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

July 29, 1981

SCHEDULE AND OUTLINE FOR DISCUSSION
256TH ACRS MEETING
AUGUST 6-8, 1981
WASHINGTON, DC

Thursday, August 6, 1981, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M.

Opening Session (Open)

1.1) Chairman's Report (CM/RFF)

1.1-1) Commission action re ACRS budget/staffing request for FY 1983

1.1-2) Status of new ACRS member

1.1-3) Status of new NRC Commissioner

(Portions of this session may be closed as necessary to discuss information of a personal nature the release of which would represent an unwarranted invasion of personal privacy.)

2) 8:45 A.M. - 12:30 P.M.

Enrico Fermi Atomic Power Plant Unit 2 (Open)

2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee (WK/PAB)

2.2) 9:15 A.M.-12:30 P.M.: Meeting with NRC Staff and Applicant

(Portions of this session will be closed as necessary to discuss information of a Proprietary nature and safeguards information specifically exempted from disclosure.)

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

3) 1:30 P.M. - 3:00 P.M.

Enrico Fermi Atomic Power Plant Unit 2 (Open)

3.1) 1:30 P.M.-3:00 P.M.: Meeting with NRC Staff and Applicant

(Portions of this session will be closed as necessary to discuss information of a Proprietary nature and safeguards information specifically exempted from disclosure.)

4) 3:00 P.M. - 8:00 P.M. <u>Waterford Steam Electric Station Unit 3 (Open)</u>

4.1) <u>3:00 P.M.-3:30 P.M.:</u> Report of ACRS <u>Subcommittee (DAW/GRQ/DB)</u>

4.2) <u>3:30 P.M.-8:00 P.M.:</u> Meeting with <u>NRC Staff and Applicant</u>

(Portions of this session will be closed as necessary to discuss information of a Proprietary nature and safeguards information specifically exempted from disclosure.)

Friday, August 7, 1981, Room 1046, 1717 H Street, NW, Washington, DC

5) 8:30 A.M. - 12:30 P.M.

Susquehanna Steam Electric Station Units 1 and 2 (Open)

5.1) 8:30 A.M. - 9:00 A.M.: Report of ACRS Subcommittee on Susquehanna Plant (WK/GY) and Mk II Type Containment Long-Term Program (MSP/PAB)

5.2) 9:00 A.M.-12:30 P.M.: Meeting with NRC Staff and Applicant

(Portions of this session will be closed as necessary to discuss information of a Proprietary nature and safeguards information specifically exempted from disclosure.)

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

6) 1:30 P.M. - 3:30 P.M.

Susquehanna Steam Electric Station Units 1 and 2 (Open)

6.1) Meeting with NRC Staff and Applicant (Portions of this session will be closed as necessary to discuss information of a Proprietary nature and safeguards information specifically exempted from disclosure.)

7) 3:30 P.M. - 4:00 P.M.

Prepare for Meeting with NRC Commissioners (Open)

7.1) Discuss topics for meeting with NRC Chairman and Commissioners

7.1-1) ACRS Review of Contract Review Panel Recommendations regarding program to evaluate alternate materials for waste disposal containers - clarification of Commission request

7.1-2) ACRS Report on Proposed NRC Safety Research Program Budget for FY 1983 - respond to Commissioners' questions

7.1-3) NRC Staff implementation of NRC Action Plan Items - discuss lack of a considered basis for decision making

7.1-4) ACRS budget/staffing requests for FY 1983 - basis for Commission action

7.1-5) Status of Appointment of New ACRS member - discuss status and basis for selection

(Portions of this session will be closed as required to discuss information of a personal nature the release of which would represent a clearly unwarranted invasion of personal privacy.)

8) 4:00 P.M. - 5:00 P.M. Meeting with NRC Chairman and Commissioners (Open)

8.1) Introduction of new Chairman/Commissioner

8.2) Discuss items noted above

(Portions of this session will be closed as required to discuss information of a personal nature the release of which would represent a clearly unwarranted invasion of personal privacy.)

9) 5:00 P.M. - 6:00 P.M. Environmental and Seismic Qualification of Electrical Equipment Important to Safety (10 CFR 50.49) (Open)

9.1) 5:00 P.M.-5:15 P.M.: Report of ACRS Subcommitte (WK/RS)

9.2) 5:15 P.M.-6:00 P.M.: Discussion with NRC Staff and representatives of the nuclear industry who may be present

10) 6:00 P.M. - 6:30 P.M. Future ACRS Schedule (Open)

10.1) Discuss anticipated ACRS Subcommittee activity

10.2) Discuss anticipated full Committee activity

11) 6:30 P.M. - 7:00 P.M. Development of Quantitative Safety Goals (Open)

11.1) Report of ACRS Subcommittee regarding status of activities regarding quantitative safety goals (DO/GRQ)

12) 7:00 P.M. - 8:00 P.M. Discuss Proposed ACRS Reports to NRC (Open)

12.1) Discuss proposed reports to NRC regarding

12.1-1) Enrico Fermi Station Unit 2

12.1-2) Waterford Station Unit 3

Saturday, August 8, 1981, Room 1046, 1717 H Street, NW, Washington, DC

13) 8:30 A.M. - 12:30 P.M. Discuss Proposed ACRS Reports to NRC (Open):

13.1) Susquehanna Station Units 1 and 2, and Mk II Containment Long Term Program
13.2) Enrico Fermi Station Unit 2
13.3) Waterford Station Unit 3

(Portions of this session will be closed as necessary to discuss Proprietary Information related to the matters being discussed and safeguards information specifically exempted from disclosure.)

12:30 P.M. - 1:3 P.M. LUNCH

14) 1:30 P.M. - 3:30 P.M. Concluding Session (Open)

14.1) Discuss proposed ACRS comments regarding 10 CFR 50.49, Environmental and Seismic Qualification of Electrical Equipment Important to Safety
14.2) Complete preparation of ACRS reports noted above

(Portions of this session will be closed as necessary to discuss Proprietary Information related to the matters being discussed and safeguards information specifically exempted from disclosure.)

Order).[1] NUREG-0737 was transmitted to each licensee and applicant by an NRC letter from my office dated October 31, 1980, which is hereby incorporated by reference. In that letter, it was indicated that although the NRC staff expected each requirement to be implemented in accordance with the schedule set forth in NUREG-0737, the staff would consider licensee requests for relief from staff proposed requirements and their associated implementation dates.

III

The licensee's submittals dated December 22, 1980, and February 20, 1981, and the references stated therein, which are incorporation herein by reference, committed to complete each of the actions specified in the Attachment. The licensee's submittals included a modified schedule for submittal of certain information. The staff has reviewed the licensee's submittal and determined that the licensee's modified schedule is acceptable based on the following:

The licensee's schedule for submittal of information in some instances does not meet the staff's specified submittal dates. Most of the information requested by the staff describes how the licensee is meeting the guidance of NUREG-0737. Therefore, this deferral of the licensee submittal will not alter the implementation of plant modifications. Therefore, plant safety is not affected by this modification in schedule for the submittal of information.

I have determined that these commitments are required in the interest of public health and safety, and therefore, should be confirmed by Order.

IV

Accordingly, pursuant to Sections 103, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, it is hereby ordered effective immediately that the licensee shall comply with the following conditions:

The licensee shall satisfy the specific requirements described in the Attachment to this Order (as appropriate to the licensee's facilities) as early as practicable but no later than 30 days after the effective date of the Order.

Any person who has an interest affected by this Order may request a hearing within 20 days of the date of publication of this Order in the Federal Register. Any request for a hearing shall be addressed to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy shall also be sent to the Executive Legal Director at the same address. If a hearing is requested by a person other than the licensee, that person shall be describe, in accordance with 10 CFR 2.714(a)(2), the nature of the person's interest and the manner in which the interest is affected by this Order. A request for hearing shall not stay the immediate effectiveness of this Order.

If a hearing is requested by the licensee or other persons who have an interest affected by this Order, the Commission will issue an Order designating the time and place of any such hearing.

If a hearing is held concerning this Order, the issue to be considered at the hearing shall be whether, on the basis of the information set forth in Sections II and III of this Order, the licensee should comply with the conditions set forth in Section IV of this Order.

This request for information was approved by OMB under clearance number 3150-0065 which expires June 30, 1983. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C.

Dated at Bethesda, Maryland this 10th day of July 1981.

This Order is effective upon issuance.

For the Nuclear Regulatory Commission.

Darrell G. Eisenhut,

Director, Division of Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. 81-21613 Filed 7-22-81; 8:45 am]

BILLING CODE 7590-01-M

[1] Attachment: NUREG-0737 Requirements, available in NRC Public Document Room.

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 August 6-8, 1981, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published in the Federal Register on June 17 and July 22, 1981.

The agenda for the subject meeting will be as follows:

Thursday, August 6, 1981

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 including selection and appointment of a new ACRS member and NRC action regarding ACRS budget and staffing requests for FY 1983.

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

8:45 a.m.-12:30 p.m. and 1:30 p.m.-3:00 p.m.: Enrico Fermi Atomic Power Plant Unit 2 (Open)

The Committee will hear the report of its Subcommittee and consultants who may be present regarding proposed operation of this unit. The Committee will also hear and discuss reports from members of the NRC Staff and representatives of the Applicant concerning this matter.

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

3:00 p.m.-8:00 p.m.: Waterford Steam Electric Station Unit 3 (Open)

The Committee will hear the report of its Subcommittee and consultants who may be present regarding proposed operation of this unit. The Committee will also hear and discuss reports from members of the NRC Staff and representatives of the Applicant concerning this matter.

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

Friday, August 7, 1981.

8:30 a.m.-12:30 p.m. and 1:30 p.m.-3:00 p.m.: Susquehanna Steam Electric Station Units 1 and 2 (Open)

The Committee will hear the report of its Subcommittees and consultants who may be present regarding proposed operation of this unit and the program for resolution of problems regarding the Mark II type containment used in this type plant. The Committee will also hear and discuss reports from members of the NRC Staff and representatives of the Applicant concerning this matter.

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

3:30 p.m.-4:00 p.m.: Discuss Items for Meeting with NRC Chairman and Commissioners (Open/Closed)

The members will consider those items to be discussed with the NRC Chairman and other Commissioners who may have an interest. Topics will include the status of the appointment of a new ACRS member, clarification of the Commission request for ACRS review of the proposed NRC program to evaluate alternate materials for radwaste disposal containers, and to respond to questions the Commissioners may have regarding the ACRS report (NUREG-0795) on the proposed NRC Safety Research Program budget for FY 1983.

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

4:00 p.m.–5:00 p.m.: Meeting with NRC Chairman and Other Commissioners (Open/Closed)

The Committee will meet with the NRC Chairman and other Commissioners who may have an interest to discuss items noted above.

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

5:00 p.m.—6:00 p.m.: Proposed NRC Regulation (10 CFR 50.49) on Environmental and Seismic Qualification of Electrical Equipment Important to Safety (Open)

The Committee will hear and discuss the report of its Subcommittee and consultants who may be present regarding this proposed NRC rule. Representative of the NRC staff and the nuclear industry will participate as appropriate.

6:00 p.m.—6:30 p.m.: Quantitative Safety Goals (Open)

The Committee will hear and discuss the report of its Subcommittee on Probabilistic Risk Assessment regarding the development and use of quantitative safety goals in the regulation of nuclear facilities.

6:30 p.m.—8:00 p.m.: Preparation of ACRS Reports (Open)

The committee will discuss proposed ACRS reports regarding projects considered during this meeting.

Saturday, August 8, 1981

8:30 a.m.—12:30 p.m. and 1:30 p.m.—3:30 p.m.: Preparation of ACRS Reports (Open)

The Committee members will discuss proposed ACRS comments/reports regarding matters considered during this meeting. The Committee will also discuss its proposed schedule for future activities.

Portions of this session will be closed as necessary to permit discussion of Proprietary Information.

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

I have determined in accordance with Subsection 10(d) Pub. L. 92-463 that it is necessary to close portions of this meeting as noted above to discuss Proprietary Information (5 U.S.C. 552b(c)(4)), information of a personal nature where disclosure would constitute a clearly unwarranted invasion of personal privacy (5 U.S.C. 552(c)(6).

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

Dated: July 20, 1981.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 81-21817 Filed 7-22-81; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50-443, 50-444]

Public Service Co. of New Hampshire, et al., (Seabrook Station, Units 1 and 2); Issuance of Director's Decision (DD-81-14)

On February 11, 1980, the Director of Nuclear Reactor Regulation denied under 10 CFR 2.206 a petition filed by the Seacoast Anti-Pollution League (SAPL). *See* DD-80-6, 11 NRC 371 (1980). While the Director's decision was pending before the Commission for possible review, SAPL filed a letter on June 30, 1980, before the Commission in support of its petition. Although the Commission declined to review the Director's decision, the Commission referred SAPL's June 30th letter to the Director for consideration as a separate petition under 10 CFR 2.206. SAPL's June 30th letter essentially reiterates its earlier request for institution of proceedings to suspend or revoke the permits issued to Public Service Company of New Hampshire for construction of the Seabrook Station. SAPL's June 30th letter raises issues concerning emergency planning and evacuation as the basis of its request. The Commonwealth of Massachusetts filed a memorandum in support of SAPL's request on March 13, 1981.

Upon consideration of the information provided by SAPL and the Commonwealth of Massachusetts, I have determined not to institute the requested proceeding. The reasons for this decision are set forth in a "Director's Decision Under 10 CFR 2.206", which is available for inspection in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C. 20555, and in the local public document room at the Exeter Public Library, Front Street, Exeter, New Hampshire 03883. A copy of the decision will also be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c).

Dated at Bethesda, Md., this 15th day of July 1981.

For the Nuclear Regulatory Commission.

Harold R. Denton,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 81-21814 Filed 7-22-81; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-312]

Sacramento Municipal Utility District, (Rancho Seco Nuclear Generating Station); Order Confirming Licensee Commitments on Post-TMI Related Issues

I. Sacramento Municipal Utility District (the licensee) is the holder of Facility Operating License No. DPR-54, which authorizes the operation of the Rancho Seco Nuclear Generating Station (the facility) at steady-state power levels not in excess of 2772 megawatts thermal. The facility is a pressurized water reactor (PWR) located at the licensee's site in Sacramento County, California.

II. Following the accident at Three Mile Island Unit No. 2 (TMI-2) on March 28, 1979, the Nuclear Regulatory Commission (NRC) staff developed a number of proposed requirements to be implemented on operating reactors and on plants under construction. These requirements include Operational Safety, Siting and Design, and Emergency Preparedness and are intended to provide substantial additional protection in the operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. The staff's proposed requirements and schedule for implementation are set

Issue Date:
April 14, 1982

MINUTES OF THE
256TH ACRS MEETING
AUGUST 6-8, 1981
WASHINGTON, DC

CERTIFIED

The 256th Meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N.W., Washington, DC, was convened by Chairman C. Mark at 8:30 a.m., Thursday, August 6, 1981.

[Note: For a list of attendees, see Appendix I. D. W. Moeller was not present Friday and Saturday; J. C. Ebersole, H. W. Lewis, and D. Okrent were not present on Saturday.

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 oral or written 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. Status of Appointment of New ACRS Member

The Chairman informed the Committee that Commissioner Gilinsky wished to interview the proposed candidates for membership on the Committee, and that the Commission would not vote on the matter until after these interviews have taken place.

B. New Member on NRC Commission

The Chairman informed the Committee that Thomas M. Roberts has been sworn in as the fifth member of the Commission. He also noted that Commissioner Bradford has indicated that he plans to leave the Commission soon.

C. Nuclear Safety Oversight Committee Comments

The Chairman noted that a letter has been promulgated from the Nuclear Safety Oversight Committee commenting on the ACRS role in the regulatory process. He recommended that the Committee should set up a mechanism to address this matter.

II. Meeting on Enrico Fermi Atomic Power Plant Unit 2 (Operating License) (Open to Public)

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

A. Subcommittee Report

W. Kerr, Subcommittee Chairman, noted that the Fermi site was visited on July 15, and the Subcommittee met on July 24, 1981 with both the Applicant and members of the NRC Staff (see Appendix IV).

B. Status of NRC Staff Review

L. Kintner, NRC Staff, identified the open issues to be completed in the supplementary Safety Evaluation Report that will be issued approximately August 31, 1981, the items that will be completed prior to the anticipated issuance of an operating license in November 1982, and those which will be completed subsequent to the issuance of an operating license and will appear as license conditions (see Appendix V).

H. Etherington noted that in the reactor heat removal system heat exchanger, coolant passes from water to steam to water in the shell. He requested that the NRC Staff analyze these conditions for potential water hammer, and report to the Committee some time in the future.

D. Okrent questioned the Staff's not reviewing this plant for potential problems from internal flooding from all causes.

C. Applicant's Presentations

1. Organization and Management

W. H. Jens, Detroit Edison (DE), discussed the organization and management for the operation of Fermi 2 (see Appendix VI).

2. Site and Plant Description

L. E. Schuerman, DE, described the plant site and location, provided the major design parameters, and noted major design changes that resulted from the resolution of generic issues (see Appendix VII).

P. G. Shewmon asked whether the NRC Staff has identified the cause of the rise in oxygen content in the water in recirculation lines in other BWR plants, and further, whether they have been able to identify the exact causes of stress corrosion cracking in the recirculation lines.

P. Matthews, NRC Staff, said that the NRC Staff has been unable to identify specific assignable causes for the problem, but that there could have been a number of contributors to the problem ranging from a burst associated with a scram, mal-operation of the cleanup system, or a condenser leak. Copper in the lines contributes to the failure problems. He added that the condenser and feedwater heater tubes are made of admiralty metal and that is believed to be the source of the copper.

W. Colbert, DE, corrected the record by noting that only the condenser tubes are admiralty metal; the steam generator tubes are not.

P. Matthews noted that these matters are being discussed with the Applicant, and that the copper content in both the feedwater and the reactor water will be monitored.

E. Leonard, DE, said that both copper content and oxygen content of the feedwater will be monitored.

3. Operator and Maintenance Personnel Selection and Training

L. E. Kanous, DE, discussed the selection procedures and training programs planned for both operators and maintenance personnel (see Appendix VIII).

4. Control Room Design

E. Lusis, DE, discussed the design of the control room for Fermi 2, and the continuing review of that design (see Appendix IX).

5. Instrumentation to Follow the Course of an Accident

L. F. Wooden, DE, discussed the instrumentation to follow the course of a serious accident, including the post accident sampling system, the means for measuring reactor vessel coolant levels, noble gas effluent monitors, iodine and particulate monitors, high range containment radiation monitors, suppression pool level monitors, dry well pressure monitors, containment hydrogen and oxygen monitors, safety and relief valve position monitors, and dry well sump level monitors, and also emergency procedures (see Appendix X).

H. W. Lewis noted his concern that dynamic conditions may give false readings of static pressure detectors used for level indication. He noted that thermocouples are neither useful nor necessary for level measurement.

M. Bender questioned the capability of the applicant to be able to sample environmental conditions following certain unexpected serious events.

D. Okrent recommended that accident sequences beyond design basis accidents should be evaluated. He also recommended that the NRC Staff should note the EPRI study on this matter.

L. Phillips, NRC Staff, noted another use for incore thermocouples that was identified in I&E Circular, 81-11, July 24, 1981, and deals with unplanned repressurizations during shutdown heat removal. He cited an incident at Dresden 3 where operators had reduced the shutdown coolant flow until insufficient vessel flow existed to provide mixing of the primary coolant and accurate temperature measurements in the recirculation pump and shutdown coolant pump could not be achieved. Because these were the only temperatures that were monitored, a slow heatup and repressurization of the reactor vessel occurred over six hours. He said that this event points out that, as time goes on, events arise where a completely instrumented system is of use.

6. GE Design Criteria for BWR Safety

In answer to questions raised during the subcommittee meeting, L. Phillips discussed the generic issue, Design Criteria for BWR Stability (see Appendix XI).

7. Plant Seismic Design

F. E. Gregor, DE, discussed the Fermi Plant seismic design, including the supplementary seismic evaluation to which the Applicant committed following the NRC Staff request of March 12, 1981, a comparison of Fermi site dependent response spectra with Lawrence Livermore and Weston Geophysical Eastern U.S. rock spectra for magnitude 5.3 earthquakes, a comparison of Fermi 2 site dependent response spectra with existing Fermi 2 design spectra, a comparison of Regulatory Guide 1.60 and the average time history spectra, and the conclusions drawn following the reevaluation (see Appendix XII).

In answer to a question regarding whether failure of non-seismic-qualified equipment could compromise seismic-qualified equipment, R. Tedesco, NRC Staff, said that the Staff has not performed a systems interactions study.

8. Reliability of Station Electrical Power

M. Featnem, DE, discussed the Fermi 2 offsite power sources, the transmission lines to the plant, the 260/130 volt d.c. distribution system, and concluded that a complete loss of d.c. power at Fermi 2 is not a credible event (see Appendix XIII). He noted that in addition to the usual redundant power sources, Fermi 2 also has four gas turbines on site, each of which can handle the inhouse load, and one of which has black-start capability.

R. Tedesco noted that the NRC Staff has not made a study of the reliability of a 3-train d.c. system. However, Task Action Plan A-44, will give consideration to such a system. Most plants today have a 2-train system. Members of the NRC Staff believe that a 3-train system is not necessarily better than a 2-train system. There are many aspects that must be considered.

9. Internal Flooding and Recurrence of SSE

(This subject was presented in answer to questions raised at previous meetings; no handouts were provided.)

F. E. Gregor noted that the Applicant's seismic consultants provided the following information regarding the recurrence period for the SSE at the Fermi site:

- If one considers the Cincinnati-Findlay arch structure as controlling, the estimated recurrence period is 2000 years.
- If the central province is considered alone, excluding the Anna, Ohio earthquake, the recurrence period is calculated to be 12,000 years.
- If the central province is considered including the Anna Ohio earthquake, the recurrence period calculates to approximately 5000 years.

With respect to internal flooding, F. E. Gregor said that both high energy and low energy pipe breaks were evaluated, and that for high energy pipe breaks, the worst case was the 20-inch feedwater pipe breaking in the steam tunnel. Assuming a full circumferential break, flooding is confined to the steam tunnel and the first floor of the auxiliary building. Maximum flooding in the auxiliary building is two feet, at which point blowout panels open, and water flows harmlessly into the turbine building. Flooding from low energy pipes would be even less severe.

In answer to a question, W. Lefevre, NRC Staff, said that internal flooding potential for this plant has been reviewed by the NRC Staff.

10 Mark I Containment Modifications

D. F. Lehnert, DE, discussed the required modifications to the Mark I Containment System in Fermi 2 (see Appendix XIV). His verbatim remarks are contained in the Appendix.

11. ATWS

E. P. Griffing, DE, discussed the training of Fermi operators with respect to the postulated ATWS events (see Appendix XV). His verbatim remarks are contained in the Appendix.

J. R. Green discussed the modifications made to the scram discharge volume and compared the current configuration with that at Browns Ferry 3 (see Appendix XVI).

12. Emergency Plans

E. Madsen, DE, discussed the emergency plans for the Fermi 2 plant (see Appendix XVII). She discussed the geography, and demography of the plant area, and noted that the areas considered to be affected by a postulated accident lie in Michigan, Ohio, and Ontario. She noted that in Michigan, emergency services are handled by State Police, who have authority to contact Canadian officials, Ohio officials, and the Coast Guard in the event of an emergency. Further, the U. S. Coast Guard has agreements with the Canada Coast Guard. It was also noted, during these discussions, that the security personnel at Fermi 2 are all Detroit Edison employees.

An additional handout on hydrogen control (see Appendix XVIII) was provided to Members, but was not discussed at the meeting.

D. Caucus

The Members indicated that they believed that they could write a favorable report on the Fermi 2 plant at this meeting.

III. Meeting on the Mark II Containment System (Open to Public)

[John C. McKinley was the Designated Federal Employee for this portion of the meeting.]

[Note: The Susquehanna Steam Electric Station was considered to be the lead plant for the Mark II containment system. However, the Committee reviewed this GE containment system separately from its review of the Susquehanna plant.]

A. Fluid Dynamics Subcommittee Report

M. S. Plesset, Subcommittee Chairman, noted that the Committee's consideration of the Mark II containment was originally planned to take place in conjunction with the review of the Shoreham plant, but since the review of Shoreham has been delayed, the Mark II containment review is being carried on simultaneously with the Susquehanna review. He said that a long period of experimental and analytical work has resolved the major safety issues for the Mark II containment. The T-quencher was developed to reduce overall loads from the safety relief valves. Fatigue questions have been resolved. The installation of vacuum breakers should reduce the means for bypass of the pressure suppression system and reduce chugging.

J. C. Ebersole noted his opinion that additional effort should be taken to remove all means for possible bypass of the pressure suppression system.

Members raised the question regarding the lack of calculations made on the effects of steam line rupture above the wet well under conditions of no steam condensation.

C. Anderson, NRC Staff, said that some of these calculations are currently being made. He said he believes that the NRC Staff has sufficient information now to proceed with licensing of the Mark II system. (For background material on the Mark II containment system studies, see Appendix XIX.)

B. Status of NRC Staff Review

C. Anderson reviewed the significant milestones in the Mark II Pool Dynamic Loads programs from the original formation of the Mark II Owners Group in 1975 through the scheduled release of NUREG-0808, Mark II Long Term Program Loads, completing the Unresolved Safety Issue A-8 Program; reviewed the experimental results of the program, both for the lead-plant program and the long-term program; provided the summary of pool swell, steam condensation, and chugging loads, identified lateral load specifications for both the lead-plant and the long-term; defined the top dynamic load; discussed the bases for the single vent load; discussed the NRC conclusions regarding the single-vent load specification; and identified the single-vent lateral load conservatisms (see Appendix XX).

C. Mark II Owners Group Presentations

1. Introduction

H. Chow, representing the Mark II Owners Group, introduced the presentations by briefly reviewing the studies conducted on questions raised regarding potential safety questions with respect to the Mark II system.

2. Chugging Fatigue Considerations

J. Kitts, Mark II Owners Group, discussed the fatigue evaluation of safety relief valve and downcomer lines in the wet well, the basis for fatigue evaluation, evaluation techniques, the frequency distribution of pressure amplitudes, assumptions regarding stress cycles, the determination of equivalent stress cycles, load combinations, and he also provided a summary of the results of the study (see Appendix XXI).

3. Vacuum Breaker Operability

M. Carne, Mark II Owners Group, discussed wet well to dry well vacuum breaker cycling during chugging, including an accident scenario, the vacuum breaker function, vacuum breaker design and location, plant operations for both normal and accident conditions, vacuum breaker qualification, and conclusions derived from the study, (see Appendix XXII).

IV. Meeting on Susquehanna Steam Electric Station Units 1 and 2 (Operating License) (Open to Public)

[Note: John C. McKinley was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

W. Kerr, Subcommittee Chairman, reported on the Subcommittee meeting held on July 23, 1981, (see Appendix XXIII). He noted that this is the first nuclear plant constructed or operated by Pennsylvania Power and Light (PP&L) Co. He added that the ACRS consultants are satisfied with the progress made to date by the Applicant with regard to both staffing and plant construction. He added that the design of the Susquehanna plants is similar to that at LaSalle. He noted that the staff of this company includes a large number of people with a good nuclear background. He noted that, prior to the TMI-2 accident, the company had ordered a simulator for training its personnel.

J. J. Ray noted his opinion that the Applicant has organized for the operation of these nuclear units in a very orderly fashion. He noted that he was impressed by both the control room layout and the design of the equipment in the control room that will be used by the operators.

B. Applicant's Presentations

1. Introduction

N. Curtis, PP&L, described the plant and site, the demography of the area, transportation routes near the site, meteorology, the ultimate heat sink, surface and groundwater considerations, geology of the area, and the impact of the Susquehanna plants on both the company and the users (see Appendix XXIV).

2. Mark II Containment System

D. Roth, PP&L, discussed the basic design of the Mark II containment system used for the Susquehanna plants, and noted the modifications made to the original design resulting from the studies made by the Mark II Owners Group (see Appendix XXV).

3. Management Structure and Management Resources

B. D. Kenyon, PP&L, discussed the organization of the nuclear department of PP&L and the background, training, and experience of the personnel involved (see Appendix XXVI). He noted that the Applicant plans to run the two units as a single facility.

4. Plant Staffing

H. Keiser, PP&L, discussed the organization, responsibility, and the staffing of the plants, noting that employment numbered 395 by the end of June, and will reach 479 by the end of 1981 (see Appendix XXVII).

5. Training Program

H. Keiser discussed the training programs for personnel at Susquehanna Units 1 and 2 (see Appendix XXVIII).

6. Philosophy of Training Functions and Structure

G. Ward, PP&L, discussed the philosophy of the training programs, the functions and structure of the programs, and provided a general overview of the programs (see Appendix XXIX).

7. Control Room Design

F. H. Cantone, PP&L, discussed the overall control room design, the studies and analyses that lead to this design, and the human factors assessments (see Appendix XXX).

8. Station Blackout

H. Keiser discussed the potential and consquences of station blackout, noting the onsite electrical distribution, the issues raised in the NRC Generic Letter 81-04, a summary of the blackout event, a simulated blackout test, the redundant systems designed to prevent the event, the scenarios involved in such an event, operator responses, and an evaluation of the test (see Appendix XXXI).

9. Decay Heat Removal

H. Keiser discussed the decay heat removal systems for use during cooldown and hot standby, the degraded mode of decay heat removal when feedwater and the normal heat sink are not available, both when HPCI and RCIC are both available and unavailable (see Appendix XXXII).

(For material provided to the Committee, but not used in the discussions, see Appendix XXXIII.)

C. Status of NRC Staff Review

R. Stark, NRC Staff, identified the open issues regarding this license application, and noted the proposed schedule for resolution (see Appendix XXXIV).

J. Hannon, NRC Staff, discussed an analysis of a postulated scram discharge system pipe failure (see Appendix XXXV).

D. Caucus

The Committee indicated that they believed they could write a favorable report on the Susquehanna 1 and 2 plants at this meeting.

V. Meeting on Waterford Steam Electric Station Unit 3 (Operating License) (Open to Public)

[Note: David Bassette was the Designated Federal Employee for this portion of the meeting.]

[Note: The minutes for this portion of the meeting were prepared by Stuart Beal.]

A. Subcommittee Report

D. A. Ward, Subcommittee Chairman, noted that two subcommittee meetings have been held to review the license application for the Waterford Steam Electric Station Unit 3, a meeting in New Orleans, LA on June 18-19, 1981 and another in Washington, DC on August 5, 1981. He noted that the plant has several unique features, namely

- The containment building, auxiliary building, and fuel handling building are all on a single foundation, forming a nuclear island. The plant is built this way because of the soil characteristics of the area.

- The ultimate heat sink, along with the diesel generators, is entirely contained on the nuclear island. This makes emergency decay heat removal independent of the rest of the site.

- The plant has an advanced control room design, employing computers and modern displays.

