTOR CORE. THE LEVEL CONTROL DISPLAY PROVIDES HIM THE KEY INFORMATION NEEDED TO DO THIS ONE THING RIGHT.

SLIDE 31 - BWR/6 SYSTEM IMPROVEMENTS

SOME IMPROVEMENTS TO BWR/6 PLANT PROTECTION SYSTEMS ARE ALSO BEING MADE OR CONSIDERED AS A RESULT OF THE RECENT GE-BWR EVALUATIONS. WATER DELIVERY SYSTEM IMPROVEMENTS INCLUDE:

- REACTOR CORE ISOLATION COOLING SYSTEM LOGIC IMPROVEMENTS -- TO IMPROVE SYSTEM AVAILABILITY AND PERMIT THE SYSTEM TO REMAIN OPERABLE DURING LOSS OF COOLANT EVENTS.
- AUTO-DEPRESSURIZATION FOR NON-BREAK EVENTS -- TO PROVIDE MORE RELIABLE ACCESS TO LOW PRESSURE WATER DELIVERY SYSTEMS IF HIGH PRESSURE SYSTEMS ARE UNAVAILABLE.
- AUTO-RESTART OF HIGH PRESSURE CORE SPRAY TO BACKUP THE OPERATOR IN THE LONGER TERM MAINTENANCE OF REACTOR WATER LEVEL FOLLOWING AN EMERGENCY.

Plant protection improvements to the BWR/6 scram system are also being made. Improvements to the scram discharge volume designs were made following evaluations of the partial scram event at Browns Ferry in 1980. In addition, ATWS modifications will be incorporated in the BWR/6 standard design consistent with NRC ATWS requirements to be established. These improvements will further enhance the overall scram reliability in BWR/6.

A significant BWR plant protection improvement is also being studied in the area of decay heat removal. We have already discussed the fact that -- provided reactor water level is maintained -- the BWR provides passive removal of decay heat to the suppression pool. This is accomplished through natural circulation internal to the reactor vessel and steam relief to the suppression pool. From there the normal decay heat removal path -- with the reactor isolated -- is through the residual heat removal system -- operating in the pool cooling mode -- to the ultimate heat sink.

The improvement being considered is to provide an alternate path for decay heat removal from the suppression pool via a containment overpressure relief vent to the atmoshpere. This path would be activated only in the event of failure of both residual heat removal systems -- and even then -- only after many hours of decay heat storage in the suppression pool and containment structures.

THE ADDITION OF CONTAINMENT OVERPRESSURE RELIEF CAPABILITY WOULD PROVIDE ANOTHER AVENUE OF PROTECTION AGAINST ONE OF THE DOMINANT BWR ACCIDENT SEQUENCES IDENTIFIED IN WASH-1400 -- NAMELY, LOSS OF ALL RESIDUAL HEAT REMOVAL SYSTEMS. IN WASH-1400 IT WAS ASSUMED THAT SUCH LOSS WOULD EVENTUALLY LEAD TO CONTAINMENT OVERPRESSURIZATION, CONTAINMENT FAILURE, LOSS OF THE SUPPRESSION POOL -- AND ULTIMATELY TO LOSS OF MAKEUP WATER TO THE REACTOR, AND CORE MELT. CONTAINMENT OVERPRESSURE RELIEF WOULD PRECLUDE THE LOSS OF RESIDUAL HEAT REMOVAL SYSTEMS FROM DETERIORATING INTO A CORE DAMAGE EVENT. IT WOULD ALSO GREATLY ENHANCE THE CAPABILITY OF THE BWR TO HANDLE A STATION BLACKOUT OF EXTENDED DURATION.

SLIDE 32 - PROBABILITY OF CORE DAMAGE

SO FAR I HAVE DISCUSSED BWR/6 PLANT PROTECTION IN A LARGELY QUALITATIVE SENSE -- DESCRIBING FEATURES WHICH OUR STUDIES HAVE IDENTIFIED AS PROVIDING EFFECTIVE PROTECTION AGAINST MANY DEGRADED EVENTS, AND PLANNED IMPROVEMENTS. WE HAVE ALSO PERFORMED QUANTITATIVE RELIABILITY ANALYSES OF THE BWR/6.

OUR RELIABILITY ANALYSES STARTED WITH THE 1975 REACTOR SAFETY STUDY (WASH-1400), AND EVALUATED RELIABILITY IMPROVEMENTS IN BWR/6 RELATIVE TO WASH-1400 -- ACCOUNTING FOR THE BWR/6 IMPROVEMENTS WE HAVE JUST DISCUSSED.

THE RESULTS OF OUR BWR/6 RELIABILITY ANALYSES ARE SHOWN HERE AND COMPARED WITH COMPARABLE RESULTS FROM WASH-1400 FOR EIGHT "CONSOLIDATED" INITIATING EVENTS. THE BARS REPRESENT PROBABILITY OF CORE DAMAGE PER REACTOR YEAR -- WITH THE HIGHER BARS BEING WASH 1400 RESULTS AND THE SMALLER BARS REPRESENTING BWR/6 RESULTS. THE BWR/6 RESULTS REFLECT A FACTOR OF 20 REDUCTION IN THE OVERALL PROBABILITY OF CORE DAMAGE.

Slide 33 - Function of Containment

ONE OF THE MORE SIGNIFICANT CONCLUSIONS FROM OUR RECENT ASSESSMENTS OF THE BWR HAS BEEN IN THE AREA OF ACCIDENT MITIGATION. SPECIFICALLY, OUR RECENT STUDIES INDICATE THAT THE OFFSITE CONSEQUENCES OF A SEVERE ACCIDENT -- EVEN ONE INVOLVING FAILURE OF THE PRIMARY CONTAINMENT -- ARE DECADES LOWER THAN PREVIOUSLY ESTIMATED.

THE FUNCTION OF A CONTAINMENT IN A NUCLEAR PLANT IS TO PROTECT THE PUBLIC FROM EXCESSIVE DOSE IN THE EVENT OF A SEVERE ACCIDENT. IN THE CASE OF BWR PRESSURE SUPPRESSION CONTAINMENTS, THIS FUNCTION IS ACCOMPLISHED TWO WAYS. THE FIRST IS THROUGH CONTAINMENT BARRIERS -- PRIMARY AND SECONDARY BARRIERS -- WHICH ARE DESIGNED TO MAINTAIN THEIR INTEGRITY FOR ALL DESIGN BASIS EVENTS, AND HAVE SUFFICIENT MARGIN TO MAINTAIN THEIR INTEGRITY FOR MOST EVENTS BEYOND THE DESIGN BASIS. THE SECOND WAY OF PERFORMING THE CONTAINMENT FUNCTION IS THROUGH FILTERED CONTAINMENT VENTING -- WHICH PROVIDES AN ADDITIONAL AVENUE OF PROTECTION FOR EVENTS WELL BEYOND THE DESIGN BASIS.

Slide 34 - Suppresion Pool Scrubbing Pathways

Effective filtration -- or scrubbing -- of potential releases from the containment is an inherent safety feature of the Mark III pressure suppression containment. Potential releases from the primary system, resulting from either degraded transient or accident events, pass through the suppression pool before reaching the main containment. The suppression pool effectively scrubs halogens and particulates from the released mixture -- yielding decontamination factors for the pool which are expected to be in the range of at least 100-1000. The suppression pool decontamination factor is multiplicative with decontamination factors arising from natural plateout mechanisms and containment sprays, so that the overall decontamination factor in a pressure suppression containment is expected to be much larger still.

Slide 35 - Yield Pressures

Our studies have shown that there is a very high likelihood that the suppression scrubbing function will be retained even in extreme accident sequences which might fail the primary containment. This arises from the very substantial structural capability of the drywell -- 160 psig for the drywell head and 280 psig for the drywell wall -- and the fact that the most probable failure point for the primary containment is near the top and would leave the suppression pool intact. This provides high assurance that the containment function -- protecting the public from excessive dose -- would be performed even

IF THE CONTAINMENT BARRIERS WERE TO FAIL.

Slide 36 - Importance of Fission Product Scrubbing

THE IMPORTANCE OF THE FISSION PRODUCT SCRUBBING MECHANISM IN THE MARK III PRESSURE SUPPRESSION CONTAINMENT IS SHOWN IN THIS SLIDE -- WHICH SHOWS LIFETIME WHOLE BODY DOSES RECEIVED BY AN INDIVIDUAL FOLLOWING A SEVERE ACCIDENT -- AS A FUNCTION OF THE INDIVIDUAL'S LOCATION DOWNWIND FROM THE SITE. THE CALCULATION WAS PERFORMED USING CORE MELT SOURCE TERMS, ASSUMING FAILURE OF THE PRIMARY CONTAINMENT BARRIER AT 4 HOURS, AND NO EVACUATION. FOR REFERENCE THE CHART SHOWS THE FATALITY THRESHOLD OF 320 REM AND THE 10CRF100 LIMIT OF 25 REM -- BELOW WHICH OBSERVABLE HEAL[illegible]CTS WOULD NOT OCCUR. THE UPPER CURVE SHOWS THE RESULTS GIVING NO CREDIT FOR SUPPRESSION POOL SCRUBBING OR OTHER DECONTAMINATION PROCESSES AS WAS DONE IN WASH-1400. WITH THIS ASSUMPTION FATAL DOSES ARE CALCULATED TO OCCUR MORE THAN 10 MILES FROM THE PLANT. THE LOWER CURVES ARE BASED ON A MORE REALISTIC TREATMENT OF SUPPRESSION POOL SCRUBBING AND OTHER DECONTAMINATION PROCESSES WHICH ARE EXPECTED TO PRODUCE OVERALL DECONTAMINATION FACTORS IN THE RANGE OF 1000-10,000. UNDER THESE MORE REALISTIC CONDITIONS NO OBSERVABLE HEALTH AFFECTS WOULD OCCUR BEYOND THE SITE BOUNDARY.

WE ARE CURRENTLY CONDUCTING A SERIES OF SMALL SCALE SUPPRESSION POOL SCRUBBING EXPERIMENTS AT GENERAL ELECTRIC TO VERIFY THE DECONTAMINATION FACTORS USED IN OUR ANALYSES AND ARE WORKING WITH THE NRC STAFF TO OBTAIN FORMAL RECOGNITION AND CREDIT FOR THE SCRUBBING CAPABILITY OF THE SUPPRESSION POOL. IN ADDITION, EPRI HAS RECENTLY INITIATED A PROGRAM WITH BATTELLE COLUMBUS TO QUANTIFY DECONTAMINATION FACTORS IN WATER POOLS IN ALL LWR'S.

IT IS CLEAR FROM THESE STUDIES THAT FISSION PRODUCT SCRUBBING PHENOMENA DRAMATICALLY REDUCE THE POTENTIAL CONSEQUENCES OF A SEVERE ACCIDENT. THESE PHENOMENA SHOULD BE CAREFULLY CONSIDERED AND COULD HAVE A MAJOR IMPACT ON THE OUTCOME OF CURRENTLY PLANNED REGULATORY ACTIONS IN THE AREAS OF SEVERE ACCIDENTS, EMERGENCY PLANNING, AND SITING CRITERIA.

SLIDE 37 - SUMMARY

WE HAVE COVERED A LOT OF GROUND THIS MORNING AND I WOULD LIKE TO END BY TAKING ONE MINUTE TO BRIEFLY SUMMARIZE MY MESSAGE.

- THE BWR/6-MARK III IS THE PRODUCT OF 25 YEARS OF DELIBERATE EVOLUTION AND SIMPLIFICATION OF THE BOILING WATER REACTOR AND PRESSURE SUPPRESSION CONTAINMENT.

- IT IS BACKED BY 400 REACTOR YEARS OF FIELD EXPERIENCE AND HAS BEEN VERY THOROUGHLY TESTED UNDER OUR "TEST BEFORE USE" PHILOSOPHY.

- IT RETAINS THE PROVEN FEATURES OF EARLIER BWR DESIGNS INCLUDING SIMPLICITY, MANY WATER DELIVERY SYSTEMS, RAPID DEPRESSURIZATION CAPABILTIY, STRONG NATURAL CIRCULATION INTERNAL TO THE VESSEL, PRESSURE SUPPRESION CONTAINMENT, AND MANY OTHERS.

- IN ADDITION, IT INCORPORATES THE VERY LATEST AND BEST SAFETY TECHNOLOGY INCLUDING RECENT IMPROVEMENTS IN THE AREA OF SCRAM SYSTEM RELIABILITY, WATER DELIVERY SYSTEM INITIATING LOGIC, OPERATOR INTERFACE IMPROVEMENTS, CONTAINMENT OVERPRESSURE RELIEF, AND SUPPRESSION POOL SCRUBBING CAPABILITY.

- THE BWR/6-MARK III DESIGN IS BACKED BY A SYSTEM ENGINEERED, TECHNICALLY INTEGRATED STANDARD NUCLEAR ISLAND DESIGN (STRIDE) AND BY GENERAL ELECTRIC'S STANDARD SAFETY ANALYSIS REPORT (GESSAR) FOR THE NUCLEAR ISLAND. BOTH STRIDE AND GESSAR HAVE BEEN ENGINEERED FOR AN ENVELOPE OF SITE CONDITIONS WHICH ENCOMPASS AN ESTIMATED 90% OF US SITES. WE BELIEVE THE STRIDE/GESSAR PACKAGE IS READY FOR ONE STEP LICENSING.

THE FIRST UNIT OF THIS NEW PRODUCT LINE -- THE KUO SHENG PLANT IN TAIWAN -- JUST RECENTLY WENT INTO OPERATION AFTER A 61 MONTH CONSTRUCTION SCHEDULE. BWR/6-MARK III HAS BEEN ENGINEERED FOR THE FUTURE AND WE EXPECT IT TO REMAIN OUR STANDARD PRODUCT OFFERING FOR SOME YEARS TO COME.

THANK YOU.

BWR FUTURE DIRECTIONS

PRESENTATION BY

D. R. WILKINS

TO

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

WASHINGTON, D.C.

JANUARY 7, 1982

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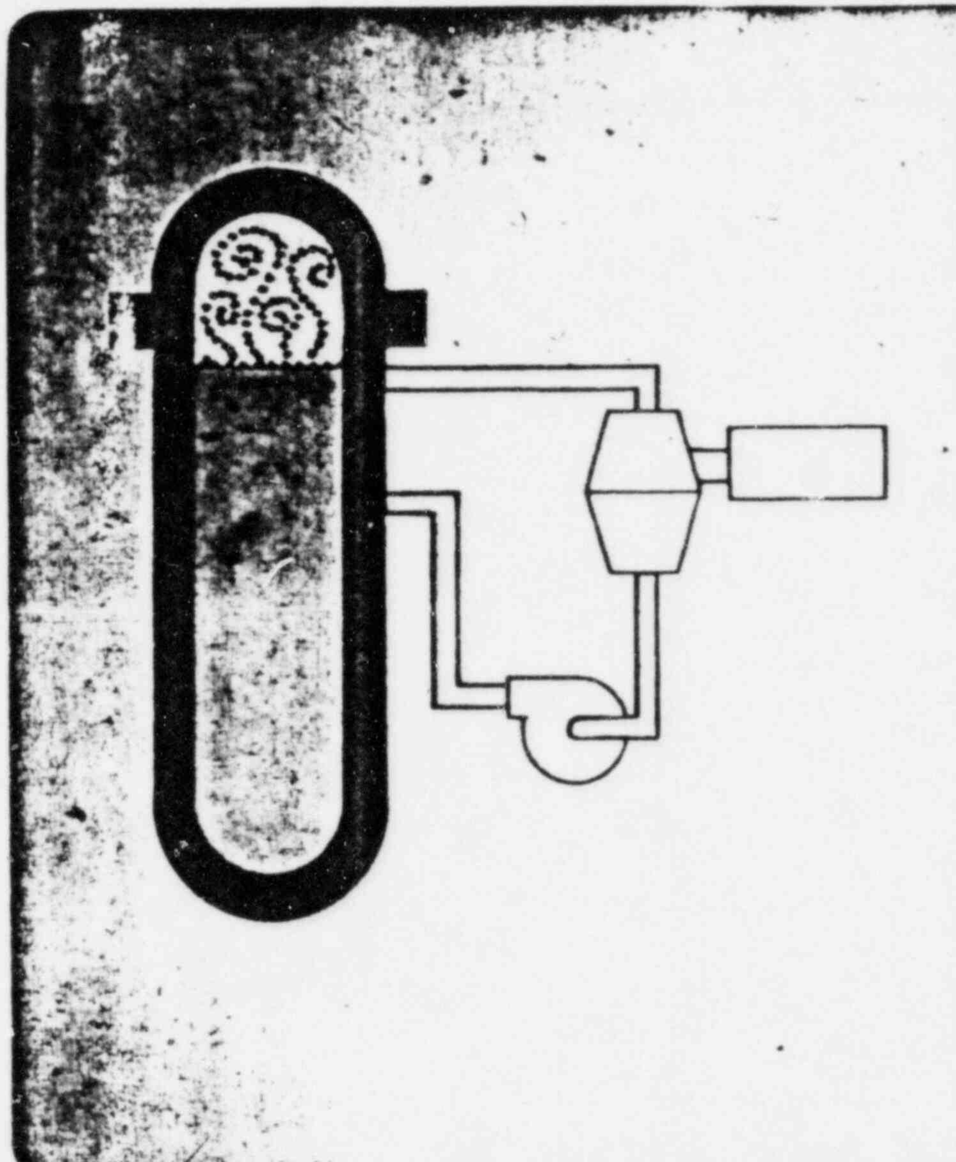

DIRECT CYCLE ADVANTAGES

- SIMPLICITY
- LOW PRESSURE
- NEGATIVE VOID COEFFICIENT
- NATURAL CIRCULATION
- POWER-FLOW COUPLING
- WATER SOURCES DIRECT TO VESSEL

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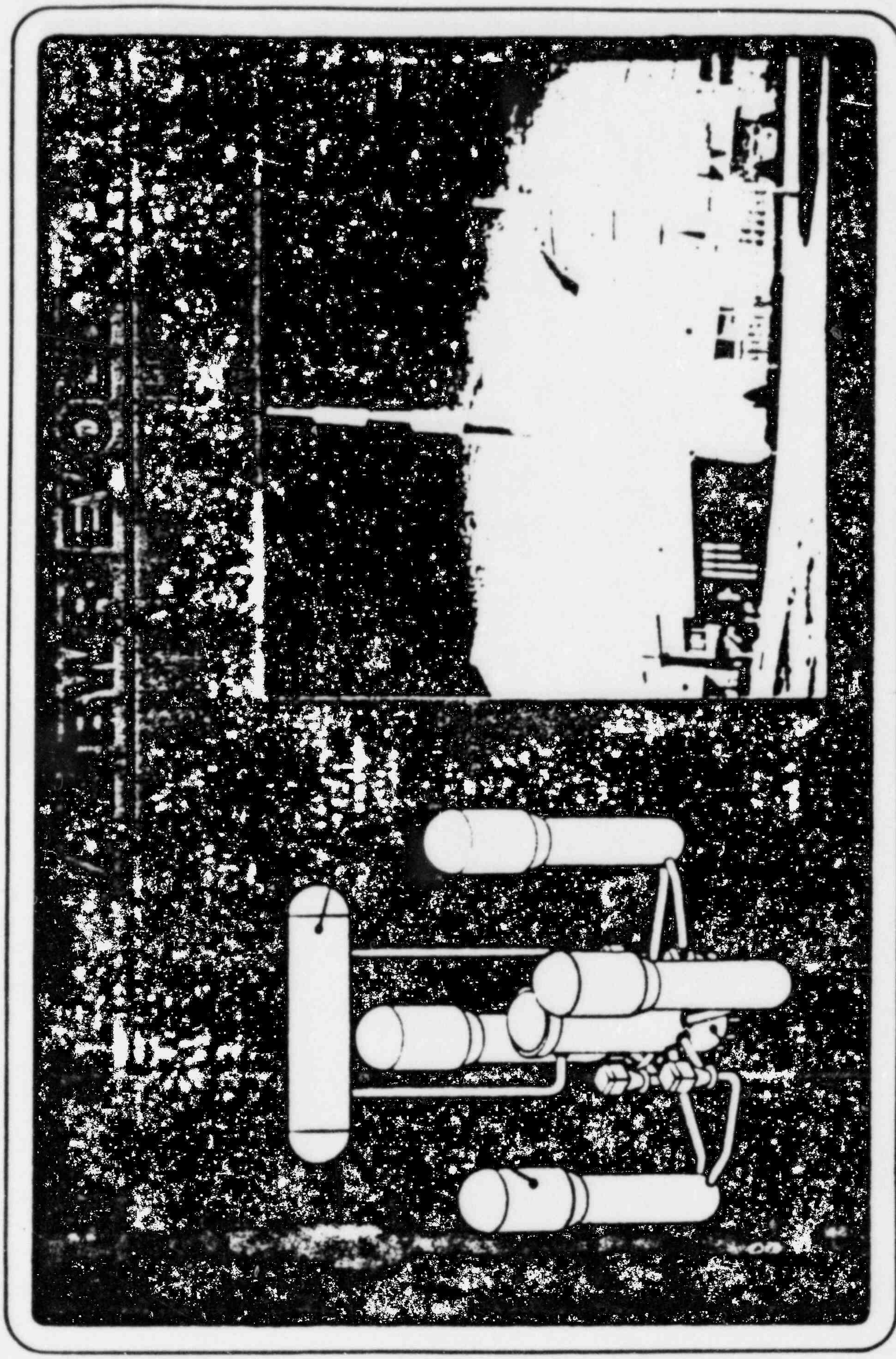

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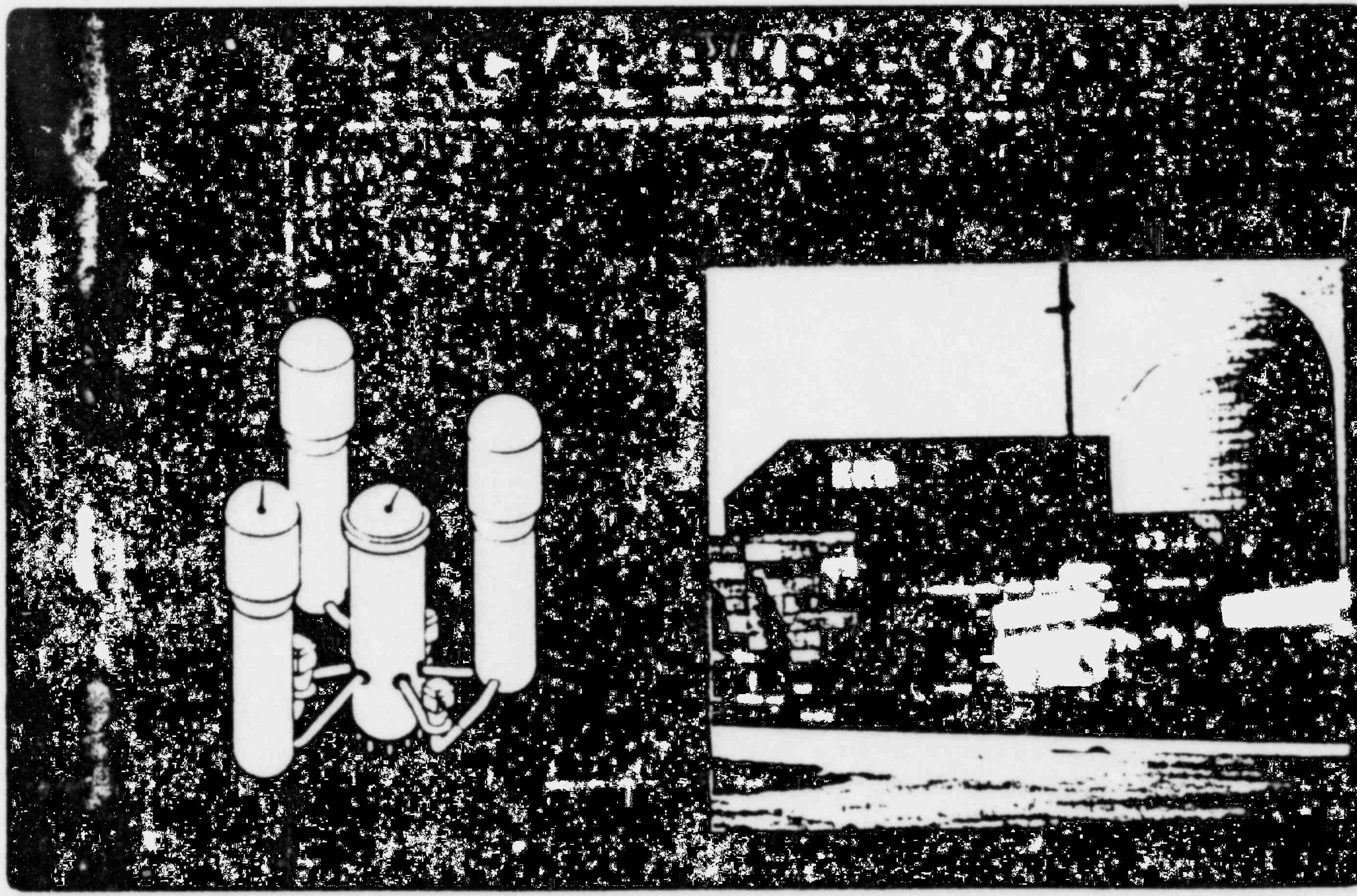

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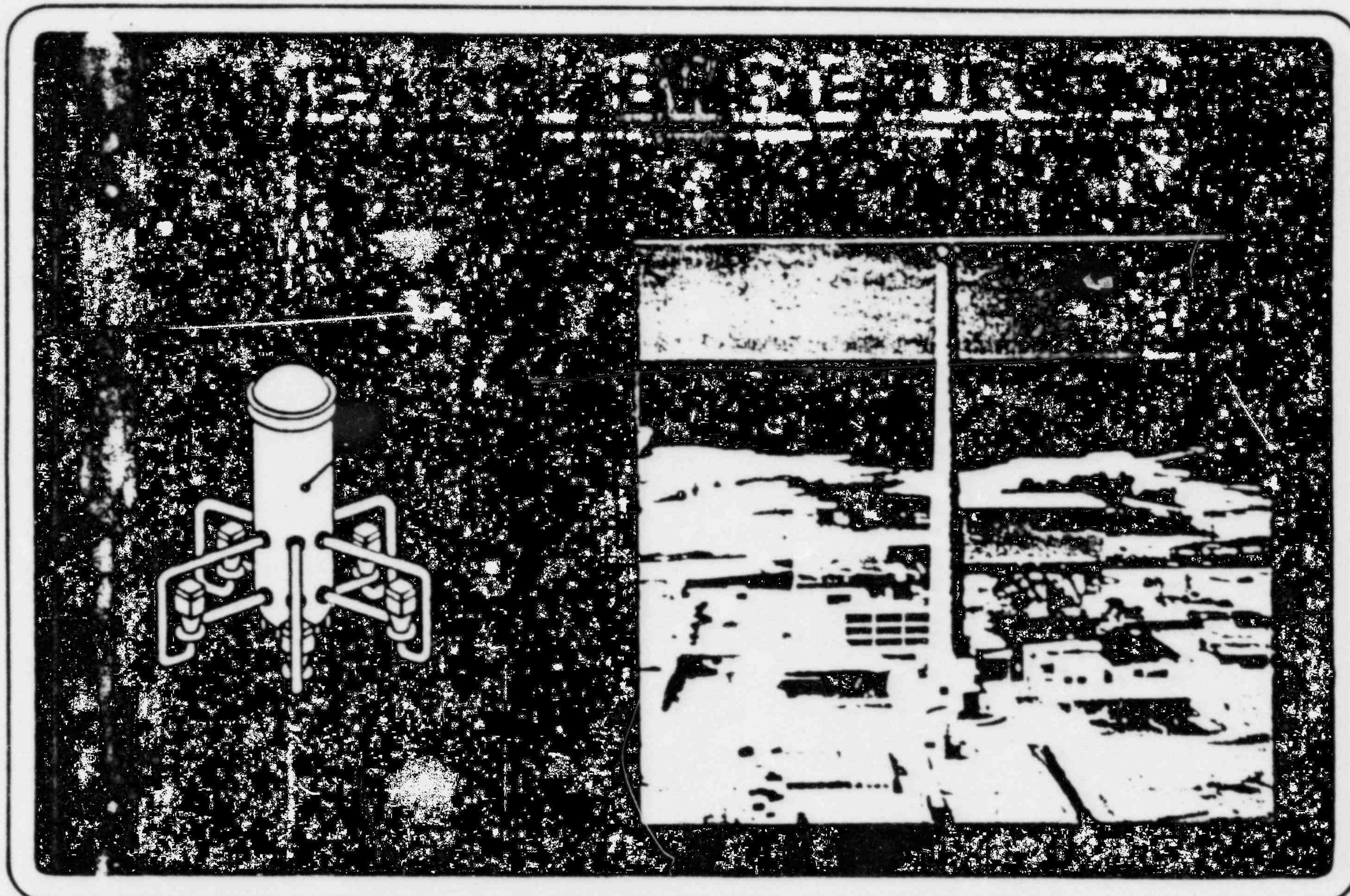

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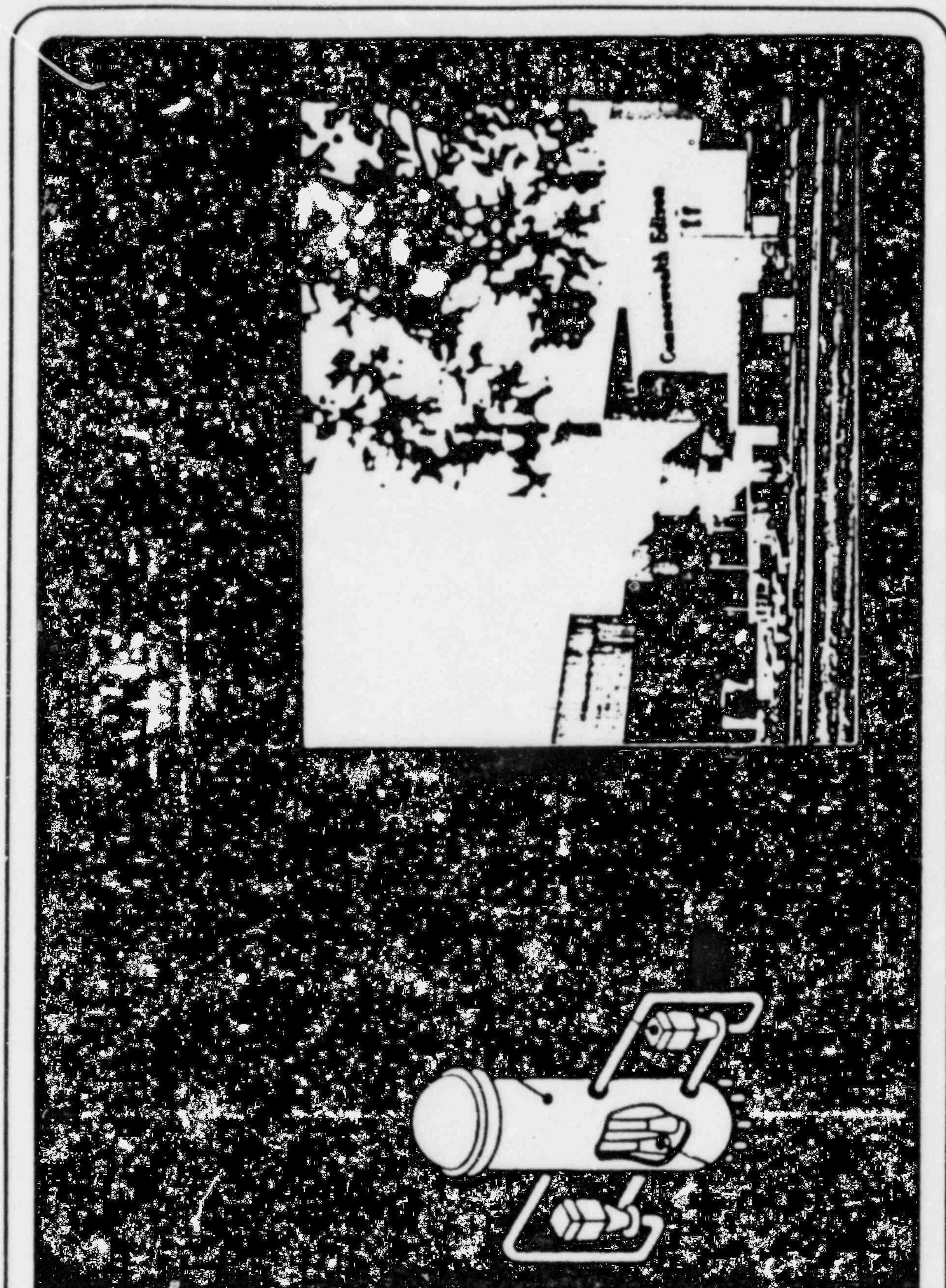

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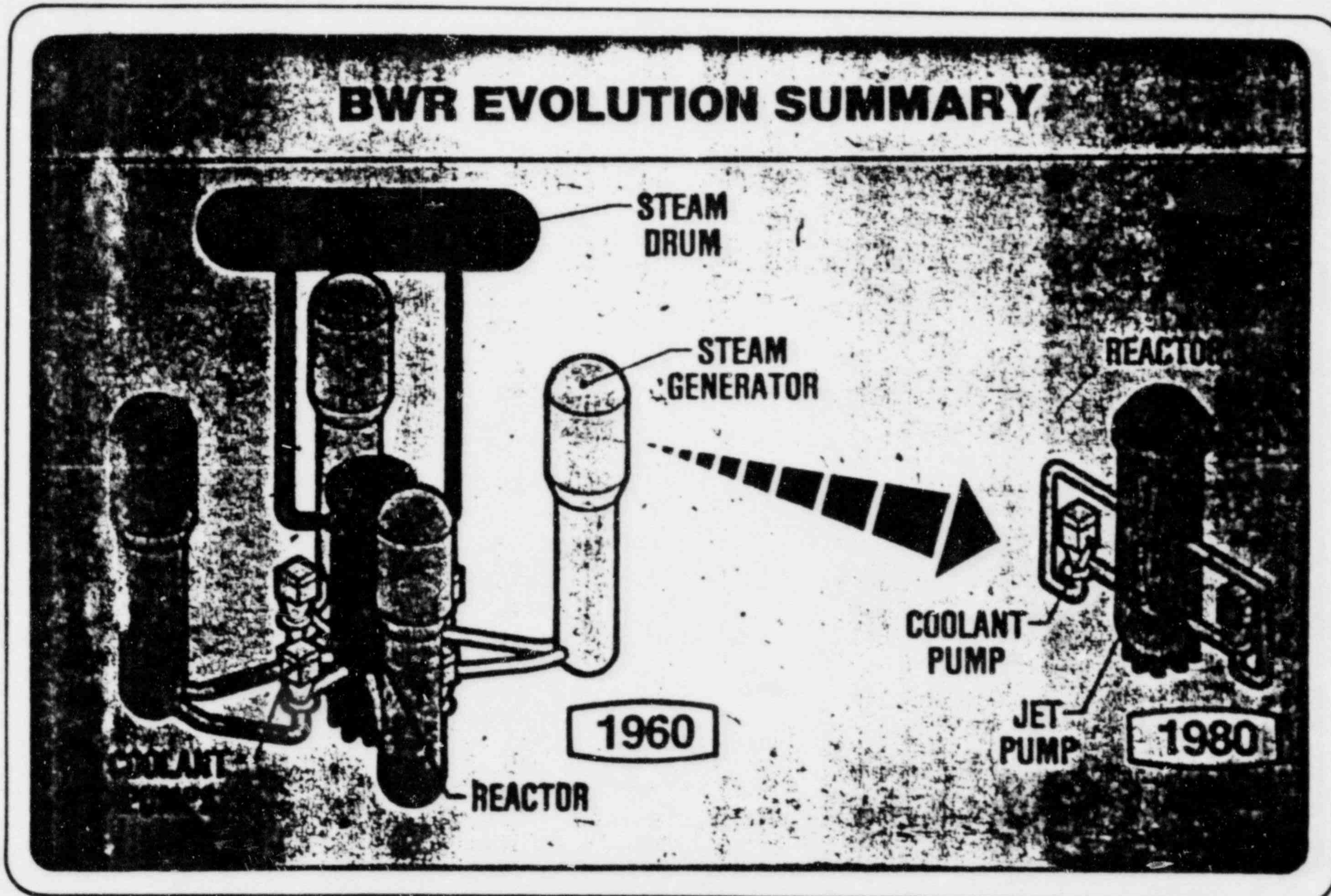
BWR EVOLUTION SUMMARY
STEAM DRUM
STEAM GENERATOR
1960
REACTOR
COOLANT PUMP
JET PUMP
1980

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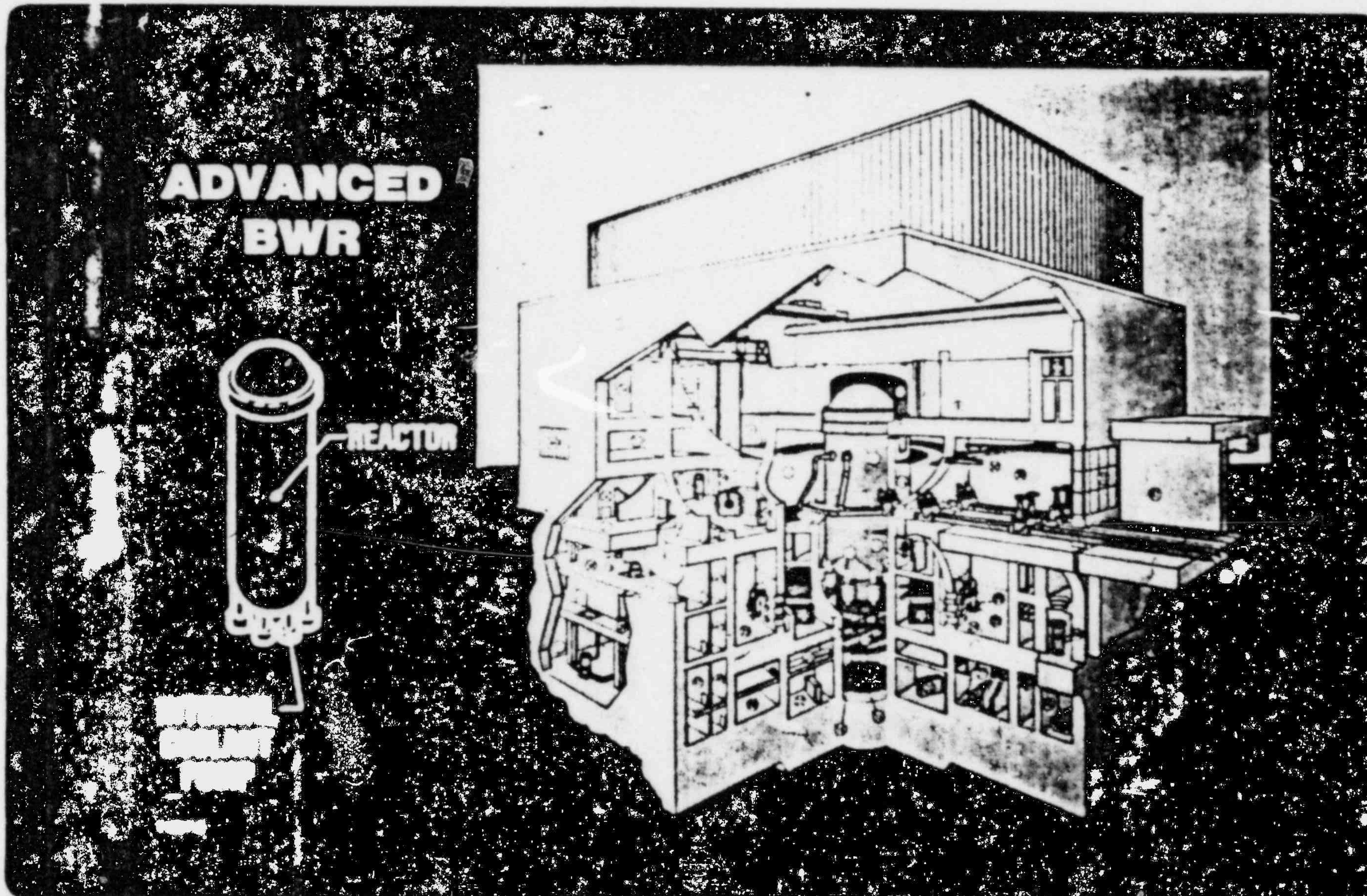

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PRESSURE SUPPRESSION ADVANTAGES

- High Heat Capacity
- Low Pressure
- LOCA Margins
- Blowdown Capability
- Fission Product Retention

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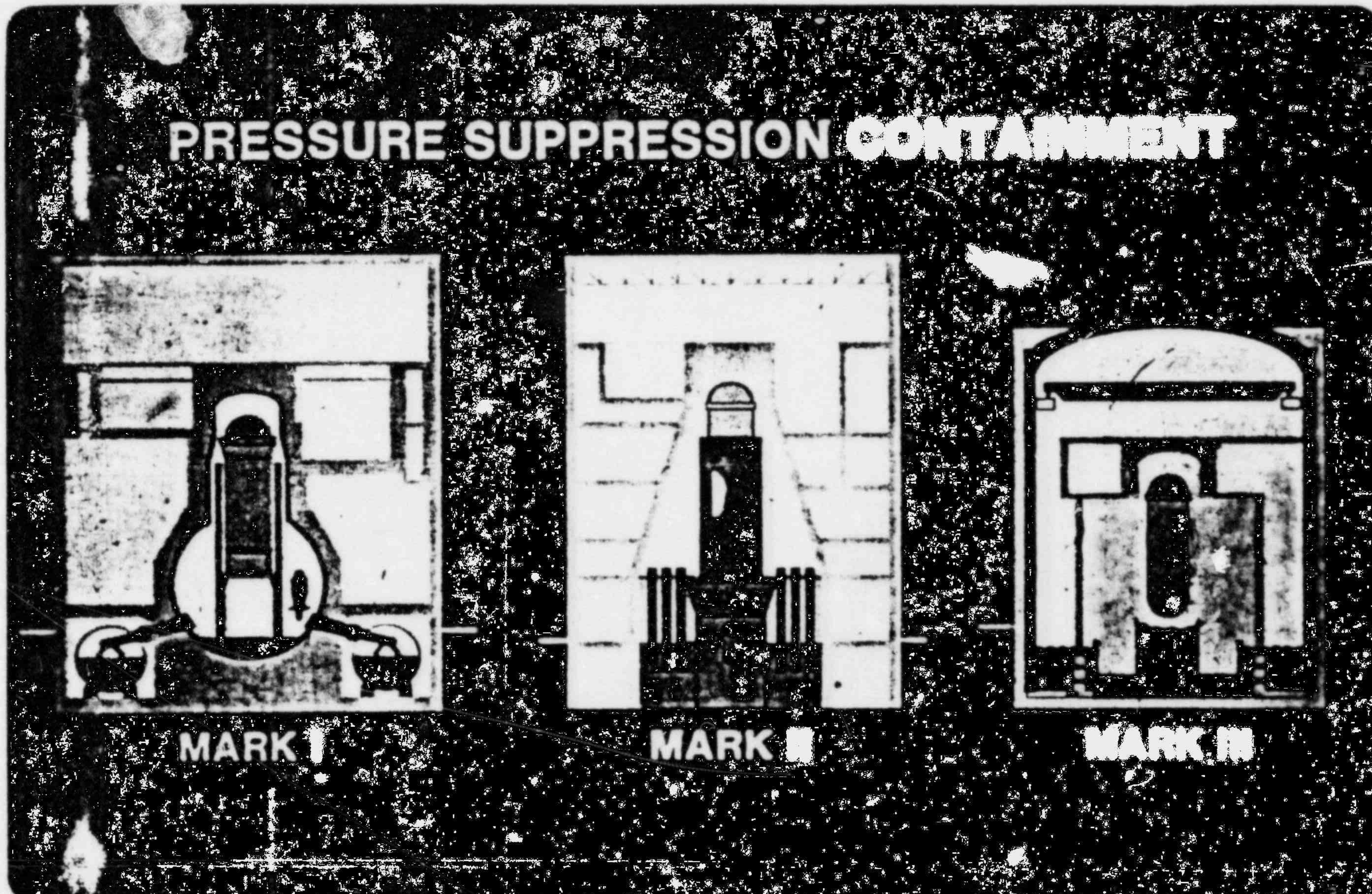

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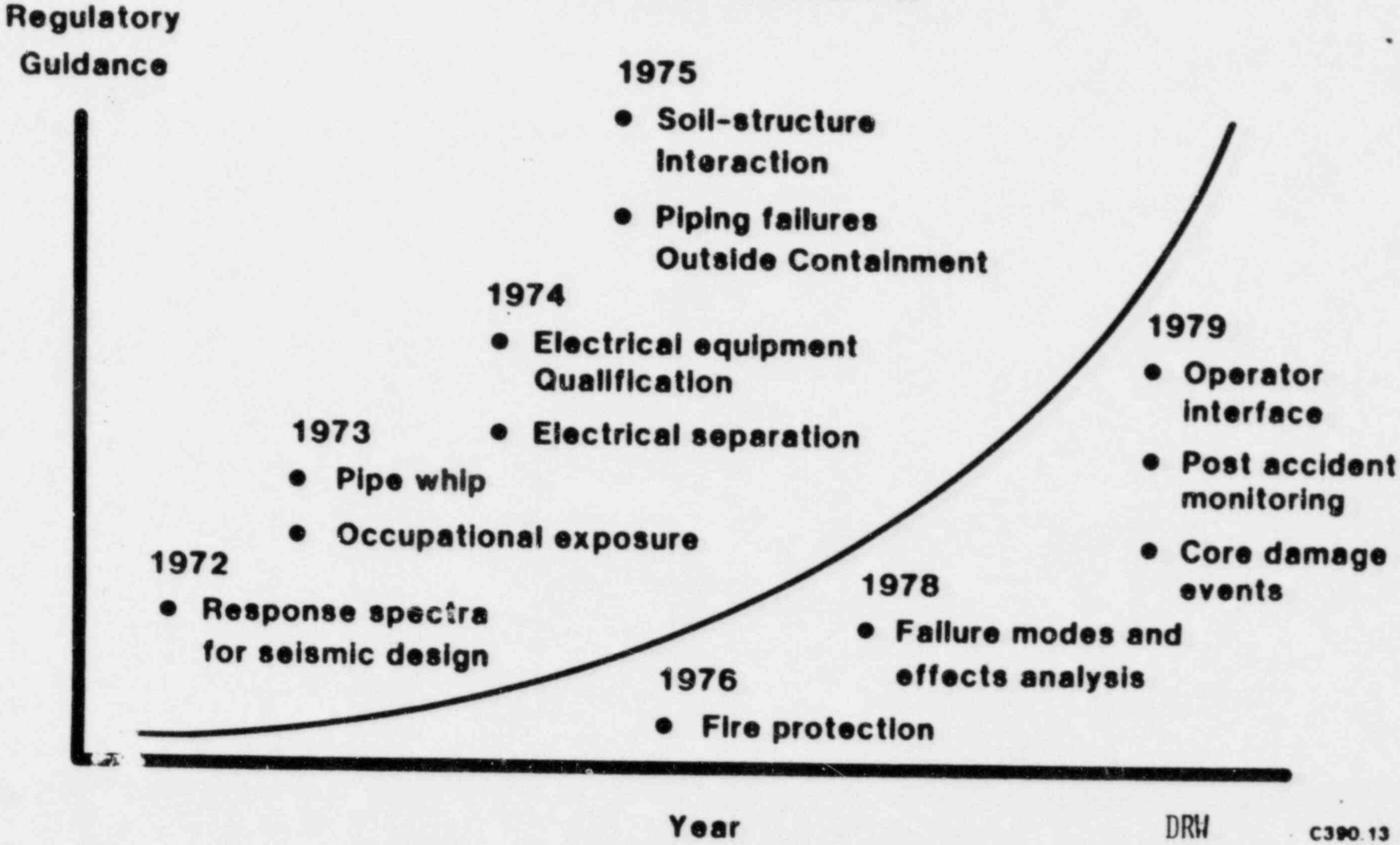
INCREASING REGULATORY EMPHASIS ON NUCLEAR ISLAND



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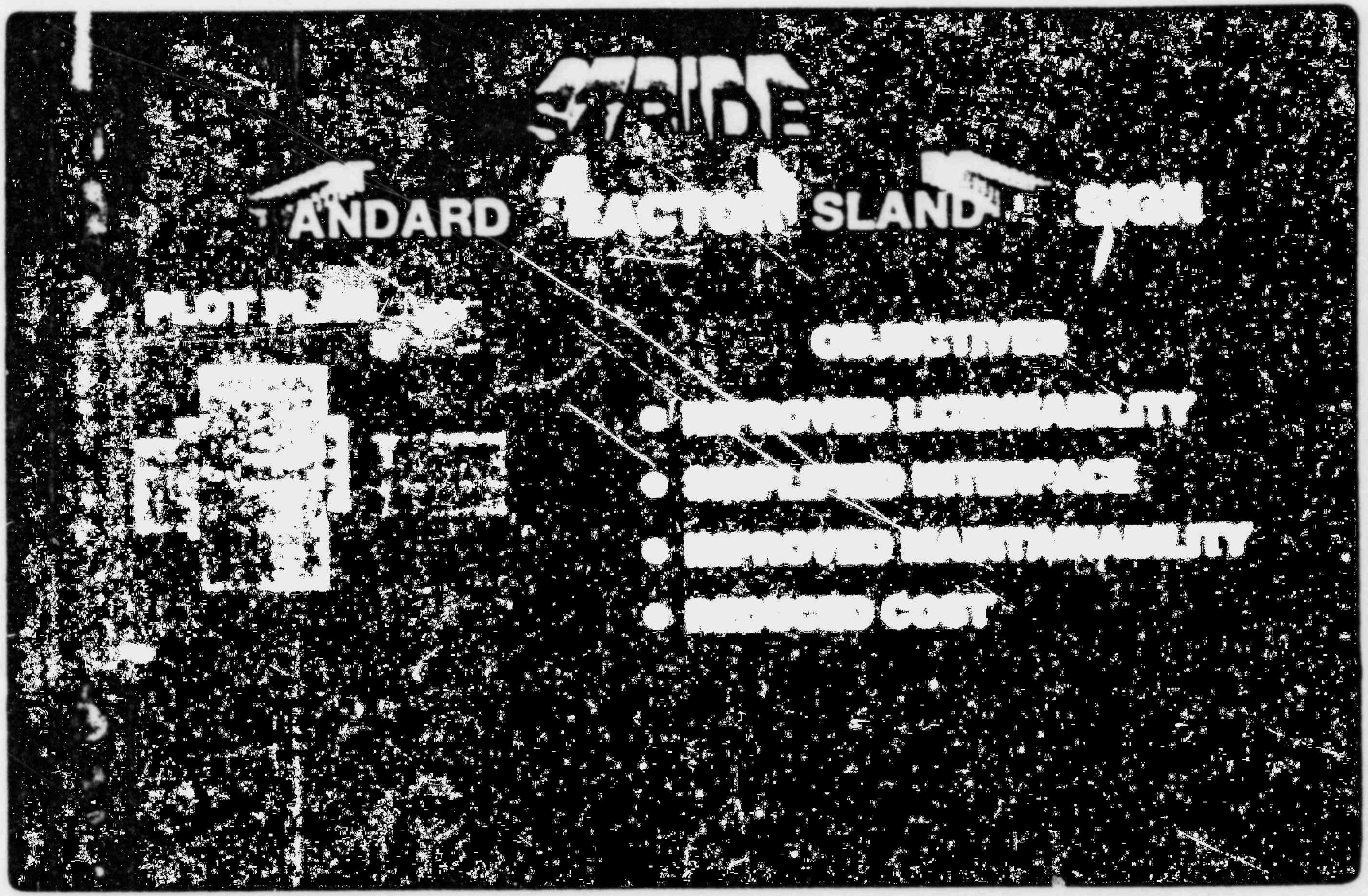

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BWR TODAY

DRW

1/7/82

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1/7/82

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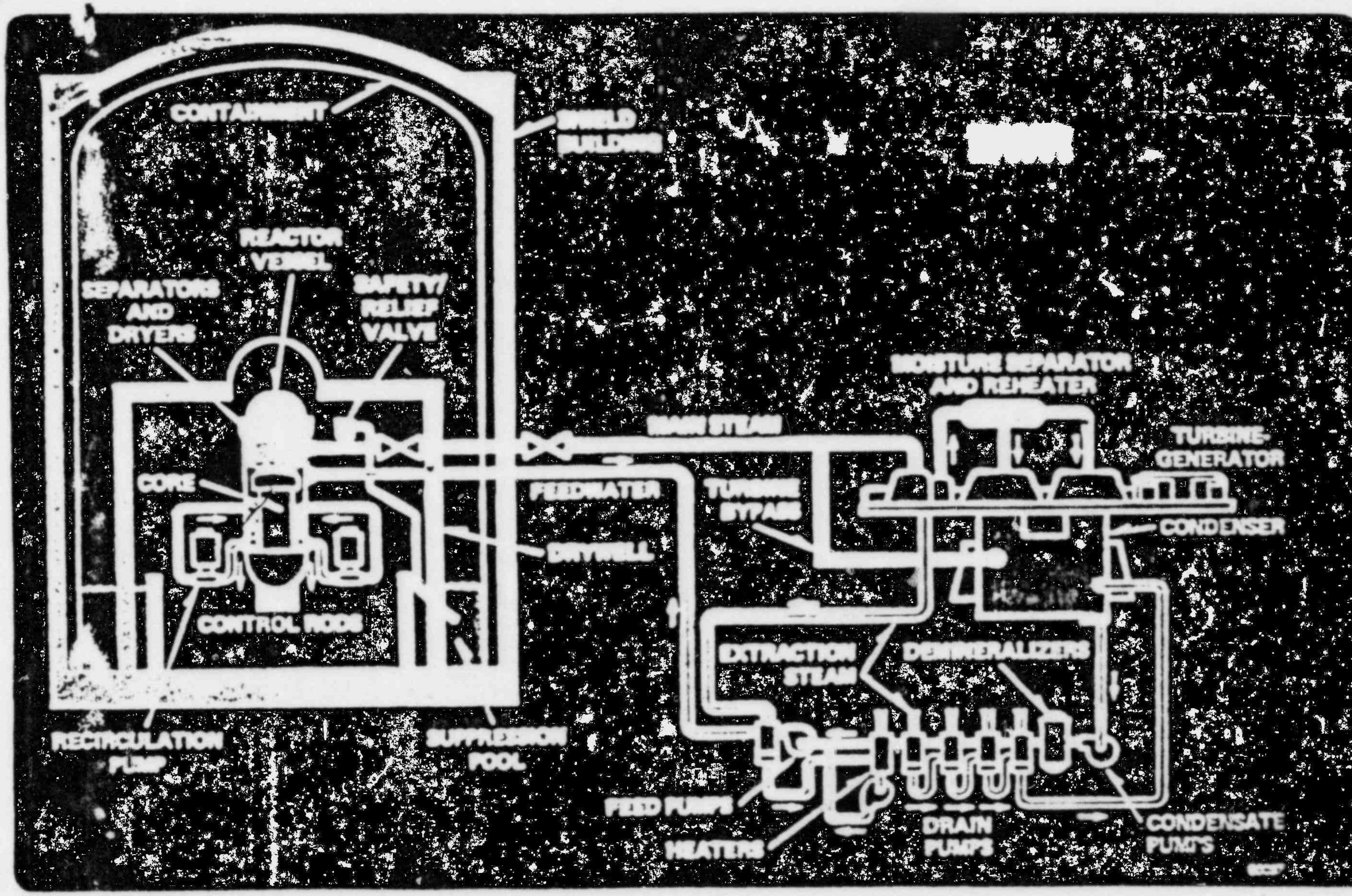

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MAJOR TEST PROGRAMS
SUPPORTING BWR/6-MARK III

- SMALL, INTERMEDIATE AND LARGE-SCALE MARK III CONTAINMENT TESTING
- FULL-SCALE FUEL BUNDLE HEAT TRANSFER TESTING
- FULL-SCALE FLOW-INDUCED VIBRATION TESTING OF REACTOR INTERNALS
- ACCELERATED FULL-SCALE PIPE TESTING FOR STRESS CORROSION PREVENTION
- EXPANDED FULL-SCALE ECCS TESTING
- BWR/6 MECHANICAL COMPONENT VERIFICATION TESTS
  - FLOW CONTROL VALVE
  - SAFETY/RELIEF VALVES
  - FAST SCRAM CRD
  - INCLINE TRANSFER TUBE
  - FEEDWATER NOZZLE/SPARGER
  - ETC.