- The Applicant is relying heavily on contractors and the Subcommittee believes they need to be more aggressive in establishing a staff capable of operating and maintaining the plants.

- The operating license is being contested by two local intervenor groups.

- The SER was published on July 9, and there are about 16 open items (aside from TMI items) still to be resolved.

D. A. Ward asked if any of the other subcommittee members had any comments. J. Ebersole noted that the plant does not have a power operated relief valve (PORV) and is relying exclusively on the auxiliary feedwater system to remove decay heat. C. Mark noted that in this respect it was similar to San Onofre and ANO-2. D. Okrent asked if the subcommittee had reviewed the plant internal security. D. A. Ward replied that they had and that the plant seemed to be well protected. (For background information, see Appendix XXXVI, for intervenor contentions, see Appendix XXXVII; for consultants' reports, see Appendix XXXVIII.)

B. Status of Staff Review

S. Black, the NRC Project Manager, gave a presentation covering the open items still remaining. She said that in her opinion the only significant one was the licensee's qualifications. D. Okrent queried the adequacy of the Applicant's response to the issue of turbine missiles. The Staff and the contractor (EBASCO) said the matter is still being pursued. D. Okrent questioned the safety factor associated with liquefaction of the foundation soils in the event of an earthquake. The Staff offered their response to this question but later in the meeting replied that the liquefaction does not represent a significant threat at this site because the sandy soil is deep and located between layers of clay.

C. Applicant's Presentation

L. Maurin, Lousiana Power and Light (LP&L), then gave an introductory presentation. He described the schedule, and the plant features, noting the similarities and differences between Waterford and other plants. At the end of L. Maurin's presentation P. Shewmon asked about the material used for support holddown bolts. The Staff and the Applicant replied that the holddown bolt material is being investigated.

J. Hart, LP&L, then gave a summary of the status of TMI open items (NUREG-0737). He stated that he was confident that they would all be resolved. L. Maurin then described LP&L's organization for Waterford.

The currently authorized staffing level is 267, and 174 persons are currently on board. He briefly described the training program and noted that all candidates for an operator's license will receive ten weeks of training at an operating nuclear plant. He noted that they are putting more emphasis on recruiting than they have in the past.

F. Drummond, LP&L, described the makeup and responsibilities of the Safety Review Committees.

M. Carbon noted that it might be a good practice to have some representation from outside the organization on these committees. F. Drummond then described the Middle South Utilities organization and the areas in which they will aid LP&L.

S. Hanauer of the NRC Staff then described his concerns with the staff and organization of LP&L.

R. Armstrong, LP&L, described the training program, and J. Edwards, the present and future simulator training. LP&L plans to have a plant specific simulator in operation in January 1985.

W. Alphonso discussed the control room and the plant computer. He noted that they are planning to install some type of core water level indicator but have not yet made a final decision as to what type.

J. Manro, EBASCO, addressed the issue of control room habitability, an area of concern because of the highly industrial nature of the site.

LP&L considered a number of industrial accidents. The principal hazards are chlorine gas and anhydrous ammonia, both of which are stored in the vicinity of the site. The control room is designed to exclude these and other toxic chemicals. Mr. Ebersole asked if electrical systems in other parts of the plant would survive the high concentrations of these gases. J. Manro replied that there have been some releases of chlorine and ammonia which required workers to evacuate but during which electrical systems continued to operate.

M. Carbon questioned the use of a single tank car as the source and asked why they did not consider more than one. J. Manor replied that multiple failures would have a lower probability but he did not know how much lower. Later, D. Michewicz, EBASCO, pointed out that once the control room is isolated, the amount of gas released does not matter very much. D. Okrent asked the NRC Staff if they had looked into the work done in Europe on the effects of explosions on nearby structures. L. Soffer, NRC Staff, said they had not.

D. Hunter, EBASCO, then described the design of the plant to allow it to withstand hurricanes, floods, and tornadoes. J. Ebersole asked, given a flood, what is the assurance that subterranean water would not seep in through cracks at a rate faster than the sump pumps could rer ve it. D. Hunter replied that groundwater leakage would show up any gross cracks. Mr. Wildis, EBASCO, then described the ultimate heat sink, which is unusual in that it does not rely on a water supply. Instead it uses cooling towers to remove heat from the component cooling water. The cooling towers are located on the nuclear island.

M. Bender asked why there is a two hour limit on station blackout; i.e., why must power be restored within two hours. Mr. O'Donnell, EBASCO, replied that the limit of the station batteries was two hours for each set and that even if they failed, the loss of DC power left the systems in a mode where decay heat could still be removed. The problem would then be lack of instrumentation to know the status of the plant. Mr. O'Donnell then noted that there is a water supply good for more than 24 hours using the auxiliary feedwater system and relieving steam through the atmospheric dump valves.

J. Ebersole asked what kept the auxiliary feedwater pump and turbine cool under these conditions. Mr. O'Donnell stated that they had not analyzed that situation in detail and agreed to look into it further.

D. Okrent asked about the condition of the main coolant pump seals during a station blackout. R. Turk, CE, replied that tests at San Onofre indicated that they were good for about 50 hours with only a small increase in the leak rate. D. Okrent said that he recalled that the French considered seal leakage to be the limiting factor during station blackout and said that he did not wish to pursue the matter further at this time, but that the Staff could anticipate interest in this question in the future.

There was some discussion of the inability to feed and bleed. J. Ebersole pointed out that this requires a more reliable auxiliary feedwater system than would otherwise be necessary. Mr. O'Donnel replied that the Waterford auxiliary feedwater system is very reliable and that the accident contributing the most risk is a loss of AC power which renders feed and bleed inoperable. J. Edwards, LP&L, added in response to another question by J. Ebersole, that there are procedures for depressurizing the steam generators so that they can provide feedwater at a lower pressure, if necessary.

At this point the Committee voted to write an interim report on the Waterford plant.

VI. Meeting with NRC Staff Regarding Environmental and Electrical Qualification of Electrical Equipment

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

[Note: The minutes for this portion of the meeting were prepared by R. P. Savio.]

A. Electrical Systems Subcommittee Report

W. Kerr reported on the activities of the Electrical Systems Subcommittee which has been reviewing a proposed rule and Regulatory Guide revision which are titled respectively, Environmental and Seismic Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, and Environmental Qualification of Electric Equipment Important to Safety for Light-Water-Cooled Nuclear Power Plants. They deal with the qualification of electrical equipment important to plant safety. For the earliest plants the qualification of this type of equipment was based essentially on the premise that the equipment used would be of the highest industrial quality. The use of the IEEE-323-1971 standard was implemented in 1971 and the use of the IEEE-323-1974 standard was implemented in 1974. The DOR guidelines, Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors and the NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, were issued in late 1979 and were implemented by the Commission Memorandum and Order CLI-80-21 on May 23, 1980. The NRC Staff has indicated that the intent of the proposed rule and Regulatory Guide is to clarify this guidance.

The NRC Staff met with industry on July 7, 1981 to discuss equipment qualification and the requirements of the proposed rule and Regulatory Guide. The provisions of the proposed rule and Regulatory Guide, if implemented, are expected by industry to have a major impact. In particular it appears that it is not clear as to the extent of the equipment which will be affected by the proposed regulations.

The Electrical Systems Subcommittee met with the NRC Staff on July 22, 1981 to discuss the proposed rule and Regulatory Guide and made a number of comments on this proposed action. The principal comments are:

1. Neither the proposed rule nor the proposed Regulatory Guide revision contains an interpretable designation of the equipment which will be affected by these regulatory actions. If some clear specification cannot be made, a rule is premature.

2. The value-impact evaluation should be improved and should address the impact of the NRC guidelines which the proposed rule and Regulatory Guide revision are intended to formalize. It would appear that compliance with the NRC guidelines would result in substantial costs. A clear case for the need for and safety benefits of this additional regulation has not been presented. The NRC Staff should consider using risk assessment in assessing the value-impact of these proposed regulations.

3. The need for and the usefulness of the proposed requirements for the maintenance of an extensive central file of qualification records should be reexamined from a cost-benefit perspective.

S. Aggarwal spoke for the NRC Staff. The NRC Staff has modified the rule in response to Committee comments. In particular the scope of the rule has been changed to clarify and narrow the scope of the equipment affected by the rule. This was discussed to some extent. It appeared that the scope of equipment which would be affected was still subject to some uncertainty and that the matter was unlikely to be resolved except on a plant by plant basis. The value impact evaluation of the program rule and Regulatory Guides were essentially unchanged. M. Bender and W. Kerr noted that the requirements of the proposed rule were quite prescriptive and questioned the need for the issuance of a rule on this matter.

The Committee noted that the matters adressed in the proposed rule and Regulatory Guide were of some significance and endorsed the issuance of the material for public comment. The Committee indicated that they would review this further after public comment had been received and considered by the NRC Staff.

Z. Rosztoczy discussed the activities of the NRC's Equipment Qualifications Branch. The ongoing work is directed toward identifying differences in operating plants and developing criteria for the qualification of equipment. A program plan addressing the activities of this Branch over the next four years is currently in draft form. The Branch is currently engaged in a review of the environmental and seismic qualification of electrical equipment with the expected issuance of a rule by December of 1981. There is a need for addressing equipment and laboratory accreditation in rules to be issued by June, 1982 and December, 1982 respectively, with the completion of those actions expected by September 1983. Work addressing the survivability of equipment in a hydrogen burn is underway and is expected to be completed in about a year and a half and will be addressed in subsequent rules.

It was noted that the multiple qualification requirements which were being developed would be accompanied by successive implementation requirement dates. The question was raised as to whether this would lead to successive modification/replacement of the same equipment components and the risk associated with the modification. This issue was left unresolved.

VII. Meeting with the NRC Commissioners (Open to Public)

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

A. Pressurized The[illegible] to Reactor Pressure Vessels

In response to Chairman Palladino's questions regarding the seriousness of the pressurized thermal shock to reactor pressure vessel problems, Members provided their opinions based on the limited information concerning the materials that are present in pressure vessels, the effects of irradiation on the conditions in the vessels, and the effects of the conservatisms used in the current calculations. Before the problem can be fully understood and then resolved, better understanding must be obtained regarding the uncertainties involved.

Chairman Palladino requested that the Committee give this problem its attention.

B. Instrumentation to Detect Inadequate Core Cooling

M. Bender called attention to the question of instrumentation to detect inadequate cooling, noting that the NRC Staff appears to be requiring instrumentation that has neither been tested nor shown to be useful. He suggested that reactor safety research could be carried out to develop criteria regarding how such instrumentation could be used.

C. Infrastructure of Nuclear Plants

M. Bender called attention to the ACRS letter to the Executive Director for Operations regarding the development of criteria for the maintenance staff of nuclear plants. He noted further that the Committee had suggested that the EDO meet with it regarding this matter, and recommended that the NRC Staff pursue the development of such criteria with high priority.

D. LOFT

Members explained to Chairman Palladino why they believe that the LOFT experiment should be shut down at the end of FY 1982. It is the opinion of the Committee that the funds used in LOFT for experiments scheduled after this date could be used better in other research projects.

M. S. Plesset added that he believes that the data that will be obtained by the tests planned for FY 1983 will not be particularly useful.

E. Radioactive Waste Disposal

Chairman Palladino clarified the Commission's request for review by the Committee regarding radioactive waste disposal. He said that there are two areas that the Commission wishes the Committee to review:

- The technical content of the contract written by the NRC Staff, and

- The capability of the proposed contractor(s).

He noted that this matter was raised originally by Commissioner Ahearne, and that he would confer with Commissioner Ahearne to determine if he concurs with the above interpretation.

VIII. Meeting with NRC Commissioners (Closed to Public)

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

[Note: Commissioners Palladino and Bradford were present for this meeting.]

A. Proposed Budgets for FYs 1982 and 1983

Proposed cuts for the proposed budgets for FYs 1982 and 1983 in order to conform to the current administration thinking regarding fiscal policy, and the areas that will be affected by these cuts were dis cussed.

IX. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. Generic Items Subcommittee

The Generic Items Subcommittee will track resolution of generic safety issues by the NRC Staff (GITS report regarding generic items and "AQUA" report regarding unresolved safety issues).

[Note: Tracking of the TMI-2 Action Plan items (APTs) will continue to be the responsibility of the TMI-2 Action Plan Subcommittee.)

2. Reactor Operations

Reactor Operations Subcommittee will review the proposed criteria for the preparation of emergency operating procedures (NUREG-0799).

3. Metal Components

The Metal Components Subcommittee, aided as necessary by members of the Waste Management Program Subcommittee, will evaluate the appropriateness of proposals and the capability of proposers of research programs to develop containers for long range disposal of high level radioactive wastes (see item VII E above).

B. Generic Safety Items

1. Loss of A.C. Power

The Committee on Waterford-3 notes that the generic evaluation of loss of all a.c. power should be expanded to include consideration of the effect of loss of space cooling on essential electric equipment and also consideration of the effect of coolant leakage from the primary system (ACRS report on Waterford-3 Operating License, August 11, 1981).

2. Evaluation of Systems Interactions

The Committee affirmed its continuing interest in the evaluation of systems interactions (both seismic and non-seismically induced) particularly in connection with ongoing operating license reviews (R. F. Fraley memo. to H. Denton dated August 12, 1981, Seismic-Induced and Other Interactions Between Non-Safety and Safety Systems).

3. Anticipated Transients Without Scram

The Committee concluded that the request by General Electric for the ACRS endorsement of the reliability of high pressure coolant injection for BWR-4 plants and high pressure coolant sprays for BWR-5 and 6 plants (letter from G.G. Sherwood to J. Carson Mark, June 25, 1981) would not be addressed in the Committee reports completed during this meeting, (Susquehanna (BWR-4) and Fermi 2 (BWR-4)) but will be left for resolution in connection with the ATWS rulemaking.

4. Potential for Steam Bypass in GE Mark II Containment

Several Members noted their continuing interest and concern regarding the potential for steam bypass in GE Mark II plants resulting from the failure of safety or relief valve discharge lines.

5. Feed-Bleed Capability in Combustion Engineering Plants

Several Members noted their concern regarding the elimination of feed-bleed capability in those Combustion Engineering plants (e.g., ANO-2 and Waterford-3) which do not have PORVs. Members suggested that this issue may warrant additional action as a generic matter. This item will be scheduled for further discussions during appropriate subcommittee and/or full Committee meetings.

C. Future Schedule

1. Future Agenda

The Committee agreed to a tentative agenda for the 257th ACRS Meeting, September 10-12, 1981 (see Appendix II).

2. Future Subcommittee Activities

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

D. Possible Testimony by ACRS Before U. S. Senate Committee on Energy and Natural Resources Regarding Application of TMI-2 Lessons Learned by DOE Facilities

The Chairman requested that the ACRS Executive Director keep him informed regarding any request for testimony by the Committee.

E. Letter from Representative Udall to Governor Thornburgh regarding the Adequacy of TMI-1 Management

The Committee agreed to not comment on this letter.

F. Guideline for Utility Management Structure and Technical Resources

It was recommended that the Committee should review NUREG-0731, Guideline for Utility Management Structure and Technical Resources, in view of the Committee's interest with respect to the competence of overall utility operating organizations.

G. Safety Research Work in Progress

The Committee agreed to periodic briefings at ACRS meetings regarding status of RES-sponsored research projects.

H. Qualification of Electrical Equipment (Regulatory Guide 1.89)

The Committee agreed with the NRC Staff that Regulatory Guide 1.89 can be released for public comment.

M. Bender noted his concern regarding the use by the NRC Staff of rules rather than case-by-case determinations for the licensing of plants. D. Ross volunteered to provide the Committee with a legal view (by H. Shapar, ELD) of the usefulness of rules vs. case-by-case determination.

I. Comments on NUREG-0739

The Committee agreed that ACRS Staff member J. M. Griesmeyer and member D. Okrent should provide the Commission with their comments regarding comments received by the Commission on NUREG-0739, An Approach to Quantitative Safety Goals for Nuclear Power Plants. It was stressed that these were the comments of the individuals who prepared the report, and not necessarily those of the Committee.

J. ACRS Reports, Letters, and Memorandum

1. Enrico Fermi Atomic Power Plant Unit 2

The Committee prepared a report to the Commissioners advising them that, subject to certain stated conditions, it believes that the Enrico Fermi Atomic Power Plant Unit 2 can be operated at a power level up to 3292 MWt without undue risk to the health and safety of the public (see Appendix XL).

2. Susquehanna Steam Electric Station Units 1 and 2

The Committee prepared a report to the Commissioners advising them that, subject to certain stated conditions, the Susquehanna Steam Electric Station Units 1 and 2 can be operated at a power level up to 3293 MWt without undue risk to the health and safety of the public (see Appendix XLI).

3. Waterford Steam Electric Station Unit 3

The Committee prepared an interim report to the Commissioners advising them that subject to certain stated conditions, and contingent on the attainment of an adequate level of management and staffing, the Waterford Steam Electric Station Unit 3 can be operated at a power level up to 3410 MWt without undue risk to the health and safety of the public (see Appendix XLII).

4. Systems Interactions

The Committee approved a memorandum from the ACRS Executive Director, to the Director, Office of Nuclear Reactor Regulation, noting that questions of interactions between safety and non-safety systems warrant attention (see Appendix XLIII).

The 256th ACRS Meeting was adjourned on Saturday, August 8th, 1981 at 2:15 p.m.

APPENDIXES
TO
MINUTES OF THE 256TH ACRS MEETING
AUGUST 6-8, 1981

ACRS - 1888

ATTENDEES
256TH ACRS MEETING
AUGUST 6-8, 1981

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

*Member Emeritus

ACRS STAFF

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

CONSULTANT

I. Catton

NRC ATTENDEES

256TH ACRS MEETING

Thursday, August 6, 1981

NUCLEAR REACTOR REGULATION

R. L. Tedesco, DL
M. Tokar, DSI
C. Willis, DSI
F. Witt, CMEB
R. Anand, CMEB
P. R. Matthews, CMEB
J. E. Kennedy, EQB
V. Benaroya, CMEB
J. Campbell, OLB
J. L. Manck, DSI
J. J. Kramer, DHFS
C. P. Tan, SGB
W. T. LeFave, ASB
J. N. Ridgely, ASB
F. Eltawila, CSB
P. P. Psomas, EPLB
P. M. Byson, RIII
D. Sells, DL
P. Sears, DMEB
L. Beltracchi, DHFS
N. Trehan, PSB
S. H. Hanauer, DHFS
W. R. Butler, CSB
J. Lave, DSB
D. R. Lunze, SGPL
J. Guttmann, RSB
C. Liang, RSB
S. Black, DL

D. E. Sells, DL
S. Black, DL
V. Nerses, DL
J. Guttmann, RSB
J. H Williams, DL
K. M. Campe
E. F. Goodwin
F. R. Inacdi
L. Wheeler
R. Benedict, LQB
A. K. Ibrahim, GSB
O. Thompson, HGEB
J. Kimball, GSB
H. E. Lefevre, GSB
B. J. Youngblook, DL
E. F. Goodwin, NRR
R. Stark
W. Hazelton, DE
W. Hodges, DSI
L. Phillips, DSI
R. Schemel, DHFS
L. Rubenstein, CCS
P. T. Kuo, SEB
H. B. Clayton, DHFS
J. W. Clifford, DHRS
R. Benedict, LQB
L. Soffer, SAB
L. Kintner

INSPECTION & ENFORCEMENT

D. Perrotti
E. F. Williams

A-2

NRC ATTENDEES

256TH ACRS MEETING

Friday, August 7, 1981

NUCLEAR REACTOR REGULATION

F. Schroeder
R. Stark
D. Terao
R. Gilbert
J. N. Hannon
E. F. Silliams, Jr.
M. M. Scosson
F. Allenspach
S. Salah
M. W. Hodges
W. LeFave, ASB
G. Zech
W. G. Kennedy
C. J. Anderson, DST/GEB
R. Tedesco, DL
A. Schwencer, DL
E. F. Goodwin
F. Eltawila
J. Wilson
J. Manck
K. Eccleston
J. O. Schiffgens
H. Gary
T. Collins
J. J. Kramer, DHFS
R. J. Schemel, DHFS
W. Long, PTRB
Z. Rosztoczy

NUCLEAR MATERIAL SAFETY & SAFEGUARDS

C. E. Gaskin
K. A. McConnell

NUCLEAR REGULATORY RESEARCH

D. Ross
S. Aggarwal

SEB/DOE

O. Rothberg

INSPECTION & ENFORCEMENT

G. Rhoads
S. Chestnut
G. W. Reinmuth

INVITED ATTENDEES

256TH ACRS MEETING

Thursday, August 6, 1981

LP&L

F. J. Drummond
J. J. Saacks
R. K. Slampdy
W. M. Alphonso
T. K. Armington
G. D. McLendon
G. R. Peelen
D. Lott
D. B. Lester
R. Azzapello
R. E. Armstrong
L. V. Maurin
J. G. Edwards
E. Sewac
T. F. Gerrets
D. L. Aswell

Combustion Engr.

J. M. Westhoven
H. B. Mulliken
R. E.Newman
R. S. Turk
G. Kling
F.Carpentino
V. Callaghan
R. Fedin
D. Young
G. Davis

TERA
B. Maguire

Shaw Pittman
N. Knowles

Law Engr. Testing Co.
J. Gustin

LMSC
W. R. Gonzalez

KMC
R. Boyd

General Electric
C. Sawyer
L. S. Gifford

EBASCO

D. J. Masiero
J. Horvath
G. G. Hofer
W. J. Krotink
D. Michewicz
A. Letizia
N. Illouits
R. Vidal
H. Parikm
M. Serbanescu
W. Livingston
M. Pavone
D. J.Lott
J. Damitz
P. V. Gvildys
G. Martin
J. Hart
G. Williams
J. Manro
Y. Boynowsky
R. Ioth
D. Bregman
R. Stampley
L. Gunther
J. Costello
A. F. Devine
B. E. Baril
D. Hunter
R. Foley
J. Tompella
J. Papalino
P. Liu
W. T. Tgng
M. Horrell
Z. Shi
B. Leter
O'Donnell

Bechtel
N. G. Chapman

Detroit Edison
E. Page

INVITED ATTENDEES

256TH ACRS MEETING

Thursday, August 6, 1981

Detroit Edison

M. Batchi
L. Schuerman
E. Madsen
E. Griffing
J. Wisniewikr
J. Green
R. Anderson
H. Taulin
R. Shaw
W. J. Fahruer
W. Jens
P. Marquardt
F. Locke
W. Colbert
F. E. Gregor
R. Curio
R. Horn
W. W. Hodges
S. Leach
M. D. Featham
M. K. Deora
L. E. Kanous
J. W. Honkala
L. F. Wooden
T. M. McKelvey
E. Lusis
A. E. Wegele
A. Shoudy
D. F. Lehnert
R. H. Duong
J. Lord
W. M. Street
E. Leonard

DECO

R. Blaudry

NUS

J Slider

Weston Geophysical Corp.

E. Levine
G. Klimkiewicz

EPRI

Leyse

General Electric

P. S. Tam

NUTECH

R. Bulhholz
A. Higginbotham

Mark II Owners Group

H. Chow
J. Kitts
M. Carne

INVITED ATTENDEES

256TH ACRS MEETING

Friday, August 7, 1981

Pennsylvania Power & Light

W. Lawthut, III
G. D. Miller
T. M. Brummins
R. W. McNamara
H. Gutshall
S. H. Cantone
P. H. Henrikson
T. E. Widner
B. Kenyon
H. Keiser
N. W. Curtis
E. R. Carlson
M. B. Detamore
D. P. Moyer
T. Coddington
H. V. Oheim
J. A. Bartos
W. J. Rhoades
D. W. Miller
D. Cobe
G. Hoams
J. Calhoun
A. Petry
W. G. Ward
S. Bella
A. Male
D. Roth
H. R. Clarke
M. R. Buring
L. D. O'Neil
C. Sprunk
R. Kelm, Dr.

General Electric

T. R. Wortham
W. M. Davis
J. H. Crow
J. W. Millard

Bechtel Power Corporation

F. Titus
E. Cornell
D. M. O'Connor
Ciaty

KMC, Inc.

E. Morris Howard
D. Knuth

PUBLIC ATTENDEES

256TH ACRS MTG.

Thursday, August 6, 1981

N. W. Curtis, Pa. Power & Light
R. Kenyon, Pa . Power & Light
H. W. Keiser, Pa. Power & Light
D. F. Greenwood, Stone & Webster
J. Behn, Gage-Babcock
J. M. Gilbert, Stone & Webster
H. H. Voigt, LeBoeuf, Lamb
E. F. Beckett, Nuclear Plus Inc.
R. W Huston, Consumers Power
J. Stampelos, Nuclear Safety Oversight Cte.
P. A. Minson, ARC
R. T. Misiaszek, Stone & Webster
A. K. Singh, Sargent & Lundy
E. T. Murphy, Westinghouse
D. P. Moyer, PP&L
C. T. Coddington, PP&L
S. T. Bells, PP&L
H. C. Schmidt, TUSI
S. Gregg
R. A. Jones, TUGCo
E. Blank, SPPT
R. Schlundt, Draper
T. Martin, NUTECH
J. C. Kuykendell, Texas Utilities
R. B. Seidel, Texas Utilities
D. W. Braswely, Texas Utilities
D. R. Black

PUBLIC ATTENDEES

256TH ACRS MTG.

Friday, August 7, 1981

J. R. Lehner, ANL
J. C. Kuykendall, TUGCO
G. T. Kitb, Sargent and Lundy
J. E. Metczlf, Stone & Webster
T. E. Dollog, Pa DER
G. Ramret, CNSNS
J. Nelson, Quadill
G. M. Beemer, PNL
G. Marr, PNL
J. E. Slider, NUS
R. G. Azzarollo, LP&L
T. R. Kishbaugh, NUTECH
D. J. Klein, NPI
R. A. Jones, T. U. G. Co.
D. W. Brasnell, TUGCO
H. Chan, LILCO
R. K. Mattu, NUS
T. Zazueta, CFE
U. P. Pornsirr, PA DRR
C. Arredondo, CNSNS
Leyse, EPRI
H. C. Schmidt, Texas Utilities
R. B. Seidel, Texas Utilities
C. A. Malourh, Stone & Webster
M. Benqa, PG&E
E. Vanaber, SElf
H. C. Stenigh, TUSI
P. A. Minson, ARC
E. T. Murphy, Westinghouse
P. Hinsberg, McGraw-Hill
R. Smith, NUCOM
M. Philips, D&L
K. Watkins, LNRA
M. D Patterson, BG&E
F. J. Drummond, LP&L
R. G. Azzonello, LP&L

APPENDIX II

FUTURE AGENDA

257th ACRS Meeting	September 1981
Briefing re Reactor Pressure Vessel Thermal Shock from Overcooling Transients - discuss ACRS position/comments	3 hrs
Briefing re Allens Creek Hydrogen Control Systems Results of Hydrogen Research Program to Date	1-1/2 hrs
Rulemaking on Reactor Siting Criteria - ACRS Comments	2 hrs
Rulemaking on Low-Level-Waste Management - ACRS Comments	1 hr
Rulemaking on Geological Criteria for Rad Waste Disposal - ACRS Comments	2 hrs
PL 96-567: Nuclear Reactor Safety Research and Development Act of 1980 - proposed DOE plan for implementation - ACRS Comments	3 hrs
Task Action Plan A-45, Decay heat Removal Systems - ACRS Comments	1-1/2 hrs
AIF Comments on NUREG-0739, <u>An Approach to Quantitative Safety Goals for Nuclear Power Plants</u> - proposed ACRS response	1-1/2 hrs
Briefing re Clinch River Breeder Reactor - Briefing re FY-82 Research Supplement (D. Ross)	1/2 hr
NSOC Comments on Deficiencies in the NRC regulatory process and proposed changes in ACRS role - discuss ACRS position/comments (D. Okrent)	1/2 hr
Scope of Annual ACRS Report to NRC on Proposed Reactor Safety Research Budget	1 hr
Briefing re Improvements in Future Westinghouse Plants Briefing (tentatively)	1 hr
Briefing re Use of Rules in Regulatory Process (H. Shapar)	1 hr
Designate Panel to Nominate ACRS Officers for calendar year 1982	1/2 hr
Evaluation of LERs, proposed changes in system - ACRS comments	1 hr

APPENDIX A

Subcommittee Reports	
ECCS: Proposed Revisions to App. K re BWRs	1/2 hr
Advanced Reactors: Proposed LMFBR Design Criteria	1/2 hr
Gas Cooled Reactor Safety Research, Long-Range Program	1/2 hr
Regulatory Activities: Proposed changes in Reg. Guides	1 hr
Procedures by which Committee decides to handle rules and regulations	1 hr
Testimony re DOE application of TMI-2 Lessons Learned to DOE facilities (tentatively)	1 hr

<u>258th ACRS Meeting</u> October 1981

Shippingport LWBR Core -- Extension of LWBR core burnup

Floating Nuclear Plant -- Manufacturing License, application of TMI-2 Action Plan

Grand Gulf Nuclear Station Units 1 & 2 -- Operating License

Shoreham Nuclear Power Station -- Operating License

<u>259th ACRS Meeting</u> November 1981

Callaway Plant Units 1 and 2 -- Operating License

Comanche Peak Steam Electric Station Units 1 & 2 -- Operating License

Watts Bar Nuclear Plant Units 1 & 2 -- Operating License

St. Lucie Plant Unit 2 -- Operating License

Sequoyah Nuclear Plant (tentatively) -- Briefing on Hydrogen Control and other outstanding issues.

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS

AUGUST

[illegible]	ECCS (Monterey, CA) (Boehnert) - Plesset, Ebersole, Etherington, Lewis, Mark, Ward (tent.). Purpose: Discuss GE's proposed revisions to Appendix K of 10 CFR 50.46 and revisions to other aspects of the ECCS EM and various topics related to NRR ECCS licensing.

SEPT.