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RECENT GE EVALUATIONS

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<u>CORE PROTECTION FUNCTIONS</u>

- SCRAM
  - SUPPLY WATER TO CORE
    - REMOVE DECAY HEAT

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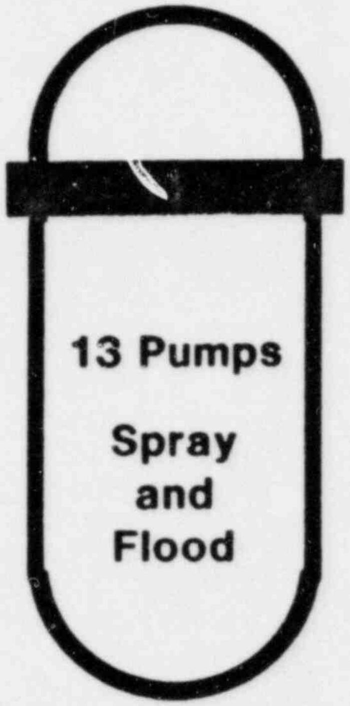
SYSTEMS TO SUPPLY WATER TO CORE



HIGH PRESSURE

← Feedwater
← Core spray
← Reactor core isolation cooling
← Control rod drive cooling

LOW PRESSURE

← Coolant injection
← Core spray
← Condensate

BWR DIRECT CYCLE PROVIDES MANY WATER SUPPLIES

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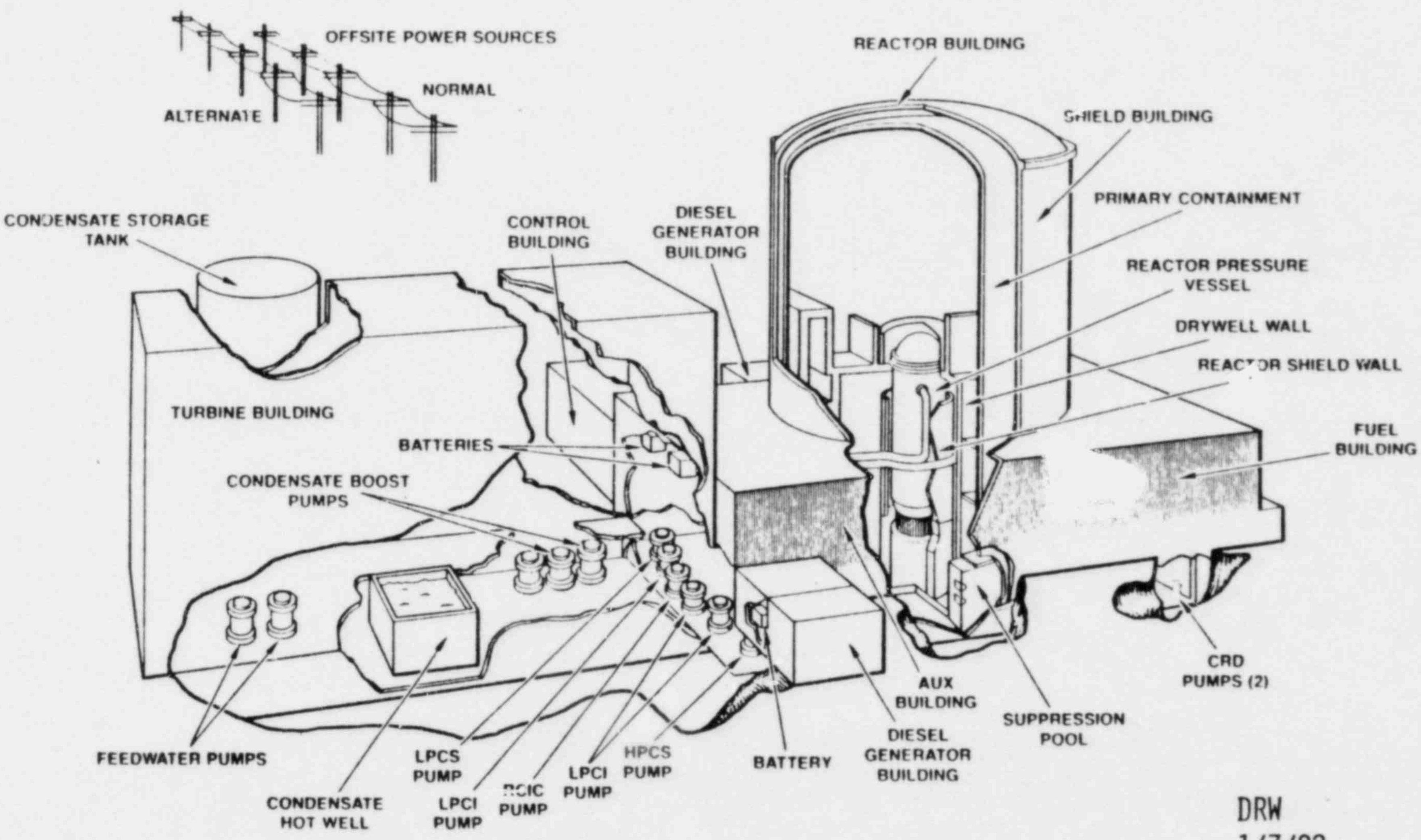
BWR/6 PLANT



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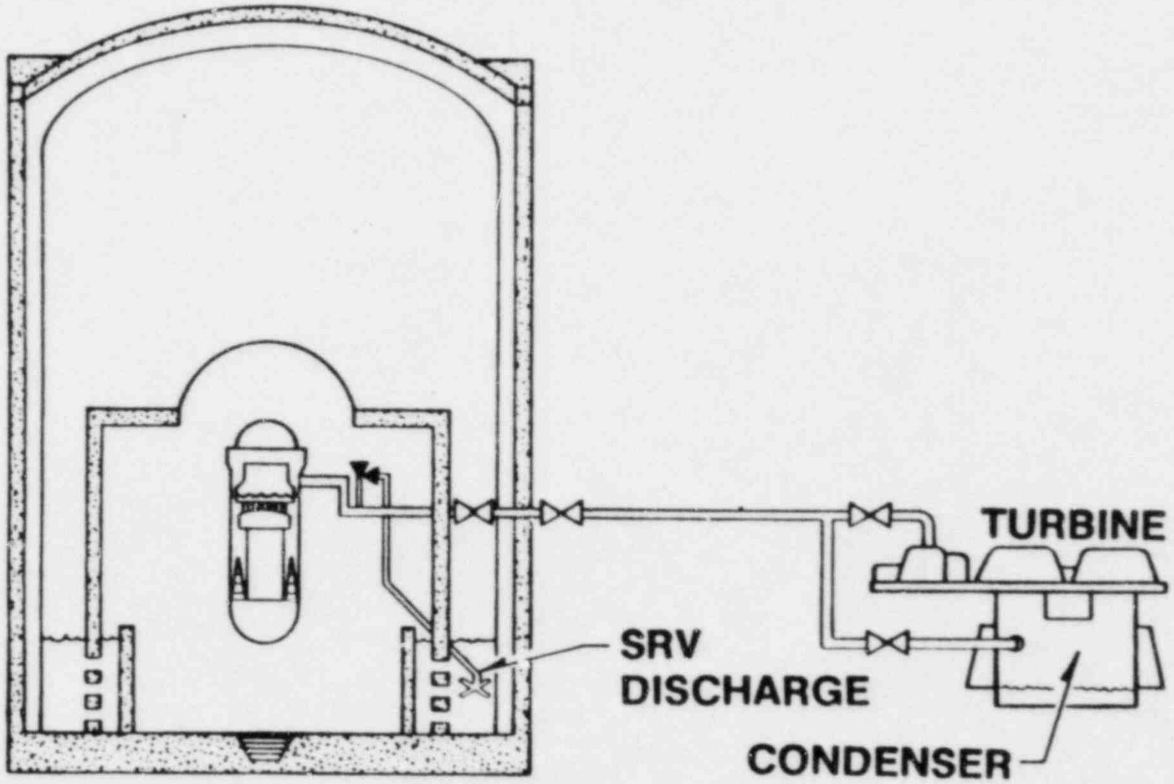
DEPRESSURIZATION



RAPID DEPRESSURIZATION CAPABILITY — ALL WATER SOURCES AVAILABLE WHEN DEPRESSURIZED

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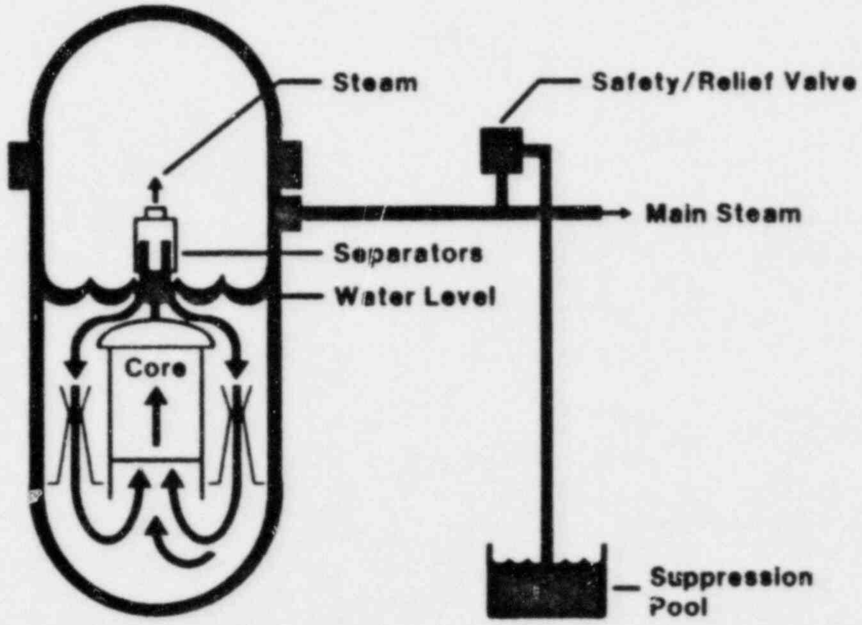
DECAY HEAT REMOVAL



NATURAL CIRCULATION + STEAM RELIEF
= PASSIVE DECAY HEAT REMOVAL

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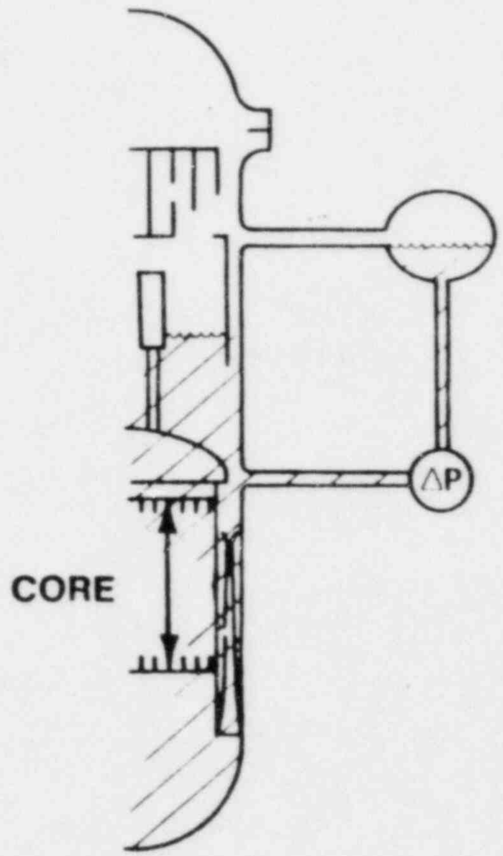

WATER LEVEL MEASUREMENT

- DIRECTLY ON VESSEL
- ΔP MEASUREMENT
- REDUNDANT

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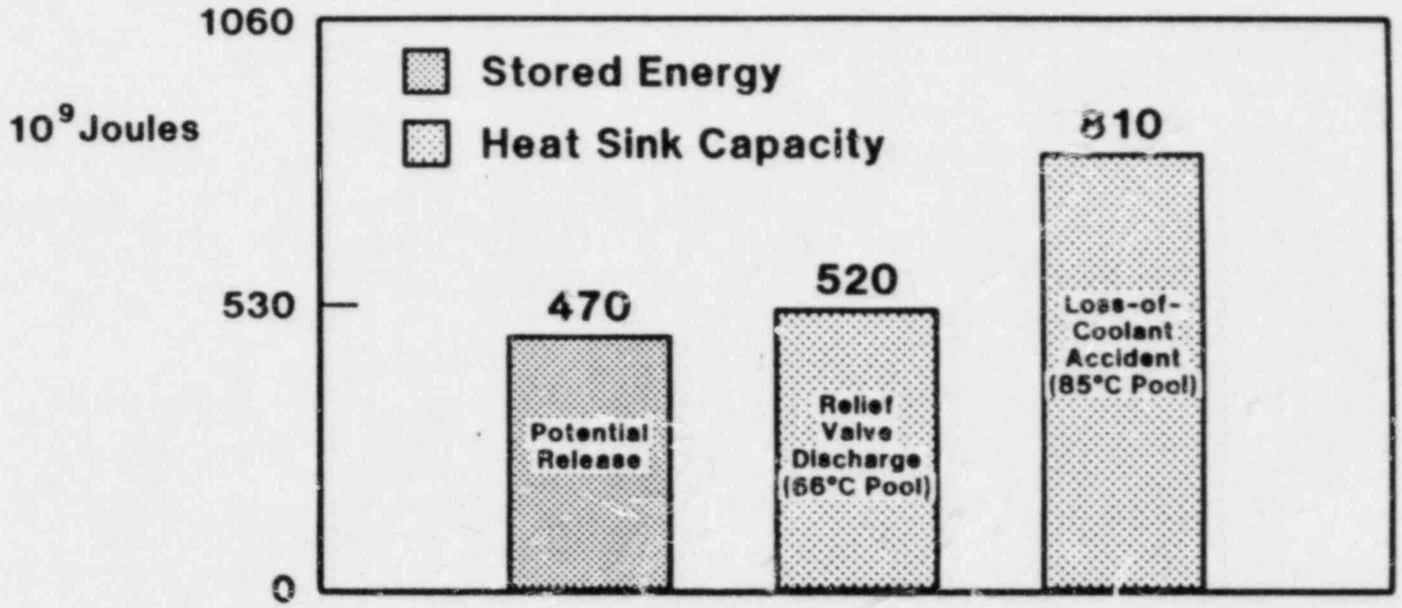
SUPPRESSION POOL PASSIVE HEAT SINK



A-162

CAPACITY

- FULL REACTOR DEPRESSURIZATION
- STORAGE OF DECAY HEAT - 6 HOURS ISOLATED
  - 2 TO 3 HOURS POST-ACCIDENT

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PLANT PROTECTION IMPROVEMENTS

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OPERATOR RESPONSE TO INVENTORY THREATENING EVENTS

REACTOR

1 MAINTAIN WATER LEVEL

- **High pressure water sources, or**
  - **depressurization, and low pressure water sources**

CONTAINMENT

PASSIVE HEAT SINK

- **Relief valves to suppression pool**

2 ESTABLISH LONG TERM HEAT SINK

- **Main condenser, or suppression pool cooling**

SIMPLE EMERGENCY GUIDELINES BASED ON SYMPTOMS

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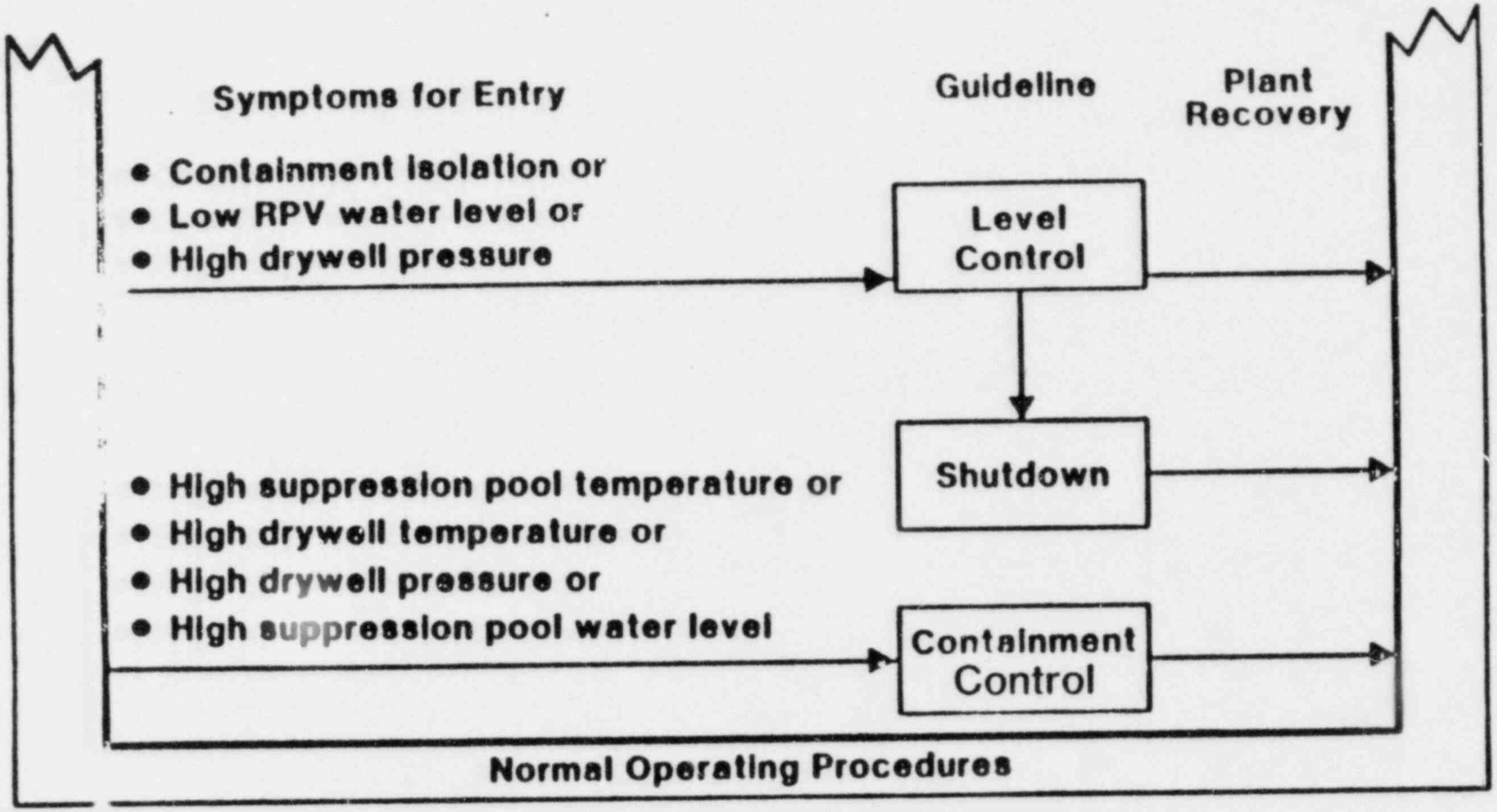
EMERGENCY PROCEDURE GUIDELINES



OPERATOR ERRORS MINIMIZED BY SYMPTOM ORIENTED GUIDELINES

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LEVEL CONTROL

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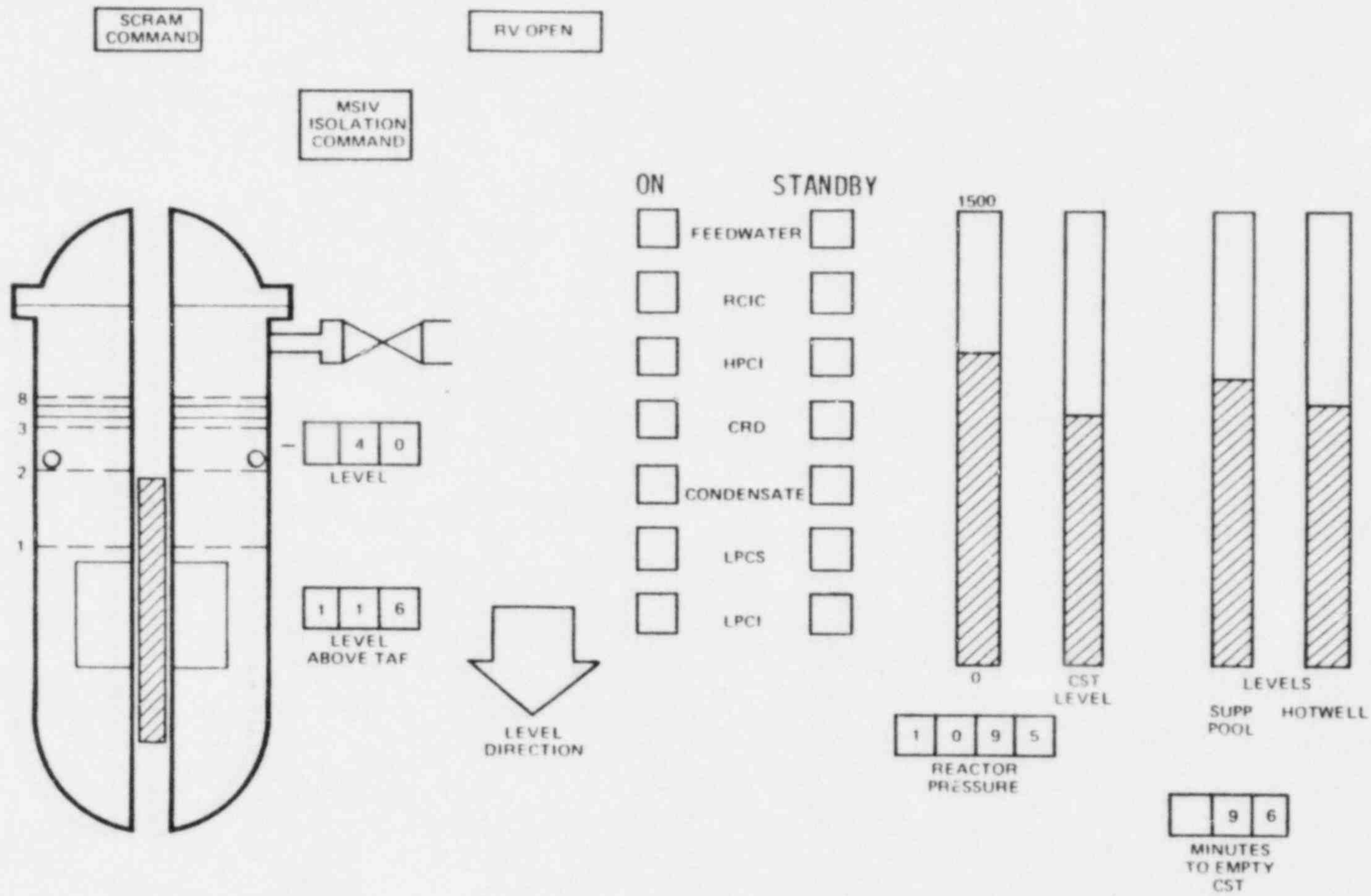

LOW WATER LEVEL
DECREASING WATER LEVEL
RELIEF VALVE OPEN
VESSEL ISOLATED
HPCI FAIL TO START
CORE SHUTDOWN

0400

01/09/81

R G B

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1/7/82

11162-08

BWR/6 SYSTEM IMPROVEMENTS

WATER DELIVERY IMPROVEMENTS

- REACTOR CORE ISOLATION COOLING SYSTEM LOGIC
- AUTO-DEPRESSURIZATION LOGIC FOR NON-BREAK EVENTS
- AUTO-RESTART OF HIGH PRESSURE CORE SPRAY

SCRAM SYSTEM IMPROVEMENTS

- SCRAM DISCHARGE VOLUME
- ATWS MODIFICATION

DECAY HEAT REMOVAL IMPROVEMENT

- CONTAINMENT OVERPRESSURE RELIEF

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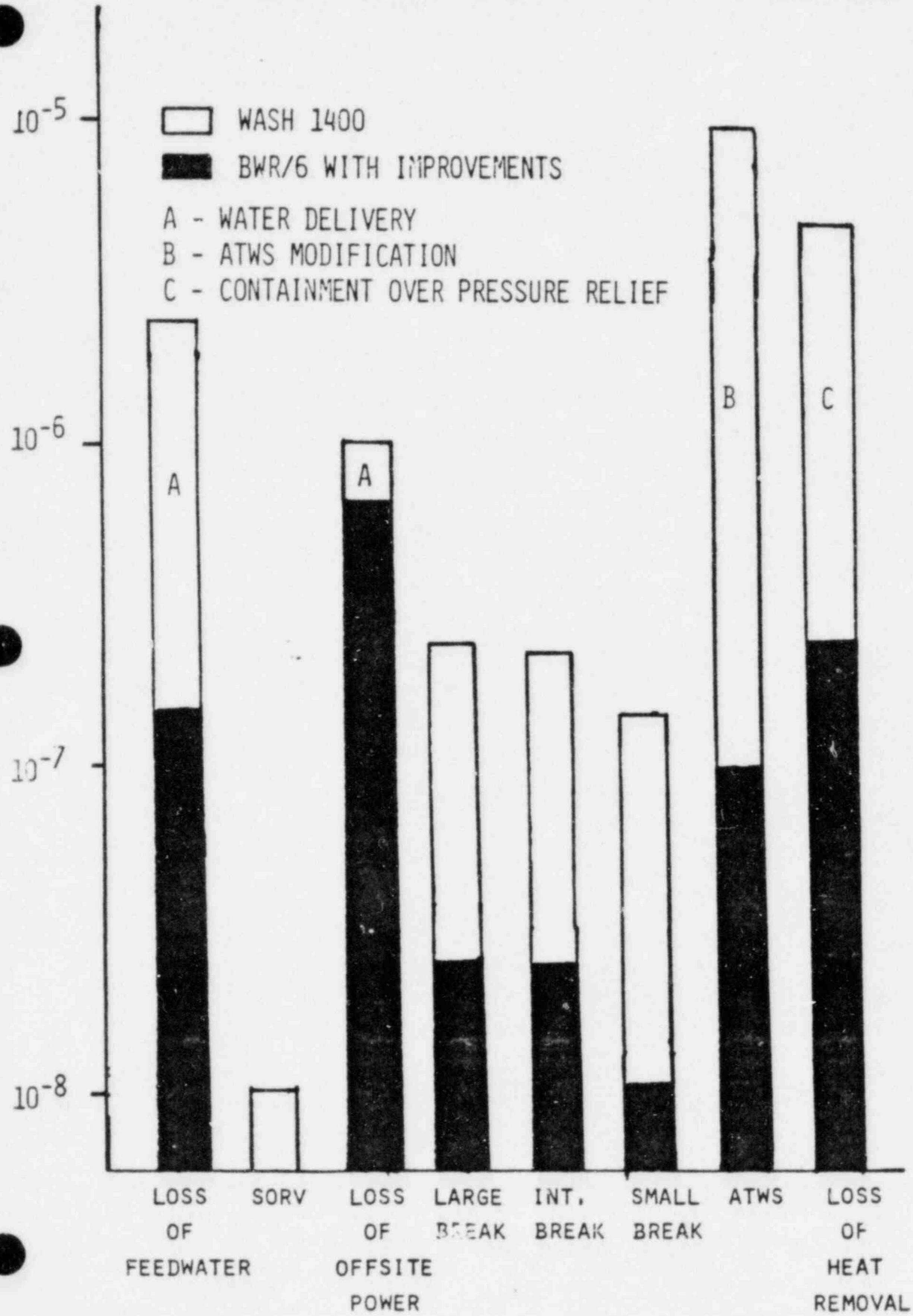
ESTIMATE OF CORE DAMAGE PROBABILITY PER REACTOR YEAR



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FUNCTION OF CONTAINENT

- FUNCTION

  PROTECT PUBLIC FROM EXCESSIVE DOSE IN SEVERE ACCIDENTS

- ACCOMPLISHED TWO WAYS

  - CONTAINMENT BARRIERS
    - . PRIMARY AND SECONDARY
    - . INTEGRITY FOR ALL DESIGN BASIS EVENTS AND MOST EVENTS BEYOND DESIGN BASIS
  - FILTERED CONTAINMENT VENTING
    - . ADDITIONAL PROTECTION FOR EVENTS BEYOND THE DESIGN BASIS

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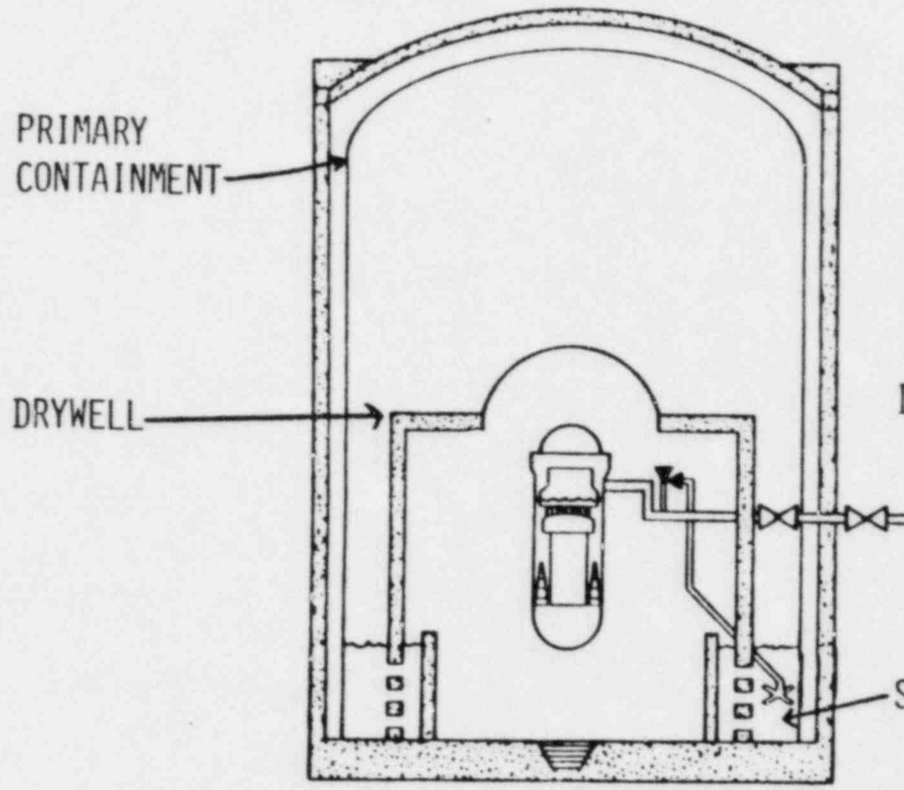
SUPPRESSION POOL SCRUBBING PATHWAYS



$$DF = DF_{POOL} \times DF_{PLATEOUT} \times DF_{SPRAYS}$$

WATER SCRUBBING FOR DEGRADED
TRANSIENTS AND ACCIDENT EVENTS

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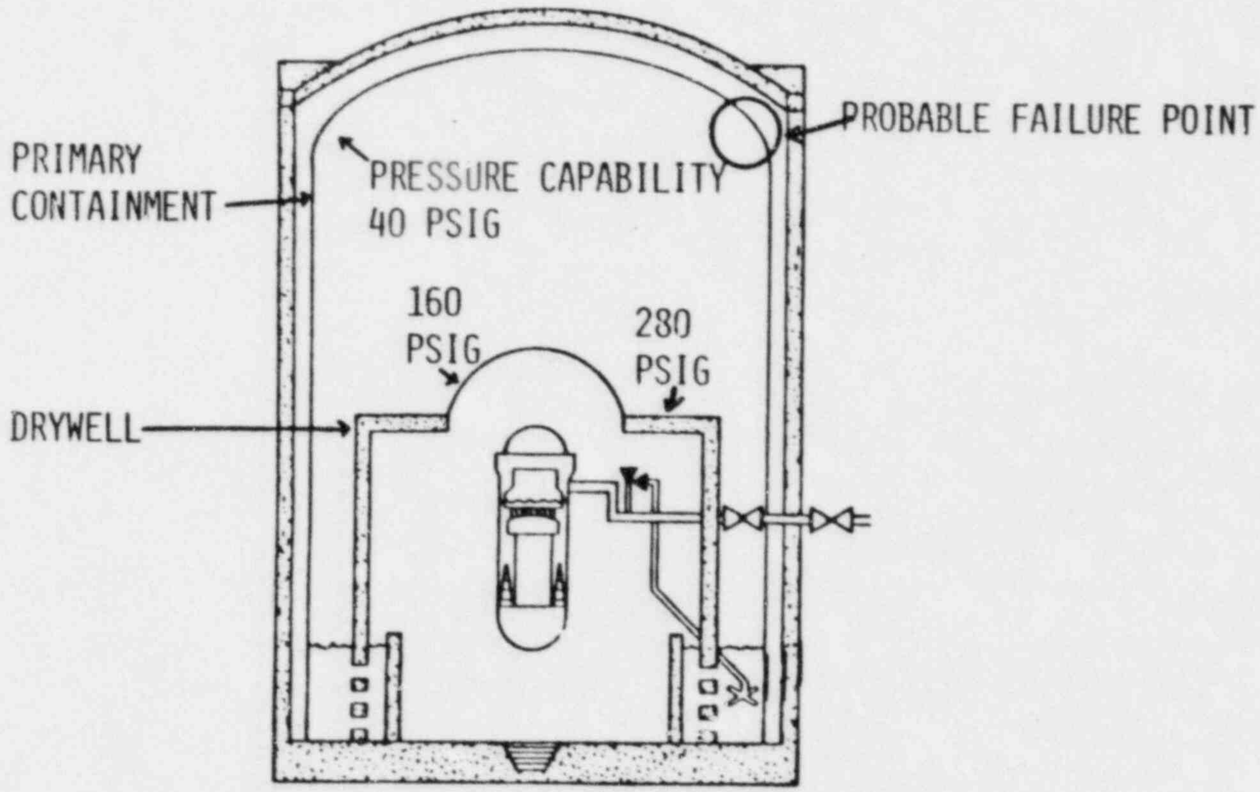
MARK III REFERENCE CONTAINMENT
APPROXIMATE YIELD PRESSURES



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RELATIVE STRUCTURAL PRESSURE
CAPABILITY INSURES POOL SCRUBBING

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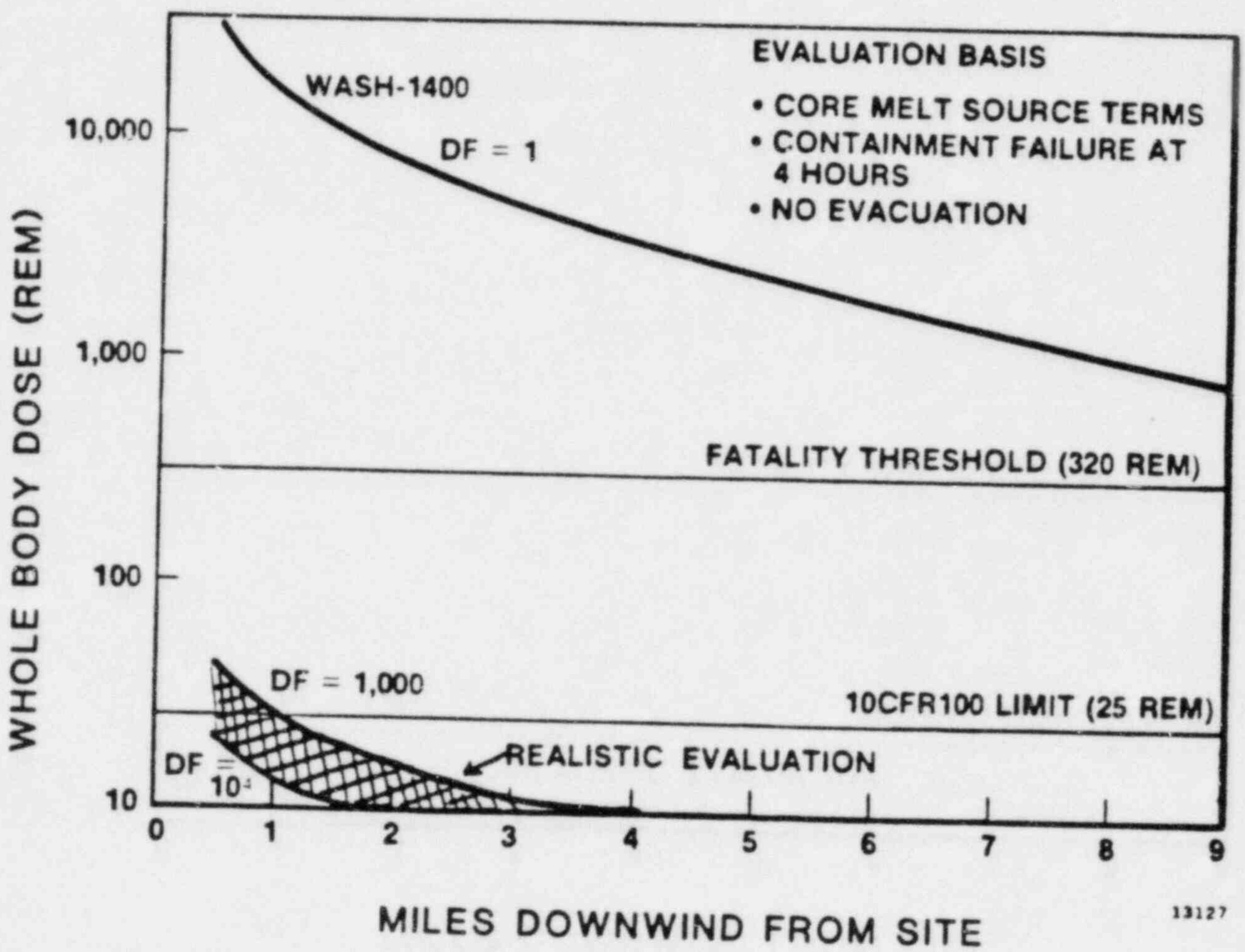
Importance of Fission Product Scrubbing in Mark III Pressure Suppression Containment



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SUMMARY

BWR/6 - MARK III

- PRODUCT OF 25 YEARS OF EVOLUTION AND SIMPLIFICATION
- BACKED BY 4ØØ R-YRS. FIELD EXPERIENCE
- MOST TESTED GE-BWR EVER
- RETAINS PROVEN FEATURES OF EARLIER BWR'S
    - SIMPLICITY
    - MANY WATER DELIVERY SYSTEMS
    - DEPRESSURIZATION CAPABILITY
    - NATURAL CIRCULATION
    - PRESSURE SUPPRESSION

      .
      .
      .
    - ETC.