2 & 3	Waste Management (Young) - Moeller, Ray. Purpose: Review of proposed rules on 10 CFR 60 and 10 CFR 61.
8 (1 p.m.)	Decay Heat Removal Systems (Savio) - Ward, Bender, Ebersole, Ray. Purpose: To continue review of the Task Action Plan A-45.
CANCELLED	Gas-Cooled Reactors
9	Evaluation of LERs (Quittschreiber) - Mathis*, Moeller*, Lewis, Okrent*, Plesset*. Purpose: To discuss recent developments in NRC's LER sequences coding and search procedure.
9 (1 p.m.) (tent.)	Shippingport (Boehnert) - Bender, Carbon, Okrent*, Siess (tent.). Purpose: Review LWBR operation up to 30,000 EFPH.
[illegible] (8:45 - 4 p.m.)	Regulatory Activities (Duraiswamy) - Siess, Bender (Part-time), Ray, Shewmon, Carbon (Part-time). Purpose: To review proposed Regulatory Guides and Regulations.
9 (2 p.m.)	National Engineering Simulator/Nuclear Manpower and Training Study (Major/Fischer) - Mathis*, Ray, Ward, Moeller*, Bender. Purpose: One-day joint working group meeting planned to discuss the draft report from DOE to the Congress.
9 (4 p.m.)	Program Management & Plan (Duraiswamy) - Siess, Mark, Okrent*, Shewmon, Plesset*. Purpose: To provide comments on DOE regarding implementation of P.L. 96-567 and comment on DOE report on conduct of research and development activities.
9 (6 - 8 p.m.)	Reliability & Probabilistic Assessment (Quittschreiber/Griesmeyer) - Okrent, Siess, Mark, Lewis, Ebersole. Purpose: Discuss NRC effort to develop safety goals.
17-18	Advanced Reactors (Igne/Savio)(Chicago, IL) (Igne) - Carbon, Mark, Shewmon. Purpose: To discuss LMFBR safety design criteria.
25	ATWS (Boehnert) - Kerr, Bender, Ebersole, Ray, Ward. Purpose: To discuss ATWS Rule proposal based on PRA.

*Conflict to be resolved.

SEPT. (Continued)

30	Shoreham (Fischer/McKinley) - Bender, Ray, Siess, Moeller. Purpose: Review application for an OL.
OCT.	
1 & 2	Transportation of Radioactive Materials (Oak Ridge/TN) (Duraiswamy) - Siess, Bender, Mark. Purpose: To review package certification procedures used by the Transportation Certification Branch of NRC.
9	Floating Nuclear Plant (Boehnert) - Moeller, Mathis, Siess, Shewmon, Okrent. Purpose: To review Supplement 4 to FNP SER.
14 (tent.)	Regulatory Activities (Duraiswamy) - Siess, Bender*, Kerr, Ray*, Carbon, Ward. Purpose: To review proposed Regulatory Guides and Regulations.
14	Reactor Operations (Major) - Mathis, Bender*, Ebersole, Okrent (tent.), Ray*. Purpose: Briefing by Div. of Human Factors Safety Div.
Date to be determined	Comanche Peak 1 & 2 (Alderman/Duraiswamy) - Bender, Okrent, Ray, Lewis (tent.). Purpose: To continue review of the application for OL.
Date to be determined	Sequoyah (Savio) - Mark, Mathis (a.m.), Siess (p.m.). Purpose: Review operating experience, ongoing hydrogen control work, and responses to ACRS request. A site visit is also planned.
Date to be determined	AC/DC Power Systems (Savio) - Ray, Ebersole, Kerr, Mathis, Okrent. Purpose: Review status of the activities associated with NUREG-0666, and AC Power Supply reliability.
12-13 or 22-23	Advanced Reactors (Chicago, IL) (Igne/Savio) - Carbon, Bender, Kerr, Mark, Shewmon. Purpose: To discuss LMFBR design criteria.
NOV.	
Prior to Nov. ACRS Mtg.	TMI-2 Action Plan (Major) - Mathis, Bender, Etherington, Lewis, Okrent. Purpose: Review Rule on Requirements for pending OL applications.

*Conflict to be resolved.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Aug 28	ECCS	(BOEHNERT) Plesset, Etherington, Mark ~~(tent)~~, Ward (tent) Lewis, Ebersole

LOCATION: Monterey, CA

BACKGROUND:

Who proposed action: M. Plesset/NRR

Purpose: Discuss GE's proposed revisions to Appendix K of 10 CFR 50.46 as well as revisions to other aspects of the ECCS EM and various topics related to NRR ECCS licensing matters.

PERTINENT PUBLICATIONS: Will be provided as available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPT. 2 & 3	Waste Management	(YOUNG) Moeller, Ray Cons: (all tent.) Healy, Muller, Orth, F. Parker, H. Parker, Steindler, Thompson

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To discuss and review the propoesd rules on:

(1) 10 CFR 60, Technical Criteria for Disposal of High-Level Radioactive Wastes in Geological Repositories, and

(2) 10 CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:.

Federal Register Notice, Vol. 46, No. 130, pg. 35280, dated Wednesday, July 8, 1981 (10 CFR 60).

Federal Register Notice, Vol. 46, No. 142, pg. 38081, dated Friday, July 24, 1981 (10 CFR 61).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sep 8 (1:00 pm)	Decay Heat Removal Systems	(SAVIO) Ward, Bender, Ebersole, Ray

LOCATION: Wash, DC

BACKGROUND:

Purpose: To continue the Subcommittee review of the Task Action Plan A-45, "Shutdown Decay Heat Removal Requirements". The Subcommittee had revised and commented on a draft of this plan at its June 6, 1981 Subcommittee meeting. It is expected that a final version of this plan will be available by this September 8, 1981 meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sep. 9	Evaluation of LERs	(Quittschreiber) Mathis, Moeller, Lewis, Okrent, Plesset. Cons. I. Catton W. Lipinski Z. Zudans

LOCATION: Washington, DC

BACKGROUND

Purpose: To discuss recent developments in NRC's LER sequences coding and search procedure.

PERTINENT PUBLICATIONS: Will be provided Later

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sept. 9 (1 p.m.) (tent.)	Shippingport	(Boehnert) Bender, Carbon, Okrent, Siess (tent.)

LOCATION: Washington, DC

BACKGROUND

Who proposed action: Naval Reactors (NR)/M. Bender

Purpose: To consider review of extention of LWBR operation from 24,000 EFPH to 30,000 EFPH. Meeting is contingent on identification of significant review issues upon receipt of information from NR and the NRC Staff.

PERTINENT PUBLICATIONS: See above

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
9/9/81 (8 a.m. to 4 p.m.)	Regulatory Activities	(DURAISWAMY) Siess (Chairman), ~~Kerr~~, Bender, Ray, Carbon, Ward, Shewmon

PURPOSE: To review the following items:

POST-COMMENT ITEMS

1. Regulatory Guide 1.33, Revision 3, "Quality Assurance Program Requirements (Operation)".

2. Regulatory Guide 1.23, Revision 1, "Meteorological Programs in Support of Nuclear Power Plants".

(Tent) 3. General Revisions to Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements".

PRE-COMMENT ITEMS

4. Proposed Regulatory Guide 1.105, Revision 2, "Instrument Set Points".

5. Regulatory Guide 1.13, Revision 2 "Spent Fuel Storage Facility Design Basis".

6. Proposed Regulatory Guide (Task No. MS-901-4) "Identification of Valves For Inclusion in Inservice Testing Program".

STATUS: The status of these items are as follows:

POST-COMMENT ITEMS

Item 1: Regulatory Guide 1.33 describes a method for complying with regard to the overall quality assurance program requirements with the Commission Regulations for the operational phase of the nuclear power plants. The previous version of this Guide endorsed, with certain exceptions, ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants".

A previous version of this Guide was reviewed by the Regulatory Activities Subcommittee in March 6, 1979 and was issued for public comment in August 1979. Subsequent to the TMI-2 accident, ANS 3.2 Standard has been revised to reflect the lessons learned from that accident. Consequently, this Guide was also revised in accordance with the Draft ANS 3.2 Standard, dated February 1980. This revised

Schedule of ACRS Subcommittee Meeting

version of this Guide was reviewed by the Regulatory Activities Subcommittee in August 6, 1980 and was reissued for public comment in November 1980. The current version of this Guide reflects consideration of public comments that were received during the public comment period of this Guide.

The NRC Staff requests ACRS concurrence in the Regulatory Positions of this Guide.

Item 2: Regulatory Guide 1.23 was originally issued in February 1972 as Safety Guide 23. It has been revised to reflect the current state of the art in meteorological measurement technology and also to accommodate some of the lessons learned from the TMI-2 accident concerning meteorological measurement programs at nuclear power plant sites.

This Guide describes meteorological measurement programs for providing meteorological data needed to estimate the potential radiation doses to the public resulting from effluent releases from nuclear power plants.

A previous version of this Guide was reviewed by the Regulatory Activities Subcommittee at the June 4, 1980 meeting and was issued for public comment in September 1980. The current version of this Guide reflects consideration of public comments that were received during the public comment period of this Guide.

The NRC Staff requests ACRS concurrence in the Regulatory Positions of this Guide.

Item 3: Revisions to Appendix G "Fracture Toughness Requirements" and Appendix H "Reactor Vessel Material Surveillance Program Requirements" are to update the requirements of Appendices G and H to be more consistent with current technology and pertinent National Standards. Some of the proposed revisions are to clarify the applicability of the requirements of Appendices G and H to old and new plants. Some other revisions are intended to grant relief to some of the existing requirements.

These proposed revisions were revised previously by the Regulatory Activities Subcommittee at the June 13, 1979 meeting and were issued for public comment in November 1980. The current version of this item reflects consideration of public comments.

Schedule of ACRS Subcommittee Meeting

The NRC Staff requests ACRS concurrence in the proposed revisions to Appendices G and H.

PRE-COMMENT ITEMS

Item 4: Proposed Regulatory Guide 1.105, Revision 2 describes a method acceptable to the NRC Staff for ensuring that instrument set points in systems important to safety are initially within and remain within the specified limits.

This Guide endorses, with certain exceptions, Instrument Society of America (ISA) Standard S67.04, "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants".

Subsequent to the Subcommittee's review, the NRC Staff may issue this Guide for public comment.

Item 5: Regulatory Guide 1.13 describes a method acceptable to the NRC Staff for implementing General Design Criterion 61, "Fuel Storage and Handling Criteria for Nuclear Power Plants", which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions; it requires also that these systems should be designed with appropriate containment, confinement, and filtering systems and be designed to prevent significant reduction in the coolant inventory of the storage facility under accident conditions.

Revision 1 to Regulatory Guide 1.13 was reviewed by the Regulatory Activities Subcommittee at the August 13, 1975 meeting and was issued for public comment in December 1975.

Subsequent to the issuance of this Guide for public comment, additional guidance for spent fuel pool design has been given in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations". NRR thought that this Guide should be updated to incorporate the information provided in the ANSI Standard, as appropriate. As a result, Revision 1 to Regulatory Guide 1.13 has never been issued as an effective Guide.

Revision 2 to Regulatory Guide 1.13 endorses, with certain exceptions, ANSI N210-1976/ANS 57.2.

Schedule of ACRS Subcommittee Meeting

Subsequent to the Subcommittee's review, the NRC Staff may issue this Guide for public comment.

Item 6: This proposed Guide is developed to provide guidance to licensees and their agents in the following areas:

1. Requirements for inservice testing of valves which are important to safety.

2. Information needed by the NRC Staff to evaluate requests for relief from ASME Section XI requirements.

3. Conditions under which testing of valves should not be performed.

The NRC Staff believes that this Guide will provide a uniform Standard for evaluating licensees' valve testing programs as well as requests for relief from code requirements. It will also accelerate the NRC Staff's review process by reducing or eliminating the waiting period caused by the need to request additional information from applicants.

Subsequent to the Subcommittee's review, the NRC Staff may issue this Guide for public comment.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPT. 9, 1981 (2:00 p.m.) rescheduled from SEPT. 1	National Engineering Simulator	(MAJOR) Mathis, Ray, Ward
	Nuclear Manpower and Training Study	(FISCHER) Moeller, Bender, Mathis Cons: Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Department of Energy (DOE), ACRS

Purpose: One-day joint working group meeting planned to discuss the draft report from DOE to the Congress that reports the results of a study on:

- the need for and feasibility of a National Engineering Simulator; and

- the sufficiency of efforts in the U.S. to provide specialy trained professionals to operate the controls of nuclear power plants and other facilities in the back-end of the nuclear fuel cycle.

The Subcommittee will also review the Nuclear Safety Research, Development, and Demonstration Act of 1980.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:.

1. Public Law 96-567

2. ACRS letter to Mr. William J. Dircks dated May 12, 1981: Requirements for Supporting Infrastructure in Nuclear Power Plants

3. Management Development Training, FPC Management Development

4. DOE report regarding implementation of Public Law 96-567 — draft copies were issued July 15, 1981 and received by the ACRS on August 5, 1981.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sept. 9 (4 p.m.)	Program Management & Plan	(Duraiswamy) Siess, Mark, Okrent, Shewmon, Plesset

LOCATION: Washington, DC

BACKGROUND:

Purpose: To provide comments on DOE report on conduct of Research and Development Activities that was developed by DOE in response to PL-96-567.

Documents:

1. DOE report in response to PL-96-567
2. PL-96-567

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sept. 9 (6 - 8 p.m.)	Reliability & Probabilistic Assessment	(Quittschreiber/Griesmeyer) Okrent, Mark, Siess, Lewis, Ebersole

LOCATION: Washington, DC

BACKGROUND:

Purpose: To discuss NRC's effort to develop Quantitative Safety Goals

PERTINENT PUBLICATIONS:

To be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sept. 17-18	Advanced Reactors	(Igne) Carbon, Mark, Shewmon Cons. Avery, Golden, Hartung, Koch, Lipinski, Siegel

LOCATION: Des Plaines, IL (Royal Court INN)

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To discuss matters relating to the development of LMFBR safety design criteria.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Sept. 25	ATWS	(Boehnert) Kerr, Bender, Ebersole, Ray, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: F. Rowesome PAS/W. Kerr

Purpose: Discuss the PRA-based proposed Rule for resolution of ATWS. This Rule was drafted by F. Rowesome (NRC/PAS) at then Chairman J. Hendrie's request.

PERTINENT PUBLICATIONS: Will be provided in near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPT. ~~29~~ 30	Shoreham	(FISCHER, McKINLEY) Bender, Ray, Siess, Moeller Cons: Catton, Lipinski, Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR request, partial SER issued 4/6/81).

Purpose: Review application for an Operating License. The NRC Staff issued a partial SER on April 10, 1981. A Supplement to the SER is scheduled to be issued in late August. The application is scheduled for ACRS review at the October 1981 meeting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:.

SER issued 4/10/81.

FSAR through Amendment 39.

Supplement to SER to be issued in late August.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Oct. 1 & 2	Transportation of Radioactive Materials	(DURAISWAMY) Siess, Bender, Mark CONS: L. Shappert, J. Langhaar, Z. Zudans

LOCATION: Oak Ridge, TN

BACKGROUND:

Purpose: To review package certification procedures used by the Transportation Certification Branch of NRC.

A-28

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Oct. 9	Floating Nuclear Plant	(Boehnert) Moeller, Mathis, Siess, Shewmon, Okrent

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Offshore Power Systems/NRC

Purpose: To review Supplement 4 to FNP Manufacturing License (ML) SER, as well as remaining ACRS outstanding issues on this project. Supplement 4 should be last SER supplement for ML. ACRS review in October is anticipated.

PERTINENT PUBLICATIONS: SER Supplement 4 (due to be issued late August)
Information on ACRS-related concerns.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Oct 14 (tent)	Regulatory Activities	(DURAISWAMY) Siess, Bender, Kerr, Ray, Carbon, Ward

LOCATION: Wash, DC

BACKGROUND:

Purpose: To review proposed Regulatory Guides and Regulations.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
OCT. 14	Reactor Operations	(MAJOR) Mathis, Bender, Ebersole, Okrent (tent.), Ray Cons: (all tent.) Buck, Keyserling, Pearson, Salvendy

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: S. Hanauer, Director, Division of Human Factors Safety, NRR

Purpose: To brief the Subcommittee on developments and programs that have been developed within the Division of Human Factors Safety over the past year. Items for discussion will include the final version of the control room design evaluation guidelines, a discussion of operator qualifications, and emergency procedures guidelines.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:.

1. Final version of control room design evaluation guidelines. (expected in Septemer)

2. NUREG-0799, "Emergency Procedures Guidelines" (June 1981).

3. Various proposals now exist for Operator Qualification.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Oct (Date to be decided)	Comanche Peak 1&2	(ALDERMAN/DURAISWAMY) Bender, Okrent, Ray, Lewis (tent)

BACKGROUND:

Purpose: To continue the review of the application for an operating license to operate Comanche Peak Units 1&2.

PERTINENT PUBLICATIONS:
1. Comanche Peak Units 1&2 FSAR
2. Comanche Peak Units 1&2 Final SER

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
October (date to be decided)	Sequoyah	(SAVIO) Mark, Mathis (a.m.), Siess (p.m.) CONS: Z. Zudans, I. Catton, W. Lipinski

LOCATION: Wash, DC

BACKGROUND:

Who proposed action: ACRS request

Purpose: To review the status of the Sequoyah Plant, operational experience, ongoing hydrogen control work, and responses to ACRS request. A site visit to the plant and to the plant simulatory facility is also planned.

PERTINENT PUBLICATIONS: Supplement 5 to the NRC Staff's SER.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
October (date to be decided)	AC/DC Power Systems	(SAVIO) Ray, Ebersole, Kerr, Mathis, Okrent

LOCATION: Wash, DC

BACKGROUND:

Purpose: To review the status of the activities assocaited with NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants" and the status of the work on the availability of AC Power. Input from the industry and public will be solicited. Foreign practices in these areas will be discussed.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Oct. 12-13 or Oct. 22-23	Advanced Reactors	(Igne) Carbon, Mark, Bender, Shewmon, Kerr Cons. Avery, Golden, Hartung Koch, Lipinski, Siegel

LOCATION: Des Plaines, IL (Royal Court INN)

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To discuss matters relating to the development of LMFBR safety design criteria.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Prior to Nov. ACRS Mtg.	TMI-2 Action Plans	(MAJOR) Mathis, Bender, Etherington, Lewis, Okrent

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: ACRS/NRC Staff

Purpose: The proposed rule on "Licensing Requirements for Pending Operating License Applications" has been sent to the ACRS for consideration. In accordance with the Memorandum of Understanding between ACRS and NRC on "ACRS Participation in Rulemaking Activities" this rule was reviewed on behalf of the Regulatory Activities Subcommittee and it is recommended that this rule be reviewed by the ACRS. On the basis of the nature of the rule, it has been recommended that this rule be reviewed by the Ad Hoc Subcommittee on TMI-2 Action Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:.

1. Copies of the Proposed Rule to 10 CFR Part 50 Licensing Requirements for Pending Operating License Application are available.

2. Copies of NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980 are available.

3. Proposed Rule: TMI Related Requirements for Operating Reactors (SECY-81-422), July 15, 1981.

Substance of the Rule:

This rule, which addresses the same set of items contained in NUREG-0737, imposes new safety requirements for operating license applications. The Commission has determined that these requirements must be met by all applicants for operating licenses. It should be noted, however, that there are many elements in the TMI Action Plan (NUREG-0660) not included in NUREG-0737, that have not yet been developed by the Staff or acted upon by the Commission. There are also items that the Commission has directed to be the subject of further study. This rule will be augmented in the future to add new requirements as they are approved. Opportunity for public comments will be provided when such additional requirements are contemplated. The Staff's review will include rules for both operating rectors and operating license applicants. The ACRS review should include both the OL applicants and OR rules.

Public comments on the Rule are being requested before August 12, 1981.

256TH ACRS MEETING
FERMI-II OPERATING LICENSE REVIEW
AUGUST 6, 1981

- PROJECT STATUS REPORT -

APPENDIX IV
FERMI 2: PROJECT STATUS REPOR

Purpose: The purpose of the meeting is to review the application of the Detroit Edison Company (Applicant) for a license to operate the Enrico Fermi Atomic Power Plant Unit 2. The Fermi-2 Subcommittee met on July 24, 1981 to review the OL application. Drs. Kerr and Zudans toured the facility on July 15, 1981.

Background: Pertinent facts concerning the Fermi Unit 2 Project include:

Location and Site: The plant is located on the western shore of Lake Erie in Frenchtown Tonwship of Monroe County Michigan about 30 miles southwest of downtown Detroit. The 120 acre site has an exclusion area radius of 915 meters (~2750 ft.). Condenser cooling water is taken from a reservoir on site and cooled by two parallel natural-draft cooling towers.. The Applicant has constructed an RHR Complex which contains 30-day water supply for the ultimate heat sink (see below).

The Fermi Unit 1 breeder reactor is also located on the site and has been decommissioned. There is a 165 MW(e) oil-fired generation station and four small peaking units on the site as well.

Plant: The NSSS is a GE BWR/4 housed in a Mark I containment generating a core power of 3292 MW(t) - 1154 MW(e). The 251-inch ID vessel will be loaded with 764 8x8 RP fuel assemblies (two water rods, natural uraninum top and bottom 6 inches of fuel rods, and slightly prepressurized). Maximum linear heat generation rate is 13.4 Kw/ft. The Mark I containment steel pressure vessel has a design pressure/temperature of 62 psig and 281°F, respectively. The plant design SSE is 0.15g; the OBE is 0.08g. Detriot Edison acted as their own AE with Sargent and Lundy employed as AE consultants. General Electric Turbine Generator Limited (England) supplied the turbine-generator.

Previous ACRS Review: The ACRS reviewed Fermi-2 for a CP license in March 1971. A copy of the Committee's CP letter is attached. Fermi-2 is similar in design to the Browns Ferry, Shorham and Hatch plants.

Subcommittee Review: Highlights of the July 24,.Subcommittee meeting are included in this Meeting Folder Section. The Subcommittee had no major reservations concerning the Project and the NRC review.

Highlights of Review Topics: The Tentative Schedule of Presentations for the meeting is attached. TMI-related topics will be the focus of the meeting: organization and management, operator training, control room redesign, hydrogen control, and human factors considerations. Other topics/plant features of interest include:

Mark I Containment Redesign: The Fermi Mark I containment was subject to extensive modification as a result of the higher than expected loads resulting from LOCA and/or SRV dischanges. The NRC USI for Mark I containments is considered resolved. NRC issued NUREG-0661 which delineates the acceptance criteria for resolution of this item. Fermi however, submitted an evaluation of their containment before the above NUREG was published. As a result, NRC is requiring a plant-unique analysis (PUA) to confirm the adequacy of the interim modifiations made. The PUA will be performed on the basis of NUREG-0661 and other criteria established as a result of the Mark I Program including some in-plant tests to confirm the SRV loads. Detroit Edison has committed to complete the PUA by May 1, 1982. This topic will be discussed at the meeting and is carried as an open item in the SER.

RHR Complex: The ultimate heat sink for Fermi-2 is the RHR complex. The complex has the capability of removing decay and residual heat under normal and accident conditions. Specificially, the RHR complex can cool the reactor by use of the RHR heat exchanger and the suppression pool. It can also cool the fuel storage pool. The complex contains two redundant divisions of equipment including the plant's diesel generators (2 in each division). The ultimate heat sink is two 3.4 million gallon reservoirs located in the "basement" of the complex. The reservoirs contain enough water to cool the plant for 30 days. Two induced-draft cooling towers per division on top of the complex are used for heat rejection.

The SER states that the complex was built to eliminate concerns expressed by the (then AEC) Staff and ACRS over the use of a planned cooling pond, given severe environmental conditions.

Unresolved Safety Issues (USI): There are 15 USIs listed as being applicable to Fermi. Eight of these issues are listed as resolved. The seven "unresolved" USIs are:

- Waterhammer
- ATWS
- Reactor Vessel Material Toughness
- Systems Interactions in Nuclear Power Plants
- Seismic Design Criteria
- Containment Emergency Sump Reliability
- Station Blackout.

In addition, four "new" USIs have been identified. These are:

- Shutdown Decay Heat Removal Requirements
- Seismic Qualification of Equipment in Operating Plants

°Safety Implications of Control Systems

°Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment.

We will discuss the majority of these issues at the meeting, either as open items or discussion items requested by the Subcommittee.

Open Items: The SER lists 22 open items (15 non-TMI, 7 TMI). As of the Subcommittee meeting there were 19 open items (see Meeting Summary). All of these items will be reviewed at the meeting. I wish to call the following items to your attention.

Seismic Reassessment of Structures, Systems, and Components Required for Safe Shutdown: As noted above, the SSE and OBE for Fermi is 0.15g and 0.08g, respectively. Fermi did not use the currently accepted Regulatory Guide 1.60 seismic design response spectrum for its original seismic design. NRC required that a new response spectrum be generated and applied to the analysis of structures, systems, and components required for safe shutdown. This NRC action is similar to what was required for the Sequoyah plant. Fermi has generated an acceptable plant-specific spectrum that is being used for the ongoing reassessment.

NRC has taken objection to some portions of the recently submitted reassessment reports. The Staff has requested the following of the applicant:

1. Analyze the buried piping and ducts using design parameters consistent with the new seismic input.

2. Combine the responses to the three components of earthquake motion using the SRSS rule in the analysis. Alternately, the absolute sum of one horizontal component and one vertical may be used.

This topic will be discussed at the meeting.

ATWS-Emergency Operating Procedure: Fermi-2 is committed to developing an ATWS procedure upon receipt of the GE Owner's Group "Reactivity Control Guidelines" which is still in preparation.

Break in Control Rod Scram Discharge Volume: This matter arose as a result of AEOD's recent report on this topic. The NRC has drafted a NUREG that addresses this topic on a generic basis. ACRS Fellow T. McKone has issued a memo discussing the status of this issue. A copy of his memo is attached.

Compliance With Appendices G&H to 10 CFR 50: NRC states that the Applicant has not demonstrated specific compliance with sections of the two subject appendices. The problem appears to be similar to that seen for other recent OL plants of this vintage i.e. the vessel was ordered and manufactured in accordance to the then-current

ASME Code requirements which preceeded publication of Appendices G&H. The Applicant plans to provide acceptable justifications to the sections in question by August 1, 1981.

Loss of Power to Instrument and Control Systems: NRC has requested that the Applicant provide procedures for attaining safe shutdown if power is lost to Class 1E or non-Class 1E buses suppling power to safety-or non-safety related instruments and controls. The Staff is also concerned whether the Applicant has addressed all the systems affected by this item.

Other Issues: The TMI-issues that are open appear for the most part to be the result of uncompleted NRC Staff review. Th: Applicant's FEMA drill for the emergency pl ı is scheduled for February 1982.

There are nine items listed as license conditions. Two items appear noteworthy. They are: (1) study of multiple control system failures. The Applicant must identify any power sources or sensors which supply power or signals to two or more control systems and show that failure of these sources or sensors will not result in consequences outside the bounds of Chapter 15 analyses or exceed the capability of the safety systems or the operator's ability to recover from the accident; (2) the licensee has taken exception to the NRC requirement for installation of in-core thermocouples. NRC has conditioned the license on their installation.

ENRICO FERMI 2 SUBCOMMITTEE MEETING
JULY 24, 1981
WASHINGTON, DC

PURPOSE

The purpose of the meeting was to review the application of Detroit Edison for a license to operate the Enric Fermi Unit 2 plant.

ATTENDEES:

Principal attendees of the meeting are noted below:

ACRS
W. Kerr, Chairman
M. Carbon
D. Moeller
J. Ray
I. Catton, Consultant
Z. Zudans, Consultant
P. Boehnert, Designated Federal Employee

NRC STAFF
L. Kintner
L. Phillips
J. Knight
E. Pedersen

DETROIT EDISON
L. Schuerman
W. Colbert
E. Griffing
E. Lusis
W. Jens
H. Talber
J. Green
L. Kanous
F. Gregor
T. McKelsey
D. Lehnert
Q. Duong
E. Page
E. Madsen
W. Hodges

GENERAL ELECTRIC
R. Hill

MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS:

1. Mr. Les Kintner (NRC LPM) presented an overview of the NRC OL review for Fermi 2. The plant is located on a 1120 acre site about 30 miles south of Detroit on Lake Erie. The BWR/4 generates a core power of 3292 MWt- 1134 MWe. He noted the OL review has been conducted in three periods: (1) from FSAR docketing in April 1975 to construction delay in September 1976; (2) from June 1978 to March 1979 (TMI-2 accident); and (3) from April 1981 to the present. There are 19 open items as of the Subcommittee meeting date (July 24, 1981 - Figures 1-2). Of these, 14 are scheduled to be addressed in an SER Supplement due on August 31, 1981 and 5 will be addressed prior to OL issuance. There are 7 items listed as license conditions (Figure 3). Fuel load is projected for November 1982, as is the OL issuance.

Discussion of the above open items led to the following questions (answers are provided in parenthesis): (1) Dr. Kerr asked how many NRC reviewers of plant emergency operating procedures are licensed operators. (One or two - most have nuclear Navy experience); (2) Dr. Moeller asked why the LaSalle routine release rate exceeds the Fermi routine release rate. (Fermi uses over two times more charcoal in its filter system - both release rates are acceptable to NRC.); (3) Mr. Ray asked what was the maximum plant power that can be generated with use of one of the two main cooling towers. (Plant suffers a derate of approxmately 33% or 700 MWe.)

2. Mr. H. Talber and Mr. W. Jens (Detroit Edison Vice Presidents) discussed the Applicant's organization and management structure. Both speakers emphasized the long history of Detroit Edison in development of nuclear power dating to the "Atoms for Peace Program" in the 1950's. Detroit Edison noted that since they are the designer and constructor of the plant, there will be extensive engineering experience in the operating organization.

   The Nuclear Operations (NO) and Safety Organizations (Figures 4 and 5) were highlighted. Mr. Jens noted the following significant organizational changes/actions:

   °Nuclear and fossile operations are completely separated.

   °All safety functions are controlled in the NO organization.

   °There is a strong emphasis on training.

   °A close relationship exists between designers and operators.

   °Detroit Edison has purchased a plant simulator.

   Detroit Edison will analyze all LERs that are screened by INPO/NSAC. In response to questions from Drs. Catton and Carbon, Mr. Jens said Detroit Edison does review the EPRI Notepad publication, but has not made a formal commitment to do a bulk review of LERs. The Nuclear Safety Committee will be composed of a Subcommittee of the Board of Directors. An Independent Review and Audit Group will review LERs and other information for potential USIs referred to it by the Onsite Review Organization (Figure 5.)

   NRC discussed concerns raised during its audit of the plant organization and management. One result will be the addition of a GE representative on each operating shift to address the Staff concern over lack of Detroit Edison's BWR operating experience.

3. Detroit Edison discussed their operator selection and training program. Detroit Edison seeks ex-nuclear Navy people preferably with eight years of experience and who are qualified engineering watch supervisors. Dr. Carbon asked if an operator is allowed to use his own judgment in

an emergency situation. Mr. Griffing (Detroit Edison) said the operator will generally follow procedures but the procedures allow some flexibility of action. Dr. Kerr asked how Detroit Edison decides if a candidate should become an operator or not. Detriot Edison replied that this is a group decision made at management level. Detroit Edison plans to use evaluation boards to aid this decision.

Detroit Edison discussed the use of a simulator in their training program. The simulator will be used for operator training and will also be used to train maintenance and I&C personnel. In addition, managers/supervisors will also train on the simulator.

The maintenance worker training program was described. All maintenance workers will be journeymen. Detroit Edison has a general maintenance journeyman program in which all personnel have a "primary" and "secondary" skill. This is designed to increase the job scope and enhance job satisfaction. Progression through the training program is tied to pay increases, and skill level evaluation is performance based.