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BWR/6 - MARK III CONT.

- INCORPORATES LATEST SAFETY TECHNOLOGY
  - WATER DELIVERY IMPROVEMENTS
  - ATWS MODIFICATION
  - OPERATOR INTERFACE
  - CONTAINMENT OVERPRESSURE RELIEF
  - SUPPRESSION POOL SCRUBBING

    .
    .
    .
  - ETC.
- BACKED BY STRIDE AND GESSAR
  - INTEGRATED NUCLEAR ISLAND DESIGN
  - SUPPORTS ONE-STEP LICENSING
- FIRST UNIT IN OPERATION
  - KUO SHENG - TAIWAN
  - 61 MONTH CONSTRUCTION

BWR/6 - MARK III --- ENGINEERED FOR THE FUTURE

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LICENSING OF GE'S BWR 6 MK III
NUCLEAR ISLAND

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JANUARY 7, 1982

GLENN SHERWOOD
DAN WILKINS
JOE QUIRK

NUCLEAR POWER SYSTEMS DIVISION
GENERAL ELECTRIC COMPANY

AGENDA

INTRODUCTION.............................. GLENN SHERWOOD

Objectives of BWR/6 MK III Licensing

BWR FUTURE DIRECTIONS......................... DAN WILKINS

BWR Evolution

Features of BWR 6 MK III

Results of BWR Evalutions

Plant Protection Features

LICENSING OF BWR 6 MK III.................. GLENN SHERWOOD

Nuclear Island Licensing

GESSAR I and II

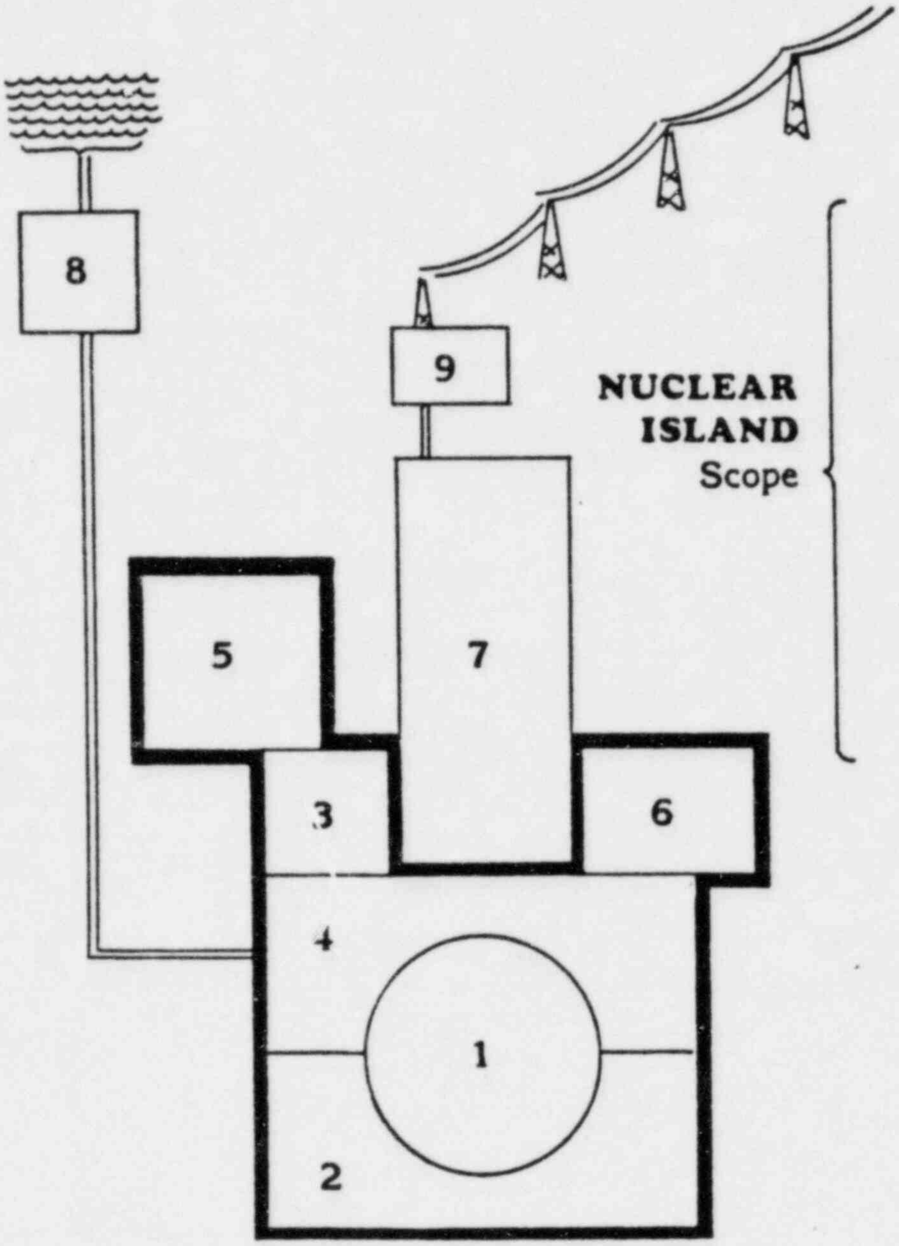
Licensing Objectives

Summary

GE'S NUCLEAR ISLAND DESIGN

- INTEGRATED DESIGN
- COVERS ALL RADIOLOGICAL SIGNIFICANT SYSETMS AND STRUCTURES
- SIMPLIFIES INTERFACES
- MAXIMIZES STANDARDIZATION AND OPTIMIZATION
- ALLOWS TIMELY IN-DEPTH SYSTEMS INTERACTION EVALUATIONS
- STRONG ENGINEERING SUPPORT
  - Design - One Organization
  - Complete Design Record
  - Detailed Plant Design Specification
- ADVANCED DESIGN FEATURES
  - Solid State NSPS
  - Improved ECCS Performance
  - Multiple Barrier Containment
  - 8 x 8 Fuel Bundle
  - Compacted Control Room

NUCLEAR ISLAND



Nuclear steam supply

(1) Reactor bldg.

Auxiliary nuclear system

(2) Fuel bldg.

(3) Diesel gen. bldg.

(4) Auxiliary bldg.

(5) Radwaste bldg.

(6) Control bldg.

Balance of plant

(7) Turbine bldg.

(8) Service water bldg.

(9) Switchyard

GESSAR PROGRAM
STATUS

<u>PRELIMINARY DESIGN APPROVAL</u> (GESSAR I)

- APPROVED 12/75 - 12/78
- EXTENSION 12/78 - 12/80

<u>FINAL DESIGN APPROVAL</u> (GESSAR II)

- TENDERED 3/80
- NRC ACCEPTANCE LETTER 12/81
- TO BE DOCKETED 1/82

NUCLEAR ISLAND
LICENSING PROGRAM

JAN-SEPT 1972	PHASE I:	AEC/ACRS ENDORSEMENT OF STANDARD PLANT CONCEPT
APRIL 1973-DEC 1975	PHASE II:	NRC/ACRS CONCURRENCE - PRELIMINARY DESIGN OF BWR/6 MARK III NUCLEAR ISLAND
MARCH 1980-DEC 1982	PHASE III:	NRC/ACRS APPROVAL - FINAL DESIGN OF BWR/6 MARK III NUCLEAR ISLAND
1975 - 1983	PHASE IV:	POWER WORTHINESS CERTIFICATE

REGULATORY ISSUES

□ REGULATORY GUIDES

- Preliminary Design Approval on Regulatory Guides Through 1.76 (March 1974)
- All Current Regulatory Guides Assessed in GESSAR II
- Final Design Approval Will Include Regulatory Guides Through 1982

□ STANDARD REVIEW PLANS (SRP's)

- Assessment of Nuclear Island Against SRP's
  - Evaluation Completed (Pre-1981 Version)
  - No Design Changes Required

□ SEVERE ACCIDENT/TMI ISSUES

- NRC to Complete Review in 1982
- Proposed Key Activities and Schedule

- Program Management Meeting	November 1981
- BWR Technology Update Meetings	Feb-April 1982
- GESSAR Severe Accident Appendix Submittal	May 1982
- ACRS Review	August 1982
- Commissioners Approval	September 1982

□ UNRESOLVED SAFETY ISSUES

25% - Not Applicable
45% - Not Open Issue (SER, Requirements specified)
30% - On Going i.e., ATWS, Station Blackout, Control System Failure

OTHER GENERIC LICENSING ISSUES

- STATION BLACKOUT

  Containment Overpressure Relief Extends Capability

- SYSTEMS INTERACTION

  Nuclear Island FMEA Will Identify Any Needed Corrections

- SABOTAGE PREVENTION

  Inherent in Nuclear Island Design

NUCLEAR ISLAND SAFETY REVIEWS

- TOTAL PLANT FAILURE MODE AND EFFECTS ANALYSIS
  - Covers 76 Safety Related Systems
  - Preliminary Analysis 80% Complete
  - No Significant Safety Problems Identified
  - To be Complete Fourth Quarter 1982

- TOTAL PLANT PROBABILISTIC RISK ASSESSMENT
  - Evaluates Core Melt Probability and Societal Risk
  - Preliminary PRA Completed in 1981
  - Meets Consensus Safety Goal
  - Plant Risk Substantially Below WASH 1400
  - To be Completed First Quarter 1982

SUMMARY

- GE DESIGN AND LICENSING EMPHASIS...BWR 6 NUCLEAR ISLAND

- NUCLEAR ISLAND RESPONSIVE TO CURRENT REGULATORY REQUIREMENTS

- NUCLEAR ISLAND LICENSING PROGRAM

    - PDA Approval in 1975
    - FDA Submitted to NRC, Docketed in January 1982
    - Requesting FDA SER in September 1982
    - Proposing Degraded Core Rule in Mid 1982

- BWR 6 MK III NUCLEAR ISLAND DESIGN TO BE GE OFFERING IN THE 1980's

- NUCLEAR ISLAND GE CONTRIBUTION TO PRE-APPROVED PLANT LICENSING REFORM

A-PWR DESIGN FEATURES (OVERVIEW)

- o REACTOR
  - LOW CORE POWER DENSITY
  - MODERATOR CONTROL
- o FLUID SYSTEMS
  - INTEGRATED SAFEGUARDS SYSTEMS (PRIMARY SIDE)
  - SECONDARY SIDE SAFEGUARDS SYSTEMS
  - AUXILIARY FLUID SYSTEMS
- o PLANT INTEGRATED CONTROL CENTER
  - PROTECTION SYSTEM
  - CONTROL SYSTEM
  - ADVANCED CONTROL ROOM
- o PRIMARY SYSTEM COMPONENTS
- o AUXILIARY FLUID SYSTEMS

LOW POWER DENSITY DESCRIPTION

Parameter	A	B Low Power Density	B/A
Core Thermal Power (MWT)	3800	3800	1.00
Fuel Assemblies	193	193	
Fuel Rods Per F/A	264	303	
Total Fuel Rods	50,952	58,479	
Fuel Rod O.D. (in.)	0.360 (1)	0.374 (2)	
Core Loading (MTU)	95.4	119.4	1.25
Equivalent Core Diameter (in.)	133	154	1.16
Active Core Length (in.)	168	168	1.00
Average Linear Power (kw/ft)	5.19	4.52	0.87
Average Specific Power (kw/kg)	39.8	31.8	0.80
Average Heat Flux (BTU/hr-ft^2)	187,900	157,600	0.84

(1) Optimized fuel rod

(2) Standard fuel rod

LOW POWER DENSITY
SUMMARY OF ADVANTAGES

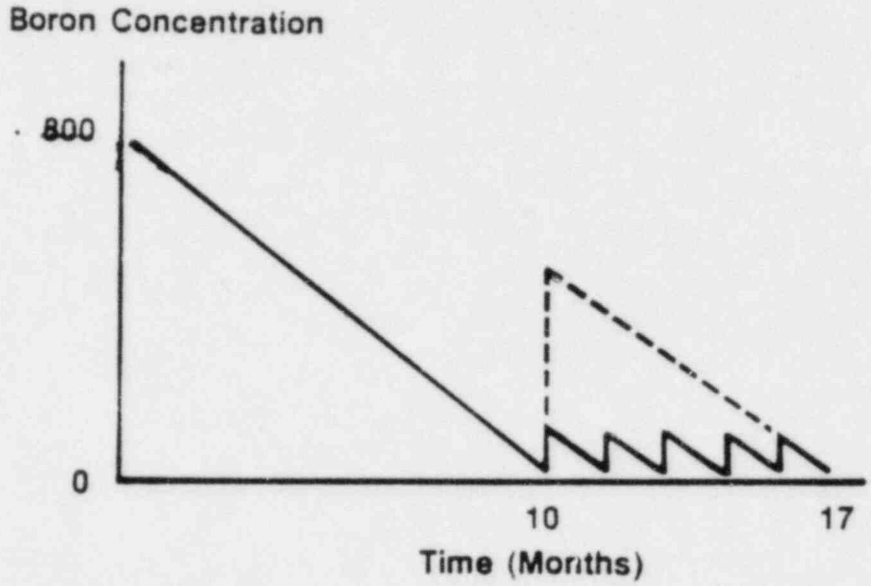
- 6% reduction in fuel cost[1]
- 9% reduction in uranium requirements[1]
- 250°F (139°C) reduction in peak clad temperature for large break LOCA
- DNB margin equivalent to 15°F (8°C) in core inlet temperature

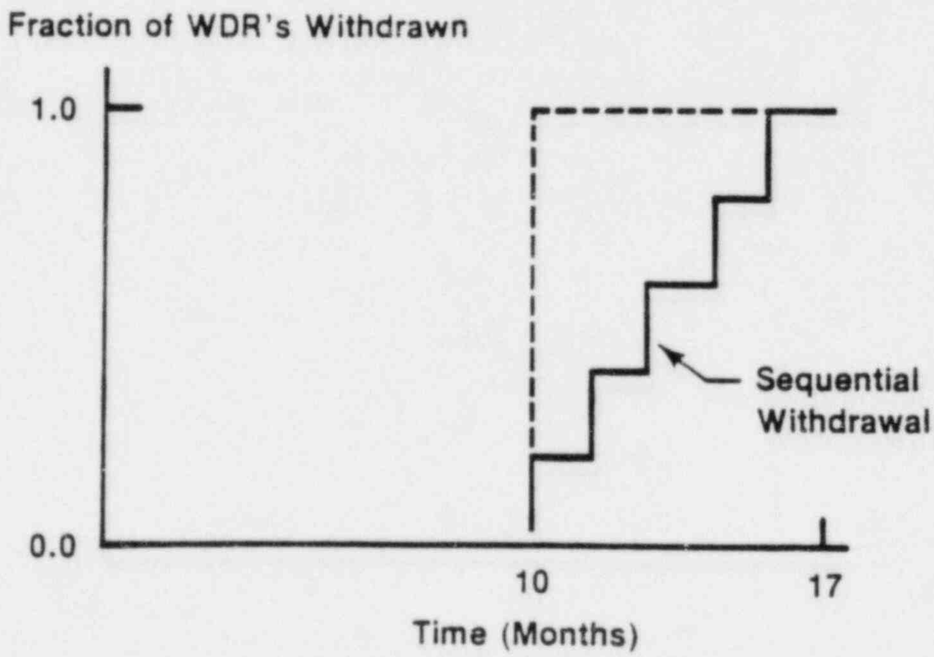
(1) Assumes 36,000 MWD/MTU discharge burnup, 18 month cycles and 75% capacity factor

MODERATOR CONTROL
GENERAL CONCEPT

- A portion of the core water volume is displaced during the first part of the cycle
  - Decreased neutron moderation
  - Increased neutron absorption in U-238
  - Increased PU production
- When the boron concentration nears 0 PPM, the displaced water is returned either gradually or at one time
  - Increased neutron moderation
  - PU production rate slows
  - Fissile material burned more efficiently
- Feed enrichments are reduced for the same energy output

BORON CONCENTRATION AND WDR WITHDRAWAL SEQUENCE VS. TIME





MODERATOR CONTROL
SUMMARY OF ADVANTAGES
(Relative to Conventional PWR)

- 10% reduction in fuel cost(1)
- 10% reduction in uranium requirements(1)
- No burnable poison rods required for cycle 1 and for reload cycles up to 2 years in duration
- DNB margin equivalent to 5°F in core inlet temperature due to more negative moderator temperature coefficients

(1) Assumes 36,000 MWD/MTU discharge burnup, 18 month cycles and 75% capacity factor

ADVANCED REACTOR
SUMMARY OF ADVANTAGES

- 19% reduction in fuel cost[1]
  5.5% net power cost reduction
  8 to 1 benefit to cost ratio
- 22% reduction in uranium requirements[1]
- Allows cycle lengths up to 2 years without burnable poison rods
- Provides LOCA margin
- Provides DNB margin which allows higher coolant temperatures and/or reduced flows
- Facilitates load follow operations

(1) Assumes 36,000 MWD/MTU discharge burnup, 18 month cycles, and 75% capacity factor

PRIMARY SIDE SAFEGUARDS SYSTEM DESIGN OBJECTIVES

- Improve system reliability
  - Simplification
  - Redundancy/diversity
  - Reduced operator action
- Achieve a high probability of no core uncovery for small break LOCA's
- Perform additional functions:
  - Emergency core cooling diverse to secondary side heat sink
  - Post-LOCA containment heat removal diverse to fan coolers
- Reduce and simplify BOP interface requirements
- Maintain cost effectiveness

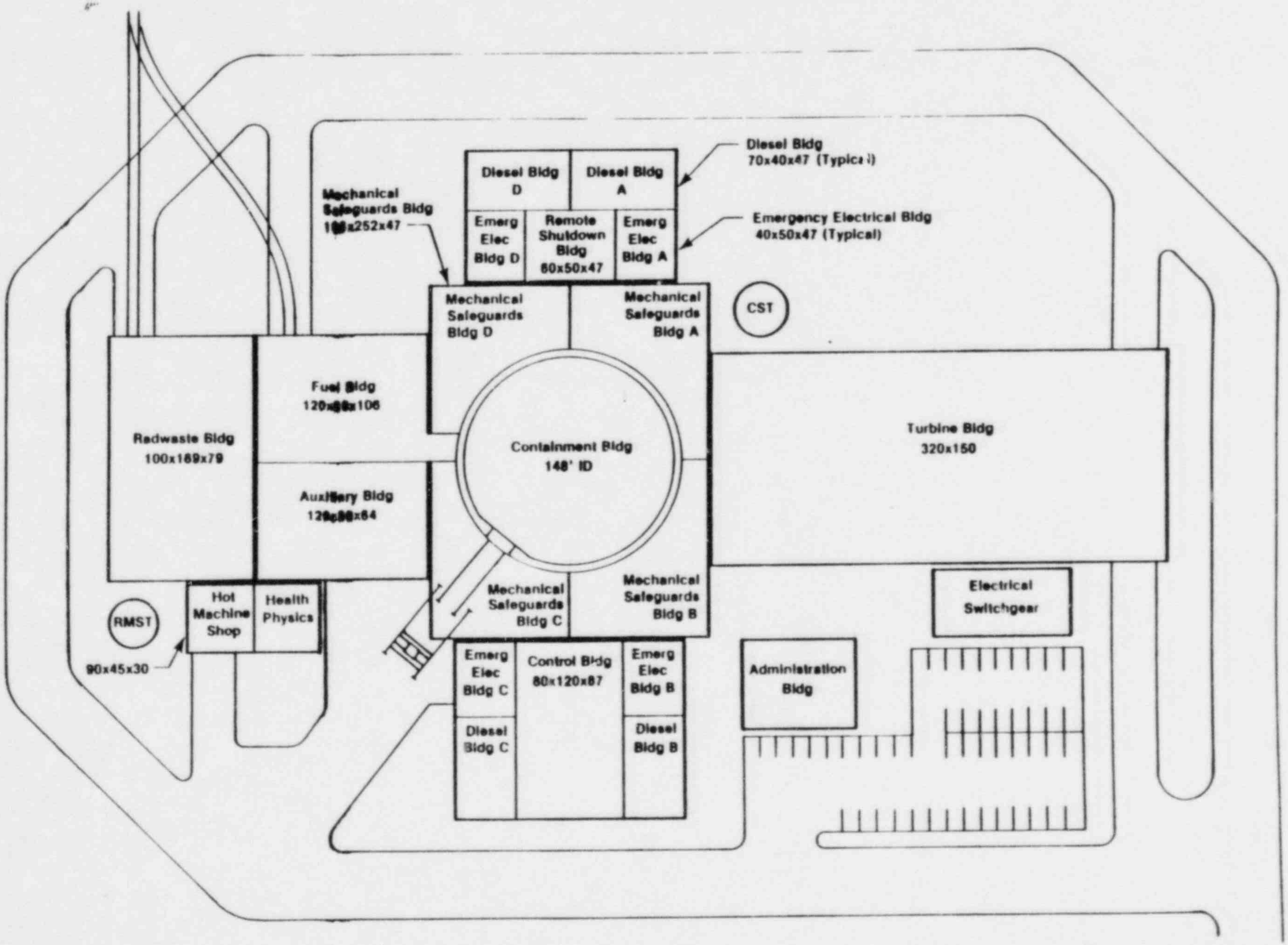
PRIMARY SIDE SAFEGUARDS LAYOUT DESIGN OBJECTIVES

- Minimize in-plant contamination and offsite releases due to radioactive spills outside containment
  - Emergency plant cooldown
  - Emergency boration/letdown
  - Post-LOCA recirculation
- Mitigate the consequences of high energy line breaks outside containment (e.g., RHR lines)
- Provide access to safeguards equipment for maintenance during long-term core cooling
- Provide greater protection against external events

16695

INTEGRATED SAFEGUARDS SYSTEM
GENERAL FUNCTIONS

- Emergency core cooling in the event of a loss of coolant accident
- Shutdown reactivity in the event of a steam break accident
- Containment spray for LOCA or steambreak
- Emergency boration and letdown
- Normal and emergency plant cooldown (below RCS cut-in temperature/pressure)
- Emergency core cooling in the event of a loss of secondary side heat sink
- Containment heat removal diverse to fan coolers in the event of a loss of coolant accident



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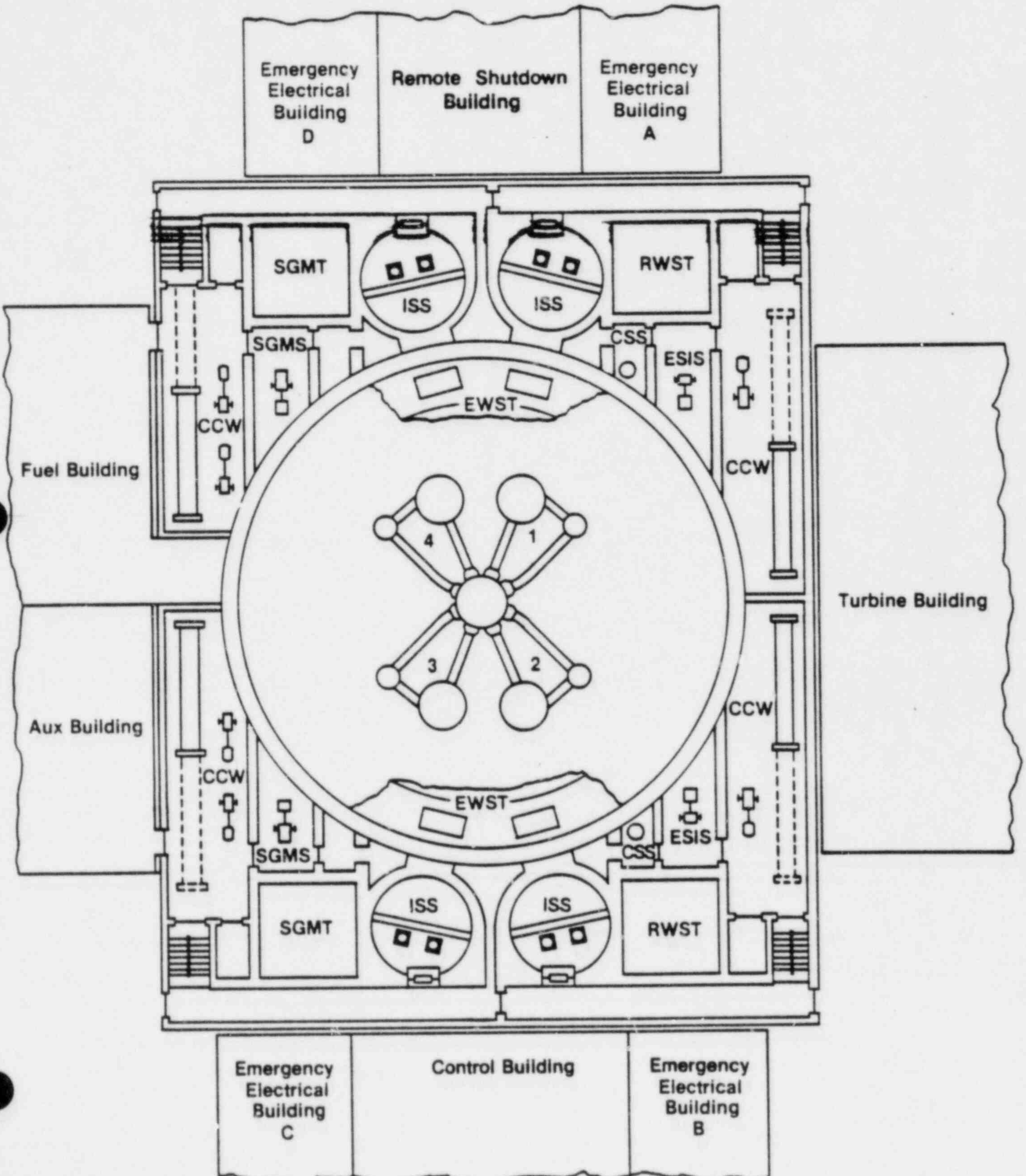
PROPOSED EQUIPMENT LAYOUT MECHANICAL SAFEGUARDS BUILDING

(Plan View - Grade Elevation)