4. The control room (CR) design was reviewed. Detroit Edison stated that they recognized early-on (1965) that man-machine interface is an important factor in CR design. The Fermit 2 CR makes extensive use of mimics, color coding, and shape coding. The CR was examined by the BWR Control Room Committee and NRC. No major human factor problems were found.

   CR habitability was reviewed. For accidents beyond the design basis accident, Detroit Edison said that the thyroid dose may double (if all the iodine is released) which would put CR doses at or near the NRC limit (30 rem). The whole body dose should be the same (100% of noble gas release already assumed) and the assumption of 1% of solids being airborne is believed by Detroit Edison to be a conservative upper limit.

5. The Subcommittee discussed installation of instrumentation to follow the course of a serious accident. The discussion centered on the requirement to install core thermocouples (T/C) in BWRs in general, and Fermi-2 in particular. GE made a presentation that argued against installing core T/Cs. The main points were: (1) T/Cs will not be useful for monitoring core cooling except when there is no ECC injection; (2) T/Cs are not cost effective and result in high man-rem dosages for installation and maintenance; (3) GE is conducting a probabilistic risk evaluation for installation of core T/Cs; (4) GE showed calculations that indicated the cost of T/C installation is high (~\$600,000 per plant) and the cost per man-rem is also very high (~\$6000/man-rem). Both Drs. Kerr and Zudans questioned and rejected the methodology used to obtain the cost-per-man-rem figure.

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Mr. Larry Phillips (NRC) discussed the requirement for incore T/Cs. Reasons given for requiring T/Cs for BWR's included: (1) diverse level indiction, (2) monitor core cooling effectiveness, and (3) operability of core spray. Dr. Carbon questioned the bases for item (3).

Dr. Moeller requested that in the near future NRC provide a written report detailing its bases for determining the costs (both $ and health) vs benefits for instrumentation required for a given plant system. The report should also discuss over what period of time costs are amortized.

6. Plant seismic design was discussed in the context of the NRC requirement for reanalysis of structures systems and components required for safe shutdown.

   On March 1981 NRC requested the seismic reanalysis using either the Regulatory Guide 1.60 spectrum shape anchored at 0.19G, or development of a site-specific ground response spectra representative of earthquake histories of magnitude 5.3 - 5.5 applicable to a rock site (Fermi-2 is founded on bedrock). For the reanalysis loss of all offsite power is assumed, but a LOCA is not assumed in combination with the earthquake. The reanalysis was completed and a final report was docketed with the Staff on July 15, 1981. Major conclusions of the study are: (1) the plant can be safety shutdown; (2) one cable tray hanger would see stresses slightly over yield; and (3) 25 items (not identified) require further evalution, requalification, retesting, or replacement.

   Mr. J. Knight (NRC) noted that an NRC review team conducted an onsite audit of the plant's seismic design last week. The preliminary conclusion of the audit team was that no major changes will be required and the reanalysis showed low stress levels in the equipment and structures.

7. Mr. W. Colbert (Detroit Edison) discussed the available normal and degraded decay heat removal modes. The normal mode is via the main condenser and, at lower power levels, the RHR shutdown system (Figure 6). In response to a question from Dr. Zudans, Detroit Edison noted that for various degraded cooling modes, a total of 8 pumps are available (2 divisions of low pressure core spray, or 2 divisions of RHR pumps - 2 pumps per division).

   In the above degraded cooling mode, the vessel is flooded to the main steam line and fluid is relieved through the S/RVs to the suppression pool from which vessel water makeup is also taken (Figure 7). Figure 8 illustrates core cooling for a LOCA situation. Detroit Edison noted that for LOCA cooling, only 1 of 4 RHR pumps and 2 of 4 service water pumps are required.

8. Station electrical power reliability was described by Mr. T. McKelvey (Detroit Edison). Fermi-2 has 2 separate offsite power systems: 120 KV and 345 KV system (Figure 8A). These systems are in turn interconnected

to at least 9 separate power stations and/or neighboring power grids. The two separate offsite systems are brought in to the plant by 5 individual lines via a 5 mile corridor. In response to questions by Mr. Ray, Detroit Edison said one 345 KV tower can disable 4 of the 5 lines into the plant, but the off-site system does remain stable under these conditions. The on site AC auxiliary system contains 2 divisions of emergency power with 2 diesel generators per division (Figure 9). The safety-related DC system consists of 2 divisions of 260/130 V batteries (Figure 10). There are three battery charges, one per each division, and a "floating spare". For a station blackout situation Detroit Edison calls on four combustion turbine generator peaking units on site. One of the four peakers is equipped for "black-start" capability.

In response to a question from Dr. Kerr, Detroit Edison said they estimate the probability of a 2 hour loss of all offsite power to be $\sim 10^{-5}$/year.

10. The status of the Mark I containment modification program was reviewed. Detroit Edison has completed most of the major modifications required as a result of the NRC generic program (Figure 11). The remaining modifications should be completed by October 1982 (Figure 12).

11. Detroit Edison's response to the requirements of NUREG-0588 - equipment environmental qualification - were discussed. The review is concentrating on equipment exposed to such harsh environments as LOCA and high energy line breaks both inside and outside primary containment. Detroit Edison will analyze, test, relocate, or change-out the impacted equipment as necessary. Edison is working with EPRI and the BWR Owner's Group on joint qualification programs of common items (used by 3 or more utilities). Detroit Edison hopes to complete this program by July 1982.

12. Mr. J. Green discussed hydrogen control measures. The containment will be inerted during operation. There is a H_2/O_2 monitoring system that is redundant (2 divisions) and is a Seismic Class I system (Figure 13). Two thermal recombiners are available, again this is an ESF system (Figure 14).

    In response to questions from Drs. Moeller and Kerr, Mr. Green said the purge system has debris screens on the exhaust openings to prevent clogging and that purging can be accomplished in about 6 hours if rapid containment access is necessary.

13. Detroit Edison discussed their ATWS emergency procedure. In the event of an ATWS, Detroit Edison relys on the above procedure, their SLCS system, and their highly trained operators. Detroit Edison noted that there is a company directive that emphasizes "safety first" above any economic considerations (in the context of SLCS actuation).

In response to questions from Dr. Kerr, Detroit Edison said their simulator will be used to test the ATWS procedure noted above. Detroit Edison also said that it would take 6-10 minutes to get to hot shutdown after SLCS activation.

Regarding the concerns raised by AEOD on the consequences of a pipe break in the SDV portion of the rod drive system, Detroit Edison said they are awaiting the issuance of a Staff NUREG that addresses the AEOD concerns and lists remedial actions to be taken by licensees. The Fermi-2 SDV system was described (Figure 14A).

14. Emergency planning for Fermi-2 was dsicussed. The principal emergency support facilities are the control room (CR), technical support center (TSC), operational support center (OSC) and emergency operations facility (EOF). Figure 15 shows the relative location of the CR, OSC and TSC. A unique item is the use of high-resolution color TV cameras in the CR to allow TSC personnel to monitor CR pannels. The EOF will be constructed on site about 3/4 mile from the plant.

15. The radiological emergency response plan was reviewed. The province of Ontario Canada just intersects the 10-mile emergency protection zone radius. In response to questions from Dr. Moeller, Detroit Edison said that FERMA handles US/Canadian coordination of emergency planning, and joint action is now underway in developing plan details. A FEMA respresentative also noted that a 1967 agreement between the US and Canada allows aircraft overflights in an emergency situation (plume tracking, etc.) as necessary. The NRC/FEMA-monitored full scale emergency plan exercise is scheduled for February 1982.

Dr. Moeller asked if there are any public drinking water intakes near the plant and what provisions exist for interdiction given a large radiological release into Lake Erie. Detroit Edison said that the Monroe County intake is near the site and the intake water is monitored via a sampling point on the pipe.

16. In response to an earlier question from Dr. Moeller, Mr. Kintner said there was a Staff differing technical opinion on fire protection provisions for the plant and this difference was resolved at the Branch level (details not discussed).

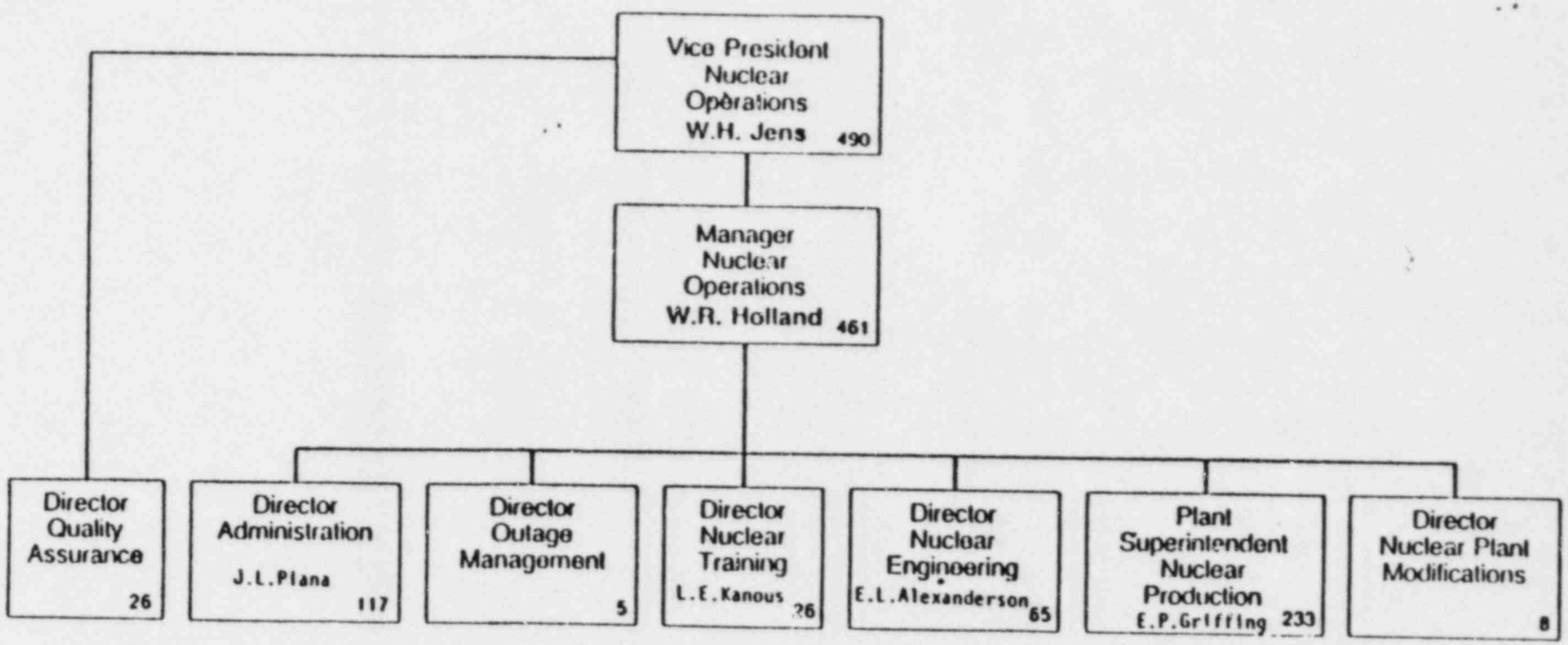
Dr. Carbon asked the NRC to address the significance of not allowing the Applicant to operate the plant under natural circulation conditions.

17. The Detroit Edison plant security plan was discussed in closed session. Detroit Edison described the security provisions installed on-stie. In response to a question from Dr. Carbon, Detroit Edison noted that the guard force will be comprised of Detroit Edison employees. Provisions for protection against insider sabotage were also discussed.

18. The Subcommittee recommended the Project be brought before the full Committee for review at the August meeting.

Nuclear Operations Organization

FIGURE 1.



*Not including 74 security and medical personnel

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FIG. 4

Nuclear Safety Organizations

FIGURE 5.

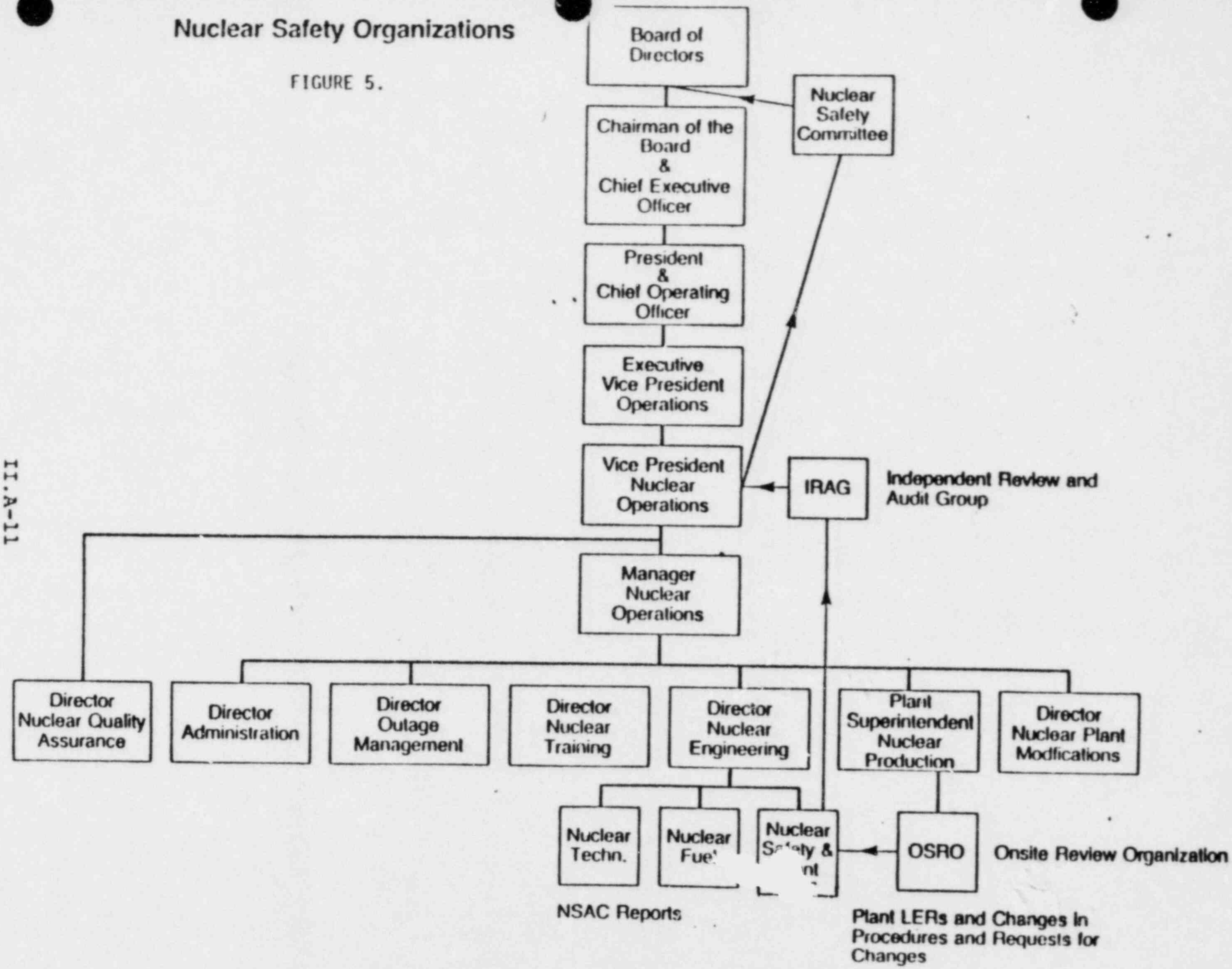

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FIG. 5

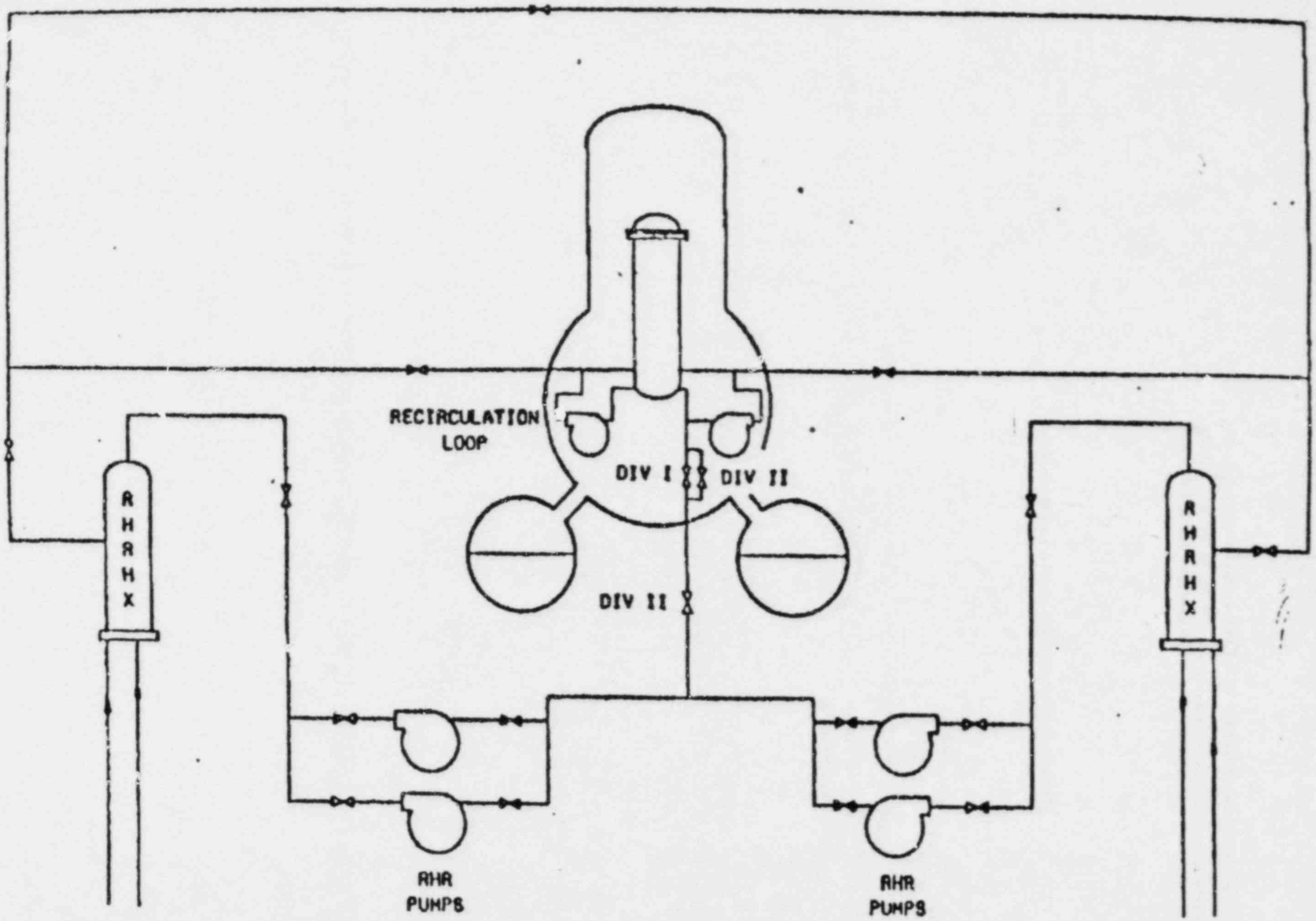


FIGURE 2: RHR SHUTDOWN COOLING MODE

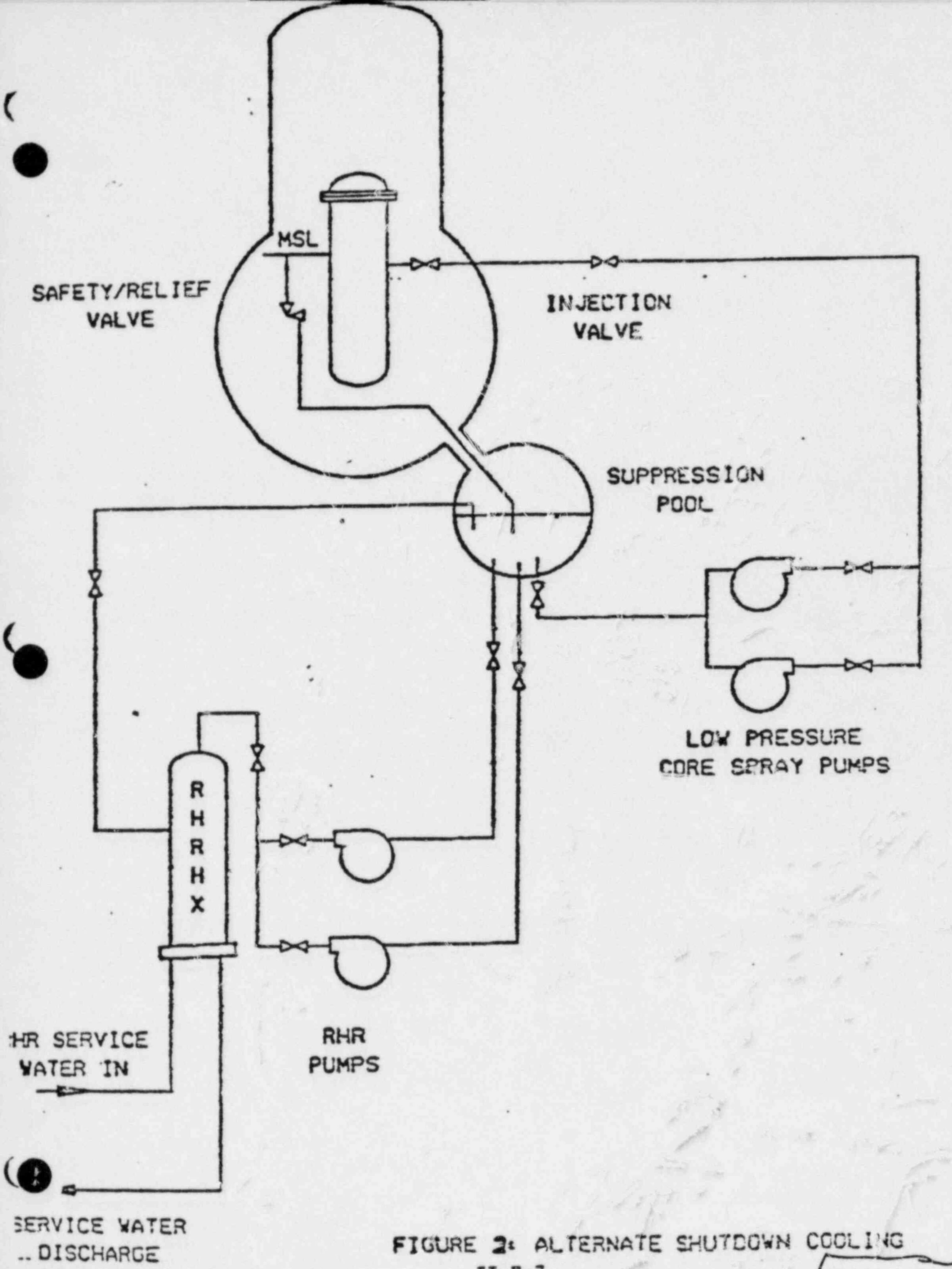
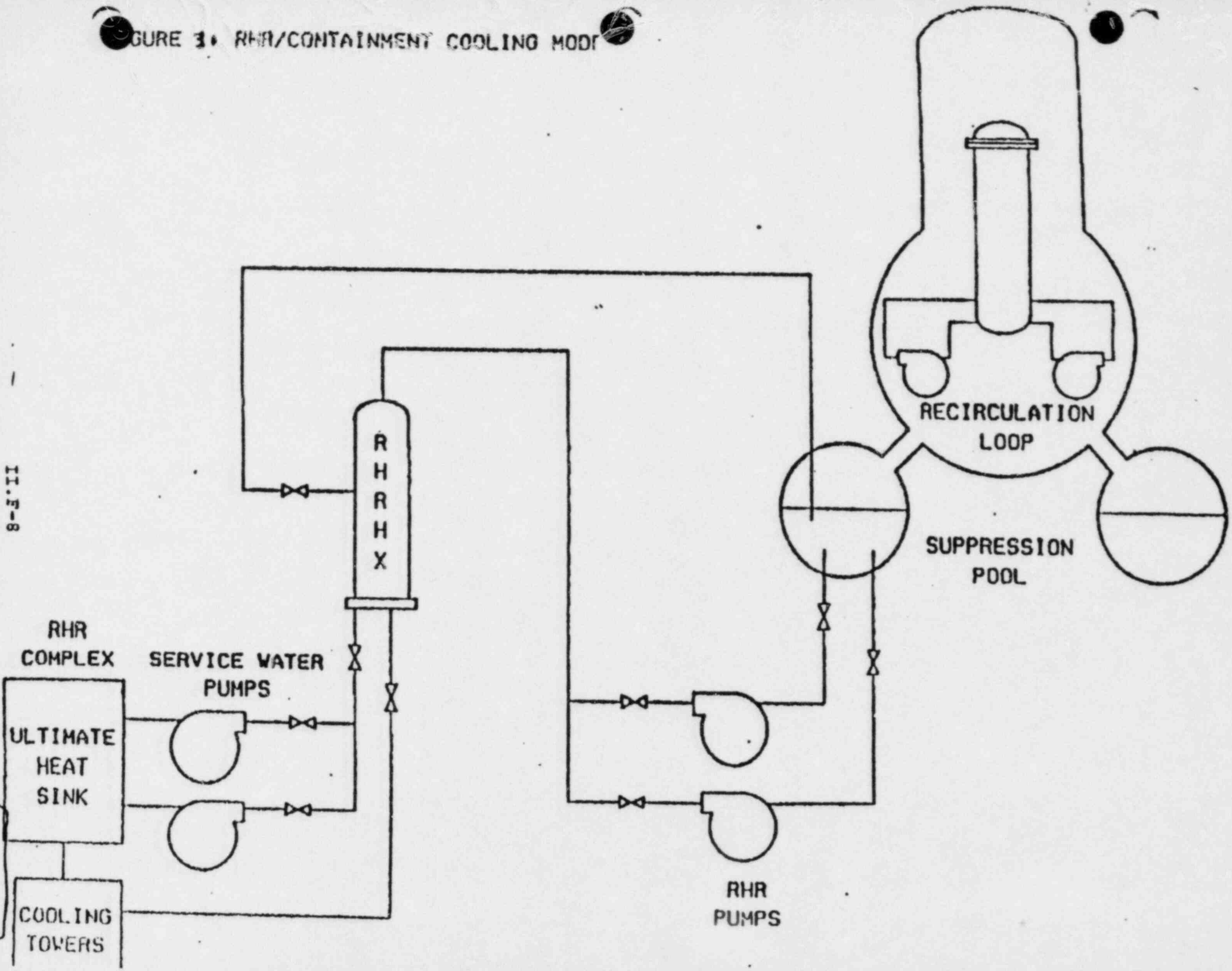


FIGURE 2: ALTERNATE SHUTDOWN COOLING

FIG. 7

GURE 3. RHR/CONTAINMENT COOLING MOD



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H.G-12

FIG. 8A

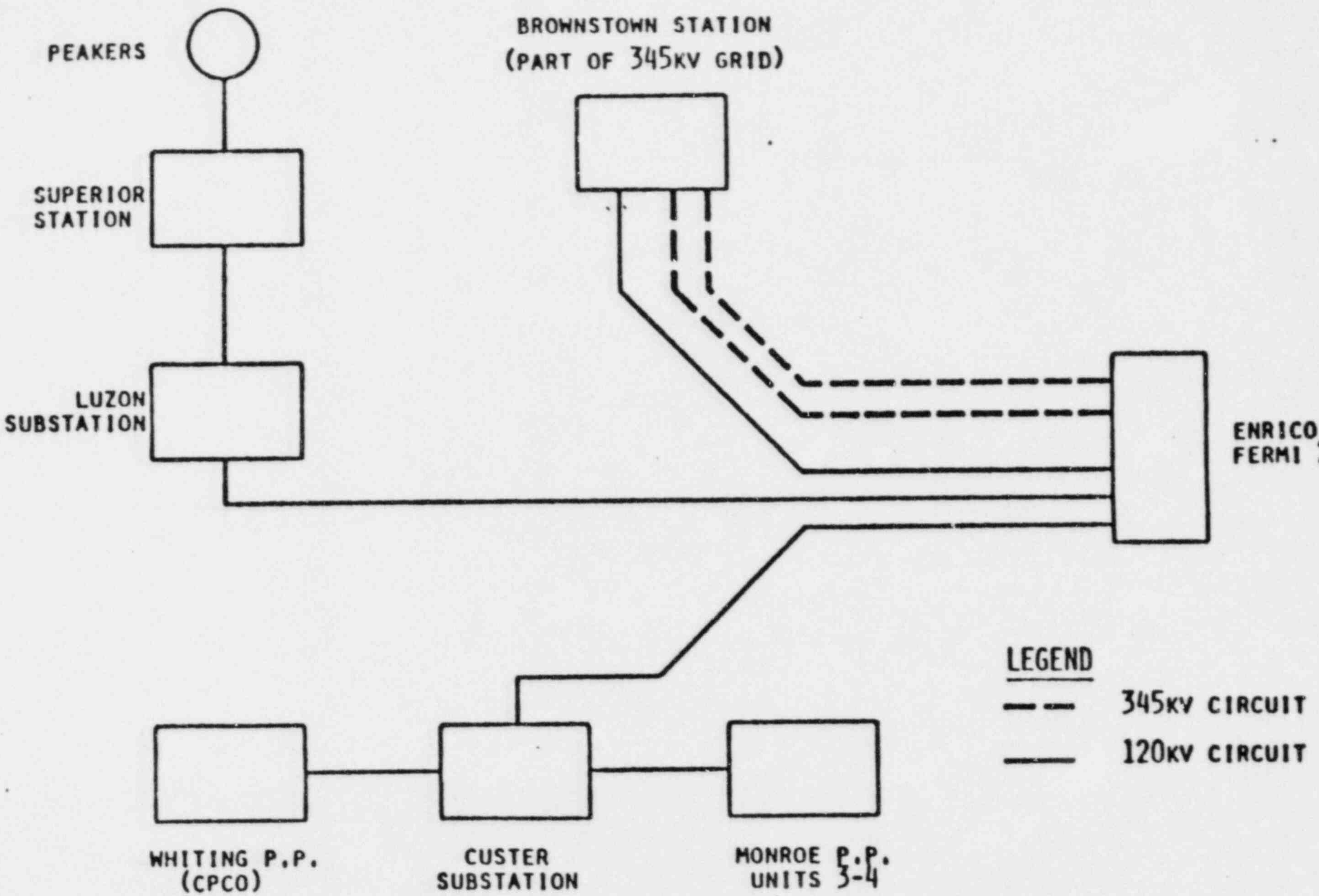

FIG. 1 ENRICO FERMI 2 OFFSITE POWER SOURCES

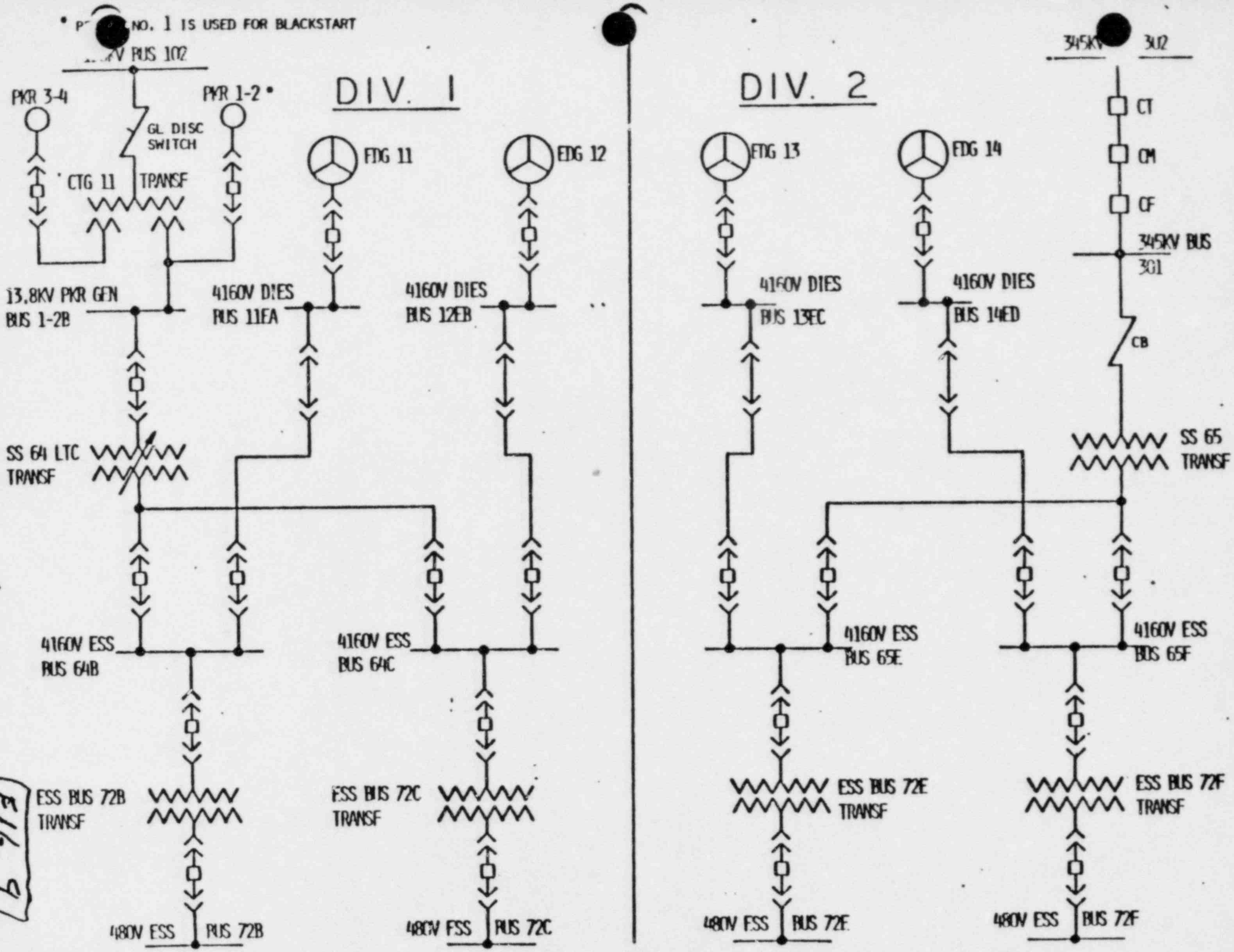

FIG. 9

FIG. 4 ENRICO FERMI 2 AUXILIARY ELECTRICAL SYSTEM

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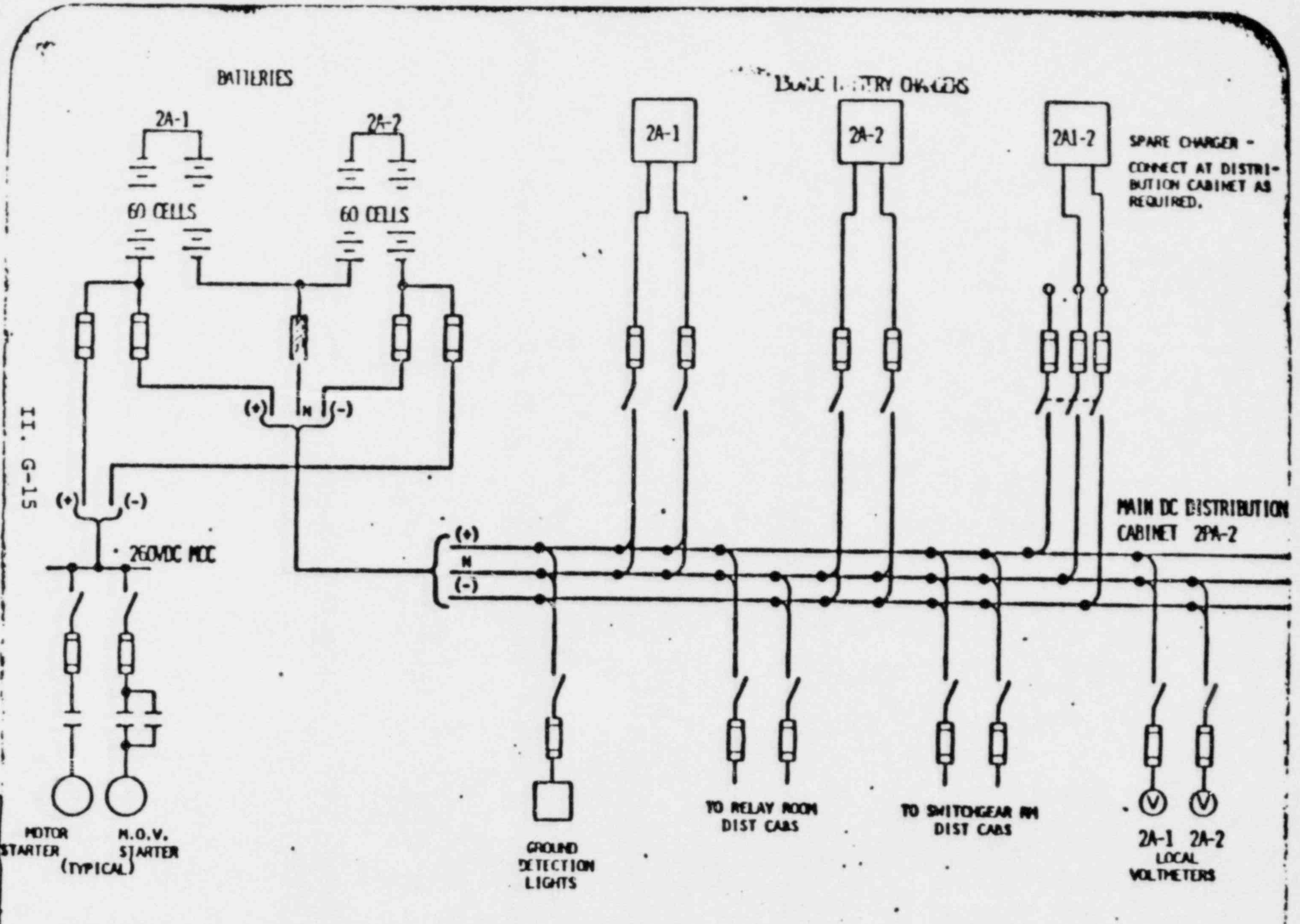

FIG. 5 ENRICO FERMI 2 260/130VDC DISTRIBUTION SYSTEM

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FIG. 10

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COMPLETED MODIFICATIONS

CATEGORY		DESCRIPTION		APPROX. MOD. DATES
MAJOR	TORUS	RING GIRDER REINFORCEMENT		6/79
MAJOR		COLUMN REINFORCEMENT		10/78
MAJOR		COLUMN CONNECTION REINFORCEMENT		12/79
MINOR	VENT SYSTEM	DOWNCOMER SHORTENING		2/80
MAJOR		VENT HEADER/DOWNCOMER STIFFENING & BRACING		11/78
MAJOR		REINFORCED EXISTING VENT SYSTEM COLUMNS & CONNECTIONS		2/79
MAJOR		VENT HEADER DEFLECTOR		2/80
MAJOR		VENT LINE/VENT HEADER STIFFENING		6/79
MAJOR		REINFORCED VACUUM BREAKER TO VENT HEADER CONNECTION		7/79
MAJOR	INTERNAL STRUCTURES	MONORAIL	ADDITIONAL SUPPORTS	5/78
MAJOR			STRENGTHEN EXISTING SUPPORTS	5/78
MINOR			EXTENDING MONORAIL	5/78
MAJOR		CATWALK	ADDITIONAL SUPPORTS	8/78
MAJOR			GRATING (DELIVER TO SITE)	3/80
MAJOR	SRV PIPING	REROUTED PIPING IN WETWELL		4/80
MAJOR		ADDITIONAL WETWELL SUPPORTS		4/79
MAJOR		REINFORCED V. L. PENETRATION		11/78
MAJOR		ADDED QUENCHER/RAMSHEAD SUPPORTS		1/80
MINOR	TORUS ATTACHED PIPING	ADDED TORUS INTERNAL SUPPORTS		4/80

CURRENT MODIFICATIONS

CATEGORY		DESCRIPTION	EST. COMPLETION	REMARKS
MAJOR	TORUS	MITERED JOINT SADDLES	2/82	FABRICATIONS AND FIELD WORK UNDERWAY
MAJOR		ADDITIONAL COLUMN ANCHOR BOLTS	4/82	FABRICATION AND FIELD WORK UNDERWAY
MAJOR	SRV PIPING	QUENCHER	9/82	MATERIAL ORDERED
MAJOR		QUENCHER SUPPORTS	9/82	MATERIAL ORDERED
MINOR	INTERNAL STRUCTURES	CATWALK GRATING INSTALLATION	9/32	

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FIG. 12

ROGEN / OXYGEN MONITORING

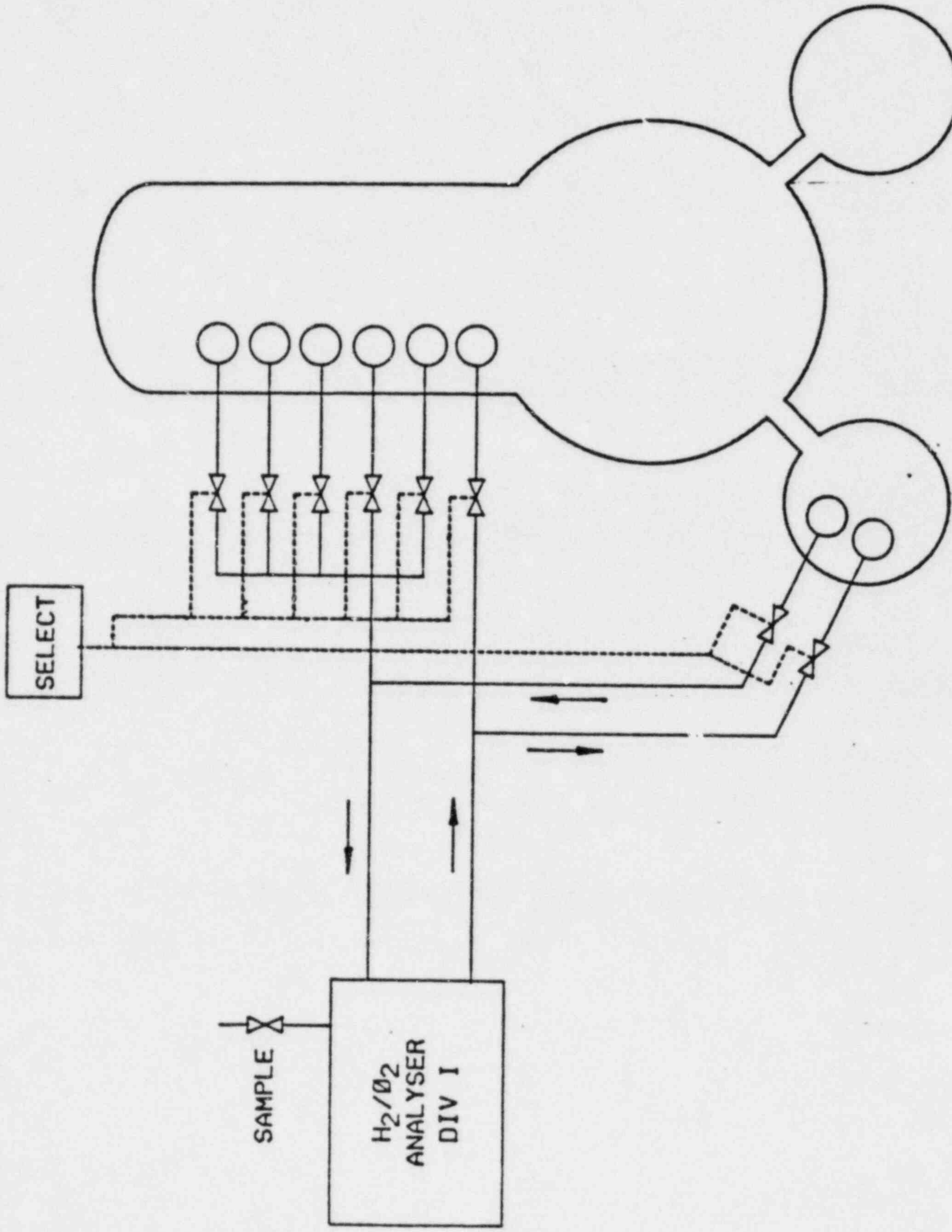
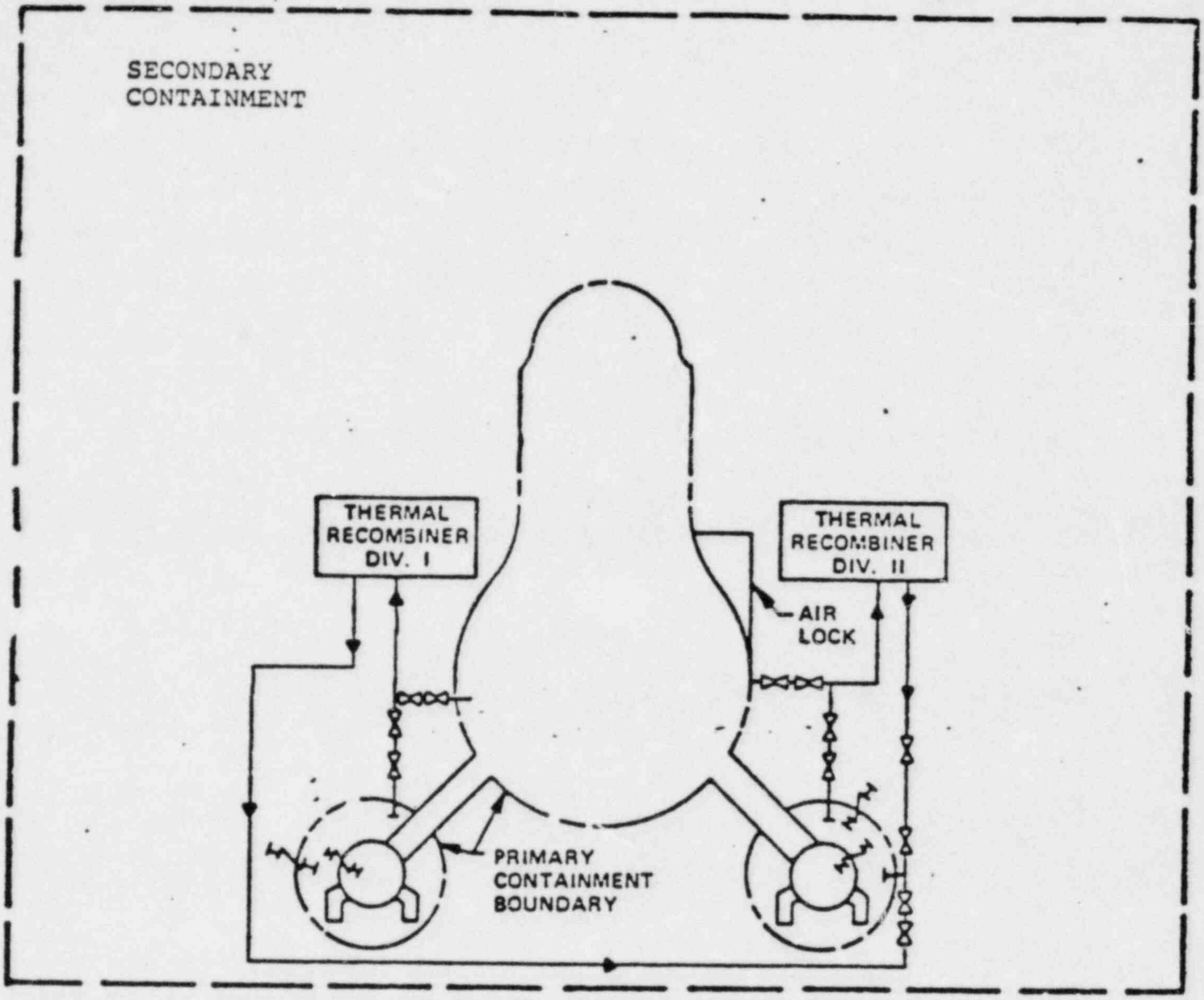
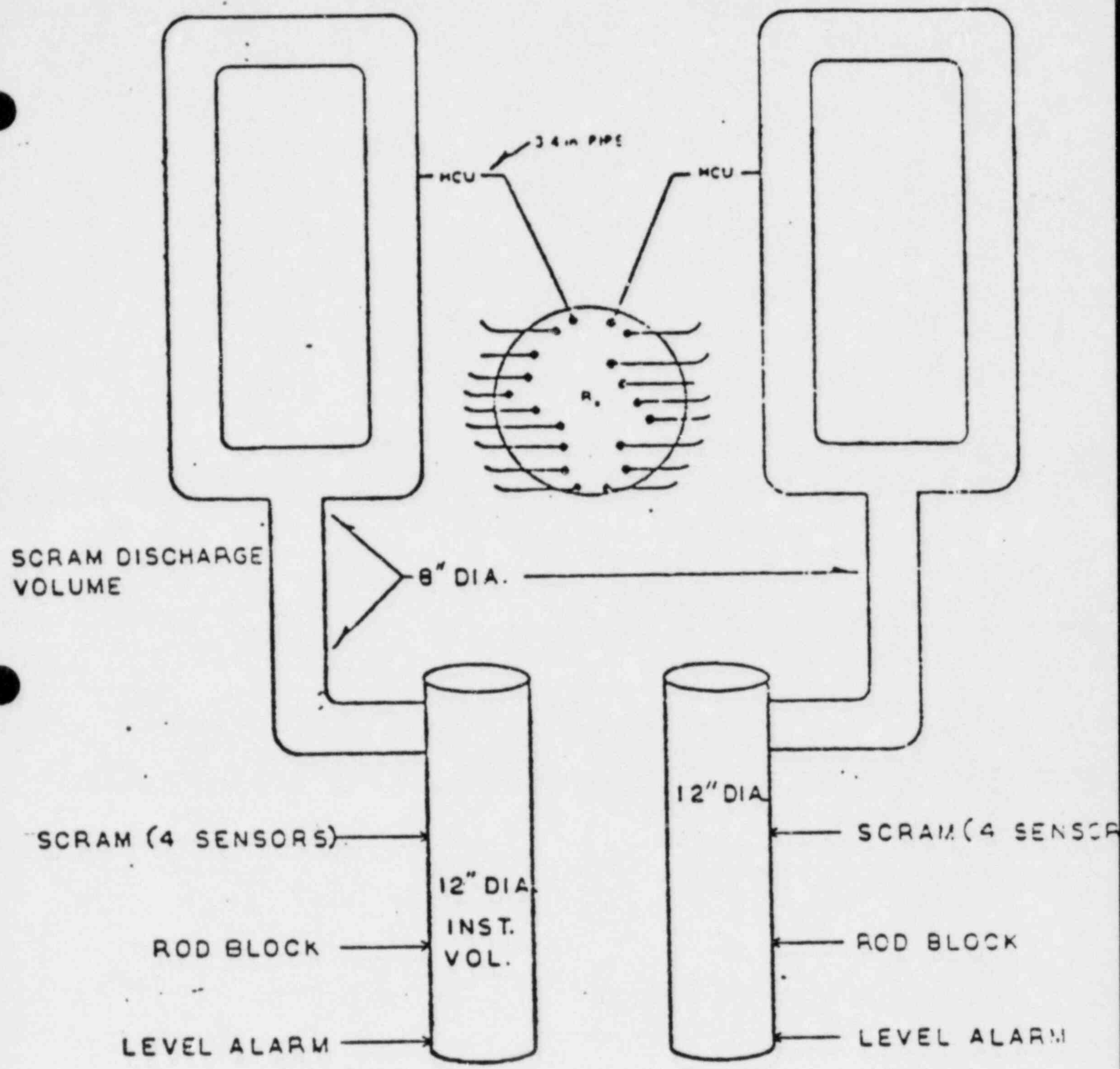


FIG. 13



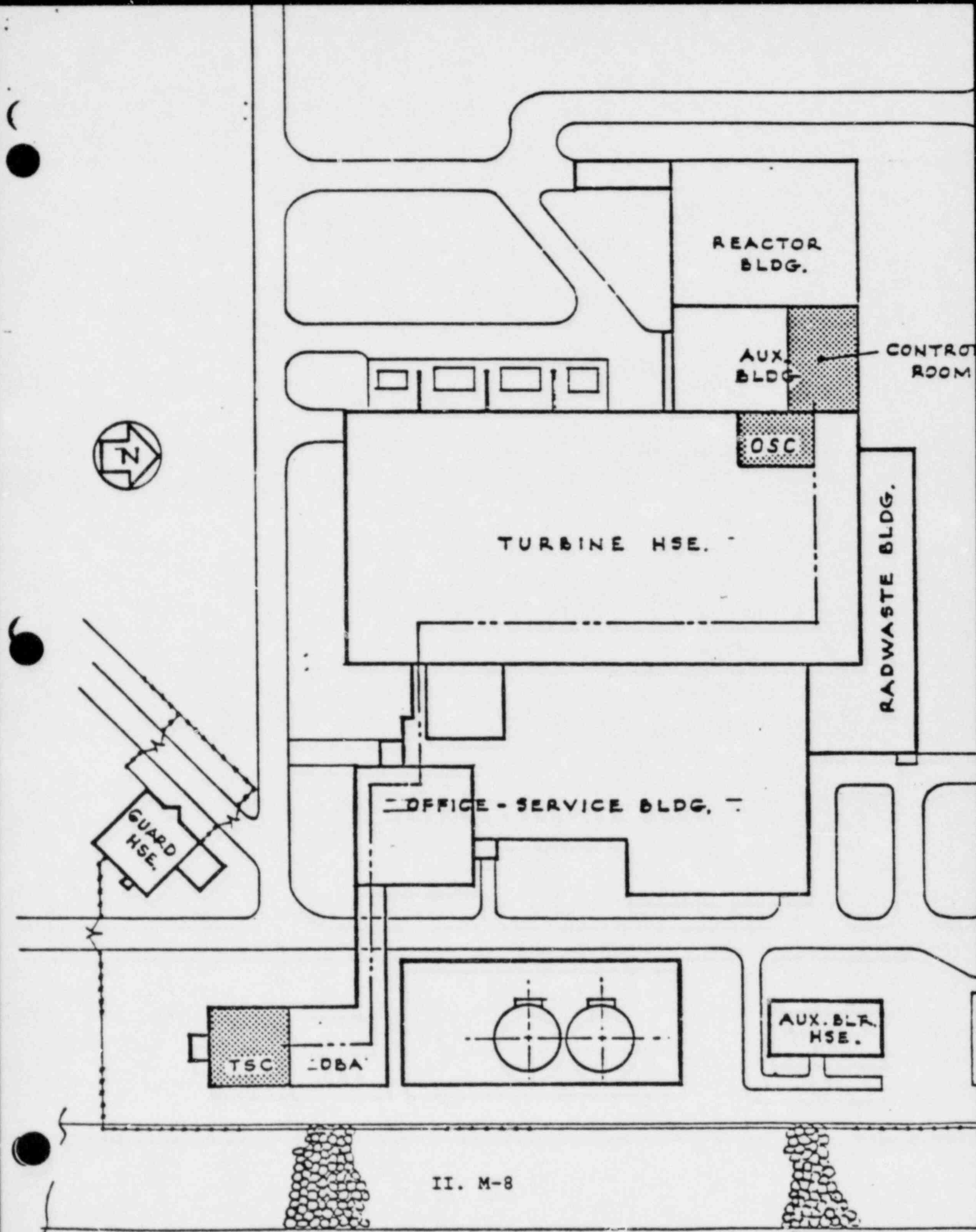
FERMI 2 THERMAL RECOMBINERS

FIG 14



FIRMI 2 SCRAM DISCHARGE VOLUME

FIG 14A



II. M-8

FIGURE A-60

FIG 15

A.2 SER OPEN ISSUES TO BE COMPLETED IN SSER (AUGUST 31, 1981)

- CONFORMANCE TO 10 CFR 20, 50, 100
- SEISMIC REASSESSMENT OF DESIGN MARGIN
- PRESERVICE TESTING OF PUMPS & VALVES
- SEISMIC QUALIFICATION REVIEW TEAM AUDIT
- BURIED PIPE FOUNDATION CONDITIONS
- CONFORMANCE TO APPENDIX G & H, 10 CFR 50
- CONTAINMENT LEAKAGE TESTS
- PROCEDURE FOR TESTING RHR ISOLATION VALVE INTERLOCKS
- LOSS OF INSTRUMENTATION & CONTROL POWER (IE BULLETIN 79-27)
- FIRE PROTECTION (CONTROL ROOM)
- PHYSICAL SECURITY PLAN
- EMERGENCY OPERATING PROCEDURES (I.C.1, I.C.8, ATWS)
- FEEDBACK OF OPERATING EXPERIENCE (I.C.5)
- CONTROL ROOM DESIGN (I.D.1)
- DEGRADED CORE TRAINING (II.B.4)
- CONTAINMENT PURGE OPERABILITY (II.E.4.2)

FIG. 1

TO BE COMPLETED PRIOR TO OL ISSUANCE (NOV. 1982)

- DESIGN OF MODIFICATION TO DIESEL ENGINES
- ENVIRONMENTAL QUALIFICATION OF EQUIPMENT
- UPGRADED EMERGENC: PREPAREDNESS
  (III.A.1.1, III.A.1.2, III.A.2)
- MARK I CONTAINMENT
  PLANT UNIQUE ANALYSIS
  TORUS - ATTACHED PIPING ANALYSIS
- SAFETY ANALYSIS AND PROCEDURES FOR TRAINING DURING LOW
  POWER TESTING (I.G.1)

FIG. 2

TO BE COMPLETED AFTER LICENSE ISSUANCE (LICENSE CONDITIONS)

- ANALYSIS OF FISSION GAS IN FUEL
- TESTS OF FUEL CHANNEL BOX DEFLECTION
- ANALYSES OF HYDRODYNAMIC STABILITY
- ANALYSIS OF MULTIPLE CONTROL SYSTEM FAILURES
- ANALYSIS OF EFFECT ON HIGH ENERGY LINE BREAK ON CONTROL SYSTEMS
- INSPECTION OF LOW PRESSURE TURBINE DISCS
- DESIGN OF POST ACCIDENT SAMPLING SYSTEM
- DESIGN OF INSTRUMENTATION FOR INADEQUATE CORE COOLING

FIG. 3

FERMI 2 ACRS MEETING

AUGUST 6, 1981

SER OPEN ITEMS

- ITEMS THAT HAVE BEEN CLOSED
- OPEN ITEMS TO BE CLOSED IN SSER
- OPEN ITEMS TO BE CLOSED PRIOR TO OL ISSUANCE
- LICENSE CONDITIONS

ISSUES THAT HAVE BEEN CLOSED

- BURIED PIPE FOUNDATION CONDITIONS
- PROCEDURE FOR TESTING RHR INTERLOCKS
- EFFECT OF HIGH ENERGY LINE BREAK ON CONTROL SYSTEMS
- FEEDBACK OF OPERATING EXPERIENCE (I.C.5)

OPEN ISSUES TO BE CLOSED IN SSER (Aug ~~SEPT. 30~~, 1981)

- CONFORMANCE TO 10 CFR 20, 50, 100 *
- SEISMIC REASSESSMENT OF DESIGN MARGIN Resolved
- PRESERVICE TESTING OF PUMPS AND VALVES Resolved
- SEISMIC AND DYNAMIC QUALIFICATION *
- CONFORMANCE TO APPENDIX G & H Resolved
- CONTAINMENT LEAKAGE TESTS *
- LOSS OF INSTRUMENTATION & CONTROL POWER *
- FIRE PROTECTION (CONTROL ROOM) *
- PHYSICAL SECURITY PLAN *
- EMERGENCY OPERATING PROCEDURES (I.C.1, I.C.8, ATWS) *
- CONTROL ROOM DESIGN (I.D.1) *
- DEGRADED CORE TRAINING (II.B.4) *
- TURBINE TRIP ANALYSIS INCLUDING EFFECT OF REHEATER *
- CONTAINMENT PURGE VALVE OPERABILITY AUDIT (II.E.4.2) *

Rest expected to be resolved Aug. 31, 1981

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* ADDITIONAL INFORMATION NEEDED FROM APPLICANT

OPEN ISSUES TO BE CLOSED PRIOR TO
OL ISSUANCE (NOVEMBER 1982)

- MODIFICATIONS TO DIESEL ENGINE LUBRICATION
- MARK I CONTAINMENT
  PLANT UNIQUE ANALYSIS
  TOURS - ATTACHED PIPING
- SAFETY ANALYSIS AND PROCEDURES FOR SPECIAL LOW POWER TESTS (I.G.1)
- UPGRADED EMERGENCY PREPAREDNESS (III.A.1.1, III.A.1.2, III.A.2)
- ENVIRONMENTAL QUALIFICATION OF EQUIPMENT
- BREAKS IN CONTROL ROD DRIVE DISCHARGE VOLUME