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INTEGRATED SAFEGUARDS SYSTEM
SUMMARY OF ADVANTAGES

- Satisfies NRC final acceptance criteria for all break sizes and locations with improved reliability
- No core uncovery for more probable small breaks (e.g., < 5") with only 2 of 4 subsystems delivering
- Provides a means for core cooling in the event of a loss of secondary side heat sink
- Provides diverse means of containment heat removal
- Reduced operator action (e.g., elimination of RWST to containment sump switchover)
- Design simplification (e.g., elimination of multi-branch lines and throttling valves)
- Minimizes pathways for off site radioactive releases
- Ensures access to safeguards equipment for maintenance during long-term core cooling
- Provides greater protection against external events
- Mitigates the consequences of high energy line breaks outside containment

SECONDARY SIDE SAFEGUARDS SYSTEM DESIGN OBJECTIVES

- Improve reliability of heat removal from the steam generators
  - Short term and long term
  - Total loss of A.C. power
  - Redundancy/diversity
  - Reduced operator actions
- Minimize the potential for radioactive releases due to atmospheric steam dump
- Improve protection of plant equipment, i.e.; reduce probability of
  - Inadequate core cooling
  - Steam generator dryout
  - Feedline cracking
- Provide closed loop, heat removal to cold shutdown conditions diverse to RHRS
- Reduce and simplify BOP interface requirements
- Maintain cost effectiveness

PASSIVE STEAM CONDENSER SYSTEM

- Closed loop, secondary side heat removal system
- 1 steam condenser in a pool of water per steam generator
- Steam from steam generator is condensed. Pool boiling provides heat removal
- Condensate returns to steam generator via natural circulation
- Extended operation can be provided by refilling pools

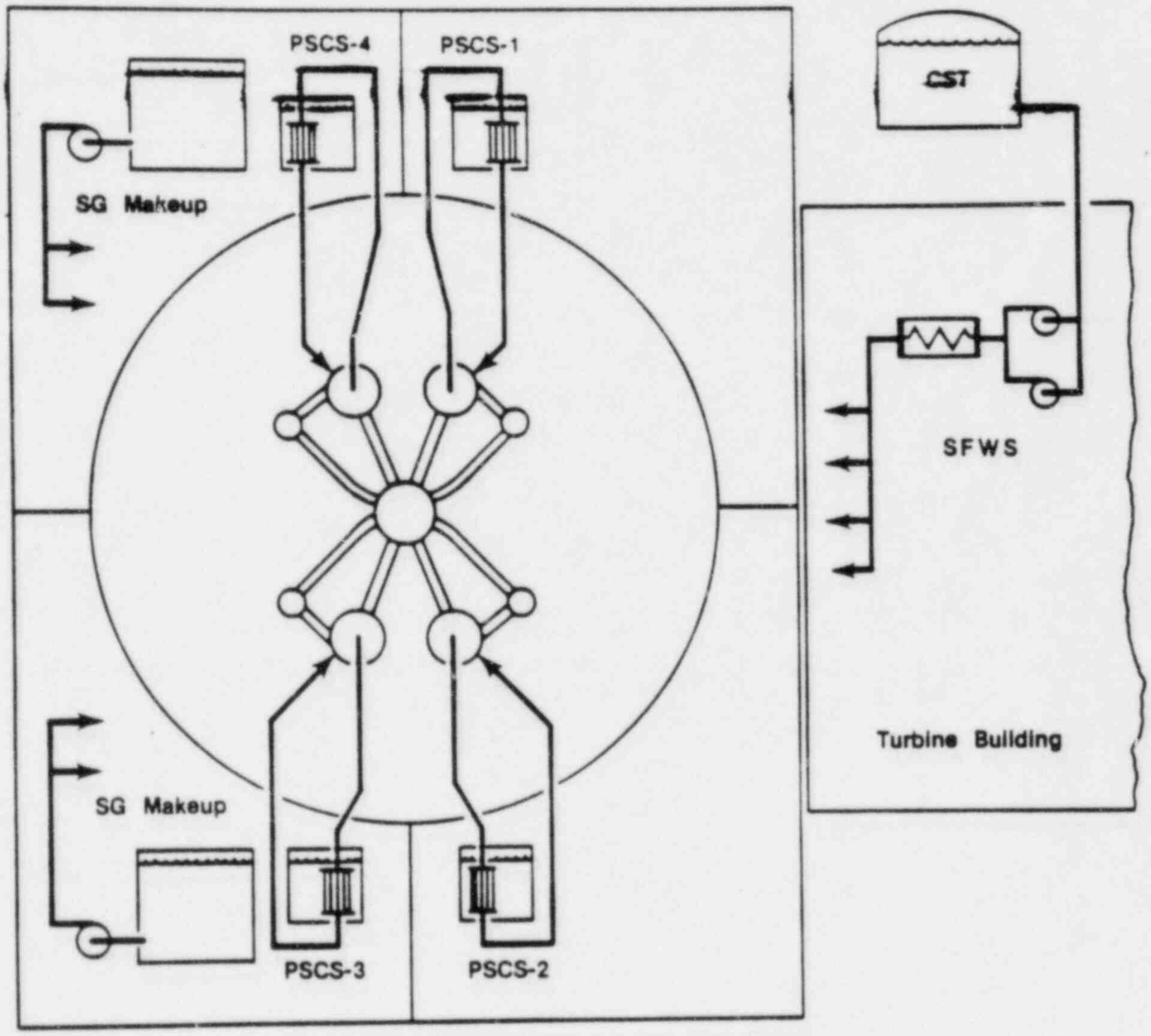
SECONDARY SIDE SAFEGUARDS FUNCTIONS

- Startup feedwater system
  - First line of defense
  - Control system
  - Plant startup and shutdown
  - Reactor trip
  - Loss of main feedwater
  - Loss of offsite power
  - Normal plant cooldown to RHR cut-in
- Passive steam condenser system
  - Second line of defense
  - Safety system
  - Faulted steam generator accidents
  - Steam gneerator tube rupture
  - Total loss of A.C. power
  - High radiation in steam lines
  - Backup to SFWS
  - Emergency plant cooldown to nearly cold conditions

SECONDARY SIDE SAFEGUARDS
DESCRIPTION

- Startup feedwater system
  - Control grade (NNS)
  - Located in non-seismic building
  - 2 Motor driven pumps
  - 1 Condensate storage tank - dearated
  - Automatic steam generator level control
  - Heated to prevent feedline cracking
- Passive steam condenser system
  - Safety grade, seismic
  - 4 Steam condensers and 4 water pools
  - 1 Condenser per steam generator
  - Long term makeup by 2 motor driven pumps and 2 storage tanks

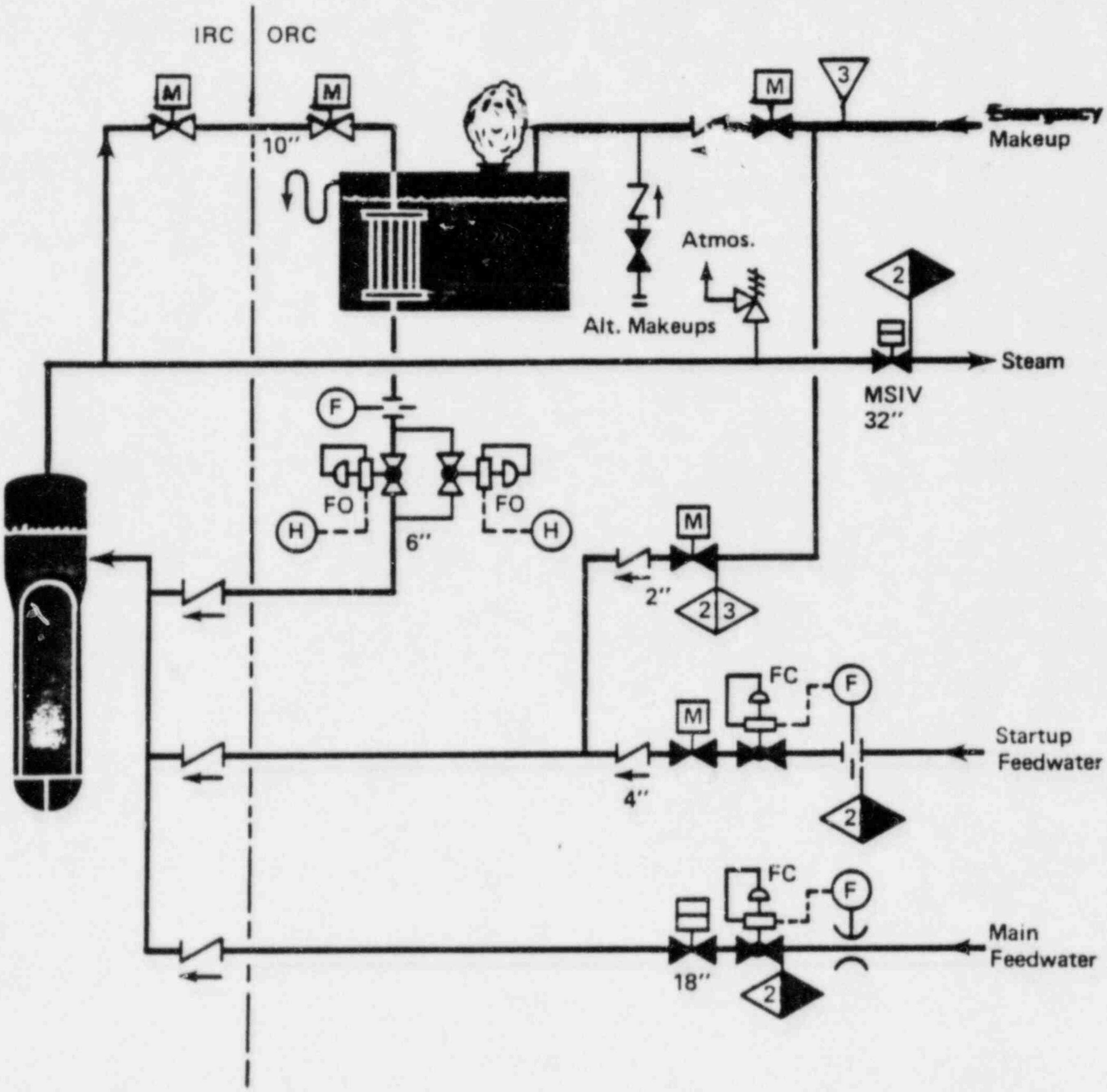
GENERAL LAYOUT OF SFWS/PSCS EQUIPMENT



PASSIVE STEAM CONDENSER SYSTEM CAPABILITIES

- Satisfies post-accident heat removal requirements with 2 of 4 operative
- Matches decay heat generation after
  - 1 minute with 4 of 4 operative
  - 3 minutes with 3 of 4 operative (Faulted steam generator condition)
- Pools are sized for 10 hours of decay heat removal with 4 of 4 operative
- RCS cooldown capability down to near cold shutdown conditions

PASSIVE STEAM CONDENSER SYSTEM
EMERGENCY OPERATION



SECONDARY SIDE SAFEGUARDS
SUMMARY OF ADVANTAGES

- Improved core heat removal reliability
  - Passive
  - Reduced demand for high quality steam generator makeup water
  - Diverse to startup feedwater system
  - Diverse to residual heat removal system
  - Reduced short-term operator actions
- Reduced potential for off-site radioactive releases
  - Accidents with leaking steam generator tubes
  - Steam generator tube rupture accident
- Spinoffs of SFWS/PSCS
  - Smaller water storage tank size
  - No diverse pump drives
  - Smaller diesel generator sizes
  - Simplified startup feedwater heating
- Improved protection of plant equipment, i.e.; reduced probability of
  - Inadequate core cooling
  - Steam generator dryout
  - Feedline cracking

PROTECTION SYSTEM

- DNB AND KW/FT PROTECTION (RESAR-414)
- BYPASS CAPABILITY
- AUTOMATIC TESTING
- SIMPLIFIED SEPARATION REQUIREMENTS
- STATUS -
  - VERIFICATION AND VALIDATION PROGRAM COMPLETE. SUBMITTED TO NRC (LATE 1980).
  - IPS PROTOTYPE.
  - BRITISH THIRD PARTY REVIEW COMPLETE.

CONTROL SYSTEM

- ADVANCED POWER CONTROL
- IMPROVED SG WATER LEVEL CONTROL
- FAULT-TOLERANT CONTROLLERS
- SIMPLIFIED SEPARATION REQUIREMENTS
- STATUS -
  - UNDER DESIGN.

ADVANCED CONTROL ROOM

o UTILIZE TOP-DOWN SYSTEMS ENGINEERING APPROACH APPLYING PRINCIPLES FROM COGNITIVE PSYCHOLOGY.

- HOW HUMANS SOLVE PROBLEMS
- HOW HUMANS PROCESS DATA
- **ALLOCATION OF TASKS BETWEEN HUMAN AND MACHINE**

o UTILIZE EXPERIENCE GAINED FROM: TECHNICAL SUPPORT CENTER (SAFETY PARAMETER DISPLAY SYSTEM); EPRI/DASS PROGRAM; PREVIOUS ACR EXPERIENCE.

o INTEGRATION OF RESOURCES

- FUNCTIONAL DESIGN
- EQUIPMENT DESIGN
- PROCEDURES
- TRAINING
- HUMAN FACTORS EXPERTS

o ADDRESS ALL MAN-MACHINE REGULATORY REQUIREMENTS

- NUREG 0700, 0696, 0801, 0835

PRIMARY SYSTEM COMPONENTS

o REACTOR VESSEL

- MODIFIED TO REFLECT INCREASED CORE, DIRECT VESSEL INJECTION

o REACTOR INTERNALS

- MODIFIED TO REFLECT CORE MODIFICATIONS

o CRDM

- ESSENTIALLY UNCHANGED

o REACTOR COOLANT PUMP

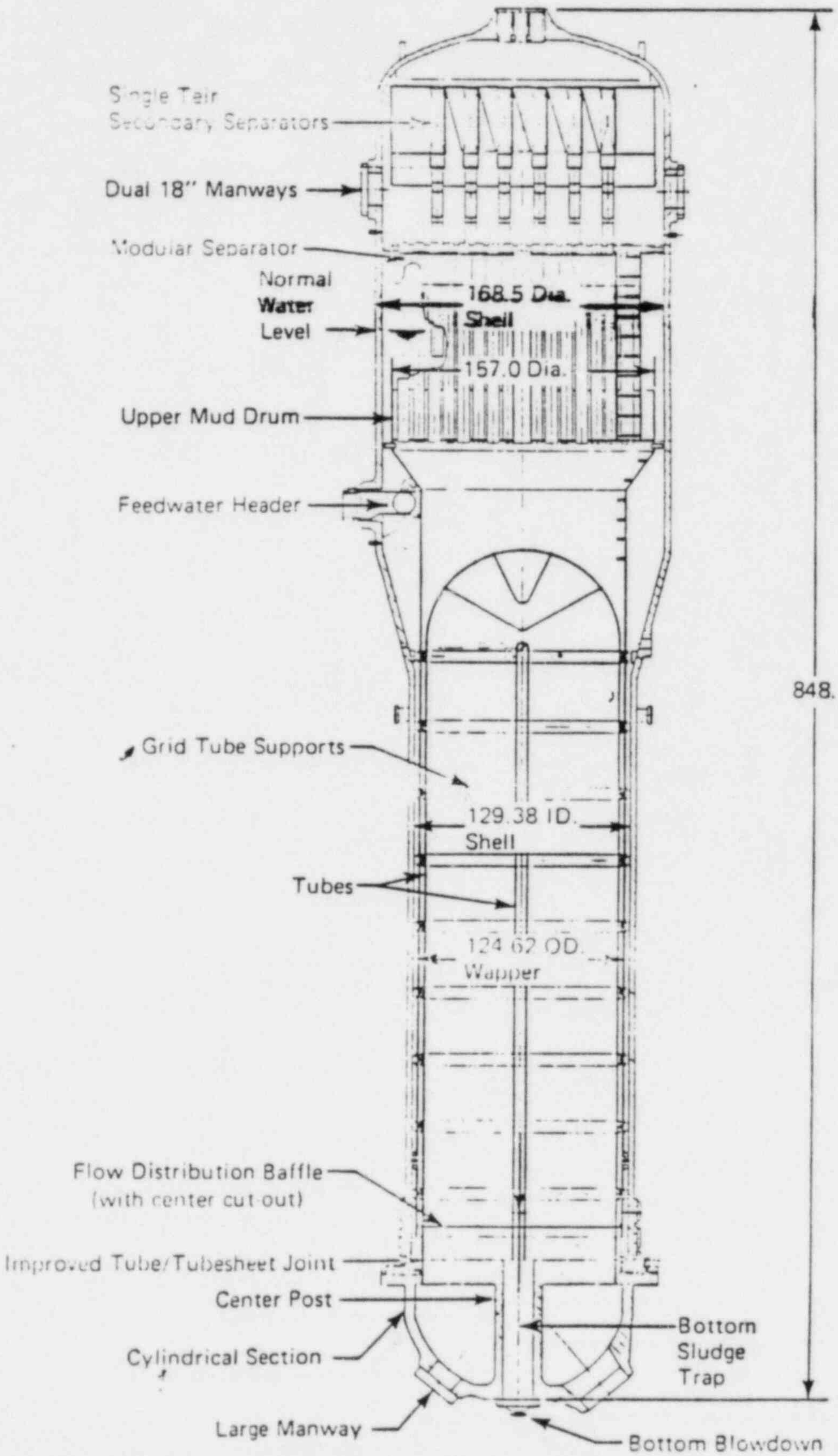
- 93A-1 ESSENTIALLY UNCHANGED
- SUPPORT SYSTEM

o PRESSURIZER

- ESSENTIALLY UNCHANGED

o STEAM GENERATOR

ADVANCED MODEL STEAM GENERATOR



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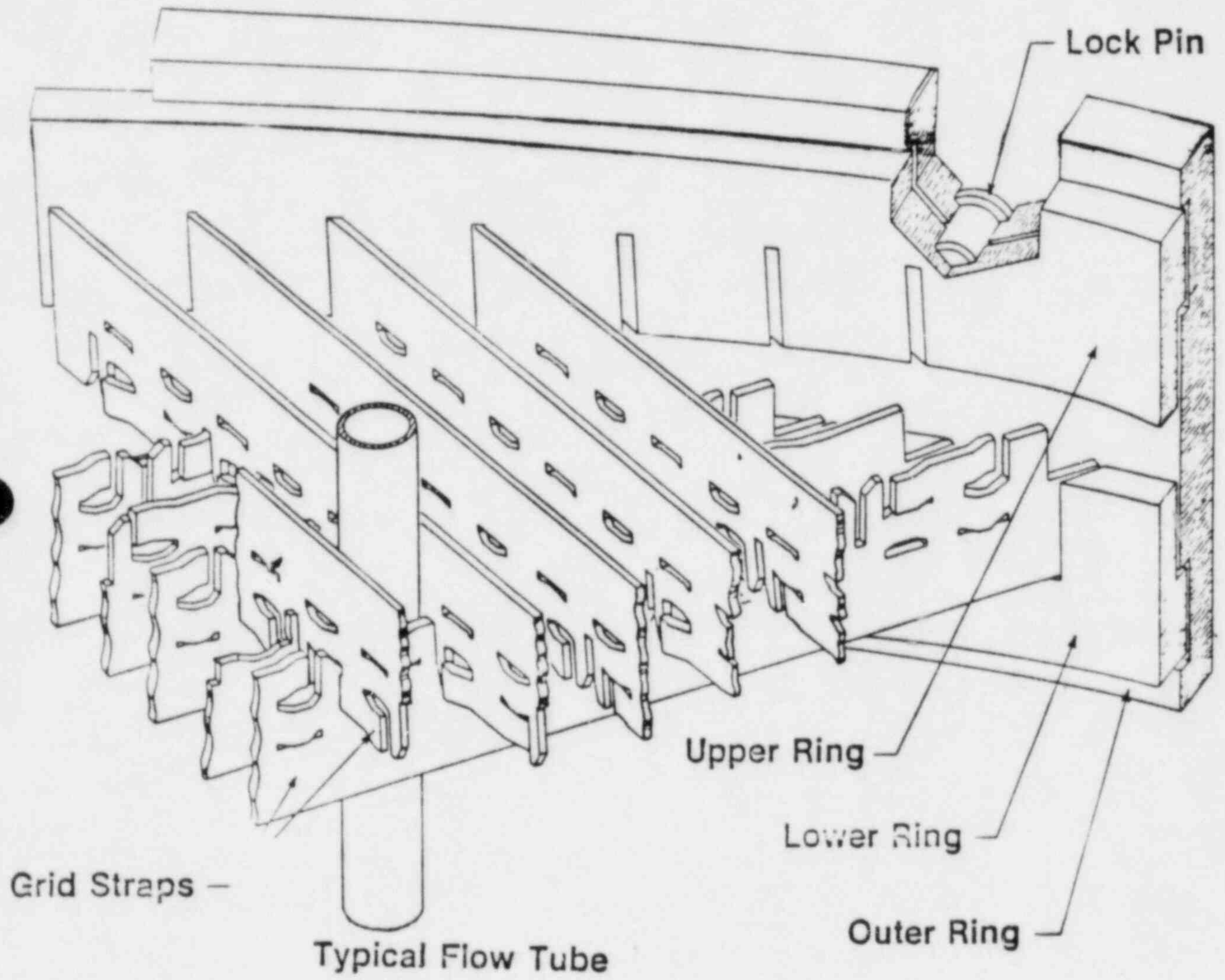
TUBE SUPPORT GRID ASSEMBLY



AUXILIARY FLUID SYSTEMS

- CHEMICAL AND VOLUME CONTROL SYSTEM
- BORON THERMAL REGENERATION SYSTEM
- BORON RECYCLE SYSTEM
- COMPONENT COOLING WATER SYSTEM
- SERVICE WATER SYSTEM
- WASTE PROCESSING SYSTEMS
- SAMPLING SYSTEM
- EMERGENCY SEAL INJECTION SYSTEM
- CONTAINMENT HEATING AND VENTILATION

LICENSING

KEY ELEMENTS

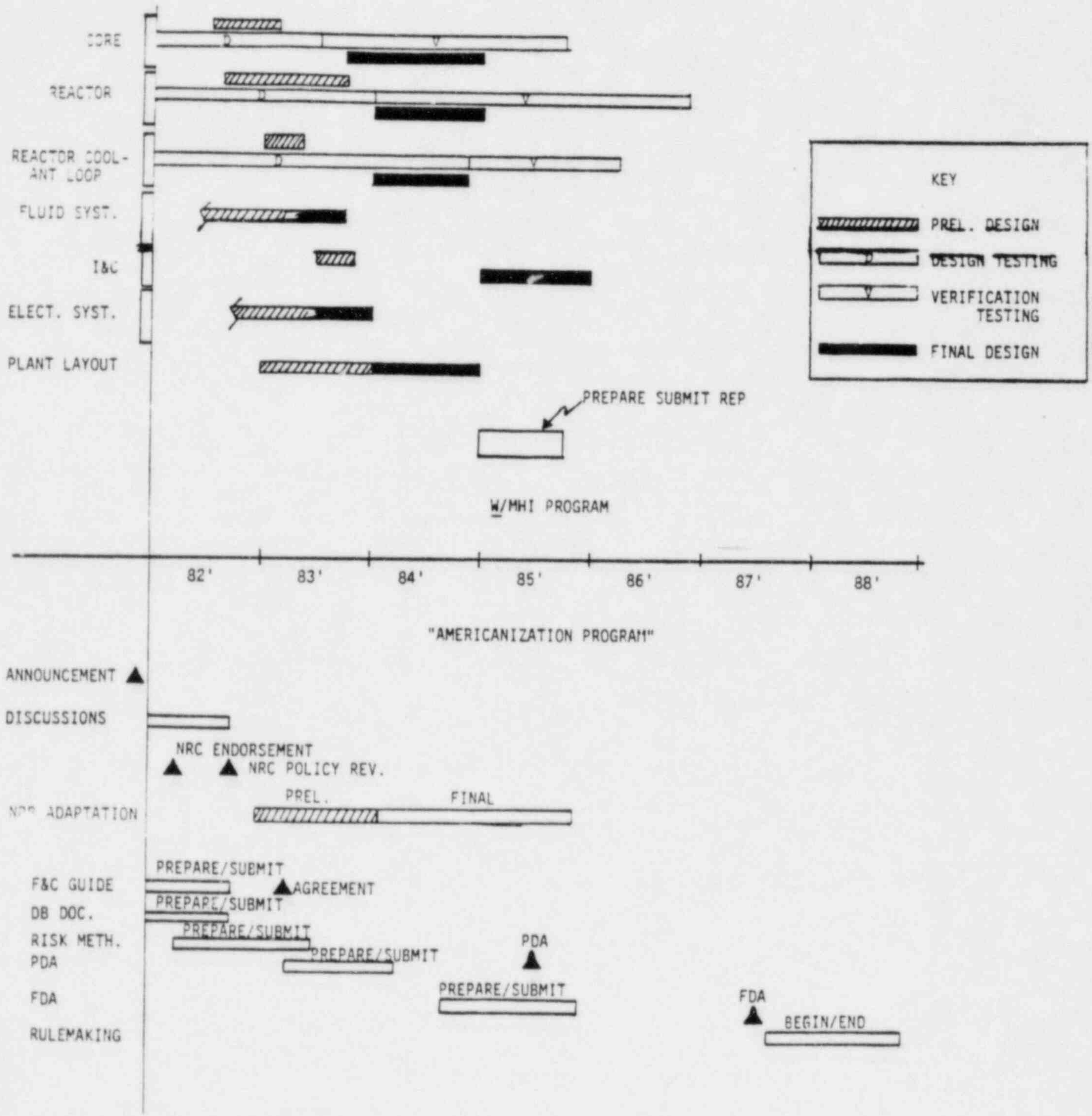
- POWER BLOCK
  - SINGLE ORGANIZATION (VENDOR) CONTROL RESPONSIBILITY FOR ALL SAFETY FEATURES
  - FLEXIBILITY ON SCOPE, I.E., DESIGN, SUPPLY, INSTALLATION, CONSTRUCTION, MAINTENANCE
- RISK ASSESSMENT
  - INTEGRAL TO THE DESIGN
  - CONTROL REASSESSMENT OF DESIGN ADEQUACY
  - FAILURE RATE DATA LIBRARY CONTINUOUSLY MAINTAINED
- SITING
  - SPECIFICATION OF SITING REQUIREMENTS BY BOUNDING CHARACTERISTICS OR METHODOLOGY FOR EVALUATION
- RULEMAKING
  - RULEMAKING UNDER PROVISIONS OF APPENDIX O TO 10CFR PART 50
  - APPLICABLE TO DESIGN, DESIGN BASES, SITING SPECIFICATION, RISK METHODOLOGY AND SAFETY GOAL

- REFERENCABILITY
  - CONSTRUCTION BEGINS AT TIME OF APPLICATION
  - DEVIATIONS FROM REFERENCE DESIGN EVALUATED UNDER "10CFR50.59"
  - APPLICANTS BUFFERED FROM NEW REGULATIONS/STANDARDS EXCEPT FOR NECESSARY DESIGN CHANGES FOR REASON OF SAFETY
  - FINAL, AS BUILT, DESIGN DOCUMENTATION PROVIDED PRIOR TO OL
- MAN/MACHINE INTERFACE
  - OPERATING LIMITATIONS INTEGRAL TO DESIGN
  - EMERGENCY/ABNORMAL OPERATING PROCEDURES INTEGRAL TO DESIGN
  - HUMAN FACTORS DESIGN BASE

PROGRAM MILESTONES

JANUARY, 1982	-	ACRS ACKNOWLEDGEMENT/ENDORSEMENT, MECHANISM FOR CONTINUING INVOLVEMENT
	-	BEGIN PRETENDERING DISCUSSIONS WITH NRC
MARCH, 1982	-	NRC REVIEW COMMITMENT, PROGRAM/PROCESS AGREEMENT
SEPTEMBER, 1982	-	NRC POLICY REVISIONS
	-	PROPOSED REVISION TO STANDARD FORMAT AND CONTENT GUIDE
	-	DESIGN BASES DOCUMENT SUBMITTAL
MARCH, 1983	-	FORMAT/CONTENT REQUIREMENTS ESTABLISHED
	-	AGREEMENT ON DESIGN BASES
JUNE, 1983	-	SUBMIT RISK METHODOLOGY AND COST/BENEFIT TRADE-OFF MODEL
MARCH, 1984	-	SUBMIT PSAR LEVEL DOCUMENTATION
JULY, 1985	-	PRELIMINARY DESIGN APPROVAL
JANUARY, 1986	-	SUBMIT FSAR LEVEL DOCUMENTATION
JULY, 1987	-	FINAL DESIGN APPROVAL
DECEMBER, 1988	-	FINAL RULEMAKING COMPLETE

DEVELOPMENT MILESTONES



NEAR TERM NEEDS

JANUARY, 1982	-	ACRS LETTER OF ACKNOWLEDGEMENT, ENCOURAGEMENT TO PROCEED
	-	ACRS PROCESS/SCHEDULE FOR CONTINUING REVIEW
MARCH, 1982	-	NRC REVIEW COMMITMENT, PROGRAM/PROCESS AGREEMENT
	-	MEETING SCHEDULE ESTABLISHED
SEPTEMBER, 1982	-	NRC POLICY REVISIONS
MARCH, 1983	-	FORMAT/CONTENT REQUIREMENTS ESTABLISHED
	-	DESIGN BASES AGREEMENT

Outline of Rule for Severe Core Damage

Assume rule must deal specifically with:

(1) Generation and dispersal in containment of an amount of hydrogen equal to significant fraction of that which would result from 100% reaction of water with core zirconium.

(2) The possibility of severe core damage beyond that which would result in (1).

(3) The possibility of release from ~~primary~~ containment of significant fraction of the radioactive material originally in the core.

Philosophy

(1) Hydrogen production is probable enough and has consequences serious enough that it must be dealt with as a design basis accident.

(2) Significant core melt is considerably less probable than generation of large amounts of hydrogen. It should be dealt with primarily by prevention.

(3) However because of the uncertainity in the capability to calculate what can be thought of and the impossibility of thinking of everything, some mitigation capability should be provided.

The Rule

(1) Hydrogen - Means must be provided which preclude, with high reliability, the collection of a detonatable mixture of hydrogen in containment. The system for insuring this must preclude containment pressure and temperature from exceeding source specified level (to be determined by further investigation). The system for dealing with hydrogen must be capable of handling an amount of hydrogen equal to 80% of that which would be produced by metal-water reaction with all the zirconium in the core region.