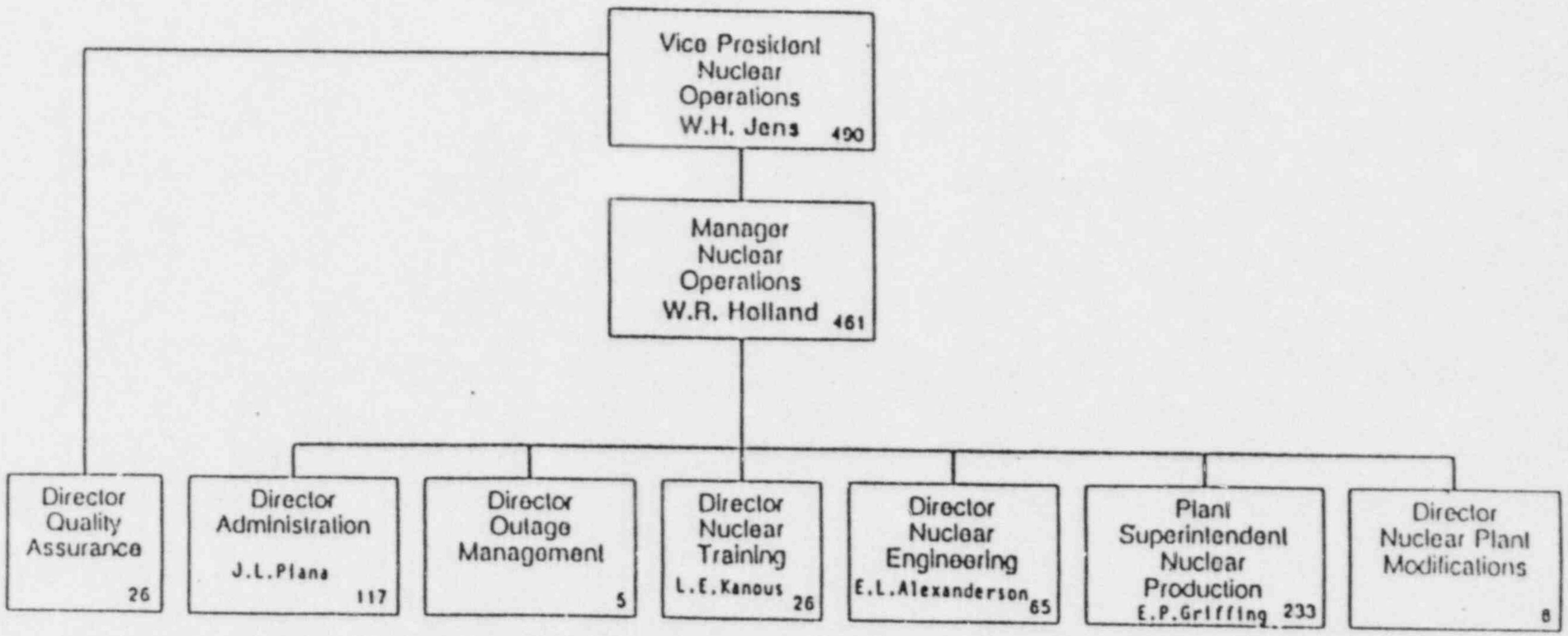
OPEN ISSUES TO BE CLOSED

AFTER LICENSE ISSUANCE (LICENSE CONDITIONS)

- ANALYSIS OF FISSION GAS IN FUEL
- TESTS OF FUEL CHANNEL BOX DEFLECTION
- ANALYSES OF HYDRODYNAMIC STABILITY
- ~~ANALYSIS OF MULTIPLE CONTROL SYSTEM FAILURES~~
- INSPECTION OF LOW PRESSURE TURBINE DISCS
- DESIGN OF POST ACCIDENT SAMPLING SYSTEM
- DESIGN OF INSTRUMENTATION FOR INADEQUATE CORE COOLING

III.A. ORGANIZATION AND MANAGEMENT

Nuclear Operations Organization FIGURE 1



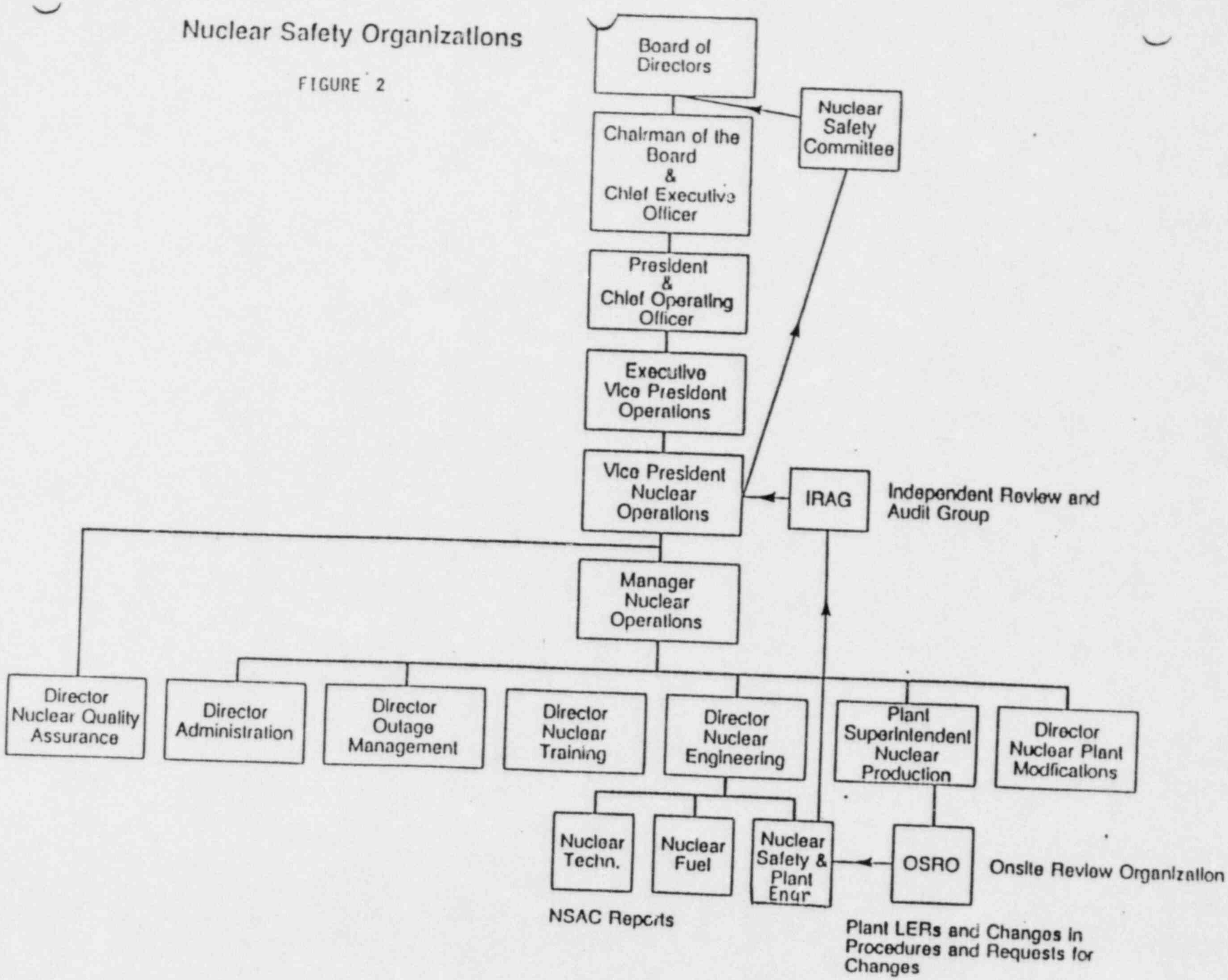
*Not including 74 security and medical personnel

A-70

III.A-1

Nuclear Safety Organizations

FIGURE 2



III.A-2

A-71

ORGANIZATION AND MANAGEMENT

TODAY MARKS A VERY LONG STRUGGLE TO REACH THIS MAJOR MILESTONE TOWARD OPERATING FERMI 2 SUCCESSFULLY. WE APPROACH IT WITH CONFIDENCE AND WITH RESPECT SINCE IT MEANS SO MUCH TO MANY OF US WHO HAVE VIRTUALLY MADE A CAREER OF THIS NUCLEAR PLANT. WE ALL KNOW HOW DEPENDENT OUR COMPANY IS ON THE SAFE AND SUCCESSFUL OPERATION OF FERMI 2.

TO EXPLAIN THE NUCLEAR OPERATIONS ORGANIZATION, I WOULD LIKE TO DEVELOP THE HISTORY OF DETROIT EDISON'S INVOLVEMENT IN NUCLEAR POWER. OUR COMPANY WAS ONE OF THE UTILITIES INVOLVED IN THE EARLY POWER DEMONSTRATION PROGRAM UNDER THE ATOMS FOR PEACE PROGRAM. INITIALLY WE WERE ASSOCIATED WITH THE DOW CHEMICAL COMPANY IN THE STUDY OF THE APPLICATION OF NUCLEAR POWER TO OUR RESPECTIVE BUSINESSES. THESE STUDIES LED ULTIMATELY TO THE DEVELOPMENT, THE DESIGN AND THE CONSTRUCTION OF THE FERMI 1 PLANT. TWO ORGANIZATIONS WERE INCORPORATED TO CARRY OUT THAT PROJECT. ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. DEVELOPED THE TECHNOLOGY AND SERVED AS THE LEAD SYSTEM DESIGNER, AND POWER REACTOR DEVELOPMENT COMPANY BUILT AND OPERATED THE PLANT.

EARLY IN 1970, AFTER FERMI 1 HAD SUCCESSFULLY OPERATED FOLLOWING THE FUEL MELTING ACCIDENT AND WAS THEM DECOMMISSIONED, MANY OF THE ENGINEERS RETURNED TO DETROIT EDISON TO BECOME INVOLVED IN THE DESIGN OF FERMI 2.

OUR RECENTLY-ELECTED CHIEF EXECUTIVE OFFICER, MR. WALTER MC CARTHY, PLAYED A LEADING ROLE IN FERMI 1. BEFORE RETURNING TO DETROIT EDISON, HE WAS THE GENERAL MANAGER OF POWER REACTOR DEVELOPMENT COMPANY. HE WAS THE FIRST PROJECT MANAGER OF FERMI 2.

I SIMILARLY SERVED IN SEVERAL MANAGEMENT POSITIONS ON THE FERMI 1 PROJECT AND, BEFORE RETURNING TO DETROIT EDISON, I WAS THE GENERAL MANAGER OF ATOMIC POWER DEVELOPMENT ASSOCIATES, INC. UPON RETURNING TO DETROIT EDISON, I SUCCEEDED MR. MC CARTHY AS THE SECOND PROJECT MANAGER OF FERMI 2.

EARLY IN 1972, IN ADDITION TO OUR INVOLVEMENT IN FERMI 2 AS THE LEAD DESIGN ORGANIZATION, OUR COMPANY INITIATED WORK ON A SECOND UNIT AT FERMI, FERMI 3, WHICH ORIGINALLY WAS TO BE A DUPLICATE OF FERMI 2. DURING THE PERIOD FERMI 3 WAS BEING DESIGNED AS A DUPLICATE, I WAS PROJECT MANAGER AND BOTH FERMI 2 AND 3 WERE MANAGED BY THE SINGLE PROJECT MANAGEMENT ORGANIZATION.

WHEN FERMI 3 WAS CONVERTED TO A BWR 6 MARK III CONTAINMENT, WE DECIDED THAT A SEPARATE PROJECT MANAGEMENT ORGANIZATION WAS REQUIRED. MR. WILLIAM FAHRNER BECAME THE PROJECT MANAGER, AND EBASCO WAS HIRED AS THE LEAD ENGINEER. WHEN FERMI 3 WAS CANCELLED IN 1975, DUE TO A FINANCIAL CRISIS, MR. FAHRNER SHORTLY THEREAFTER BECAME THE PROJECT MANAGER FOR FERMI 2.

OUR OTHER INVOLVEMENT IN NUCLEAR POWER ALSO OCCURRED IN 1972, WITH THE INITIATION OF THE DESIGN OF TWO NUCLEAR UNITS, GREENWOOD 2 AND 3, AT OUR GREENWOOD ENERGY CENTER. THESE UNITS WERE BEING ENGINEERED BY BECHTEL. THE UNITS WERE TERMINATED IN 1980.

IN ORDER TO DISCHARGE ITS RESPONSIBILITIES, OUR COMPANY NOT ONLY HAD TO STAFF FOR ITS DIRECT RESPONSIBILITY FOR FERMI 2, BUT OUR COMPANY ALSO HAD TO MANAGE THESE OTHER NUCLEAR PROJECTS BEING DESIGNED BY OUTSIDE A/E's. OUR INVOLVEMENT CONSISTED OF WRITING THE SPECIFICATIONS FOR THE PLANT'S PERFORMANCE, CONDUCTING THE LICENSING WORK AND EVALUATING THE DESIGN WORK BEING DONE FOR THE PROJECT. TO CARRY OUT THESE RESPONSIBILITIES, WE HAD TO HIRE AND TRAIN AN ADDITIONAL TECHNICAL STAFF.

FOLLOWING THE TERMINATION OF THE WORK ON FERMI 3 AND GREENWOOD 2 AND 3, EDISON PERSONNEL INVOLVED IN THESE PROJECTS THEN BECAME A VERY VALUABLE RESOURCE TO HELP THE PROJECT MANAGEMENT ORGANIZATION COMPLETE THE DESIGN AND CONSTRUCTION OF FERMI 2. THESE HUMAN RESOURCES WILL ALSO BE VALUABLE IN THE OPERATION OF THE FERMI 2 PLANT.

ONE OF THE OTHER VALUABLE CONSEQUENCES OF CANCELLING FERMI 3 AND GREENWOOD 2 AND 3 IS THAT THE COMPANY HAS ONLY ONE PURPOSE IN NUCLEAR POWER AT THIS TIME. THAT IS THE SUCCESSFUL COMPLETION OF FERMI 2 AND ITS SUCCESSFUL AND SAFE OPERATION. THERE IS NO OTHER DIVERSION OF THIS TALENT, AND WE CAN CONCENTRATE ALL THE RESOURCES ON THIS ONE NUCLEAR PLANT.

A LITTLE OVER A YEAR-AND-A-HALF AGO, WHEN I HELD A POSITION VERY SIMILAR TO MR. HARRY TAUBER (I WAS THE ASSISTANT VICE PRESIDENT-ENGINEERING AND CONSTRUCTION AND THE FERMI 2 PROJECT MANAGER REPORTED TO ME), OUR COMPANY ASKED ME TO EVALUATE AND RECOMMEND WHAT IT SHOULD DO ABOUT THE OPERATION OF FERMI 2 IN VIEW OF THE EXPRIENCE AT THREE MILE ISLAND AND OTHER NUCLEAR PLANTS. UP TO THAT TIME, IT HAD BEEN PLANNED THAT FERMI 2 WOULD BE OPERATED AS ANOTHER ONE OF OUR POWER PLANTS IN THE PRODUCTION DEPARTMENT.

TO MAKE THE STUDY, I FELT THAT I NEEDED OUTSIDE HELP AND I ENGAGED TWO CONSULTANTS. ONE OF THE CONSULTANTS WAS MR. LOUIS RODDIS, WHO WAS INTIMATELY INVOLVED IN THE THREE MILE ISLAND RECOVERY AND REORGANIZATION OF GENERAL PUBLIC UTILITIES. HE ALSO SERVED AS THE DIRECTOR OF THE DIVISION OF REACTOR DEVELOPMENT IN THE ATOMIC ENERGY COMMISSION, VICE CHAIRMAN OF THE BOARD OF CONSOLIDATED EDISON AND PRESIDENT OF PENNSYLVANIA ELECTRIC, ONE OF THE GENERAL PUBLIC UTILITIES COMPANIES.

THE SECOND CONSULTANT WAS MR. LAWRENCE MINNICK, FORMER PRESIDENT OF YANKEE ATOMIC ELECTRIC, WHO I FELT REPRESENTED A UNIQUE ORGANIZATION INVOLVED WITH THE DESIGN, TESTING AND OPERATION OF SEVERAL VERY SUCCESSFUL NUCLEAR PLANTS. BOTH CONSULTANTS WERE INVALUABLE RESOURCES IN DEVELOPING THE ORGANIZATION FOR THE OPERATION OF FERMI 2, OUR ONLY NUCLEAR UNIT.

OUR COMPANY IS TECHNICALLY VERY STRONG. WE HAVE 11,000 EMPLOYEES AND 1,100 OF THOSE ARE PROFESSIONALS. WE HAVE, THROUGH THE DESIGN OF FERMI 2, DEVELOPED AN IN-DEPTH CAPABILITY IN NUCLEAR PLANT DESIGN. WE HAVE A STRONG ENGINEERING RESEARCH DEPARTMENT THAT IS ORIENTED TO PROBLEM SOLVING FOR OUR OPERATING PLANTS AND ELECTRICAL SYSTEM. THE DEPARTMENT HAS TESTING LABORATORIES TO ASSIST IN ITS PROBLEM SOLVING FUNCTION. THESE RESOURCES WERE ALL CONSIDERED TO BE AVAILABLE FOR THE OPERATION OF FERMI 2.

THE STUDY LED TO CERTAIN CONCLUSIONS AND RECOMMENDATIONS. ALL WERE ACCEPTED AND ARE NOW BEING IMPLEMENTED.

WE RECOMMENDED THAT WE SEPARATE OUR NUCLEAR FROM OUR FOSSIL OPERATION, THAT WE INTEGRATE ALL OF OUR NUCLEAR ACTIVITIES UNDER ONE ORGANIZATION WHEN FERMI 2 IS PLACED IN OPERATION,

THAT ALL SAFETY-RELATED FUNCTIONS BE CONTROLLED BY THE NUCLEAR ORGANIZATION AND THAT GREAT EMPHASIS BE GIVEN TO TRAINING IN THE NEW ORGANIZATION. THIS MEANT THAT THE TRAINING FUNCTION HAD TO BE ELEVATED TO A HIGH LEVEL IN THE ORGANIZATION.

FURTHER, SINCE WE ONLY HAD ONE NUCLEAR PLANT TO FOCUS ATTENTION ON, IT PERMITTED US TO LOCATE THOSE HUMAN RESOURCES AT THE PLANT SITE. THE LOCATION OF THE NUCLEAR ENGINEERING STAFF AT THE PLANT WOULD ALLOW US TO DEVELOP A VERY INTIMATE RELATIONSHIP BETWEEN THOSE ENGINERS WHO DESIGNED AND SPECIFIED THE PLANT AND WHO KNEW HOW THEY INTENDED IT TO PERFORM WITH THOSE ENGINEERS AND OPERATORS WHO WILL HAVE TO TEST, OPERATE AND MAINTAIN THE PLANT. AS AN EXAMPLE OF THE IMPORTANCE OF DEVELOPING THIS RELATIONSHIP BETWEEN THESE TWO GROUPS, WE PUT TOGETHER A TEAM OF OPERATING AND DESIGN ENGINEERS, INCLUDING SHIFT SUPERVISORS AND THE PLANT SUPERINTENDENT, TOGETHER WITH OUR PROJECT ENGINEER AND MANY OF OUR SYSTEM ENGINEERS UNDER THE DIRECTION OF MR. WILLIAM COLBERT. THE TEAM TOGETHER MADE AN INDEPENDENT SAFETY REVIEW OF THE FERMI 2 DESIGN FOLLOWING THE ACCIDENT AT THREE MILE ISLAND. THE VALUE OF DEVELOPING THIS INTIMATE RELATIONSHIP WAS DEMONSTRATED DURING THAT STUDY IN CONVEYING INFORMATION FROM THOSE WHO DESIGNED THE PLANT TO THOSE WHO WILL HAVE TO OPERATE IT. IT IS INTENDED THAT THIS RELATIONSHIP BE CONTINUED THROUGH THE ORGANIZATIONAL STRUCTURE THAT WAS RECOMMENDED.

WE ALSO RECOMMENDED THAT WE BUY A PLANT UNIQUE SIMULATOR AND THAT WE BUILD A NEW BUILDING IN WHICH TO HOUSE THE SIMULATOR AND ALL THE HUMAN RESOURCES WE NEED TO SUPPORT THE OPERATION OF THE PLANT. A NEW HUNDRED-THOUSAND-SQUARE-FOOT FACILITY IS PRESENTLY BEING BUILT FOR THIS PURPOSE.

IN ADDITION, WE RECOMMENDED THAT THE COMPANY COMMIT SOME OF THE BEST PEOPLE AVAILABLE IN THE COMPANY TO NUCLEAR OPERATIONS. I CAN ASSURE YOU FROM MY EXPERIENCE TO DATE THAT THAT COMMITMENT IS ALSO BEING MET.

SLIDE 1 SHOWS THE ORGANIZATION THAT WAS RECOMMENDED AND ACCEPTED. ALTHOUGH THE ORGANIZATION WAS ARRIVED AT INDEPENDENT OF THE GUIDANCE IN NUREG-0731, IT IS VERY SIMILAR, WHICH IS INDEED FORTUNATE SINCE THE ORGANIZATIONAL STRUCTURE WAS READILY ACCEPTABLE TO THE NRC STAFF.

BASICALLY YOU WILL NOTICE THAT THE TRAINING FUNCTION IS ELEVATED TO THE SAME LEVEL AS THE PLANT OPERATING ORGANIZATION. I THINK THAT IS SOMEWHAT DIFFERENT THAN HAS BEEN DONE IN THE PAST, AND IT PLACES A GREAT DEAL OF EMPHASIS IN THE MINDS OF EVERYONE IN NUCLEAR OPERATIONS THAT TRAINING IS VERY IMPORTANT AND THAT PROCEDURES MUST BE FOLLOWED CAREFULLY.

THE NAMES IN THE BOXES INDICATE THE POSITIONS PRESENTLY FILLED. THE ONLY THREE POSITIONS NOT FILLED ARE THE DIRECTOR-OUTAGE MANAGEMENT, THE DIRECTOR-NUCLEAR PLANT MODIFICATIONS

AND THE DIRECTOR-QUALITY ASSURANCE. WE SHOULD HAVE NO DIFFICULTY IN SELECTING ALL THREE DIRECTORS FROM THE MANY QUALIFIED PEOPLE NOW ASSIGNED TO THE PROJECT MANAGEMENT ORGANIZATION TO FILL THESE POSITIONS.

THE PEOPLE WHO WILL STAFF THESE DEPARTMENTS ARE PRESENTLY ENGAGED IN FINISHING THE DESIGN AND CONSTRUCTION OF FERMI 2. THE PEOPLE WHO WILL SCHEDULE THE OUTAGE WORK ARE SOME OF THE SAME PEOPLE NOW SCHEDULING THE DESIGN, CONSTRUCTION AND TESTING WORK FOR THE PROJECT. THE WORK NECESSARY TO CARRY OUT DESIGN AND CONSTRUCTION FOR PLANT MODIFICATIONS IS VERY SIMILAR TO THE WORK REQUIRED OF THE PROJECT MANAGEMENT ORGANIZATION TO DESIGN AND BUILD THE PLANT. SINCE THESE RESOURCES ARE STILL REQUIRED TO COMPLETE THE PLANT, WE DO NOT WANT TO CALL ON THEM IN THE NUCLEAR OPERATIONS ORGANIZATION UNTIL ABSOLUTELY NECESSARY BECAUSE THEY STILL HAVE A BIG JOB TO COMPLETE THE CONSTRUCTION OF THE PLANT.

ONE OF THE LESSONS LEARNED AT THREE MILE ISLAND IS THE NECESSITY TO TRANSFER INFORMATION WITHIN OUR INDUSTRY AND WITHIN OUR ORGANIZATION.

OUR COMPANY WILL REVIEW AND ASSESS BOTH INTERNAL AND EXTERNAL OPERATING EXPERIENCE TO ASSURE THAT INFORMATION PERTINENT TO PLANT SAFETY IS CONTINUALLY SUPPLIED TO OPERATORS AND OTHER PERSONNEL AS APPROPRIATE AND IS UTILIZED TO EFFECT

DESIGN AND PROCEDURAL CHANGES TO CORRECT GENERIC OR SPECIFIC DEFICIENCIES AND TO ENHANCE PLANT SAFETY WHEN WARRANTED.

THE REVIEW OF EXTERNALLY GENERATED OPERATING EXPERIENCE WILL BE CARRIED OUT PRIMARILY BY INDIVIDUALS WITHIN THE NUCLEAR SAFETY AND PLANT ENGINEERING GROUP. THIS EXPERIENCE WILL INCLUDE, BUT NOT BE LIMITED TO, GENERAL ELECTRIC NSSS REPORTS, NSAC REPORTS, LER's FORWARDED FROM THE INPO PROGRAM "SIGNIFICANT EVENT EVALUATION AND INFORMATION NETWORK" (SEE-IN) AND NRC BULLETINS, CIRCULARS AND NOTICES. INFORMATION WHICH IS CONSIDERED TO BE OF A NATURE SUCH THAT URGENT DISPOSITION IS NECESSARY WILL BE FORWARDED TO THE APPROPRIATE PLANT PRODUCTION STAFF IMMEDIATELY.

OPERATING EXPERIENCE WHICH IS CONSIDERED TO WARRANT FURTHER EVALUATION IS ASSIGNED TO INDIVIDUALS WITHIN THE NUCLEAR SAFETY AND PLANT ENGINEERING GROUP OR PLANT STAFF AS APPROPRIATE FOR EVALUATION AND RECOMMENDATIONS. THE CONCLUSIONS AND RECOMMENDATIONS ARE RETURNED TO THE NUCLEAR SAFETY AND PLANT ENGINEERING GROUP FOR APPROVAL AND THEN ASSIGNED FOR IMPLEMENTATION TO THE NUCLEAR PLANT MODIFICATIONS GROUP FOR DESIGN CHANGES AND/OR TO THE APPROPRIATE PLANT STAFF AND TRAINING ORGANIZATIONS FOR PROCEDURAL MODIFICATIONS. PROCEDURAL CHANGES WILL BE REVIEWED AND APPROVED BY THE ON-SITE REVIEW ORGANIZATION (OSRO) WHICH IS MADE UP OF THE SECTION LEADERS OF THE NUCLEAR PRODUCTION STAFF AND THE PLANT SUPERINTENDENT. OPERATORS AND SHIFT TECHNICAL ADVISORS

ARE NOTIFIED OF IMPLEMENTED CHANGES AS WELL AS THE NUCLEAR TRAINING DEPARTMENT.

THE REVIEW OF INTERNALLY GENERATED OPERATING EXPERIENCE (PRIMARILY LER's) BEGINS WITH THE TECHNICAL ENGINEER WHO PREPARES PLANT LER's. THE TECHNICAL ENGINEER SUBMITS THE LER TO OSRO. SHIFT SUPERVISORS AND SHIFT TECHNICAL ADVISORS WILL BE NOTIFIED OF ALL PLANT LER's. OSRO WILL EVALUATE THE LER TO DETERMINE WHETHER PROCEDURAL AND/OR DESIGN CHANGES ARE CALLED FOR OR WHETHER FURTHER ANALYSIS IS NECESSARY. FOR STRAIGHTFORWARD PROCEDURAL MATTERS, OSRO WILL RECOMMEND AND APPROVE THE CHANGES AND ASSIGN THEM TO THE PLANT STAFF FOR IMPLEMENTATION, INCLUDING TRAINING PROGRAM MODIFICATIONS. FOR LER's WHICH RELATE TO PLANT DESIGN OR REQUIRE IN-DEPTH ANALYSIS, OSRO WILL TRANSFER THE INFORMATION TO THE NUCLEAR SAFETY AND PLANT ENGINEERING GROUP ALONG WITH THEIR RECOMMENDATION, IF ANY. THIS GROUP WILL DETERMINE THE APPROPRIATE RESPONSE AND ASSIGN THE CHANGE FOR IMPLEMENTATION TO THE NUCLEAR PLANT MODIFICATIONS GROUP OR PLANT STAFF AS APPROPRIATE.

THE NUCLEAR SAFETY AND PLANT ENGINEERING GROUP WILL RECEIVE ALL PLANT LER's AND INTERNAL OPERATING EXPERIENCE AS A MATTER OF COURSE AND, THEREFORE, MAY CHOOSE TO TAKE ACTION ON AN LER EVEN IF OSRO ELECTED NOT TO REFER IT TO THEM.

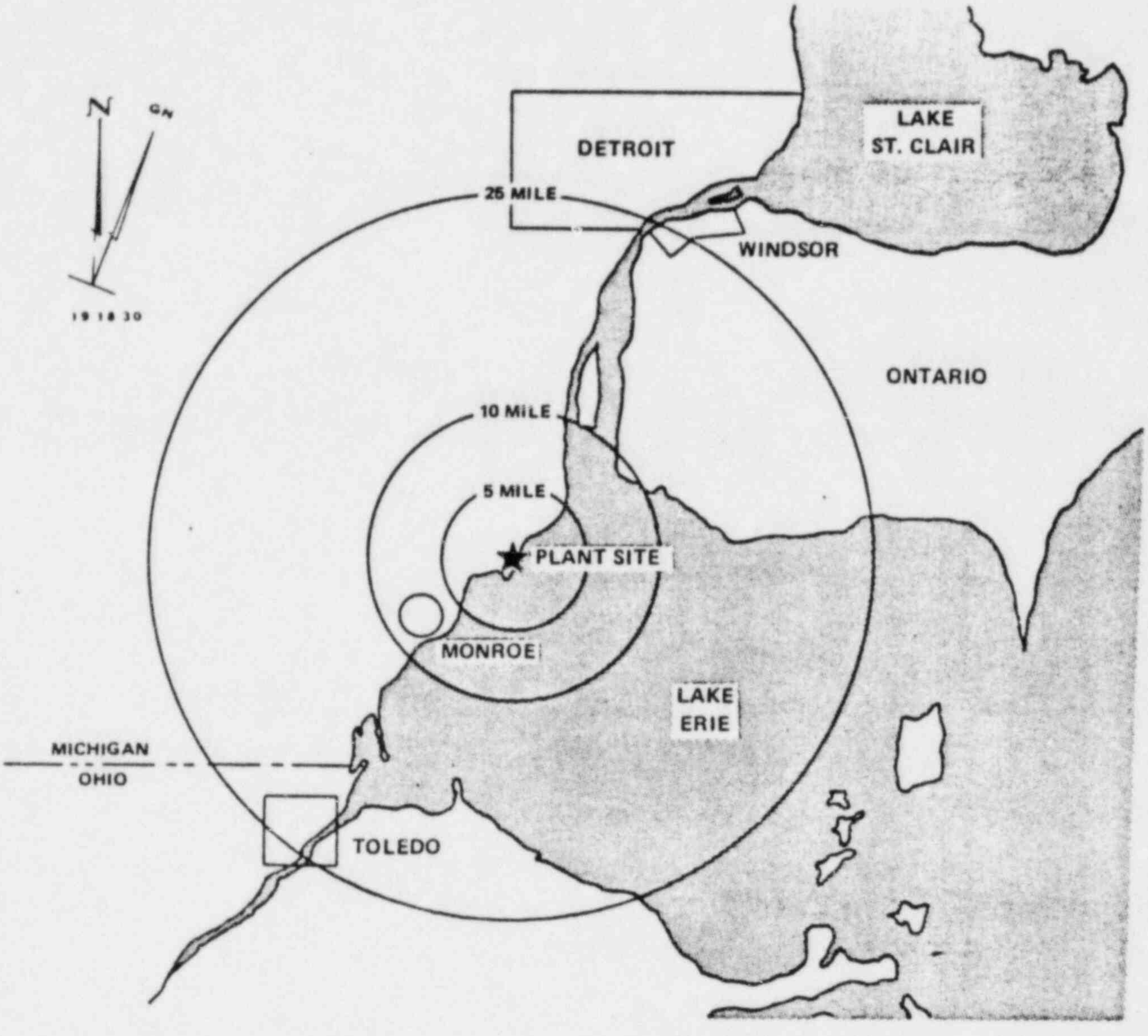
ANY CHANGE WHICH CONSTITUTES AN UNREVIEWED SAFETY QUESTION WILL BE REVIEWED AND APPROVED BY THE INDEPENDENT REVIEW AND AUDIT GROUP (IRAG) PRIOR TO IMPLEMENTATION. IRAG CONSISTS OF KEY NUCLEAR OPERATORS, SUPERVISORS AND EXPERTS FROM WITHIN AND OUTSIDE THE COMPANY.

SLIDE 2 INDICATES HOW THE NUCLEAR OPERATIONS ORGANIZATION REPORTS WITHIN DETROIT EDISON, HOW FERMI 2 EXPERIENCE IS EVALUATED AND HOW CHANGES ARE APPROVED IN PROCEDURES AND IN THE DESIGN. CHANGES IN PLANT PROCEDURES ARE APPROVED BY OSRO. IF THE CHANGES INVOLVE AN UNREVIEWED SAFETY QUESTION OR IF OSRO RECOMMENDS A DESIGN CHANGE, THE CHANGE IS REVIEWED BY THE PERMANENT AND FULL TIME NUCLEAR SAFETY AND PLANT ENGINEERING GROUP. AFTER THEIR REVIEW AND SAFETY ANALYSIS ARE COMPLETED, THE PREVIOUSLY UNREVIEWED SAFETY ISSUES INVOLVING CHANGES ARE THEN REVIEWED BY IRAG. IRAG WILL ALSO CONDUCT PERIODIC AUDITS OF ALL DEPARTMENTS WITHIN NUCLEAR OPERATIONS TO REVIEW THEIR OVERALL SAFETY PERFORMANCE.

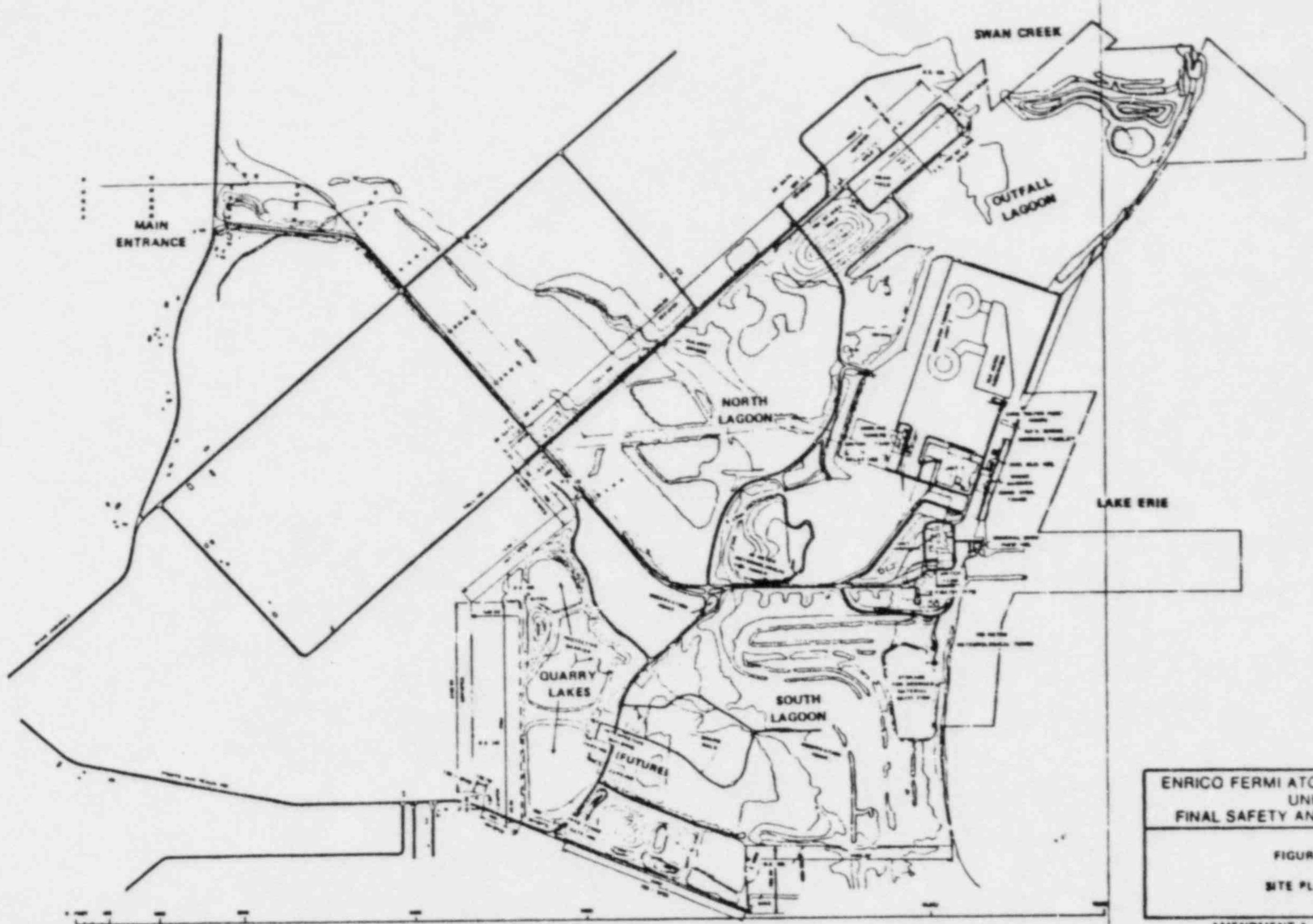
OUR BOARD OF DIRECTORS HAS EXPRESSED AN INTEREST IN REVIEWING THE PERFORMANCE OF NUCLEAR OPERATIONS. IT IS OUR PRESENT INTENTION TO NAME A SUBCOMMITTEE OF THE BOARD OF DIRECTORS, CONSISTING OF SEVERAL MEMBERS OF THE BOARD AND PERHAPS A CONSULTANT. THIS SUBCOMMITTEE WILL ESTABLISH THE OVERALL NUCLEAR SAFETY PHILOSOPHY AND WILL REVIEW SIGNIFICANT EVENTS, TRENDS AND OPERATING EXPERIENCE OF NUCLEAR OPERATIONS.

THAT COMPLETES MY PRESENTATION. I WILL NOW BE HAPPY TO ANSWER ANY QUESTIONS YOU MAY HAVE.

II.A. SITE AND PLANT DESCRIPTION



ENRICO FERMI ATOMIC POWER PLANT UNIT 2 FINAL SAFETY ANALYSIS REPORT
FIGURE 1 GENERAL REGION OF THE FERMI SITE



DETROIT EDISON COMPANY DWG. NO. 6A721-2102

ENRICO FERMI ATOMIC POWER PLANT
UNIT 2
FINAL SAFETY ANALYSIS REPORT

FIGURE 2

SITE PLOT PLAN

AMENDMENT 1 - NOVEMBER 1975

SUMMARY OF PLANT DESIGN

DESIGN FEATURE	FERMI 2
RATED THERMAL POWER (MW_{th})	3292
ECCS DESIGN POWER (MW_{th})	3430
GROSS ELECTRICAL OUTPUT (MW_e)	1154
MAINSTREAM FLOW RATE (lB/HR)	$14.156(10^6)$
REACTOR TOTAL FLOW RATE (lB/HR)	$100.0(10^6)$
SYSTEM PRESSURE (PSI)	1005
VESSEL SIZE (DIAMETER IN)	251
VESSEL DESIGN PRESSURE (PSI)	1250
NUMBER OF FUEL ASSEMBLIES	764
FUEL TYPE (8x8)	62 + 2
RECIRC. LOOP INSIDE DIAMETER (IN)	28
MAXIMUM LINEAR POWER GENERATION (KW/FT)	13.4
MAXIMUM FUEL TEMPERATURE ($^{\circ}F$)	3435
TOTAL PEAKING FACTOR	2.43
CORE HEIGHT (IN)	150
NUMBER CONTROL RODS	185
MAIN CONDENSOR CAPACITY (BTU/HR)	7547×10^6

DESIGN FEATURE (CONTINUED)	FERMI 2
CIRCULATING WATER PUMPS	5
LPCS (NUMBER + FLOW RATE)	2 @ 6250
HPCI (NUMBER + FLOW RATE)	1 @ 5000
LPCI (NUMBER + FLOW RATE)	3 @ 10000
ADS	5 VALVES
RHR (NUMBER LOOPS + FLOW RATE)	4 @ 7200
RHR HEAT EXCHANGERS (NUMBER + DUTY)	2 @ $41.6(10^6)$
RCIC (GPM)	600
PRIMARY CONTAINMENT TYPE	MK I, STEEL
PRIMARY CONTAINMENT DESIGN PRESSURE (PSI)	
EXTERNAL	2
INTERNAL	56
DRYWELL VOLUME (FT^3)	163,780
TORUS AIRSPACE (FT^3)	130,900
SUPPRESSION POOL VOLUME (FT^3)	117,450
DRYWELL TEMPERATURE (^oF)	281

SITE AND PLANT DESCRIPTION

ENRICO FERMI ATOMIC POWER PLANT, UNIT 2 (FERMI-2) IS LOCATED ON A 1120-ACRE SITE, APPROXIMATELY 30 MILES SOUTH OF DETROIT AND ABOUT 25 MILES NORTHEAST OF TOLEDO, OHIO AS SHOWN IN FIGURE 1. THE SEVERAL CONTINGUOUS BUILDINGS COMPRISING THE FERMI-2 PLANT ARE SITUATED ON THE WESTERN SHORE OF LAKE ERIE IN FRENCHTOWN TOWNSHIP, MONROE, MICHIGAN, AS SHOWN IN FIGURE 2.

FERMI 2 IS CO-OWNED BY THE DETROIT EDISON COMPANY AND TWO COOPERATIVES, NORTHERN MICHIGAN ELECTRIC COOPERATIVE INC., AND WOLVERINE ELECTRIC COOPERATIVE, INC. APPROXIMATELY 90% OF THE LAND AREA WITHIN TEN MILES OF THE PLANT LIES WITHIN MONROE COUNTY; THE REMAINING 10% IS IN WAYNE COUNTY. OF THE TEN-MILE AREA IN MONROE, APPROXIMATELY 55% CONSISTS OF FARMLAND. WITHIN A 50-MILE RADIUS OF THE SITE ARE ALL, OR PORTIONS OF, ELEVEN COUNTIES IN MICHIGAN, TEN IN OHIO, AND TWO IN ONTARIO, CANADA. THE 1980 CENSUS DATA SHOWED A POPULATION OF 84,~~8~~600 WITH IN TEN MILES OF THE PLANT AND 5.5 MILLION WITHIN 50 MILES.

FERMI 2 UTILIZES A BWR-4 BOILING WATER REACTOR DESIGNED AND SUPPLIED BY GENERAL ELECTRIC AND A MARK I CONTAINMENT. THE REACTOR CONTAINS THE CORE, CONTROL

RODS, INSTRUMENTATION, STEAM SEPARATOR AND DRIER ASSEMBLIES, JET PUMPS, AND THE CONTROL ROD DRIVE MECHANISMS, WHICH ARE MOUNTED ON THE BOTTOM OF THE REACTOR PRESSURE VESSEL.

THE REACTOR CORE CONTAINS 764 FUEL ASSEMBLIES AND 185 CONTROL RODS ARRANGED IN AN UPRIGHT CYLINDRICAL CONFIGURATION. EACH FUEL ASSEMBLY CONSISTS OF AN 8 x 8 ARRAY OF RODS, 62 OF WHICH CONTAIN FUEL AND TWO OF WHICH CONTAIN WATER. THE 100% RATED THERMAL POWER LEVEL OF THE REACTOR FOR WHICH DETROIT EDISON IS REQUESTING AN OPERATING LICENSE IS 3292 MEGAWATTS.

THE EMERGENCY CORE COOLING SYSTEM IS OF CONVENTIONAL DESIGN, EMPLOYING A HIGH PRESSURE COOLANT INJECTION SYSTEM, A LOW PRESSURE COOLANT INJECTION SYSTEM, AND A CORE SPRAY SYSTEM, AND AN AUTOMATIC DEPRESSURIZATION SYSTEM TO BRIDGE THE CAPABILITIES OF THE HIGH PRESSURE AND LOW PRESSURE SYSTEMS. THE RESULTING SYSTEM PROVIDES REDUNDANCY AND DIVERSITY.

THE IN-HOUSE ELECTRICAL DISTRIBUTION SYSTEM IS DESIGNED TO PROVIDE SUFFICIENT NORMAL AND STANDBY SOURCES OF ELECTRICAL POWER TO PERMIT SAFE SHUTDOWN AND TO MAINTAIN THE PLANT IN A SAFE CONDITION UNDER ALL CREDIBLE CIRCUMSTANCES. IN

ADDITION, THE POWER SOURCES ARE ADEQUATE TO ACCOMPLISH ALL ESF FUNCTIONS REQUIRED UNDER POSTULATED DESIGN BASIS ACCIDENT CONDITIONS.

THE FERMI 2 FACILITY EMPLOYS A RADWASTE SYSTEM DESIGNED TO LIMIT THE DOSE TO THE GENERAL PUBLIC DUE TO RADIOACTIVE EFFLUENTS TO LEVELS WHICH MEET THE DESIGN OBJECTIVES OF 10 CFR 50, APPENDIX I AND ARE AS LOW AS REASONABLY ACHIEVABLE. IN ADDITION, THE FERMI 2 FACILITY DESIGN INCORPORATES FEATURES WHICH MINIMIZE THE OCCUPATIONAL RADIATION EXPOSURE UNDER NORMAL AND POSTULATED ACCIDENT CONDITIONS.

THE DETROIT EDISON COMPANY POSSESSES A LARGE POOL OF QUALIFIED PERSONNEL TO SUPPORT THE OPERATION OF FERMI 2. OPERATING PERSONNEL CAN DRAW UPON THIS RESOURCE AS NEEDED. MOST OF THE NUCLEAR OPERATIONS PERSONNEL ARE LOCATED NEARBY THE PLANT SITE. IN ADDITION, SIGNIFICANT OVERSIGHT IS PROVIDED BY DETROIT EDISON MANAGEMENT TO ASSURE THAT OPERATIONS WILL BE SAFE AND EFFICIENT.

FERMI 2 IS SIMILAR TO A NUMBER OF MODERN BWR POWER REACTORS THAT ARE PRESENTLY OPERATING IN THE UNITED STATES, SUCH AS HATCH AND BROWNS FERRY. THE ENCLOSED TABLE SUMMARIZES KEY DESIGN PARAMETERS OF FERMI 2.

WITH ISSUANCE OF A CONSTRUCTION PERMIT FOR FERMI 2, THE ACRS IDENTIFIED A NUMBER OF OUTSTANDING ITEMS WHICH MUST BE RESOLVED PRIOR TO OPERATION OF FERMI 2. THESE ITEMS, ADDRESSED IN APPENDIX B OF THE FERMI 2 FSAR, HAVE BEEN RESOLVED. IN ADDITION, SINCE CONSTRUCTION OF FERMI 2 HAS BEEN STARTED, A NUMBER OF GENERIC ISSUES HAVE BEEN IDENTIFIED, REQUIRING DESIGN CHANGES, AS FOLLOWS:

A. IGSCC (1974-78) RESULTED IN MATERIAL, PROCESS, WELD, AND SYSTEM CHANGES: RR, CRD, CORE SPRAY SPARGERS.

B. FIRE AT BROWNS FERRY RESULTED IN COMPLETE FIRE HAZARDS ANALYSIS, INSTALLED FIRE DETECTION, AND PROTECTION SYSTEMS UPGRADED: SAFE SHUTDOWN ANALYSIS AND SOME PLANT CHANGES: FIRE BARRIERS, FIRE STOPS, MORE FIRE PROTECTION APPARTUS.

C. SECURITY REQUIREMENT - EDISON'S PLAN ORIGINATED FOR FERMI 2 CONTROLS ACCESS TO AN ESSENTIALLY LOCKED PLANT WHICH USES NATURAL PLANT STRENGTH AS THE BARRIER TO INTRUSION AND HAS EXTERNAL WARNING PERIMETER WITH MULTIPLE SENSING TO ALERT GUARD FORCES.

D. CRD SYSTEM REFINEMENT WITH REMOVAL OF CRD RETURN LINES TO VESSEL.

E. FEEDWATER NOZZLE CRACKING PROBLEMS (1975-1978) AT OTHER PLANTS RESULTED IN SPARGERS CHANGED OUT TO LATER DESIGN.

III.B.1 OPERATOR SELECTION AND TRAINING

SERIOUS ACCIDENTS BEYOND D.B.A.

TOPICS

1. CORE COOLING MECHANICS
2. POTENTIALLY DAMAGING OPERATING CONDITIONS
3. GAS/STEAM BINDING ON CORE COOLING
4. RECOGNIZING CORE DAMAGE
5. HYDROGEN HAZARDS DURING ACCIDENT CONDITIONS
6. MONITORING CRITICAL PARAMETERS DURING ACCIDENT CONDITIONS

    INCORE INSTRUMENTATION

    EXCORE INSTRUMENTATION

    PROCESS COMPUTER

    HIGH RADIATION SAMPLING
7. RADIATION HAZARDS AND RADIATION MONITOR RESPONSE
8. CRITERIA FOR OPERATION AND COOLING MODE SELECTION
9. INFREQUENT ABNORMAL AND EMERGENCY OPERATING PROCEDURES
10. THERMODYNAMICS AND HEAT TRANSFER
11. RECRITICALITY POTENTIAL
12. EMERGENCY PLAN

TRAINING FOR SERIOUS ACCIDENTS BEYOND D.B.A.

IN ACCORDANCE WITH NUREG-0737, NUREG-0660, APPENDIX H OF THE ENRICO FERMI 2 FSAR, AND THE MARCH 28, 1980 NRC STAFF DIRECTIVE (DENTON LETTER), DETROIT EDISON HAS ESTABLISHED A PROGRAM TO TRAIN ITS PERSONNEL IN CONTROLLING AND MITIGATING ACCIDENTS BEYOND THE DESIGN BASED ACCIDENT (DBA). THIS PROGRAM WILL MEET BOTH THE INTENT AND REQUIREMENTS OF THE ABOVE DIRECTIVES, AND THE RECOMMENDATIONS OF THE INSTITUTE OF NUCLEAR POWER OPERATION (INPO) DOCUMENT TITLED "TRAINING GUIDELINES FOR RECOGNIZING AND MITIGATING THE CONSEQUENCES OF SEVERE CORE DAMAGE," DATED JUNE 30, 1980.

THE TABLE SHOWN PRESENTS A TOPICAL SUMMARY OF THE TRAINING TO BE REPRESENTED IN THE DETROIT EDISON TRAINING PROGRAM. CONTACT HOURS OF INSTRUCTION RECEIVED ARE NOT LISTED. IT IS THE DETROIT EDISON VIEWPOINT THAT KNOWLEDGE ATTAINMENT/RETENTION IS THE KEY TO THE VALIDITY OF A TRAINING CURRICULUM. THEREFORE, OUR COURSE DESIGNS WILL BE SUCH TO INSURE THAT THE PERSONNEL HAVE ATTAINED THE LEVEL OF EXPERTISE TO BE ABLE TO APPROPRIATELY RECOGNIZE AND RESPOND, IRRESPECTIVE OF THE TIME NEEDED TO REACH THAT LEVEL.

DETROIT EDISON'S LICENSED OPERATOR TRAINING PROGRAM HAS RECEIVED EXTENSIVE MODIFICATION AND AUGMENTATION SINCE THREE MILE ISLAND TO MEET THE PRESENT GUIDELINES. TRAINING IN INFREQUENT, ABNORMAL, AND EMERGENCY PROCEDURES HAS BEEN MODIFIED TO EMPHASIZE THE BIG PICTURE, AND OPERATOR UTILIZATION OF ALL INPUT DATA FOR MAINTAINING THE CORE COVERED AND THE CONTAINMENT INTACT. TRAINING IN PROCESS COMPUTER OPERATIONS IS BEING EXPANDED TO INCLUDE USE OF THE COMPUTER DURING DEGRADED CORE CONDITIONS FOR ATTAINING AND ANALYZING INFORMATION DESCRIBING THE ACTUAL CORE STATUS.

NEW COURSES HAVE BEEN PLACED INTO THE PROGRAM TO INSURE THAT ALL OPERATORS HAVE THE KNOWLEDGE TO PROPERLY EVALUATE AND RESPOND TO THESE EMERGENCY CONDITIONS. THEORETICAL TRAINING, WITH PRACTICAL APPLICATIONS, IN THERMODYNAMICS AND HEAT TRANSFER HAS BEEN IDENTIFIED AS NECESSARY. LIKEWISE, A COURSE IN MITIGATION OF CORE DAMAGE HAS BEEN INCLUDED. AT THIS TIME, DETROIT EDISON IS EVALUATING THE CAPABILITIES OF VARIOUS TRAINING ORGANIZATIONS TO SUPPORT OUR TRAINING OBJECTIVES IN THIS AREA. ESTABLISHMENT OF A FINAL CURRICULUM HAS NOT BEEN COMPLETED. HOWEVER, ALL APPROPRIATE SUBJECTS TO INCLUDE UTILIZATION OF INCORE AND EXCORE INSTRUMENTATION, CORE COOLING MECHANICS, PRIMARY CHEMISTRY, RADIATION MONITORING, GAS GENERATION AND HAZARDS, AND RECRITICALITY POTENTIAL WILL BE COVERED.

TRAINING IN THIS AREA WILL BE GIVEN, AS APPROPRIATE, TO ALL OPERATING PERSONNEL AT ENRICO FERMI 2, FROM THE PLANT SUPERINTENDENT TO THE LICENSED OPERATOR. SHIFT TECHNICAL ADVISORS (STA'S) HAVE BEEN ESTABLISHED TO SUPPLEMENT THE OPERATING CREWS' CAPABILITIES DURING ABNORMAL CONDITIONS. THEIR TRAINING WILL INCLUDE ALL OF THE ABOVE MENTIONED CURRICULUM, AND BE EXPANDED TO GREATER DEPTH. TRAINING FOR INSTRUMENT MAINTENANCE AND RAD-CHEM PERSONNEL WILL BE CONDUCTED ON BOTH A THEORETICAL AND PRACTICAL LEVEL FOR THE ACCOMPLISHMENT OF HIGH RADIATION SAMPLING.

FINALLY, ALL PLANT PERSONNEL WILL BE TRAINED TO A DEGREE COMMENSURATE WITH THEIR DUTIES ON EMERGENCY PLAN IMPLEMENTATION. THIS WILL INCLUDE SUCH TOPICS AS ACTIVATING THE ONSITE TECHNICAL SUPPORT CENTER AND THE OFFSITE EMERGENCY OPERATIONS FACILITY, ALERT NOTIFICATION PROCEDURES AND RESPONSIBILITIES, EVACUATION PROCEDURES, AND SUPPORT PROCEDURES.

IN SUMMARY, WE BELIEVE WE SATISFY THE CRITERIA FOR TRAINING FOR SERIOUS ACCIDENTS BEYOND D.B.A.

MAINTENANCE WORKER TRAINING

ALL WORKERS WILL BE JOURNEYMEN

GENERAL MAINTENANCE JOURNEYMAN PROGRAM

- INCREASED JOB SCOPE → INCREASED JOB SATISFACTION
- FACILITATE MANPOWER UTILIZATION
- PRIMARY VS SECONDARY SKILL

GMJ APPRENTICE TRAINING PROGRAM

- BASED ON TASK ANALYSIS
- MODULAR FORMAT
- SELF-PACED, BUT TIED TO PAY INCREASES
- PERFORMANCE BASED EVALUATION

ENRICO FERMI MAINTENANCE SELECTION PROCESS

- CONSTRAINED BIDDING SYSTEM
- GMJ MUST WANT TO WORK AT FERMI
- GMJ MUST BE HIGH PERFORMER

ENRICO FERMI SPECIFIC TRAINING

"BIG PICTURE"

- NEW APPLICATION OF OLD SKILLS
- NEW TOOLS AND EQUIPMENT
- NEW SKILLS DUE TO NUCLEAR ENVIRONMENT
- DEVELOPMENT OF PROPER WORKING HABITS

ATTACHMENT 1
TRAINING PRESCRIPTION

				GMJ	Welding Engineer	Shop Welding Foreman	Field Welding Foreman	Maintenance Foreman-Plant	Asst. General Foreman	General Foreman	Senior Engineering Technician	Engineer	Senior Engineer	Asst. Maintenance Engineer	Maintenance Engineer	
				●	●	●	●	●	●	●	●	●	●	●	●	General Employe
				●	●	●	●	●	●	●	●	●	●	●	●	Emergency Plan
				●	●	●	●	●	●	●	●	●	●	●	●	Security Plan
				●	●	●	●	●	●	●	●	●	●	●	●	Fire Protection Plan
				●	●	●	●	●	●	●	●	●	●	●	●	First Aid
				●	●	●	●	●	●	●	●	●	●	●	●	Cardiopulmonary Resucitation
				●	●	●	●	●	●	●	●	●	●	●	●	Plant Administrative Procedures
				●	●	●	●	●	●	●	●	●	●	●	●	Maintenance Administrative Procedures
				●	●	●	●	●	●	●	●	●	●	●	●	QA/QC Indoctrination
				●				●	●							Nuclear Mechanical Seals
				●	●	●	●	●	●	●						Mobile Equipment Operation
				●	●	●	●	●	●	●						Nuclear Precision Tools
				●	●	●	●	●	●	●	●	●	●	●	●	Safety Tagging
				●	●	●	●	●								Cutting & Burning
								●	●							MSIV Maintenance
								●	●							Safety Relief Valve Maintenance
								●	●							Recirc. Pump Seal Maintenance
								●	●							CRD Hydraulic System Maintenance
				●				●	●	●						Emergency Diesel Generator Maintenance
				●				●	●							Dexter Valve Reseater Operation
				●				●	●							Wacks Cutter Operation
				●				●	●							Hydraulic Torquing Eq. Operation
				●				●	●							Limitorque Valve Actuator Maintenance
				●				●	●							Powell Valve Maintenance
				●				●	●							Rototorque Valve Maintenance
				●	●	●	●	●	●	●	●	●	●	●	●	Generic Systems and Procedures
				●	●	●	●	●	●	●	●	●	●	●	●	Respirator Use
				●	●	●	●	●	●	●	●	●	●	●	●	Radiation Worker Skills
				●	●	●	●	●	●	●	●	●	●	●	●	Decontamination Skills
				●	●	●	●	●	●	●	●	●	●	●	●	Housekeeping
				●	●	●	●	●	●	●	●	●	●	●	●	Nuclear Codes and Standards
				●	●	●	●	●	●	●						Tool Calibration
				●	●	●	●	●	●	●						Visual NDE Examination
				●	●	●	●									Visible Dye Penetrant Examination

SELECTION AND TRAINING OF MAINTENANCE PERSONNEL

THE DETROIT EDISON COMPANY IS FIRMLY COMMITTED TO THE CONCEPT OF MAXIMIZING THE SAFETY OF NUCLEAR POWER PLANTS THROUGH THE USE OF WELL-TRAINED PERSONNEL. ALL CRAFT WORKERS ASSIGNED TO THE MAINTENANCE DEPARTMENT AT THE ENRICO FERMI ATOMIC POWER PLANT UNIT 2 (EF2) WILL BE JOURNEYMEN. OVER A DECADE AGO, THE DETROIT EDISON COMPANY INSTITUTED A PROGRAM IN TRAINING MAINTENANCE CRAFT WORKERS TO BECOME GENERAL MAINTENANCE JOURNEYMEN (GMJ). DEPARTING FROM THE TRADITIONAL ONE-MAN, ONE-CRAFT CONCEPT, DETROIT EDISON REALIZED THAT INCREASING JOB SCOPE TO INCLUDE MORE THAN ONE CRAFT WOULD BENEFIT BOTH THE WORKER AND THE COMPANY. THE INCREASED VARIETY OF SKILLS THAT THE WORKER WAS CALLED UPON TO PERFORM WOULD INCREASE JOB SATISFACTION, WHICH WOULD LEAD TO HIGHER QUALITY WORK. THIS WOULD OBVIOUSLY BENEFIT THE COMPANY, BUT ALSO THE INCREASED DIVERSITY OF SKILLS WOULD BENEFIT THE COMPANY IN MANPOWER UTILIZATION.

EACH GMJ IS TRAINED IN TWO SKILLS, A PRIMARY SKILL AND A SECONDARY SKILL. THE GMJ IS ABLE TO PERFORM AND LEAD OTHER WORKERS IN HIS PRIMARY SKILL, IS ALSO COMPLETELY TRAINED TO PERFORM ANY TASK RELATED TO HIS SECONDARY SKILL AND FURTHERMORE IS ABLE TO ASSIST IN ANY CRAFT. THE TRAINING PROGRAM TO PRODUCE A GMJ IS A STATE-OF-THE-ART PROGRAM. BASED ON TASK ANALYSIS AND CLOSELY CRITERION-REFERENCED, THE PROGRAM HAS

BEEN EXTREMELY SUCCESSFUL SINCE ITS INCEPTION. THE GMJ APPRENTICE TRAINING PROGRAM IS MODULAR IN FORMAT TO ALLOW FOR DIFFERENT ENTRY LEVELS AND IS PERFORMANCE BASED. THE TRAINEE IS ALLOWED TO PROGRESS THROUGH EACH TRAINING MODULE AT HIS/HER OWN PACE WITHIN A DEFINED SCHEDULE WHICH IS TIED TO INCREMENTS IN PAY. WHEN THE TRAINEE ASCERTAINS THAT HE/SHE HAS COMPLETED THE OBJECTIVES OF THE MODULE, THEY ARE GIVEN A FORMAL EVALUATION BY THE TRAINING AND/OR SUPERVISORY STAFF. THIS EVALUATION REQUIRES THAT THE TRAINEE DEMONSTRATE THE SKILLS LEARNED AND IS EVALUATED AGAINST MEASURABLE, OBSERVABLE CRITERIA. THE TRAINEE IS NOT ALLOWED TO CONTINUE TO MORE ADVANCED MODULES UNTIL HE/SHE HAS DEMONSTRATED MASTERY OF THE PREVIOUS MODULE. INTEGRAL WITH THE CLASSROOM PORTION OF THE TRAINING PROGRAM IS EXTENSIVE PLANNED, ORGANIZED AND EVALUATED ON-THE-JOB EXPERIENCE, APPLYING ALL OF THE SKILLS PREVIOUSLY LEARNED.