(2) The probability of melting of 80% of the core must be shown by analysis to be less than 10-4 per year. This demonstration must include at least a careful qualitative consideration of sabotage and demonstration that the system relied upon to remove decay heat must have a failure rate (assuming automatic actuation and operation) of no more than 10-3 per demand.

Outline of Rule for Severe Core Damage (Cont'd)

(3) The probability of release of more than 10% of the radioactivity contained in the core at shutdown within the first 96 hours after shutdown must be shown to be less than 10-2.

(4) Emergency planning should consider the residual risk associated with (2) and (3).

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

January 6, 1982

APPENDIX XIV
SUBCOMMITTEE ON HUMAN FACTORS MEETING OF
JANUARY 5, 1982: PROJECT SUMMARY

MEMORANDUM FOR: David Ward, Chairman
ACRS Human Factors Subcommittee

FROM: Richard Major, Staff Engineer Richard Major

SUBJECT: SUBCOMMITTEE ON HUMAN FACTORS MEETING OF JANUARY 5, 1982
PROJECT SUMMARY

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

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
K. Kirby, ACRS Fellow
C. Ryder, ACRS Fellow
E. Case, NRR
E. Goodwin, NRR
H. Thompson, NRR
J. Kramer, DHFS
V. Moore, DHFS
L. Beltracchi, DHFS
E. Blackwood, ROGR
J. Norberg, RES

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PROPOSED SUMMARY OF THE JANUARY 5, 1982
MEETING OF THE ACRS SUBCOMMITTEE ON
HUMAN FACTORS

Purpose: The purpose of the meeting was to continue Subcommittee discussions on three NUREG documents prepared by the Division of Human Factors Safety. The documents are:

1. NUREG-0700, "Guidelines for Control Room Design Reviews."
2. NUREG-0801, "Evaluation Criteria for Detailed Control Room Reviews."
3. NUREG-0835, "Acceptance Criteria for the Safety Parameter Display System."

The Subcommittee expects to present these NUREGs to the full ACRS during the January 1982 meeting for information and to give the Committee a chance to comment on these NUREGs if it so desires. Also, the results of the CRGR (Mr. Stello's Committee to Review Generic Requirements) review of Human Factors programs will be presented to the full Committee. The full ACRS may wish to discuss in a generic sense the interaction between the Committee and the CRGR.

Attendees:

ACRS

D. Ward, Chairman
W. Mathis, Member
J. Ray, Member
W. Keyserling, ACRS Consultant
R. Pearson, ACRS Consultant
A. Debons, ACRS Consultant
R. Major, ACRS Staff
J. MacEvoy, ACRS Fellow
K. Kirby, ACRS Fellow

NRC

H. Thompson, DHFS
M. Greenberg, "
J. Kramer, "
L. Beltracchi, "
R. Froelich, "
— V. Moore, "
H. Thompson, "
S. Weiss, "
D. Tondi "
R. Schemel, "
R. Echenrode, "
— E. Blackwood, ROGR
K. Goller, RES
J. Norberg, RES
P. Williams, NRR
J. Jenkins, RES

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Attendees (Continued):

A. Morse, Jr., Stone & Webster
R. Faust, Jr., Stone & Webster
J. Miller, Westinghouse
J. Gallagher, "
J. Little, "
H. Price, Human Factors Society
J. Preston, Lawrence Livermore Natl. Lab.
R. Huston, Consumers Power Co.
T. Tipton, AIE
S. Harris, EEI
L. Lund, Lund Consulting, Inc.
P. Diety, INPO
L. Kolonay, B&W
W. Coley, Duke Power, Chairman, AIF Control Room Subcommittee
R. Leyse, EPRI
H. Rooher, NUS Corp.

Meeting Highlights, Agreements and Requests:

1. Mr. E. Blackwood of the Regional Operations and Generic Requirements Staff presented the CRGR's review to date of Emergency Response Capabilities and Facilities. Part of the responsibility of the CRGR includes a review of all new requirements to be placed on utilities in order to ensure an integrated approach to the requirements coming from the NRC and to ensure that the number of new requirements are not excessive. The Committee is headed by Victor Stello, Deputy EDO, with representatives from the various NRC divisions.

Attachment 1 is a copy of the CRGR's preliminary comments. These comments and views will be finalized later this month. CRGR suggests that when the basic requirements for emergency response capabilities and facilities are finalized, they should be transmitted to licensees via a generic letter from NRR, promulgated to NRC Staff and incorporated in the Standard Review Plan. The basic requirement proposed by CRGR is that the SPDS (Safety Parameter Display System) be established as a regulatory requirement that should be implemented independent of the other initiatives (emergency response facilities, control room improvements, etc.). The SPDS is to be an operator aid in the control room. The SPDS should allow operators to rapidly and reliably determine the safety status of the plant. The SPDS will not have seismic or class IE qualifications and will not have to meet the single failure requirements. The SPDS will present a limited number of variables. There will be no future changes to the SPDS required.

Regarding control room improvements, the CRGR believes that licensees should be required to review the available human engineering documents (including NUREG and EPRI documents). Licensees should be required to describe modifications they wish to make to the NRC, and other than for reasons of unreviewed safety questions or a technical specification change which requires Staff approval according to 10 CFR 50.59, licensees should be free to implement control room improvements based on their own review. In general, the CRGR believes Reg. Guide 1.97 should be published and implemented although there are some exceptions. Requirements for equipment environmental qualifications should be handled in rulemaking. The CRGR believes only wind speed, wind direction and atmospheric stability from the variables listed in Reg. Guide 1.23 need be required for the purposes of emergency response.

The CRGR will decide which items from the published material on emergency support facilities (Technical Support Center, Operations Support Center, and Emergency Operations Facility) are important enough to be basic requirements to be imposed on licensees. These requirements include size, structure, habitability, communications and documentation, and staffing and activation. No seismic or class IE requirements or single failure requirements are felt to be necessary for emergency facilities. The CRGR believes implementation of requirements should be done interactively between licensees and the NRC. There should be flexibility in the implementation schedules.

2. NUREG-0700 deals with guidelines for control room design reviews. It is basically a handbook of human engineering principles for control rooms. The control room design review guidelines were initiated as a result of studies stemming from the accident at Three-Mile Island. Other recommendations included improvements in training, emergency operating procedures, and management and organization. The Staff feels all these

items are equally important. Due to resource limitations and since the control room is the nerve center of the plant, the Staff felt it was the logical place to begin implementing recommendations from the TMI experience. Initiatives in other areas of recommendations are under way.

3. The Subcommittee raised a concern over the amount of emphasis on the design of the control room. Noting studies of LERs, it was pointed out that only a small fraction of human errors could be traced to the design details of the control room instrumentation and controls. It was noted it is possible that this concern could have arisen due to inadequacies in the LER data base.

4. The Staff explained that the current control room reviews will be based on function and systems analysis and the integration of control room design with procedures and the use of the SPDS. Function and task analyses are performed to determine what the operator's tasks are and what he needs in the control room to accomplish them.

5. The Staff noted that, in general, the present control rooms that display the best human engineering are those that were designed with a lot of input from the operating organization. In the future, the Staff expects licensees to have access to people with human factors experience in addition to control room design input from their operating organization.

6. The Staff presented some estimated figures for control room reviews and upgrades. For the control room reviews the cost would be approximately $500,000. For implementing enhancements (paint, label, and tape changes) estimates were around $100,000. An average number for equipment additions

and relocations was $300,000 although this number varied greatly among different licensees giving estimates. The cost of a basic SPDS would be between $1/2 M to $1 M, although it is expected many licensees will go beyond basic requirements. Upgrading emergency operating procedures will cost about $560,000.

7. Modification verification is emphasized in NUREG-0700 and NUREG-0801 to guard against faulty modifications. Licensees will use simulators and mock-ups to minimize the risk of faulty modifications. The NUREGs also recommend that licensees develop programs for following the performance of the operators to ensure, through feedback, that the changes are appropriate. Retraining of operators to ensure recognition of new modification is also stressed. The Staff recognizes that there is a possibility of some short-term degration in performance, however, they believe the overall effect on the remaining life of the plant will be positive.

8. NUREG-0700 has two purposes. The short-term purpose is performing detailed control room reviews over the next two years. A longer-term purpose is to use this document in the design of new plants. In the short term NUREG-0700 will be updated, as necessary, based on surveys of four to six plants at various stages of their control room review. Over the long term NUREG-0700 would be updated by experience and used in the design of future plants.

9. NUREG-0801, "Evaluation Criteria for Detailed Control Room Reviews," gives the criteria for evaluating the results of the utilities' control room reviews based on NUREG-0700. NUREG-0801 does contain one method for categorizing and evaluating the potential or probability of an individual human engineering discrepancy causing a safety problem.

10. The Staff intends to review utility program plans before the control room design review actually begins. The Staff will ensure the utility has people qualified to make human factors decisions. The Staff estimates seven man-days of Staff effort on each program plan (which may apply to more than one unit) and eleven days of Staff consultant time. The Staff estimates the second phase of their review (the evaluation of the results of the utilities' control room design review with proposed changes to the control room, and justification for not making changes to correct individual human engineering discrepancies such as training or procedures rather than hardware fixes) would require 15 staff-days and 30 Staff consultant-days for this review.

11. Regarding Subcommittee consultant concerns over whether enough qualified human factors specialists would be available to perform control room reviews, the Staff believes there would be. The first year would have about 80 cases each requiring 3-4 months of time from a specialist. The Staff feels there are resources available to meet this demand.

12. Linda Lund of Lund Consulting, Inc. made a public statement which expressed a concern over a perceived narrow focus to human factors applications by the Staff. A copy of a letter from Ms. Lund to Chairman Palladino is Attachment 2.

13. The Staff discussed NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System." The function of the SPDS is to assist control room operators in detection of abnormal operating conditions which may impact on safety. The Staff hopes to accomplish this by integrating a minimum set of plant parameters into a display from which plant safety status may be assessed by control room operators.

14. ACRS consultants questioned the Staff on SPDS display formats. It was suggested that a top-level display (those critical eight to ten parameters) be continuously displayed either on a separate CRT or on a hardwired display. The Staff noted they had such a requirement but dropped it, to allow additional information to be presented such as trend plots. Trend plots could be called up on secondary formats. The Staff was cautioned not to allow an SPDS which could involve an operator in depth on a particular variable on a secondary display to the point where he looses sight of the overall plant status. In general, the Staff noted that multiple-screen CRT displays are being proposed for SPDS designs.

15. The Staff explained that NUREG-0835 responds to industry SPDS design trends which have universally elected to use CRT formats. The Staff

believes this a cost-effective development of regulations. However, the Staff has not disallowed any display technology.

16. The Human Factors Staff believes that a disturbance analysis system is a potential future addition to the SPDS. However, additional research is needed to evaluate the need and if necessary develop functional requirements for disturbance analysis systems.

17. Westinghouse briefed the Subcommittee on their SPDS design. Westinghouse noted that they have sold 9 units to 5 utilities. Display formats were given along with the human engineering input to the design.
The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions. As a secondary function it is recognized that upon detection of an abnormal plant status, it may be desirable to provide additional information to analyze and diagnose the cause of the abnormality, execute corrective actions, and monitor plant response. The Westinghouse evaluation of safety parameter display concepts, and technical audit of their SPDS concept was presented.

18. The majority of Subcommittee consultants were in favor of issuing the NUREG documents and applying their reviews to the industry, but urged flexibility in the Staff's approach. The consultants also noted that control room design was only a part of the total human factors picture. Emergency and maintenance operating procedures, training, and management organization are all equally important.

Future Meetings: The full ACRS will hear an abbreviated presentation on the three NUREG documents and CRGR review on Friday, January 8, 1982.

A-242

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

DEC 29 1981

MEMORANDUM FOR: Harold Denton, Director
Office of Nuclear Reactor Regulation

Richard DeYoung, Director
Office of Inspection and Enforcement

Robert Minogue, Director
Office of Nuclear Regulatory Research

John Davis, Director
Office of Nuclear Material Safety and Safeguards

FROM: Victor Stello, Jr.
Deputy Executive Director
Regional Operations and Generic Requirements

SUBJECT: EMERGENCY RESPONSE CAPABILITY AND FACILITIES

Emergency response activities within NRC were discussed at the CRGR meetings on December 3 and 10, 1981. The outcome of those meetings included the need to articulate the basic requirements for emergency response facilities and related initiatives. Other issues include the extent to which (1) information and guidance in NUREGs and Regulatory Guides may have been imposed as requirements and (2) NRC has effectively managed coordination and integration of initiatives.

The enclosed package proposes basic requirements and a way in which they could be implemented. Note that plant-specific schedules would be developed with licensees to take advantage of their previous efforts to implement the initiatives. I expect that adoption of firm, basic requirements with flexibility to implement them on a realistic schedule will reduce the degree of uncertainty that now exists among licensees.

I would appreciate the benefit of your thoughts on the issues and comments on the enclosed package so that the CRGR in its next meeting may formulate its recommendation to the EDO. Please forward comments to me or E. B. Blackwood to be received by January 5, 1982.

RMinogue

for Victor Stello, Jr.
Deputy Executive Director
Regional Operations and Generic Requirements

Enclosure:
Emergency Response Capability

cc: W. Dircks
CRGR Members

ATTACHMENT 1

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EMERGENCY RESPONSE CAPABILITY

Studies that followed the accident at TMI identified the need to improve the on-site and off-site management capability for responding to accidents. The fundamental weakness revealed during these studies was the lack of attention devoted to the "man" in the "man -- machine equation." We must not detract from this finding. Well trained operating staff with clearly defined emergency roles is the cornerstone to accident response. Preplanning by utility, industry and governmental representatives is necessary. Well thought out and practiced emergency procedures both on site and off site are required. Following the accident at TMI, the President directed that the off-site responsibility for emergency response would be under the cognizance of the Federal Emergency Management Agency. Significant progress has been made in this area with all operating plants required to have fully implemented emergency plans by April 1, 1981, and our review and evaluation of these plans completed 1 year from then.

Progress for improving the on-site capability has been slower. There has been confusion within and outside the NRC regarding additional features and equipment needed as part of this emergency response. These include the Safety Parameter Display System, the Technical Support Center, Emergency Operations Facility, the On-Site Operational Support Center, revised emergency procedures, control room reviews, the use of Regulatory Guides 1.97 and 1.23. Consideration of these features, if done in a fragmented and uncoordinated manner, can weaken accomplishing the improvements cited in the preceding paragraph. Therefore, review and incorporation of these facilities as aids to various personnel that respond to emergencies must be accomplished and integrated into the overall emergency plan. Flexibility of equipment and facility arrangement is mandatory and must remain as an overriding principle to assure a timely and fully coordinated emergency response capability.

Based on a review of existing requirements and guidance on multiple aspects of emergency response capabilities and facilities, the CRGR believes that the need exists to articulate the substantive (basic) requirements for power reactor licensees. The information presented in the CRGR meeting on December 3, 1981, underscored the need within the NRC staff for more effective management, coordination and integration of initiatives related to emergency response.

Numerous comments received from nuclear industry groups and individual NRC licensees of operating reactors reflect widespread uncertainty regarding the extent to which information and guidance published by NRC are being applied as regulatory requirements. The CRGR believes that the differences between requirements and information/guidance and how they are applied, need to be reiterated to the industry and NRC staff.

When the basic requirements for emergency response capabilities and facilities are finalized, they should be transmitted to licensees via a generic letter from NRR, promulgated to NRC staff and incorporated in the Standard Review Plan. The letter to licensees should request that licensees submit a proposed schedule for completing actions to comply with the basic requirements. Each licensee's proposed schedules would be reviewed by its Licensing Project Manager, who would discuss them with the licensee and mutually agree upon schedules and completion dates.

The following sections describe CRGR recommendations for basic requirements, their interrelationships and NRC actions to improve management of emergency response regulation.

Use of Existing Documentation

The CRGR recommends that NRC issue a policy statement such as:

The following NUREG documents are to be used as information only, and the Regulatory Guides are to be considered as guidance or a possible approach to meeting formal requirements. Under no circumstances should the items in these documents be misconstrued as requirements to be levied on licensees or as inflexible criteria to be used by NRC staff reviewers.

NUREGs	0654 -	Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants
	0696 -	Functional Criteria for Emergency Response Facilities
	0700 -	Guidelines for Control Room Design Reviews
	0799 -	Draft Criteria for Preparation of Emergency Operating Procedures
	0801 -	Evaluation for Control Room Design Reviews
	0814 -	Methodology for Evaluation of Emergency Response Facilities
	0835 -	Human Factors Acceptance Criteria for SPDS
Reg. Guides	1.23 -	Meteorological Measurement Program for Nuclear Power Plants
	1.97 -	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

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Coordination and Integration of Initiatives

Recommendations

1. The SPDS is not contingent on control room reviews, R.G. 1.97, 1.23, TSC, EOF, OSC or NDL. SPDS should be established as a regulatory requirement that should be implemented independent of the other initiatives listed above. The NRC does not plan to impose future additional requirements on licensees regarding SPDS.

2. Implementation of part or all of Regulatory Guides 1.97 and 1.23 represents a control room improvement. This and any other control room improvements are separate and independent of TSC, EOF, OSC and NDL initiatives in terms of content and sequence of implementation.

3. Emergency response facilities (TSC and EOF) are related in terms of communication and instrumentation needs among the TSC, EOF and control room. TSC and EOF structures are independent of each other. The OSC is independent of TSC and EOF.

4. The three groups of initiatives discussed above (1-SPDS, 2-control room improvements and 3-emergency response facilities) are independent of each other except for the following interrelationships:

   (a) The SPDS is an improvement in the control room because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention. The SPDS influences other control room improvements that licensees may consider and to an extent could obviate the need for extensive modifications to control rooms.

(b) New instrumentation that may be added to control rooms should be considered for inclusion in the design of the TSC and EOF only to the extent that such instrumentation is essential to TSC and EOF functions.

(c) The SPDS and control room improvements are essential elements in operator training programs and the final plant-specific emergency operating procedures.

5. Specific implementation plans and reasonable, achievable schedules should be defined by mutual agreement between NRC and each individual licensee. The program office responsible for implementing each initiative should propose procedures identifying:

(a) The respective roles of NRR and IE Headquarters and Regions in checking licensee rate of progress and verifying compliance, including the extent to which NRC approval (review and inspection) is necessary during implementation.

(b) Procedural methods and enforcement measures that could be used to assure NRC staff and licensee attention to meeting mutually agreed upon schedules without significant delays and extensions.

Safety Parameter Display System (SPDS)

Actual Regulatory Requirements

None

Functional Statement

The SPDS should provide a concise display of critical plant variables to the control room operators in order to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations and all classes of emergencies, the primary purpose of the display is to aid the operators in monitoring the safety status of the plant during anticipated transients and the initial phase of accidents.

Recommended Requirements

1. Each operating reactor shall be provided with a Safety Parameter Display System conveniently located within the control room. This system will serve to concentrate and continuously display a minimum quantity of information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

2. The principal purpose and function of the SPDS will be to aid control room personnel in timely detection and assessment of abnormal conditions for the reactor which must be subsequently controlled and corrected through human actions to avoid damage to the reactor core.

3. The SPDS need not meet requirements of the single failure criterion, and it need not be qualified per class IE requirements if suitably isolated from equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified.

4. The important plant functions relevant to the information display of the SPDS shall include, but not be limited to:

   ° Reactivity control

   ° Reactor core cooling and heat removal from the primary system

   ° Reactor coolant system integrity

   ° Containment conditions

   Guidance on specific variables that may usefully serve the SPDS purpose are the Type A and B variables listed in Tables 1 and 2 of RG 1.97.

Basic Reference Documents

NUREG-0600	--	Need for SPDS identified
NUREG-0737	--	Specifies SPDS
NUREG-0696	--	Functional criteria for SPDS
NUREG-0835	--	Specific acceptance criteria keyed to 0696
RG 1.97	--	Support document for variables to be used on SPDS

Detailed Control Room Design Review (DCRDR)

Actual Regulatory Requirements

As specified in item I.D.1 in NUREG-0737, implementation schedule to be developed.

Functional Statement

To reduce human error in the control room of a nuclear power plant.

Recommended Requirements

Licensees shall review the human engineering handbook being compiled by EPRI and consider its content as one input in deciding on control room improvements. Licensees shall submit to NRC a brief description of modifications planned as a result of the review. No NRC approval of these changes is necessary other than that which would otherwise be required pursuant to 10 CFR 50.59.

Basic Reference Documents

NUREG-0585	Statement that licensees should conduct review.
NUREG-0660	States that NRR will require for ORs, OLs.
NUREG-0737	States that requirement was issued 6/80, final guidance not yet issued.
NUREG-0700	Final guidelines for DCRDR
NUREG-0801	Draft for comment; Staff Evaluation Criteria

Regulatory Guides 1.97 and 1.23
Application to Emergency Response Facilities

Actual Regulatory Requirements

None

Functional Statement

Provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

Recommended Requirements

Control Room

- RG 1.97 (Rev. 2) - Those variables listed in Types A, B, C, D, E except those variables (such as BWR thermocouples) pending resolution or development as requirements are not required.
- Reliable indication of meteorological variables as specified in RG 1.97 for site meteorology with data accuracy as specified by RG 1.23 regarding stability, wind speed and direction.
- Meteorological system availability of approximately 0.9 or greater is acceptable.

TSC

- RG 1.97 (Rev. 2) - guidance for A, B, C. D, E variables with same exceptions as for control room above.
- Class 1E and seismic qualifications and single failure criterion need not be met for this instrumentation in the TSC. Environmental qualification (EQ) is not required in Emergency Response Facilities as EQ will be addressed in future rulemaking.

- Meteorological data as specified in RG 1.97 consisting of wind direction, wind speed and atmospheric stability for site (local) meteorology with data accuracy of RG 1.23 and regional data regarding stability, wind direction and wind speed.

EOF

- RG 1.97 (Rev. 2) guidance but only for selected variables enabling the EOF estimation of containment failure and releases of radioactivity from the plant.
- Class 1E and seismic qualifications and single failure criterion need not be met by the EOF data system. Environmental qualification will be addressed in a future rulemaking.
- Meteorological data as specified in RG 1.97 consisting of wind direction, wind speed and atmospheric stability for site (local) meteorology with data accuracy of RG 1.23 and regional data regarding stability, wind direction and wind speed.

Additional Recommendations

1. Issue RG 1.97 in near term after ensuring that RG 1.23 does not conflict with RG 1.97.
2. Do not issue RG 1.23 in near term until:
   (a) Justification and implications of proposed availability goals are more firmly established.

   (b) Backup meteorological data system is further justified in light of relative insensitivity of regional meteorological data to the overall risk and consequence predictions.

(c) RG 1.23 is reviewed and changed if necessary to make it consistent with RG 1.97.

Upgrade of Emergency Operating Procedures (EOP)

Actual Regulatory Requirements

10 CFR 50.34(b)(6)(v)

Functional Statement

To improve human reliability and the ability to diagnose and cope with multiple failure conditions at a nuclear power plant.

Recommended Requirements

Revise EOPs to make them symptom-oriented and consistent with SPDS, control room improvements and emergency response facilities.

<u>Emergency Response Facilities</u>

<u>Actual Regulatory Requirements</u>

10 CFR 50.47(b) (for OLs)	Requirement for emergency facilities and equipment to support emergency response.
10 CFR 50.54(g)	Requirement for prompt communications among principal response organizations to emergency personnel and to the public.
	Requirement that adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.
10 CFR 50.54(g) (for ORs)	Same requirements as 10 CFR 50.47b plus Appendix E
10 CFR Part 50, Appendix E	Requirement for:

1. Equipment at the site for personnel monitoring;
2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;
3. Facilities and supplies at the site for decontaination of onsite individuals;

4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;
5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on site;
6. Arrangements for transportation of contaminated injured individuals for the site to specifially identified treatment facilities outside the site boundary;
7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;
8. A licensees onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;
9. At least one onsite and one offsite communications system; each system shall have a backup power source.

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

a. Provision for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and amont the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power

	reactor control room, the onsite technical support center, and the near-site emergency operations facility. such communications shall be tested monthly.
Denton Letter 10/30/79	Clarification of requirements and implementation schedule.
Eisenhut Letter 4/25/80	Clarification of requirements.
Eisenhut Letter 2/18/81 (previously deleted from NUREG-0737)	Description of location, habitability and staff required for emergency response facilities. Request and deadline for submittal of conceptual design of emergency response facilities

Technical Support Center (TSC)

Functional Statement

When activated, the TSC will be the onsite technical operations center for predesignated technical, engineering and senior licensee management personnel; any other licensee predesigned personnel; and five NRC predesignated personnel. Once activated, the TSC will operate uninterrupted to perform the following functions until it is deactivated:

- Provide plant management and technical support to plant operations personnel.
- Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations.
- Perform EOF functions for the Alert Emergency class and for the site Area Emergency class and General Emergency class until the EOF is functional.
- Provide Technical Support to the EOF.

Recommended Requirements

1. Be located within the site protected area to facilitate necessary interaction with CR, OSC, EOF and other personnel involved with the emergency.
2. Be sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation in the center.

3. Be structurally built in accordance with the National Uniform Building Code.

4. Be environmentally controlled to provide normal room air temperature, humidity and cleanliness.

5. Have available radiological protection in accordance with 10 CFR Part 20 "Standards for Protection Against Radiation" for personnel coming to, leaving from or located in the center.

6. Be capable of uninterrupted voice, data and hard copy communications with CR and EOF and uninterrupted voice communication with OSC and NRC Operations Center.

7. Be capable of uninterrupted data collection, storage, analysis, display and communication sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

   (a) the variables in the appropriate Table 1 or 2 of RG 1.97 Revision 2, except those variables not required; and

   (b) the meteorological variables in RG 1.97 for site locale and region as accurate as is indicated in RG 1.23, Revision 1.

   Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

8. Have available accurate, complete and current plant records essential for evaluation of emergency conditions.

9. Be staffed by predesignated personnel under the direction of a predesignated senior licensee official and be operational within approximately 1 hour after activation.

Operations Support Center (OSC)

Functional Statement

When activated, the OSC will be the on-site area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by the CR, TSC and EOF.

Recommended Requirements

1. Be located on site to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

2. Be capable of uninterrupted voice communications with CR, TSC and EOF.

Emergency Operations Facility (EOF)

Functional Statement

The EOF is a licensee controlled and operated support center. The EOF will have facilities for:

- Management of overall licensee emergency response,
- Coordination of radiological and environmental assessment,
- Determination of recommended public protective actions, and
- Coordination of emergency response activities with Federal, State, and local agencies.

When the EOF is activated, it shall be staffed by predesignated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display, and evaluation of all radiological, meteorological, and containment failure data required to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radioactive releases from the plant and to determine dose projections.

Recommended Requirements

1. Be located within 20 miles of the site to facilitate necessary interaction with the CR, TSC, OSC and other personnel involved with the emergency.
2. Be sufficient to accommodate and support Federal, State, local and licensee predesignated personnel, equipment and documentation in the EOF.

3. Be structurally built in accordance with the National Uniform Building Code.

4. Be environmentally controlled to provide normal room air temperature, humidity and cleanliness.

5. Have available radiological protection in accordance with 10 CFR Part 20, "Standards for Protection Against Radiation for personnel coming to, leaving from or located in the EOF.

6. Have uninterrupted voice, data and hard copy communications facilities to the TSC and control room, and uninterruptable voice communication facilities to NRC, State and local emergency operations centers. The normal communication path between the EOF and the control room will be through the TSC.

7. Be capable of uninterrupted collection. storage, analysis, display and communication of data addressing containment failure, radiological release and meteorological data sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF function shall be available in the EOF:

   (a) variables from the appropriate Table 1 or 2 of RG 1.97, Revision 2, except those variables not required; and

   (b) the meteorological variables in RG 1.97 for site locale and region as accurate as is indicated in RG 1.23, Revision 1.

Principally those data must be available that would enable evaluation of incident sequence, determination of mitigating actions, evaluation of damages, and determination of plant status during recovery operations.

8. Have ready access to current plant records, procedures, and emergency plans needed to perform EOF functions.

9. Be staffed by predesignated personnel under the direction of a predesignated senior licensee official and be operational within approximately 1 hour after activation.

10. Be provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

Basic Reference Documents

10 CFR 50.47(b)	Requirements for emergency facilities and equipment for OLs.
10 CFR 50.54(g) and Appendix E	Requirements for emergency facilities and equipment for ORs.
NUREG-0660	Description of and implementation schedule for TSC, OSC and EOF.
Eisenhut letter 9/13/79	Request for commitment to meet requirements.
Denton letter 10/30/79	Clarification of requirements and implementation schedule.
Eisenhut letter 4/2[illegible]/80	Clarification of requirements.
NUREG-0696	Functional criteria for emergency response facilities.
NUREG-0737	(deleted for document)
Eisenhut letter 2/18/81	Description of location, habitability and staff required for emergency facilities

	Request and deadline for submittal of conceptual design of facilities.
NUREG-0814	Methodology for evaluation of emergency response facilities.
RG 1.97	Guidance for variables to be used in selected emergency response facilities.
RG 1.23	Guidance for meteorology.

Nuclear Data Link (NDL)

Actual Regulatory Requirements

None

Functional Statement

To provide for transmission of reactor data to an NRC facility in the event of an emergency.

Recommended Requirements

Data management facilities available in the EOF shall include the capability for future modification to provide for transmittal of the data to another site.

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January 4, 1982

Nunzio Palladino, Chairman
US Nuclear Regulatory Commission
Washington, DC 20555

Subject: December 29, 1981 letter from V. Stello
"Emergency Response Capability and Facilities"

Dear Chairman Palladino:

As a result of the accident at Three Mile Island the nuclear utility community and the NRC have recognized the importance of consideration of the area of human factors in the design and operation of nuclear power plants. We, members of this expert interdisciplinary community, are extremely concerned over the contents of the subject letter which totally disregards the need for human factor consideration in nuclear regulations.

Specifically, we note that while Mr. Stello cites a "fundamental weakness revealed during these (TMI) studies was the lack of attention devoted to the "man" in the "man-machine equation," all of the proposed regulations promoted by Mr. Stello are "machine" solutions to the "man" problem.

The human factor community has been working closely with members of the NRC's Division for Human Factor Safety to educate, provide guidance and promote consideration of human factors in control room instrumentation, use of procedures, training needs, staffing considerations, etc.

We encourage a multi-disciplinary approach of human factors experts and utility personnel to provide an integrated and correct consideration of the man-machine interface in nuclear power plants.

We feel that a fragmented and equipment oriented approach as described by the subject letter is both incorrect and misguided.

Sincerely,

Linda O. Lund

Linda O. Lund
President

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
J. Davis
T. Murley

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ATTACHMENT 2

While the attached letter represents the sole expressed views of the undersigned Lund, Inc., several concerned human factor experts have serious concerns with the Dec. 29 letter.

Though our decision was to comment individually to pursue timeliness, my colleagues and I share the concern that the CRGR recommendations lack human factor consideration.

Each of us would be willing to be contacted to provide comments collectively at NRC's request:

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Statement on December 29, 1981 CRGR Letters
"Emergency Response Capability and Facilities"

As human factors specialists working in the nuclear industry, we have long been calling for an integrated approach to human factors efforts. We feel strongly that SPDS the TSC, EOF, OSC, revised EOP's, CR reviews and Reg. Guides 1.97 and 1.23 must not be considered "in a fragmented and uncoordinated manner".

We acknowledge and support the finding that the "fundamental weakness" revealed during post-TMI studies was the "lack of attention to the 'man' in the 'man-machine equation' ". We also concur with the statement that "well trained operating staff with clearly defined emergency roles is the cornerstone to accident response." However, we would add that:

- Well-designed and operational Control Rooms and
- Well-written and validated EOP's

are the necessary aides to the reactor operating crews and their supervisors during an emergency.

Although we support the initial statements, the remainder of this document appears to be in direct conflict with our knowledge of TMI-2 and the current state of affairs at US Nuclear Plants. Indeed, the recommendations focus in this document exclusively on the "machine' side of the "man-machine equation".

If a "well trained operating staff" is a "cornerstone of accident response" then the almost total lack of reference to training in the recommendations is a definite shortcoming. For example, recommendations for the Emergency Response Facilities deal almost exclusively with the equipment to be installed (the "machines") and make no reference to the staffing (the "men") in these facilities or the

- organization
- role assignment and
- training

of the staff members. Moreover, the supposed slowness of the on-site emergency response needs to be balanced by the fact that on-site, normal as well as emergency response is a concern.

The statement is made that "The SPDS and control room improvements are essential elements in operator training programs...." It is true that training programs for operators will have to familiarize operators with control room improvements and train them in the use of an SPDS. In addition, training programs will have to familiarize operators with changes made to EOP's. However, nowhere in this document is there a full understanding of the impact an SPDS will have on an operator and the implications that it will have on training.

An SPDS brings with it some definite problems. Not only are there difficulties in the development of the equipment itself, but,

- Where it is placed in the control room
- How its use is integrated in EOP's
- How it is used as a diagnostic tool vis-a-vis the Control Board and
- How the operator is trained to use the two diagnostic modalities (SPDS and the Control Board)

are important considerations.

The above four issues regarding SPDS are either ignored or glossed over lightly in the December 29, 1981 document.

To date a SPDS is still a device of the future. Vendors are still working on the design of such a system; so the validity of the usefulness of such a device has no basis to determine the actual usefulness of a SPDS to its trained user(s) during normal, abnormal or emergency conditions.

As with the Emergency Response Facilities, equipment ("machine") needs are focused on in the recommendations for a SPDS. Moreover, a SPDS as a solution to other control room modifications is "the cart before the horse".

The potential danger in this document, if implemented is that it follows the pre-TMI philosophy that it purports to deplore. Lund, being involved with the daily operations of nuclear facilities recognizes the need for Human Factor considerations in control room design, instrumentation, staffing, training, procedure use, etc.

This multi-disciplinary human factor approach is recognized and welcomed as long overdue by the personnel at nuclear power generating stations. The implementation of a SPDS does not cover all of these diverse areas and without Human Factor considerations will be more of a hinderance to, than a helpful device for the, nuclear community. That is, by encouraging the addition of an SPDS and misunderstanding the importance of the Control Room review and systematically developed EOP's yet another piece of machinery is to be added to the Control Rooms without substantial analysis of its impact on the operator or its use in overall plans to mitigate accidents.

A "man-machine equation" does not refer to the simple sum of two numbers, but to a dynamic interrelationship. The present design of the control room must be reviewed to identify existing human factor deficiencies and to assess the best solution: hardware, procedures or training. Also, since the need is for not just the best design or format, but the assessment of the functional utility of:

1. Control Board
2. SPDS
3. EOP's

a systems-oriented functional task analysis of the Control Room, the SPDS and the EOP's in relation to the crew that use them must be performed.

There is no basis for this document's statement that "The SPDS is an improvement in the control room because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention". Indeed if the type of systems-functional-task analysis referred to above is not done, an SPDS could easily detract from, not enhance operator performance.

As human factors specialists in the nuclear industry we have witnessed much confusion over human factors issues as presented in many of the cited NUREGs. This confusion in many cases comes from an unfamiliarity with the human factors field and its methodologies. We would urge more open communication between HF specialists and the NRC to help simplify and coordinate the HF effort in the nuclear industry. As concerned HF scientists we deplore the move back to a pre-TMI philosophy as illustrated by this document.

Submitted by the staff of Lund, Inc.

L.O.Lund, President

C. Sherwood
Dr. P. Haymond
G. Opetosky

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KEMENY COMMISSION

CRITICIZED NRC

- NO EXAMINATION OF MAN/MACHINE INTERFACE
- EMPHASIZED EQUIPMENT, IGNORED HUMAN BEINGS
- TOLERATED OUTDATED TECHNOLOGY IN CONTROL ROOM

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ROGOVIN COMMITTEE RECOMMENDATION

GREATER APPLICATION OF

HUMAN FACTORS ENGINEERING,

INCLUDING BETTER

INSTRUMENTATION DISPLAY AND

IMPROVED CONTROL ROOM DESIGN

LESSONS LEARNED TASK FORCE

- LICENSEES REVIEW & IMPROVE CONTROL ROOMS USING NRC HUMAN ENGINEERING GUIDELINES

- LICENSEES INSTALL SPDS (CALLED SAFETY STATE VECTOR)

NUREG-0660 : THE NRC ACTION PLAN

TASK I.D. - CONTROL ROOM DESIGN

OBJECTIVE: IMPROVE OPERATOR CAPABILITY TO PREVENT OR TO COPE WITH ACCIDENTS.

TASK I.D.1 - CONTROL ROOM DESIGN REVIEWS

- [illegible]
- ACCEPTANCE CRITERIA

TASK I.D.2 - PLANT SAFETY PARAMETER DISPLAY CONSOLE

- FUNCTIONAL CRITERIA
- ACCEPTANCE CRITERIA

NUREG-0700 GUIDELINES FOR CONTROL ROOM DESIGN REVIEWS

RECOMMENDED PROCEDURES FOR:

- PLANNING
- THE REVIEW PROCESS
- ASSESSMENT AND IMPLEMENTATION
- REPORTING

PLUS — HUMAN ENGINEERING GUIDELINES

INTEGRATION

- EMERGENCY OPERATING PROCEDURES
- SAFETY PARAMETER DISPLAY SYSTEM
- EMERGENCY SUPPORT FACILITIES
- OPERATOR TRAINING
- CONTROL ROOM STAFFING
- IMPROVED INSTRUMENTATION

THE REVIEW PROCESS

- DOES THE CONTROL ROOM PROVIDE THE MEANS FOR EFFECTIVE ACCOMPLISHMENT OF OPERATOR TASKS?

- ARE THERE CHARACTERISTICS THAT CAN DETRACT FROM OPERATOR PERFORMANCE?

THE REVIEW —

FOUNDATION PROCESSES

- OPERATING EXPERIENCE REVIEW
- SYSTEMS FUNCTION REVIEW AND ANALYSIS OF OPERATOR TASKS
- INVENTORY OF CONTROL ROOM INSTRUMENTATION AND EQUIPMENT

THE REVIEW –

<u>INVESTIGATIVE PROCESSES</u>

- CONTROL ROOM SURVEY
- VERIFICATION OF TASK PERFORMANCE CAPABILITY
- VALIDATION OF CONTROL ROOM FUNCTIONS

ASSESSMENT

- ANALYZE AND EVALUATE THE PROBLEMS THAT COULD ARISE FROM THE HUMAN ENGINEERING DISCREPANCIES

- DEVELOP MEANS OF CORRECTING DISCRGPANCIES THAT COULD LEAD TO SUBSTANTIAL PROBLEMS

REPORTS

- TWO STEP PROCESS -

- PROGRAM PLAN

  AT COMPLETION OF
  PLANNING PHASE

- DETAILED CONTROL ROOM
  DESIGN REVIEW REPORT

  AT COMPLETION OF
  DESIGN REVIEW

EVALUATION CRITERIA
NUREG-0801

AID TO NRC STAFF &
LICENSEE IN DETERMINING
THE ADEQUACY OF THE
CONTROL ROOM
UPGRADE

NUREG - 0801

ADEQUACY OF:

- PLAN & TEAM QUALIFICATIONS
- IDENTIFICATION OF DISPLAY & CONTROL NEEDS
- IDENTIFICATION OF HEDs
- ASSESSMENT OF HEDs
- CORRECTIONS

NUREG-0801

- JUSTIFICATION FOR HEDS NOT CORRECTED
- SCHEDULE FOR CORRECTIONS

NUREG-0835

HUMAN FACTORS ACCEPTANCE CRITERIA
FOR THE
SAFETY PARAMETER DISPLAY SYSTEM

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SAFETY PARAMETER DISPLAY SYSTEM

FUNCTION: TO ASSIST CONTROL ROOM OPERATORS IN DETECTION OF ABNORMAL OPERATING CONDITIONS WHICH MAY IMPACT SAFETY.

METHOD: INTEGRATE A MINIMUM SET OF PLANT PARAMETERS INTO A DISPLAY FROM WHICH PLANT SAFETY STATUS MAY BE ASSESSED BY CR OPERATORS

SPDS

DEFINED:	NUREG-0585
	LLTF - FINAL REPORT
FUNCTIONAL CRITERIA;	NUREG-0696
	EMERGENCY RESPONSE FACILITIES
ACCEPTANCE CRITERIA:	NUREG-0835
	HUMAN FACTORS ACCEPTANCE CRITERIA

ISSUED FOR COMMENTS

ACCEPTANCE CRITERIA SCOPE

- RESPONDS TO DESIGN CRITERIA IN NUREG-0696
- LIMITED TO CRT TYPE DISPLAYS
- OTHER TYPE DISPLAYS NOT RULED OUT

GENERAL ACCEPTANCE CRITERIA

- ENHANCED OPERATOR PERFORMANCE IN ASSESSING SAFETY STATUS OF PLANT

- DISPLAY OF ABNORMAL OPERATING CONDITIONS SIGNIFICANT TO SAFETY

DISTINCTLY DIFFERENT FROM

- DISPLAY DEPICTING NORMAL OPERATING CONDITIONS

GENERAL ACCEPTANCE CRITERIA

- DISPLAY PATTERNS

- SCALING OF DISPLAYS

- PARAMETER IDENTIFICATION

- PERCEPTUAL AIDS

    COLOR

    SYMBOLS AND MIMICS

    [illegible]

    SETPOINTS

    BLINKING AND FLASHING

SPECIFIC ACCEPTANCE CRITERIA

ITEMS IN NUREG-0696

- FUNCTIONS
- DATA SET
- DATA VALIDATION
- DISPLAY
- LOCATION AND SIZE
- STAFF
- PROCEDURES
- ALARMS
- DESIGN CRITERIA

STAFF REVIEW PROCESS

- EVALUATE LICENSEE/APPLICANT V & V PROCESS
- AUDIT ELEMENTS OF DESIGN AND TEST
- ONE STEP REVIEW

EST. LICENSEE COSTS

CONTROL ROOM REVIEWS

REVIEW [illegible]	$500K
ENHANCEMENTS - hum. factor no wiring	$100K
RELOCATIONS & ADDITIONS	$300K
	$900K

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C.R. REVIEW	$900K
SPDS	$750K
UPGRADE EOPs	$560K
	$2.210K

APPENDIX XVI
PRESENTATION BY THE COMMITTEE TO REVIEW
GENERIC REQUIREMENTS (CRGR)

EMERGENCY RESPONSE

ACRS SUBCOMMITTEE ON HUMAN FACTORS

JANUARY 5, 1982

COMMITTEE TO REVIEW GENERIC REQUIREMENTS (CRGR)

- CLARIFY REQUIREMENTS
- USE OF NUREGS AND REGULATORY GUIDES
- IMPROVE COORDINATION AMONG ACTIVITIES
- REQUEST DEDROGR STAFF DEVELOP:
  - BASIC REQUIREMENTS
  - IMPLEMENTATION PLAN

A-301

SCOPE

SPDS

CONTROL ROOM IMPROVEMENTS

- DESIGN REVIEW
- R.G. 1.97
- R.G. 1.23

EMERGENCY RESPONSE FACILITIES

- TSC
- EOF
- OSC
- NDL

A-302

MILESTONES

REVIEW BY PROGRAM OFFICES	1/5/82
REVIEW AT CRGR MEETING	1/7/82
CRGR RECOMMENDATION TO EDO	
EDO DECISION	
COMMISSION INFORMATION/ACTION	
IMPLEMENTATION	

PROPOSED BASIC REQUIREMENTS

SPDS

- OPERATOR AID IN CONTROL ROOM
- SEPARATE FROM OTHER ACTIVITIES
- NO SEISMIC OR CLASS 1E QUALIFICATION OR SINGLE FAILURE REQUIREMENTS
- LIMITED NUMBER OF VARIABLES
- NO FUTURE CHANGES REQUIRED

A-303

CONTROL ROOM IMPROVEMENTS

- REVIEW HUMAN ENGINEERING DOCUMENTS
- DESCRIBE MODIFICATIONS
- NO NRC APPROVAL OTHER THAN 10 CFR 50.59
- R.G. 1.97 LIST ONLY WITH EXCEPTIONS; TREAT EQ IN RULEMAKING
- R.G. 1.23 THREE PARAMETERS FOR EMERGENCY RESPONSE

A-304

A-305

EMERGENCY RESPONSE FACILITIES

TSC/EOF/OSC

- INSTRUMENTATION ESSENTIAL TO FUNCTION; SPDS NOT NECESSARY
- LOCATION
- SIZE
- STRUCTURE
- HABITABILITY
- COMMUNICATIONS AND DOCUMENTATION
- STAFFING AND ACTIVATION
- NO SEISMIC OR CLASS 1E QUALIFICATION, NO SINGLE FAILURE CRITERION

NDL - DATA TRANSMISSION TO REMOTE SITE

A-306

IMPLEMENTATION

- NUREGS AND REG. GUIDES ARE NOT REQUIREMENTS
- GENERIC LETTER
- SCHEDULES DEVELOPED INTERACTIVELY
- FLEXIBILITY
- INTEGRATION WITH EOPs

Please delete the following pages A-307 thru A-310 as deletion:

DELETION 1

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

December 16, 1981

MEMORANDUM FOR: Mr. William J. Dircks
Executive Director for Operations

FROM: Mr. Raymond F. Fraley
Executive Director
Advisory Committee on Reactor Safeguards

Subject: OCCUPATIONAL EXPOSURES AT PALO VERDE NUCLEAR GENERATING STATION AND OTHER SYSTEM 80 PLANTS

During its review of the CESSAR-80/Palo Verde Nuclear Generating Station, Units 1, 2, and 3, the ACRS was provided estimates of the annual collective occupational dose associated with the operation of each unit at the Palo Verde Station which may average well over one-thousand person rem. In view of the fact that these units are based on a standard design which supposedly incorporates application of the ALARA principle, the members expected somewhat lower dose estimates.

In this connection, it should be noted that the occupational dose estimates may have been unduly conservative and therefore misleading. The Committee's review did not provide an opportunity to examine the basis for these dose estimates in detail and this should be done to determine if they result from the CESSAR-80 design, the balance of plant design, the proposed method of operation, or other factors.

The Committee urges attention to this matter regarding the Palo Verde Station and the CESSAR-80 standardized plant design. The ACRS Subcommittee on Reactor Radiological Effects would be pleased to discuss this matter further with the NRC Staff.

Raymond F. Fraley

Raymond F. Fraley
Executive Director
ACRS

cc: C. Mark, ACRS
H. Denton, NRR
E. Goodwin, NRR
D. G. Eisenhut, NRR
W. E. Kreger, NRR

A-311

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

December 14, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ACRS REVIEW AND REPORTS ON NRC SAFETY RESEARCH PROGRAMS

Dear Dr. Palladino:.

In our letter of October 20, 1981 we expressed our belief "that reviewing the LRRP would not be an effective use of our time unless a more meaningful plan is developed." Although we anticipate significant improvements in the LRRP, it is perhaps too late to use the new LRRP as a basis for our report to Congress on the FY 1983 program since that report is well under way, and we have not yet received the new plan. Nevertheless, we intend to review the plan and, to the extent needed and practicable, provide you and the Commissioners with our comments. It is likely that our comments this year can be based primarily on the reviews we have carried out in preparation for our report to Congress; extensive interaction with the RES Staff should not be necessary. Nevertheless, we will consider ways in which our review of the FY 1984 Safety Research Program can be carried out in order to provide you with timely and useful comments on the LRRP and, at the same time, provide us with the information and insights we need to prepare our report to the Congress.

With regard to a review and report to the Commission in July on the RES budget request, we said in our letter of October 20, 1981 that we will continue to provide comments on funding levels, in detail or in general, and on specific portions of the program. In doing so, however, we would expect to limit our interaction with the RES Staff; this would be possible if there is an easily identifiable relation between their budget request and the needs and programs described in the LRRP. Moreover, we would not intend to elaborate on the bases for our recommendations if it is possible to relate them to comments made previously in connection with the LRRP and our report to Congress.

We will continue to make both general and specific recommendations to the Commission and to the RES Staff. It would be helpful to us in our continuing review of the Safety Research Program, if RES would respond in writing to each recommendation, general or specific, made in our report to the Congress.

In summary, we believe that procedures can be developed to provide the information requested in your letter of December 10, 1981.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

A-312

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

December 15, 1981

Mr. Jerry D. Griffith, Acting Director
Office of Nuclear Power Systems
Office of Nuclear Energy
Department of Energy
Washington, DC 20545

SUBJECT: ACRS REPORT ON THE FINAL DRAFT OF THE DEPARTMENT OF ENERGY'S RESPONSE TO PUBLIC LAW 96-567

Dear Mr. Griffith:

During its 260th meeting, December 10-12, 1981, the Advisory Committee on Reactor Safeguards reviewed the final draft of the Department of Energy (DOE) response to Public Law 96-567, "Nuclear Safety Research, Development, and Demonstration Act of 1980." A meetin- of ACRS Working Groups was held in Washington, D.C. on December 9, 19 1 to consider this matter. During its review, the Committee had the benefit of discussions with representatives of DOE and the NRC Staff. Our general comments on the final draft response to Congress appear below.

I. Assessment of the Need for and Feasibility of Establishing a National Reactor Engineering Simulator Facility

We believe that our comments, as contained in our September 16, 1981 report on the first draft of the DOE response to Public Law 96-567, have been adequately considered by DOE. While we agree that a national simulator facility is not justified, we believe that a cohesive national light water reactor systems simulation program should be considered by the Simulation Working Group organized by DOE.

We recommend that the Simulation Working Group define early in its deliberations the uses which it believes to be appropriate for simulation. The results of this effort should be available before much is done toward the development of a program.

II. A Study of the Desirability and Feasibility of Creating a Federal Nuclear Operations Corps

Although we believe that our comments relating to the desirability and feasibility of creating a Federal Nuclear Operations Corps have been adequately considered by DOE, and although we concur with the conclusion that such a Corps is not needed, we want to offer several comments.

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The current draft of the report states that the "Nation's academic and nonacademic institutions, outside the nuclear industry, have a large and expandable capability in place to provide training in nuclear fundamentals and to augment specific training by utilities." While this may be true, we believe it is important to recognize that this capability, particularly in terms of graduate education in nuclear engineering and radiation protection, has been declining in recent years. There is no assurance that the necessary resources and students will be available to enable the existing training capability to be fully utilized. Similarly, we believe it is overly optimistic to state that the "Institute of Nuclear Power Operations (INPO) has outlined an overall plan for an industry-wide program to provide adequately trained personnel to perform operational and supervisory functions." While commendable, the INPO plan, unless modified, appears to us to be capable of providing only a portion of the total number of people that will be required. We believe the report should acknowledge these deficiencies as well as the need to take action to correct them.

III. Program Management Plan for the Conduct of a Research, Development, and Demonstration Program for Improving the Safety of Nuclear Power Plants

The Program Management Plan is unchanged from that in the first draft of the report. We continue to believe that it constitutes an appropriate and potentially successful approach to the development and execution of a research, development, and demonstration program for improving the safety of nuclear power plants.

Such a program has not been developed. However, Working Groups with representation from industry, NRC, and DOE have been established in the several areas addressed by the Act. These Working Groups have been meeting to identify issues and plan to develop National Programs in each area and recommend measures for their implementation. We believe that this approach is an acceptable way to develop meaningful programs with appropriate participation by the various organizations.

We wish to be kept informed of the efforts of DOE and its various Working Groups related to the implementation of Public Law 96-567.

Please let us know if we can be of further assistance.

Sincerely,

J Carson Mark

J. Carson Mark
Chairman

A-314

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

December 15, 1981

MEMORANDUM FOR: W. J. Dircks, Executive Director for Operations

FROM: R. F. Fraley, Executive Director, ACRS

R F Fraley

SUBJECT: ACRS RECOMMENDATIONS REGARDING PROPOSED REGULATORY GUIDE ON "QUALIFICATION AND ACCEPTANCE TESTS FOR SNUBBERS USED IN SYSTEMS IMPORTANT TO SAFETY" (TASK NO. SC 708-4)

During its 260th meeting, December 10-12, 1981, the ACRS considered the recommendations of its Subcommittee on Regulatory Activities regarding this proposed Regulatory Guide and agreed to defer further consideration and action until it has been reviewed by the Committee to Review Generic Requirements (CRGR). The following comments are reasons in part for deferring consideration pending review by the CRGR.

The objective of this Guide, to improve the quality and performance of snubbers, is commendable, and the requirement for functional requirements, qualification tests, and acceptance tests will help achieve that objective. However, the costs in terms of applicant or licensee resources have not been evaluated quantitatively or comprehensively. Nor have the benefits, especially those relating to the health and safety of the public, been evaluated adequately.

As a consequence, the ACRS does not believe that the proposed requirements have been justified on a cost-benefit basis or in relation to other improvements that could be made to reduce risk.

The ACRS believes that this proposed action falls clearly within the charter of the CRGR and that evaluation by that Committee based on the information and criteria required by its charter will provide a more adequate basis for a decision by the ACRS regarding the need for this Guide.

cc: V. Stello, EDO
T. Murley, EDO
H. Denton, NRR
E. Goodwin, NRR
R. Minogue, RES
W. Morrison, RES
W. Anderson, RES

A-315

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

December 15, 1981

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REPORT ON THE PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3

Dear Dr. Palladino:

During its 260th meeting, December 10-12, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Arizona Public Service Company, the Salt River Project Agricultural Improvement and Power District, the El Paso Electric Company, the Public Service Company of New Mexico, and the Southern California Edison Company (Applicants) for a license to operate the Palo Verde Nuclear Generating Station Units 1, 2, and 3. The joint applicants have designated the Arizona Public Service Company as the Project Manager and Operating Agent with full authority to construct and operate the power station. The project was considered at a Subcommittee meeting in Phoenix, Arizona on November 23-24, 1981, and members of the Committee toured the facility on November 23, 1981. In its review the Committee had the benefit of discussions with representatives of the Arizona Public Service Company, Combustion Engineering, Inc., Bechtel Power Corporation, the NRC Staff, and members of the public. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for the Palo Verde Nuclear Generating Station Units 1, 2, and 3 in a report dated November 12, 1975 to the NRC Chairman.

The Palo Verde application is submitted in accordance with the Commission's regulations as described in Appendix O to Part 50, "Licensing of Production and Utilization Facilities," and Section 2.110 of Part 2, "Rules of Practice," of Title 10 of the Code of Federal Regulations. NRC policy stated in the Federal Register (42 FR 34395 and 43 FR 38954) allows for a reference system that involves an entire facility design or major fraction of a design outside the context of a license application. For this application the reference system is the Combustion Engineering standard nuclear steam supply system known as its Standard Reference System 80. This design has been reviewed by the ACRS and discussed in its report dated December 15, 1981, "Final Design Approval for Combustion Engineering, Inc. Standard Nuclear Steam Supply System (Standard Reference System 80)".

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This power station is located in a sparsely populated section of Maricopa County, Arizona, about 36 miles west of the nearest boundary of Phoenix, Arizona. The nearest densely populated center is Sun City, Arizona, about 35 miles east-northeast of the site, which had a 1980 population of about 57,800 persons. Palo Verde is the first commercial nuclear power station to be operated by Arizona Public Service Company and the first in the state of Arizona.

The Palo Verde Nuclear Generating Station uses three System 80 pressurized water nuclear steam supply systems designed by Combustion Engineering, Inc. Each of these has a design core power output of 3800 MWt. The turbine generators are oriented so as to minimize plant damage should turbine failure occur. The containment is a steel-lined, prestressed concrete cylindrical structure with a hemispherical dome and a design pressure of 60 psig. The cooling tower makeup is supplied from treated sewage effluent from the city of Phoenix.

The Committee's review included consideration of the management organization and capability, and the operator training program. The organizational plan for technical support of the operating plant is still being formulated. The Committee notes that the Arizona Public Service Company management personnel have extensive experience in both commercial and other nuclear plant operation and construction. The utility anticipates using many of its installation surveillance staff members as part of the technical support team. The ACRS encourages this organizational arrangement, but believes the Applicant should promptly analyze the skill requirements needed to support operations and make certain that the necessary capabilities will be available when needed. In order that the Committee be kept informed, we request an update on the organizational arrangement in about one year from this date.

The Committee notes that Arizona Public Service Company has a training simulator in operation at the Palo Verde site. The Committee's review indicated that the training program is being developed and that use of the plant simulator is still in the process of being integrated into the program. The Committee recommends that Arizona Public Service Company examine industry-sponsored programs concerning effective use of simulators for training and make certain that its approach takes account of current understanding of simulator training limitations.

Discussion with the Arizona Public Service Company staff indicated that emergency operating procedures for dealing with off-normal plant behavior are incomplete. Development of such procedures should be expedited to provide maximum time to make use of them in the operational training program.

In the Palo Verde design the primary system does not include capability for rapid, direct depressurization when the plant has been shut down. This places extra importance on the reliability of the auxiliary feedwater system and makes it necessary that the NRC Staff and the Applicant assure the availability and dependability of this system for a wide variety of transients. It also places extra requirements on the continued integrity of the two steam generators as the only method of heat removal immediately after shutdown. The ACRS recommends that the NRC Staff and the Arizo Public Service Company give additional attention to the matter of shutdown heat removal for Palo Verde and develop a detailed evaluation and justification for the position judged to be acceptable. The Committee wishes to be kept informed.

Arizona Public Service Company should expand its studies on systems interactions and systems reliability.

A number of items have been identified as Outstanding Issues, Confirmatory Issues, and proposed License Conditions in the NRC Staff's Safety Evaluation Report dated November 1981. The ACRS is satisfied with the progress on these topics and believes that they should be resolved in a manner satisfactory to the NRC Staff.

Our approval of the operation of this plant is contingent upon the satisfactory completion of construction and preoperational testing. For this reason, we request that, prior to fuel loading on Unit 1, a report be provided to the Committee describing significant construction deficiencies and their disposition, effectiveness of the quality assurance program, and results of the preoperational test program. In addition, a review of the startup experience on Unit 1 should be made prior to fuel loading on Unit 2 and the Committee kept informed.

We believe that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that Palo Verde Nuclear Generating Station Units 1, 2, and 3 can each be operated at power levels up to the design core power output of 3800 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS member M. Bender and ACRS members H. W. Lewis and M. S. Plesset are presented below.

Sincerely yours,

J Carson Mark

J. Carson Mark
Chairman

Additional Comments by ACRS Member M. Bender

The NRC requirements for instrumentation to follow the course of an accident have been generally outlined in Regulatory Guide 1.97. The ACRS has concentrated most of its attention on instrumentation to detect inadequate core cooling, sometimes called pressure vessel coolant level measuring instrumentation. The Regulatory Guide 1.97 requirements and the emphasis on measurement of vessel coolant levels both seem to have confused the real accident diagnosis requirements.

The proposed coolant level indicators could only have value under quiescent conditions. The proposed devices, differential pressure indicators and heated junction thermocouples, require considerable information about hydraulic conditions, pressure distribution, and density variations in the primary coolant circuit to be useful for unambiguous interpretation of changing coolant inventory in the reactor core. A full understanding of mass and energy distribution and related physical behavior of the nuc'ear system would be needed to make such information diagnostically useful under most accident conditions. The main value would appear to be for conditions where the system has been depressurized and the coolant state is known, for example, prior to refueling. Such knowledge does not appear relevant to the circumstances of primary concern such as accident conditions comparable to the TMI-2 event.

Regulatory Guide 1.97 has a mixture of requirements, some directed to pre-accident symptom identification, some to actual surveillance of rapidly changing transients, and some to surveillance of accident recuperation conditions. Although all of these requirements could be justified under some circumstances, it is likely that, if everything listed in the guide were provided, the operators could be overwhelmed by the informational detail and their diagnostic capability actually impaired.

At a time when unambiguous accident diagnostic information is urgently needed, a maze of indicating and analytical devices that might confuse the operators hardly makes sense. I propose the following criteria as a basis for determining accident diagnostics adequacy.

1. Does the operator have a well-defined set of signals to guide his emergency response to important accidents?

2. Do the emergency procedures enable the operator to avoid misinterpretation of those signals under circumstances where accident diagnosis is needed in conjunction with emergency actions?

3. In accident recovery is the sensor capability adequate to enable the operators to establish whether a stable and safe operating condition is being maintained until the system can be brought to cold shutdown and reliable decay heat removal functions assured?

4. If fuel failures occur, is there capability to determine whether the failures are of minor or major significance (clad reaction

with water and fuel melting); whether bulk quantities of radioactive nuclides have been released to the primary coolant circuitry, the containment interior, or are leaking from containment; and whether the containment boundary is jeopardized by overpressure or overtemperature?

Only a few additions to the pre-TMI-accident instrumentation appear necessary to address these considerations. However, to be certain that necessary information is available, the actions required of operators during accidents must be thoroughly examined. Emergency procedure guidance is now being developed by the nuclear steam supply equipment vendors. This guidance must be converted into usable procedures that may be testable on nuclear plant simulators. Palo Verde and a few other installations have simulators that might be used for this purpose. Those operating organizations having appropriate simulation equipment should give priority attention to proving the effectiveness of the diagnostic equipment in conjunction with proposed emergency procedures in order to verify diagnostic adequacy. No serious effort in this direction appears to have been initiated up to this time.

Additional Comments by ACRS Members H. W. Lewis and M. S. Plesset

We do not wish to belabor the points we made in our addendum to the ACRS letter dated November 17, 1981 on the St. Lucie Plant Unit 2, but they are as relevant here as there. The Staff continues to accept instruments that do not provide an unambiguous measure of liquid level in the pressure vessel, and continues to lack an adequate rationale therefor. We do not find fault with the Applicants for their efforts to be responsive to the Staff, but are concerned about the proliferation of inadequately considered requirements, of which this is only one example. To sanctify an ambiguous indication of core water level is to play with fire. In this particular case (heated thermocouples in a separator tube), not only dynamic effects, but a pressure vessel full of high-void-fraction water will spoof the instrument, and tend to lull the operator into a false sense of security about the coolant inventory. In that specific case, the instrument will indicate that the vessel is nearly full.

None of the above is meant to suggest that we oppose the provision of instrumentation to follow the course of an accident or to detect the onset of inadequate core cooling - unambiguous diagnosis of accident conditions through improved instrumentation and training is a high priority. Our concern is a piecemeal and incoherent approach to the problem, as exemplified here.

References:

1. Arizona Public Service Company, "Palo Verde Nuclear Generating Station, Final Safety Analysis Report," with Amendments 1 through 6.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Palo Verde Nuclear Generating Station, Units 1, 2, and 3," NUREG-0850, dated November 1981.

3. Combustion Engineering, Inc., "System 80 CESSAR FSAR," with Amendments 1 through 5.
4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," NUREG-0852, dated November 1981.

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

December 15, 1981

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REPORT ON FINAL DESIGN APPROVAL FOR COMBUSTION ENGINEERING, INC. STANDARD NUCLEAR STEAM SUPPLY SYSTEM (STANDARD REFERENCE SYSTEM 80)

Dear Dr. Palladino:

During its 260th meeting, December 10-12, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of Combustion Engineering, Inc. for final design approval for its Standard Reference System 80 described in CESSAR. A Subcommittee meeting was held with representatives of the Applicant and the NRC Staff in Windsor, Connecticut on November 19, 1981. The Committee also had the benefit of the documents listed. The Committee's report on the preliminary design approval for this standard nuclear steam supply system (NSSS) was provided in a letter to the NRC Chairman dated September 17, 1975.

The System 80 design consists of a reactor system with a design rated core output of 3800 MWt and includes the reactor coolant system, reactor protection system, engineered safety features actuation system, chemical and volume control system, shutdown cooling system, safety injection system, and fuel handling system. The System 80 design provides safety-related interface requirements information essential to the design of the balance of plant. Combustion Engineering provides, at the option of the user, certain other nonstandard safety-related systems and services which are outside the scope of the System 80 design. Such systems will need to be dealt with in each user's Safety Analysis Report. The regulations governing the review of standard plant designs under the "reference system" option described in the Federal Register (42 FR 34395 and 43 FR 38954) are contained in paragraph 2.110 of 10 CFR Part 2 and Appendix O to 10 CFR Part 50.

CESSAR provides information required to ensure that the balance of plant is designed to protect the System 80 from site-related hazards. It envelops all plant sites approved to date for Combustion Engineering nuclear steam supply systems. When the System 80 design is applied, the related site must be evaluated to establish its acceptability within the System 80 envelope. For multiple reactor units at a single site, the reference design requires that each important safety-related item be separately provided for each reactor unit. The first plant using the System 80 design will be Palo Verde Nuclear Generating Station, Units 1, 2, and 3, of which Unit 1 is scheduled to load fuel during November 1982.

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Because the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related requirements have been met, the NRC must review these matters with the utility-applicants on a case-by-case basis. The ACRS believes that Combustion Engineering should be required to evaluate the adequacy of the implementation of interface requirements, including such items as the influence of plant control system performance and reliability on NSSS integrity and function.

In recent years, the availability of reliable shutdown heat removal capability for a wide range of transients has been recognized to be of great importance to safety. The System 80 design does not include capability for rapid, direct depressurization of the primary system or for any method of heat removal immediately after shutdown which does not require use of the steam generators. In the present design, the steam generators must be operated for heat removal after shutdown when the primary system is at high pressure and temperature. This places extra importance on the reliability of the auxiliary feedwater system used in connection with System 80 steam generators and extra requirements on the integrity of the steam generators. The ACRS believes that special attention should be given to these matters in connection with any plant employing the System 80 design. The Committee also believes that it may be useful to give consideration to the potential for adding valves of a size to facilitate rapid depressurization of the System 80 primary coolant system to allow more direct methods of decay heat removal. The Committee wishes to review this matter further with the cooperation of Combustion Engineering and the NRC Staff.

System 80 employs some new design features for the steam generators, the core outlet flow region, control rod guidance and shrouding, and the core support structure. These appear to be acceptable, but, because they are new features, they should be monitored during early operation to determine if they perform as expected.

A number of items have been identified as Outstanding Issues and Confirmatory Issues. These include some TMI-2 Action Plan requirements. Progress on these matters is satisfactory, and we believe these issues can be resolved in an acceptable manner. The Committee wishes to be kept informed.

The manner of applying preliminary and final design approvals of the type proposed for System 80 will not be completely defined until System 80 has been used for several licensing actions at both the construction permit and operating license stages. The Committee believes that standard designs such as System 80 can be useful in assuring acceptably safe plants. However, a policy to establish when and how changes will be permitted to new or previously licensed plants is needed.

The Committee believes that, subject to the above comments and approval of the balance-of-plant designs, the System 80 design can be incorporated into nuclear power plants that can be operated without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Combustion Engineering, Inc., "System 80 CESSAR FSAR," with Amendments 1 through 5.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," NUREG-0852, dated November 1981.

Please delete the following pages A-325 thru A-326 as deletion:

DELETION 1

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

November 17, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON COMANCHE PEAK STEAM ELECTRIC STATION UNITS 1 AND 2

Dear Dr. Palladino:

During its 259th meeting, November 12-14, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Texas Utilities Generating Company (TUGCO), Dallas Power and Light Company, Texas Electric Service Company, Texas Power and Light Company, Texas Municipal Power Agency, Brazos Electric Power Cooperative, Inc. and Tex-La Electric Cooperative for a license to operate the Comanche Peak Steam Electric Station Units 1 and 2. The Units are to be operated by the Texas Utilities Generating Company. A Subcommittee meeting was held in the Dallas/Fort Worth area on June 29, 1981 to consider this project. A tour of the facility was made by Subcommittee members on June 29, 1981. An additional Subcommittee meeting was held in Washington, D.C. on November 11, 1981. During its review, the Committee had the benefit of discussions with representatives of the Applicant and the NRC Staff. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for this station in its report dated October 18, 1974 to AEC Chairman Dixie Lee Ray.

The Comanche Peak Station is located in Somerville County in North Central Texas about 40 miles southwest of Fort Worth, Texas, the nearest city having a population in excess of 25,000 persons.

Each Comanche Peak Unit is equipped with a Westinghouse pressurized water reactor having a rated power level of 3425 MWt. Each unit is housed in a steel-lined, reinforced concrete, dry containment building with a design pressure of 50 psig.

The Reactor Protection System will use N-16 gamma radiation detectors to provide a signal for reactor trip. Because this system has not been proven in commercial applications, we recommend that the NRC Staff closely follow its implementation and operation. The Committee wishes to be kept informed.

This is the first commercial nuclear power plant to be operated by TUGCO and the first in the state of Texas. The Committee's review included consideration of the management organization and capability and the operator training

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program. The training program is well planned and comprehensive, and includes simulator training at other facilities. We were favorably impressed with the training program, general competence, and responsive attitude of the utility's operating organization. Nevertheless, there is a significant lack of hands-on experience with large commercial nuclear power plants that will only be corrected by the operation of the Comanche Peak Plant. The NRC Staff is requiring the utility to strengthen its own organization with on-shift personnel having experience with large commercial PWR operations until suitable experience has been developed by the operating staff. We endorse the NRC Staff requirement but recommend that attainment of 100% rated power should not be the only consideration in determining that operational proficiency has been achieved.

The Committee also recommends that the operating organization establish a list of technological matters which may have to be faced in future operation of the nuclear plant and identify sources of skilled personnel and expertise that ought to be available to address these matters when needed. The Committee wishes to be kept informed.

The Station Operations Review Committee, the Independent Safety Engineering Group, and the Operations Review Group should include personnel from outside the operating organization who are experienced in the operational management of large PWRs and related technology as well as other independent advisors with mature judgment about public safety matters.

TUGCO should expand its studies on systems interaction and probabilistic assessment so that it will have a better understanding of the Comanche Peak nuclear systems.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation report supplement dated October 1981. The ACRS is satisfied with the progress on these topics and believes that they should be resolved in a manner satisfactory to the NRC Staff.

TUGCO is evaluating potential methods of providing instrumentation for detection of inadequate core cooling as discussed in the ACRS letter to the Executive Director for Operations dated June 9, 1981. The Committee believes that this equipment should not be installed until it is well established that the instruments will provide reliable information of significant value beyond that provided by the instrumentation which is already installed.

We believe that if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing and preoperational testing, there is reasonable assurance that Comanche Peak Steam

Electric Station Units 1 and 2 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Sincerely yours,,

J. Carson Mark

J. Carson Mark
Chairman

References:
1. "Final Safety Analysis Report for the Comanche Peak Steam Electric Station Units 1 and 2," including Amendments 1 through 23.
2. U.S. Nuclear Regulatory Commission "Safety Evaluation Report related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," USNRC Report NUREG-0797, dated July 1981 and Supplement No. 1 dated October 1981.
3. Letter from Citizens for Fair Utility Regulation To S. Duraiswamy, ACRS, regarding the licensing of Comanche Peak, dated July 18, 1981

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

November 17, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE CALLAWAY PLANT UNIT NO. 1

Dear Dr. Palladino:

During its 259th meeting, November 12-14, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Union Electric Company (the Applicant) for a license to operate the Callaway Plant Unit No. 1. A tour of the facility was made by members of the Subcommittee on November 4, 1981, and a Subcommittee meeting was held in Columbia, Missouri on November 4 and 5, 1981. During its review, the Committee had the benefit of discussions with representatives of the Nuclear Regulatory Commission (NRC) Staff and with representatives and consultants of the Applicant, Westinghouse Electric Corporation, and Bechtel Power Corporation. The Committee also had the benefit of the documents listed below. The Committee commented on the construction permit application for this plant in its report dated September 17, 1975 to NRC Chairman William A. Anders.

The Callaway Plant application was one of four submitted in response to the Commission's standardization policy as described in Appendix N to Part 50 of Title 10 of the Code of Federal Regulations. This option allows for a simultaneous review of the safety-related parameters of a limited number of duplicate plants which may be constructed within a limited time span at a multiplicity of sites. The five utilities that originally joined together designated their common design the "Standardized Nuclear Unit Power Plant System" (SNUPPS). At the present time, in addition to the Callaway Plant Unit No. 1, only the Wolf Creek Generating Station remains an active SNUPPS project.

The Callaway Plant is located in a rural section of Missouri about 80 miles west of St. Louis. The site is approximately 5 miles north of, and about 325 feet above the flood plain of, the Missouri River. The nearest population center is Jefferson City (estimated 1980 population about 34,000), which is 25 miles west-southwest of the Plant.

The Plant will use a Westinghouse, four-loop, pressurized water reactor, nuclear steam supply system having a rated power level of 3425 MWt. The Plant employs a cylindrical, steel-lined, reinforced, post-tensioned concrete containment structure with a free volume of 2.5 million cubic feet. The containment design pressure is 60 psig.

The Callaway Plant will be the first commercial nuclear power plant in the state of Missouri, and is the first nuclear power plant to be operated by the Union Electric Company. The Committee reviewed the Applicant's management organization, experience, and training program. We were favorably impressed by the general competence and attitude of the Applicant's personnel, but we believe their commercial nuclear experience is less than desirable. The NRC Staff is requiring the utility to augment its own organization with on-shift personnel having experience with large commercial PWR operations until suitable experience has been developed by the operating staff. We endorse the NRC Staff requirement but recommend that attainment of 100% rated power should not be the only consideration in demonstrating operational proficiency. We also recommend that a highly competent, senior individual with considerable professional experience on commercial PWRs be assigned to assist the Plant Superintendent as an advisor through at least the first year of full power operation.

The Committee recommends that the operating organization establish a list of technological matters which may have to be faced in future operation of the nuclear plant and identify sources of skilled personnel and expertise that ought to be available to address these matters when needed. The Committee wishes to be kept informed.

The Onsite Review Committee, Nuclear Safety Review Board, and Independent Safety Engineering Group should include personnel from outside the operating organization who are experienced in the operational management of large PWRs and related technology as well as other independent advisors with mature judgment about public safety matters.

During our review, it was noted that Shift Technical Advisor training in the areas of Plant Systems and especially Transient/Accident Analysis appears marginal. It is recommended that the NRC Staff evaluate this matter and apply the results to those nuclear plants where they are generically applicable.

Discussion with the Applicant indicated that emergency operating procedures for dealing with off-normal plant behavior are incomplete. However, the Applicant is endeavoring to develop such procedures utilizing new and promising approaches, and we encourage such efforts. The Committee wishes to be kept informed.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the NRC Staff's Safety Evaluation Report dated October 1981; these include some TMI Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff and recommend that this be done.

The Committee believes that, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Callaway

Plant Unit No. 1 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Additional comments by Dr. M. W. Carbon, ACRS Member, are presented below.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

Additional Comments by Dr. M. W. Carbon, ACRS Member

It is my belief that the NRC Staff's requirement for experienced, on-shift personnel during the early operation of the plant is inadequate. I therefore recommend that a licensed Senior Reactor Operator (SRO), who has been previously licensed as an SRO on another Westinghouse PWR, be available on each shift in an advisory capacity through the first year of full-power operation. I also believe that the advisor to the Plant Superintendent should have an educational background at least equal to a Bachelor of Science degree in engineering or a related discipline.

References:

1. "Final Safety Analysis Report for Standardized Nuclear Unit Power Plant System," including Revisions 1 through 7.

2. "Final Safety Analysis Report for Standardized Nuclear Unit Power Plant System, Callaway Plant Units No. 1 and 2 Addendum," including Revisions 1 through 4.

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1," NUREG-0830, dated October 1981.

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

November 17, 1981

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON ST. LUCIE PLANT UNIT NO. 2

Dear Dr. Palladino:

During its 259th meeting, November 12-14, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of the Florida Power and Light Company (the Applicant) for authorization to operate the St. Lucie Plant Unit No. 2. The project was considered at a Subcommittee meeting in West Palm Beach, Florida on October 30-31, 1981 and members of the Committee toured the facility on October 30, 1981. In its review the Committee had the benefit of discussions with representatives of the Applicant, Combustion Engineering, Inc., Ebasco Services, Inc., the NRC Staff, and members of the public. The Committee also had the benefit of the documents listed. The Committee commented on the construction permit application for St. Lucie Plant Unit No. 2 in a report dated December 12, 1974 to AEC Chairman Dixie Lee Ray.

St. Lucie Plant Unit No. 2 is located on Hutchinson Island adjacent to Unit No. 1, which went into commercial operation in December 1976. Both units use Combustion Engineering nuclear steam supply systems with a rated core power of 2560 MWt. The two units are nearly identical.

A number of items have been identified as Outstanding Issues, Confirmatory Issues, and License Conditions in the NRC Staff's Safety Evaluation Report dated October 1981. These include some TMI-2 Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff. We also recommend resolution of concerns on instrumentation for detection of inadequate core cooling expressed in the ACRS letter to the Executive Director for Operations dated June 9, 1981.

Discussion with the Florida Power and Light Company Staff indicated that emergency operating procedures for dealing with off-normal plant behavior that might develop during the operation of St. Lucie Plant Unit No. 2 are incomplete. We recommend that a concentrated effort be made by the Florida Power and Light Company staff to complete emergency operating procedures which take advantage of new information and approaches developed during the past two years. This matter should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

At the time this site was initially approved, the population density was relatively low, and the projected increase during the life of the plant was not unusually large. Since that time, the growth in population has been much more rapid than predicted, and current estimates predict continued growth at relatively high rates. Although the present population and that predicted for the next several years are not a cause for concern, it now seems possible that the population density in portions of the surrounding area could reach a level, during the lifetime of the St. Lucie Plant, that might then warrant additional measures. We recommend that the Applicant and the NRC Staff periodically review the actual and projected population growth. If required as a result of these reviews, plans for appropriate preventive or remedial measures could then be made in a considered but timely manner.

We recommend that the Staff give due regard to the special nature of this site in evaluating the final emergency plan.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the St. Lucie Plant Unit No. 2 can be operated at core power levels up to 2560 MWt without undue risk to the health and safety of the public.

Additional comments by Members H. W. Lewis and M. S. Plesset are presented below.

Sincerely yours,

J. Carson Mark

J. Carson Mark
Chairman

Additional Comments by Members H. W. Lewis and M. S. Plesset

In the aftermath of the accident at Three Mile Island Unit 2, which dramatically emphasized the importance of instrumentation to follow the course of an accident, the NRC Staff has required applicants for an Operating License to demonstrate specific capability to detect the onset of inadequate core cooling. For PWRs this has come to mean in practice the provision, inter alia, of an instrument which can be called a water-level indicator for the pressure vessel. (Although the NRC Action Plan allows for alternatives, none appear to have been seriously contemplated.) A number of such devices have been accepted and/or proposed, some of which measure differential pressure, some average void fraction in a part of the pressure vessel, some cooling rate at a number of places in the vessel. All can give spurious response because of dynamic effects.

Many of these views have been previously expressed in the Committee letter of June 9, 1981.

We are concerned that, in the commendable eagerness to avoid a repetition of TMI, the NRC Staff is requiring ill-defined instrumentation without any clear picture of the contribution of that instrumentation to the prevention or mitigation of accidents - considerations which must necessarily be scenario dependent. If it were really true that core water level were the important parameter, then differential pressure indicators would appear to be preferable, provided the coolant is quiescent. If instead cooling capacity is important, then some form of heated wire or thermocouple would appear to be preferable. Since either may be acceptable, we are left with the inference that the NRC Staff has not really clarified the role of this instrumentation.

We believe that, before, not after requiring these instruments for all the new plants, the NRC Staff should develop a position regarding their utility. This position, which should be based upon accident analysis and risk assessment, would lead to a much clearer understanding of just what instrumentation, if any, is needed.

REFERENCES:

1. Florida Power and Light Company, "St. Lucie Plant, Unit No. 2 Final Safety Analysis Report," with Amendments 1 through 6.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of St. Lucie Plant, Unit No. 2," Docket No. 50-389, USNRC Report NUREG-0843, dated October 1981.
3. Letter from Betty Lou Wells to the Chairman of the Advisory Committee on Reactor Safeguards, dated October 28, 1981.
4. Written statement by Joette Lorian, Research Director for the Center for Nuclear Responsibility.

Please delete the following pages A.336 thru ___________ as deletion:

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SECRETARIAT RECORD COPY

DEC 2 3 1981

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: STAFF COMMENTS ON THE ACRS REPORT ON ST. LUCIE UNIT NO. 2

You asked for the staff to comment on ACRS concerns with inadequate core cooling instruments. The concerns were stated in the November 17, 1981 ACRS letter, including the "additional comments" by members H. W. Lewis and M. S. Plesset. The staff developed a response for inclusion in the St. Lucie Unit 2 Supplemental Safety Evaluation Report. We have incorporated that SER input into our enclosed response to your request.

Since that time, the December 15 ACRS letter on Palo Verde contained similar concerns. ACRS member Mike Bender suggested in his "additional comments" some criteria for determining accident diagnostics adequacy. Addressing these criteria may help the staff and ACRS come to resolution of their differences of opinion. We will keep you informed of progress along these lines.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosure:
Response to ACRS Concerns -
Instrumentation Requirements for Detection of ICC

cc: Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OPE
OGC
SECY

Contact: L. E. Phillips, DSI:CPB
X-29472

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J. Ledbetter
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EGoodwin
12/16/81

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DATE	12/14/81	12/14/81	12/14/81	12/14/81	12/17/81	12/ /81	12/ /81

A337

OFFICIAL RECORD COPY

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: STAFF COMMENTS ON THE ACRS REPORT ON ST. LUCIE UNIT NO. 2

The staff has previously prepared a response to the ACRS concerns stated in the November 17, 1981 ACRS letter, including the "additional comments" by members H. W. Lewis and M. S. Plesset. That response was provided for inclusion in the St. Lucie Unit 2 Supplemental Safety Evaluation Report. We have incorporated that SER input into our enclosed response to your request regarding this matter.

William J. Dircks
Executive Director for Operations

Enclosure:
Response to ACRS concerns - Instrumentation Requirements for Detection of ICC

cc: Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts
OPE
OGC
SECY

Contact: L. E. Phillips, DSI:CPB
X-29472

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USGPO: 1981—335-960

RESPONSE TO ACRS CONCERNS

INSTRUMENTATION REQUIREMENTS FOR DETECTION OF

INADEQUATE CORE COOLING

Introduction

The NRC staff requirements for implementation of TMI Action Plan Item II.F.2, "Additional Instrumentation for Detection of Inadequate Core Cooling," have been discussed in various ACRS and ACRS subcommittee meetings (including meetings for review of TMI-1 restart and for review of licensing for several plants). The NRC staff and contractors, vendors of proposed instrumentation systems, applicants and licensees, and other interested organizations such as EPRI have provided testimony at these meetings. Subsequently, the ACRS has expressed concerns regarding staff positions and efforts to implement this action plan item.

It has been clear since initiation of this action plan item that many applicants and licensees have strong reservations concerning the need for and value of the additional instrumentation. Their resistance has been strengthened by schedule requirements which necessitated selection and ordering of the instrumentation systems prior to completion of the final design and testing and, hence, prior to NRC approval of the available systems. Their question of need has been based on the conviction that their existing instrumentation systems provide adequate protection against design scenarios. Little cognizance has been given to the fact that the intended purpose for this instrumentation is to provide in-depth protection against unpredictable scenarios involving multiple failures. Testimony against the requirement has consistently cited problems revealed by early testing of available systems without recognition that those problems have been long since resolved during the design and development evolution of proposed instrumentation systems.

The staff believes that the ACRS concerns primarily reflect the expressed views of applicants and licensees. The staff has tried but apparently failed to communicate to ACRS the progress of this program during its development and implementation. In addition, no recent effort has been made to debate or justify the need for the instrumentation, which the staff believes was resolved at the outset during establishment of Item II.F.2 in the Action Plan.

Our response to the ACRS Report on St. Lucie and plans for resolution of the ACRS concerns follow.

A339

Resolution of Comments in ACRS Report on St. Lucie

The ACRS, in their letter of November 17, 1981 to Chairman Palladino, "Report on St. Lucie Plant Unit 2," recommended resolution of concerns on instrumentation for detection of inadequate core cooling previously expressed in the ACRS letter to the Executive Director for Operations dated June 9, 1981.

The staff provided an initial response to the ACRS concerns in the memorandum for Chairman J. Carson Mark from William J. Dircks, Executive Director for Operations, dated July 10, 1981. These concerns may be summarized as follow:

(1) Installation schedule including relation to the schedule for development, testing, evaluation, and qualification of reactor vessel level monitoring instrumentation;

(2) utility of the information provided by the ICC monitoring system as determined by accident analyses and development of emergency procedures to deal with various specific accident scenarios; and

(3) possible consequences of misleading information to the operator due to dynamic effects on level monitoring instrumentation as a result of improper attention to the first two items in an over eager response to TMI.

With respect to the first item, the staff has provided details of the implementation status for vessel level monitoring instrumentation, including the HJTC system proposed for St. Lucie 2, in the Commission paper SECY 81-582 dated October 7, 1981. That paper addressed the testing programs to be completed in advance of staff approval of these systems and describes our review program and review status to support our preliminary conclusions concerning the prospects for acceptability of the systems. The paper also recommended that the staff be given permission to delay the January 1, 1982 installation requirement on a case by case basis as warranted by the equipment development and procurement and installation constraints. Based on the status reported in the paper, the staff expects that this would result in installation of the systems on most plants (possible exceptions are B&W vintage PWRs) by the first refueling outage after January 1, 1983.

By letter dated November 16, 1981 the Commission has approved our scheduling recommendation for Westinghouse and CE designed reactors committed to the Westinghouse or CE vessel level monitoring systems. We believe that this schedule relief in conjunction with the information provided on the equipment development, testing and evaluation program is responsive to and resolves the concerns of Item (1) above, to the extent possible prior to completion of our generic review.

A-340

With respect to the second item, as explained in the July 10 memorandum to Chairman Mark, evaluation of specific water level monitoring systems has included analyses of specific accident scenarios (e.g., Westinghouse NOTRUMP code analyses of small break LOCA (WCAP-9753) events and the response of their dp system under ICC conditions; CE analyses of small break LOCA scenarios (CEN-117); analyses to predict instrument test performance under simulated small break LOCA conditions, etc.).

However, these scenarios are not all inclusive. In accordance with TMI-2 lessons learned recommendations and current (post-TMI-2) practice on all emergency procedures, guidelines and procedures are to be symptom oriented. All process signals indicative of ICC conditions (saturation margin, coolant inventory, coolant or fuel temperature, etc.) are useful to confirm the need for emergency operator actions. The staff has offered to meet with the ACRS to inform them of progress in our generic review of the level monitoring systems and in the development of associated guidelines for emergency procedures for detection and recovery from a condition of inadequate core cooling. Presently, that meeting is expected to take place in February 1982. The staff expects that this meeting will provide a basis for resolution of the second concern. In any case, emergency procedures relating to vessel water level instrumentation are not required to be in place prior to issuance of an operating license for St. Lucie 2.

With respect to the third item of concern, the staff agrees that the best available ICC monitoring systems (Westinghouse and CE) are not perfect. We are attempting to identify deficiencies, including any which are related to dynamic effects, and believe that once identified, they can be neutralized by design of the data processing and display systems coupled with proper operating instructions. This concern can be resolved by thorough testing and design evaluation. The staff will keep the ACRS informed on the results of our review.

Response to Additional Comments - ACRS Report On St. Lucie Plant Unit 2

Members Lewis and Plesset of ACRS have expressed the opinion that, before, not after requiring specific instrumentation to detect the onset of ICC, the NRC staff should develop a position regarding their utility. The position should be based upon accident analysis and risk assessment.

To place these comments in perspective, it should be remembered that the requirement is a product of the TMI Lessons Learned Task Force short term recommendations in NUREG-0578 (July 1979) and subsequent related documents and was supported by most, if not all, of the organizations reviewing that aspect of the TMI accident (especially the ACRS which insisted upon the water level instrumentation). The need for the instrumentation was thoroughly examined in many ACRS and Commission meetings. There is no general disagreement today on its utility. The original implementation date for this additional instrumentation was January 1, 1981. This recommendation and concurrent recommendations and studies relating to accident analyses and emergency procedure development called for a symptom oriented approach and provision of diverse information to the operator as the basis for operator actions and to monitor plant status. On this basis, the staff believes that saturation margin, coolant level and inventory, and fuel cladding temperature (as inferred by coolant superheat) are all important parameters for evaluation of core cooling adequacy, and that there is no unique emergency procedure governing which parameter should be used to initiate operator actions and what parameters should be used to verify the effectiveness of these actions and to monitor the course of the event.

Since establishment of the II.F.2 requirement, staff effort has been directed to development of acceptable instrumentation and the earliest feasible implementation of an ICC monitoring system which is consistent with objectives of the lessons learned recommendations. While a rigorous mathematical risk assessment evaluation was not a part of the decision process in developing this requirement, it seems clear that the cost of installation (both dollars and dose) is greater where back fits are required (more backfits are required when the definition of specific instrumentation is delayed), and that the benefits derived from risk reduction after installation of the system are a function of the useful life remaining for those operating reactors which must be backfitted. Therefore, there are advantages to be derived from the earliest feasible implementation consistent with development of acceptable systems, and that has been the staff goal since the lessons learned need for the system was established.

In view of the advanced status of implementation of these systems and the earlier decision making process described above, the staff does not believe that consideration of multiple additional accident scenarios (aside from the development of guidelines for emergency procedures) or risk assessment analyses would be useful at this time.

Please delete the following pages A-343 thru __________ as deletion:

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APPENDIX XX

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, E. F. Goodwin to R. F. Fraley, Revised Proposed NRR Agenda Items for the February, March and April 1982 ACRS Meeting, Jan. 6, 1982

2. Memorandum, M. L. Ernst to R. L. Tedesco, Systems Interaction, Jan. 5, 1982

3. Handouts in support of RES Presentation by D. Ross and R. B. Minogue regarding the ACRS report to the U.S. Congress regarding the proposed NRC Safety Research Program for FY-83, January 7-8, 1982

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

January 9, 1982

APPENDIX XXI
LTR, J. J. Ray, Acting Chairman, ACRS to Admiral Rickover, dated January 9, 1982

Admiral H. G. Rickover
Naval Sea Systems Command
Navy Department
Washington, DC 20362

Dear Admiral Rickover:

The Advisory Committee on Reactor Safeguards has been informed of your forthcoming retirement later this month from active duty with the United States Navy. The Committee has long recognized your dedication to quality of design and construction of naval reactors and the selection, training and motivation of well-qualified operating crews. Your concern for quality and safety has been exemplary and has resulted in an outstanding performance record for the naval reactors program.

Your long and distinguished service in developing the "nuclear Navy" and application of related technology to the commercial nuclear power program placed you in a position where you could provide valuable insight regarding nuclear plant design, construction, and operation. The Committee will miss productive discussions with you concerning reactor design and operation and their impact on safety.

We wish to take this opportunity to congratulate you on your distinguished career and accomplishments. Your work in the nuclear reactor field is truly noteworthy. The Committee wishes both to take this opportunity to recognize your accomplishments and to wish you well in your future activites.

Sincerely,

J. J. Ray

J. J. Ray, Acting
ACRS Chairman