ONCE THE INDIVIDUAL HAS COMPLETED THE GMJ PROGRAM, HE/SHE IS ALLOWED TO BID ON A POSITION AT EF2. THE BIDDING SYSTEM USED IS A CONSTRAINED SYSTEM; NOT ALL BIDDERS ARE ACCEPTED. IN ORDER TO BE ACCEPTED FOR A MAINTENANCE POSITION AT EF2, THE INDIVIDUAL MUST NOT ONLY SHOW AN INTEREST IN THE POSITION, BUT ALSO MUST HAVE DEMONSTRATED HIGH ACHIEVEMENT IN HIS/HER PREVIOUS TRAINING AND EXPERIENCE.

EACH GMJ WHO IS ACCEPTED AT EF2 THEN BEGINS A SECOND PHASE OF TRAINING TO PROVIDE HIM/HER WITH THE ADDITIONAL SKILLS AND ABILITIES NECESSARY TO WORK AT A NUCLEAR POWER PLANT. AFTER COMPLETING TRAINING PROGRAMS TO SHOW THE GJM HOW HE/SHE FITS INTO THE "BIG PICTURE", TRAINING BEGINS IN EACH OF THE FOLLOWING FOUR AREAS CONCURRENTLY.

MANY OF THE SKILLS THAT THE GMJ ALREADY POSSESS WILL BE USED AT THE NUCLEAR PLANT, BUT APPLIED IN NEW WAYS. GMJ PIPEFITTERS, FOR EXAMPLE, ARE WELL TRAINED IN REPACKING VALVES, BUT AT EF2 THEY WILL HAVE TO BE INSTRUCTED IN HOW TO PERFORM THIS TASK IN A GLOVE BOX.

AS FERMI 2 WILL BE THE ONLY OPERATING NUCLEAR POWER PLANT IN THE EDISON SYSTEM, THE GMJ WILL BE REQUIRED TO WORK ON TYPES OF EQUIPMENT THAT WERE NOT ADDRESSED IN THEIR APPRENTICE TRAINING PROGRAM, PERHAPS WITH TOOLS THAT ARE NOT USED IN OTHER POWER PLANTS. PROGRAMS TO ENSURE PROFICIENCY IN THESE TASKS WILL BE GIVEN TO THESE WORKERS.

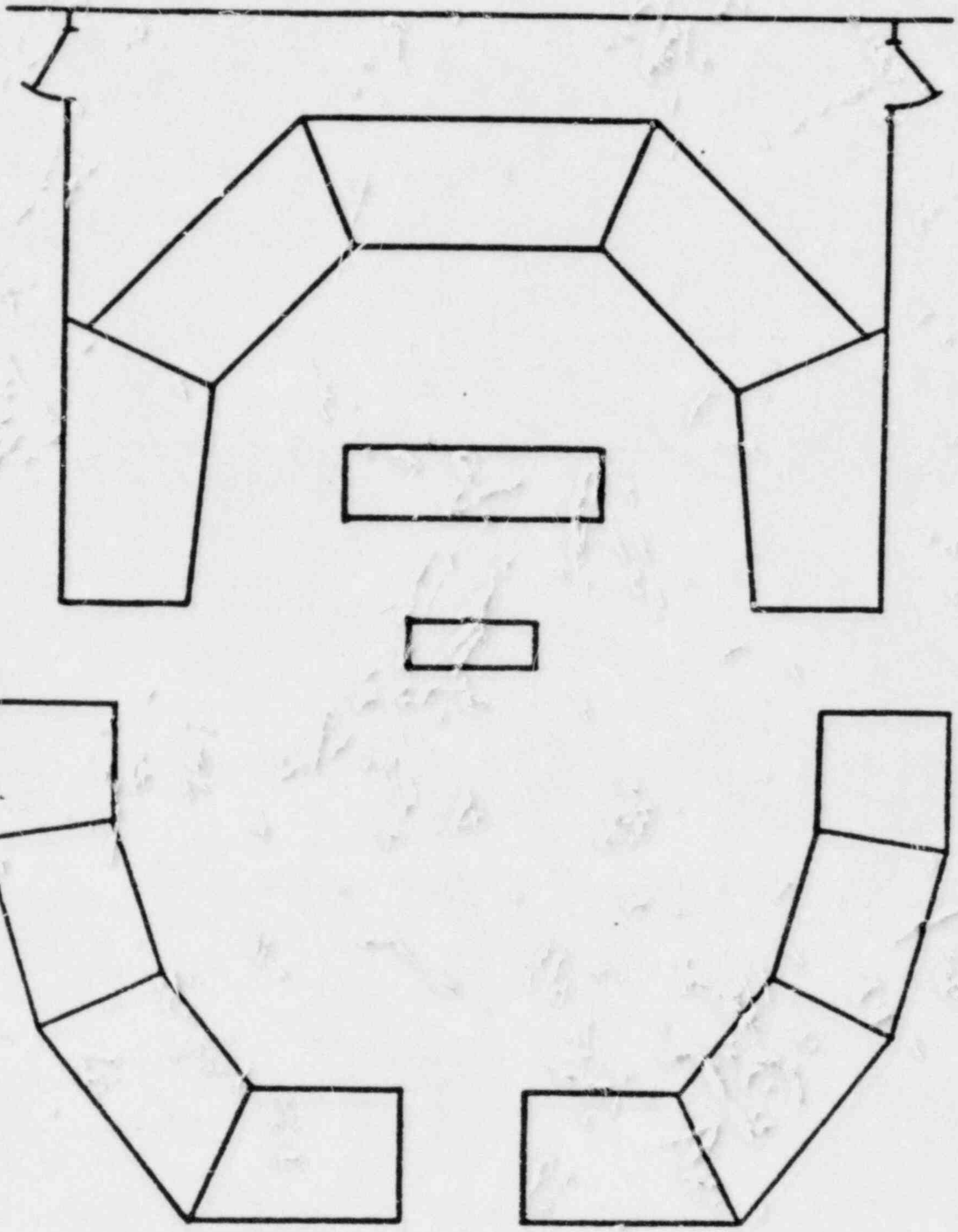
WORKING AT A NUCLEAR POWER PLANT IS A DIFFERENT ENVIRONMENT FROM THAT OF A FOSSIL PLANT. NEW SKILLS NEEDED TO ENSURE THE SAFETY (BOTH RADIATION AND INDUSTRIAL) OF THE GMJ AND OTHER PERSONNEL WILL BE REQUIRED. EACH GMJ WILL BE TRAINED TO BE PROFICIENT IN DONNING AND REMOVING PROTECTIVE CLOTHING, SPILL PREVENTION, CLEANUP AND DECONTAMINATION, JUST TO NAME A FEW.

TO ENSURE THAT THE GMJ DEVELOP THE PROPER WORKING HABITS AND AN UNDERSTANDING OF THE NUCLEAR ENVIRONMENT, SPECIFIC PROGRAMS ADDRESSING THE DISCIPLINED NATURE OF NUCLEAR WORK ARE BEING DEVELOPED. NUCLEAR CODES AND STANDARDS, PLANT AND MAINTENANCE ADMINISTRATIVE PROCEDURES, QA/QC INDOCTRINATION, HOUSEKEEPING AND SECURITY PLAN TRAINING ARE EXAMPLES OF SUCH PROGRAMS.

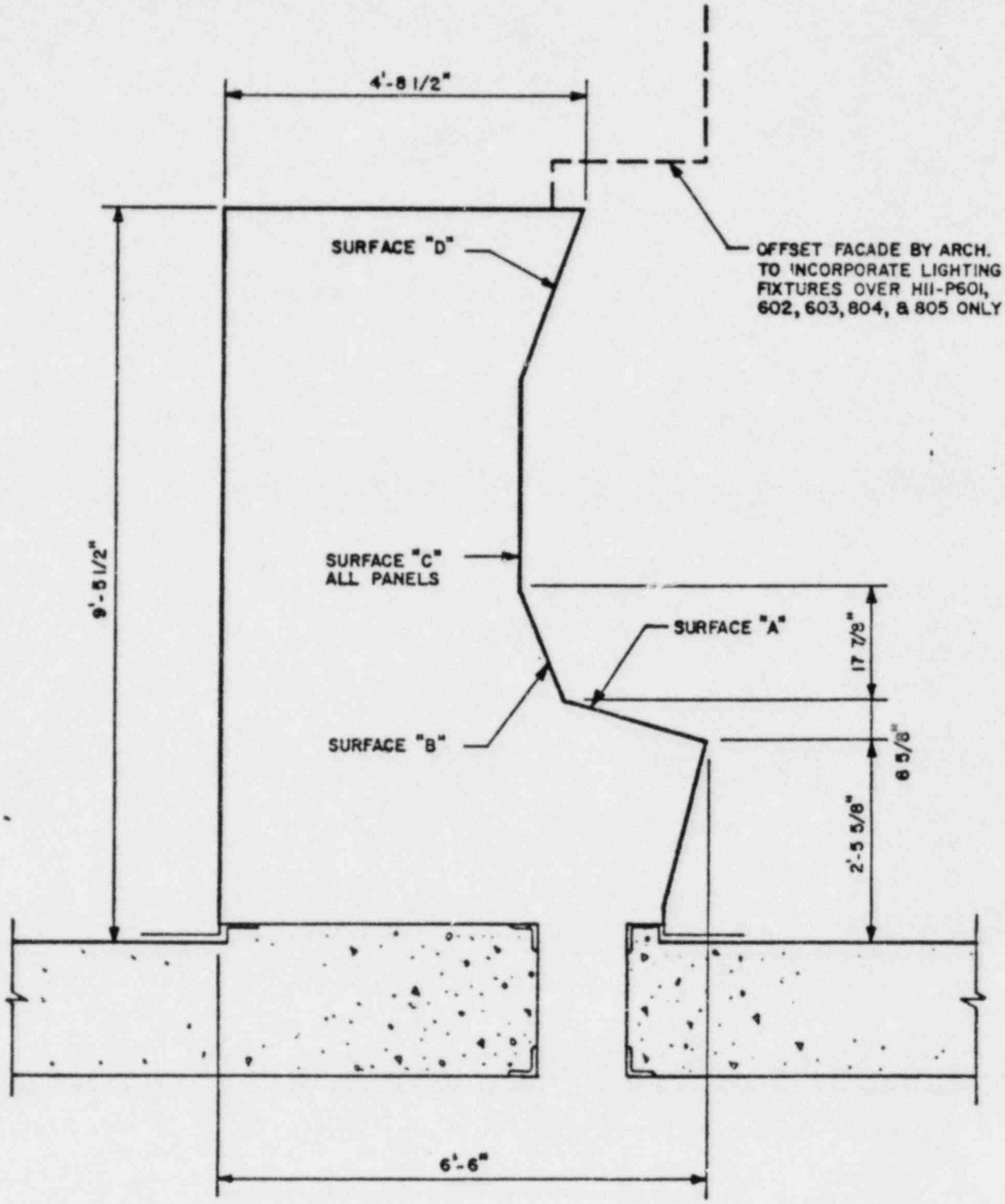
THESE NUCLEAR SPECIFIC TRAINING PROGRAMS ARE BEING DEVELOPED USING THE SAME HIGH STANDARDS USED IN THE GMJ APPRENTICE TRAINING PROGRAM. TASK ANALYSES, CRITERION-REFERENCING AND FORMALIZED PROFICIENCY DEMONSTRATION EVALUATIONS WILL BE EXTENSIVELY USED.

ATTACHMENT 1 DETAILS THE TYPES OF TRAINING PROGRAMS THAT WILL BE USED FOR EACH JOB TITLE IN THE MAINTENANCE DEPARTMENT.

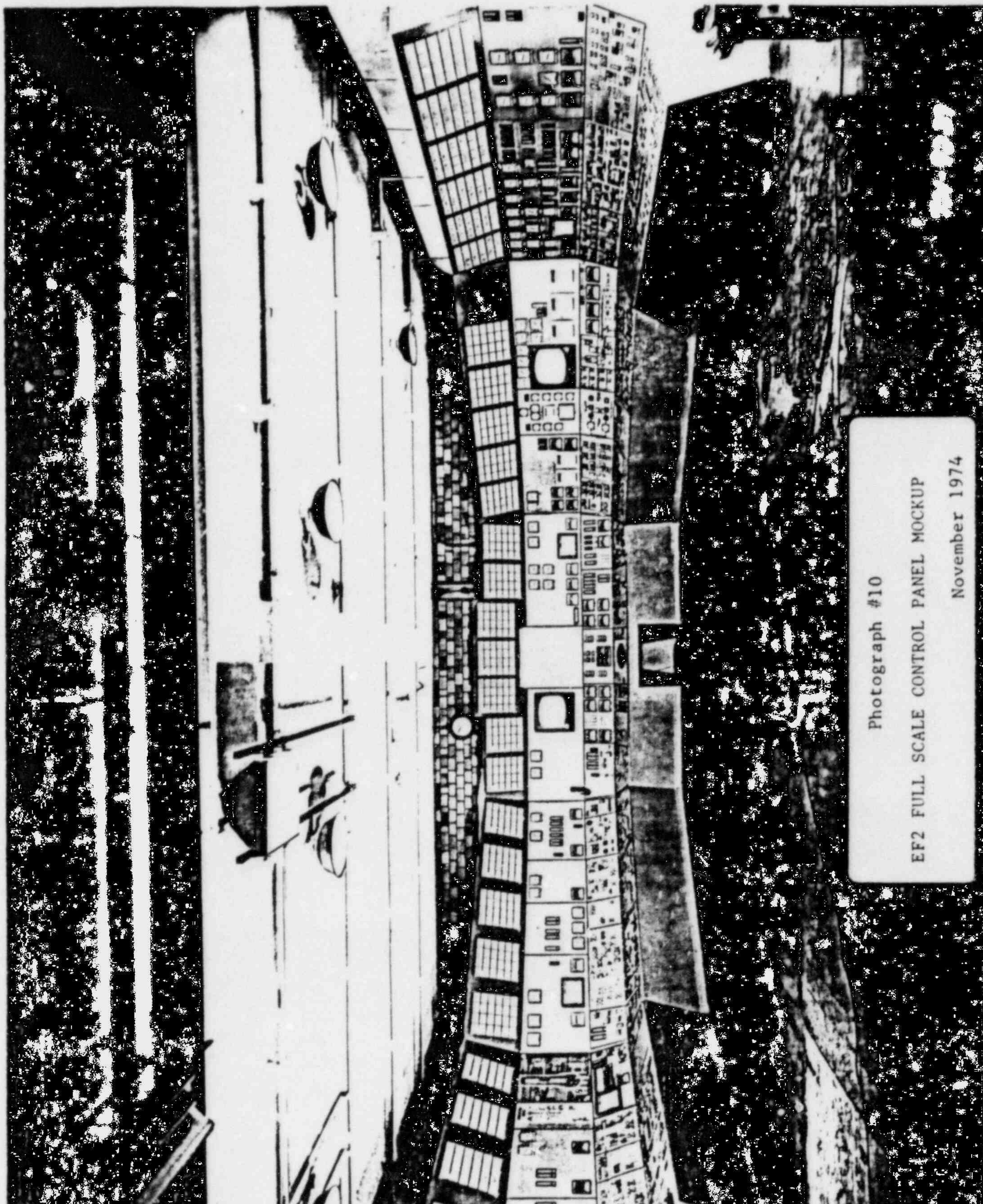
III.C. CONTROL ROOM DESIGN REVIEW



III.C-1



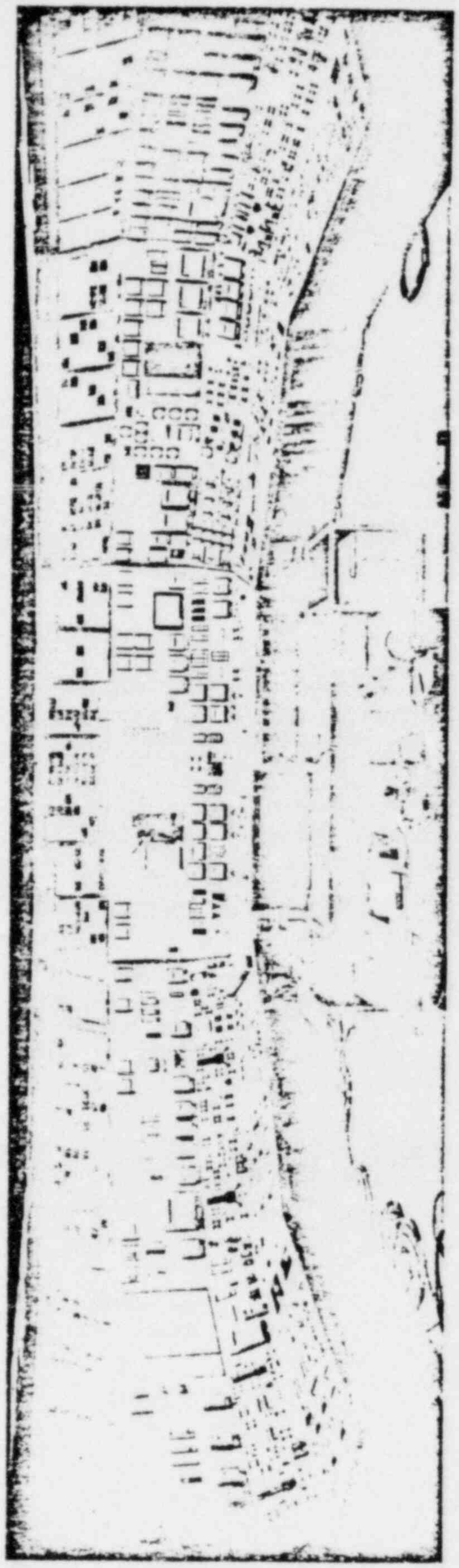
EF2 CONTROL PANEL PROFILE

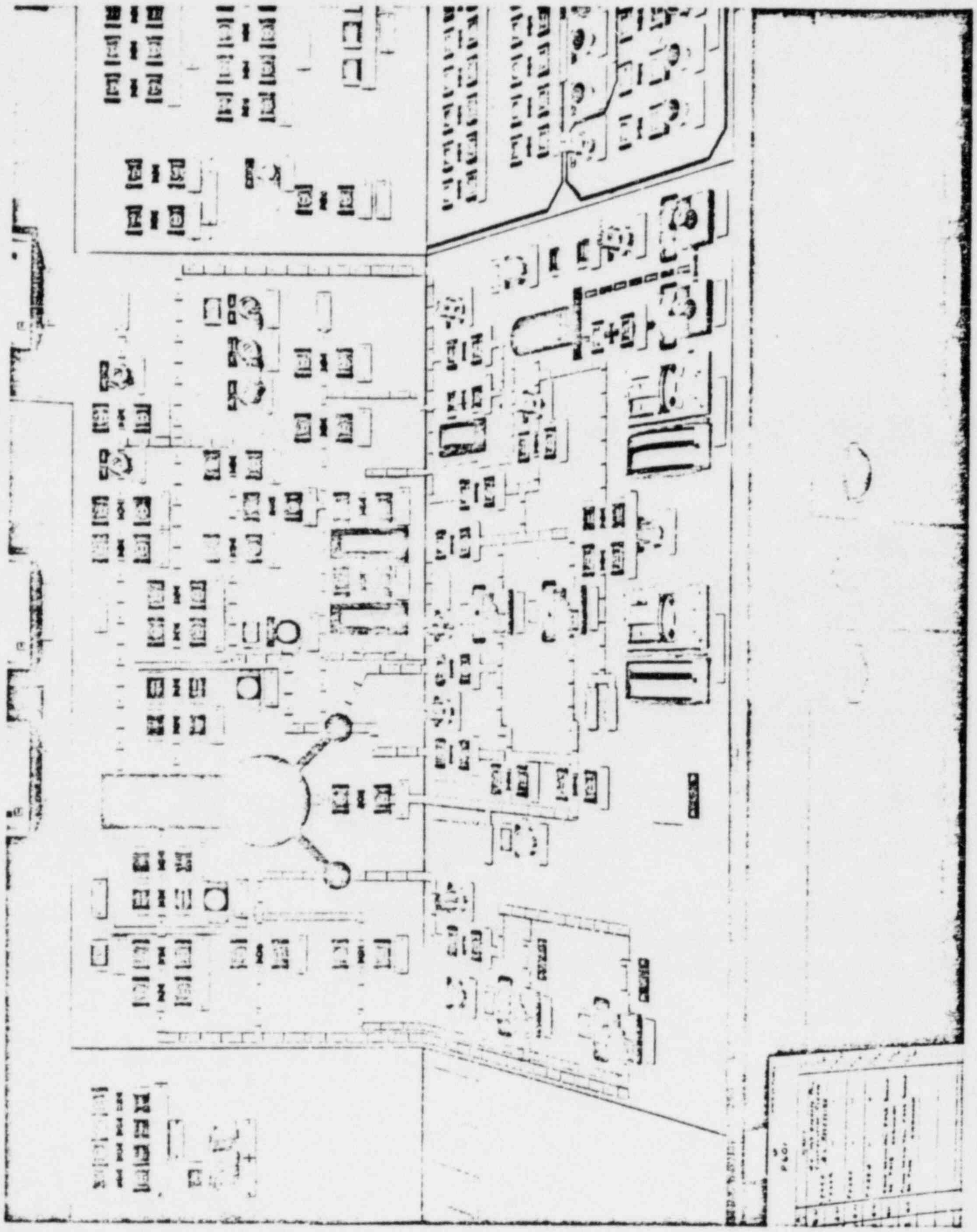


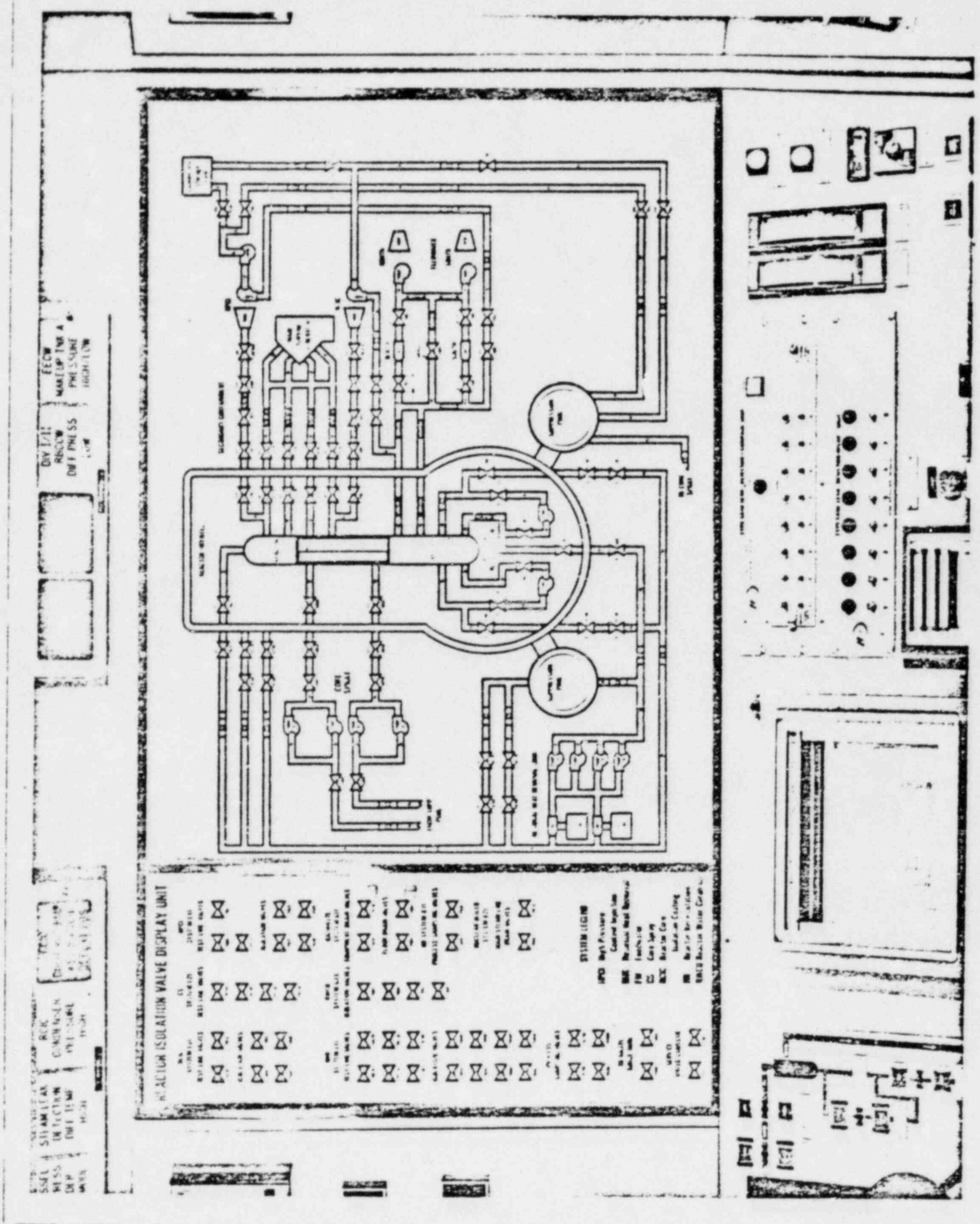
Photograph #10

EF2 FULL SCALE CONTROL PANEL MOCKUP

November 1974






REACTOR ISOLATION VALVE DISPLAY UNIT
SYSTEM LEGEND
EECW MAKEUP TNK A PRESSURE HIGH/LOW
DIV I/II RBCCW DIFF PRESS LOW
RCIC CONDENSER PRESSURE HIGH
STEAM LEAK DETECTION DIFF TEMP HIGH

CONTROL ROOM DESIGN REVIEW FOR HUMAN FACTORS

THE DETROIT EDISON COMPANY HAS BEEN AWARE OF THE PROBLEMS THAT A MAN-MACHINE INTERFACE CAN CAUSE IN THE COMPLEX MODERN POWER PLANTS, CONTROLLED FROM A LOCATION REMOTE FROM THE EQUIPMENT. TO COPE WITH THESE PROBLEMS THE COMPANY DECIDED IN 1965 TO CONVENE A TASK FORCE WHENEVER A NEW CONTROL ROOM WAS BEING DEVELOPED.

THE FERMI 2 CONTROL ROOM TASK FORCE (CRTF) COMPRISED TWO OPERATING EMPLOYEES, REPRESENTATIVES FROM THE THREE ENGINEERING DISCIPLINES: (MECHANICAL, ELECTRICAL AND I&C), AND A PSYCHOLOGIST, TO PROVIDE WHAT IS NOW CALLED THE HUMAN ENGINEERING INPUT. SYSTEM ENGINEERS, FAMILIAR WITH THE FUNCTIONAL REQUIREMENTS OF EACH SYSTEM, WERE CALLED IN WHEN THEIR ASSIGNED SYSTEMS WERE DISCUSSED.

TO COMPENSATE FOR THE LIMITED EXPERIENCE IN NUCLEAR OPERATION, THE CRTF VISITED THE MORRIS SIMULATOR TO GAIN SOME "HANDS ON" EXPERIENCE WITH THE NUCLEAR POWER PLANT SYSTEMS. OTHER NUCLEAR PLANTS WERE ALSO VISITED TO SEE THEIR CONTROL ROOMS, AND TO OBTAIN COMMENTS FROM THEIR OPERATORS. SOME OF THE BASIC DECISIONS REACHED BY THE CRTF, AND LATER IMPLEMENTED IN THE FERMI 2 CONTROL ROOM WERE:

- ALL NORMAL AND ABNORMAL OPERATION OF THE PLANT SHOULD BE DONE FROM THE CONTROL ROOM.

- A PRIORITY SYSTEM FOR EASY ACCESS TO CONTROL COMPONENTS: THE EMERGENCY OPERATION FIRST, THEN STARTUP AND SHUTDOWN, AND THEN THE NORMAL OPERATION.

- PANELS ARRANGED IN A DOUBLE HORSESHOE, WRAPPED AROUND AN OPERATOR'S DESK.

- THE CROSS SECTION OF THE PANEL TO BE CONTOURED RATHER THAN FLAT, TO AFFORD A MAXIMUM WORK SPACE ACCESSIBLE TO THE REACH OF THE OPERATOR.

- FUNCTIONAL SYSTEMS CONTROL COMPONENTS ARRANGED TOGETHER WITH THE INFORMATION REQUIRED FOR OPERATING THEM.

- CONSISTENT USE OF SHAPE CODING: ALL VALVES OPERATED BY MASTER SPECIALTIES PUSBUTTONS, ALL OTHER EQUIPMENT BY CMC SWITCHES: FLOW INDICATORS DIFFERENT SHAPE THAN TEMPERATURE OR PRESSURE INDICATORS, ETC.

- THE DETROIT EDISON STANDARD USE OF COLOR FOR INDICATING LIGHTS AND BACKLIGHTED SWITCHES WAS CONSISTENTLY APPLIED: RED FOR ENERGY FLOW, GREEN FOR NO FLOW, WHITE FOR ABNORMAL WARNING, AMBER FOR CAUTION.

- MIMICS USED WHEREVER POSSIBLE TO HELP THE OPERATORS IN LOCATING THE COMPONENTS MORE EASILY.

- THE ANNUNCIATOR WINDOW LETTERING LARGE ENOUGH TO READ FROM THE CENTER OF THE HORSEHOE.

- USE OF COLOR CODED ANNUNCIATOR WINDOWS.

- MARKING INSTRUMENT SCALES WITH COLOR BANDS TO DENOTE NORMAL, ABNORMAL AND EMERGENCY SITUATIONS.

A FULL SIZE MOCKUP OF THE CONTROL ROOM WAS CONSTRUCTED, AND EACH PLANT FUNCTIONAL SYSTEM LAID OUT ON THE MOCKUP. AFTER THE TASK FORCE HAD AGREED ON THE LAYOUT, THE OPERATORS PERFORMED MOCK OPERATION, STARTUP, SHUTDOWN, AND EMERGENCIES. THE MOCKUP WAS THEN ADJUSTED FOR DEFICIENCIES OBSERVED BY THE TASK FORCE.

THE FINAL PANEL LAYOUT WAS COMPLETELY REVIEWED BY THE PROJECT SYSTEM ENGINEERING GROUP, BEFORE SENDING IT TO THE MANUFACTURER.

AFTER THREE MILE ISLAND, DETROIT EDISON REVIEWED THE CONTROL ROOM COMPARING IT WITH EPRI REPORT NP-1118-54, AND ISSUED GED TECHNICAL REPORT NO. 154 IN MARCH 1980. IT INDICATED CLOSE CONFORMANCE TO THE PRINCIPLES DEFINED BY EPRI.

DETROIT EDISON HAS ALSO ACTIVELY PARTICIPATED IN THE BWR CONTROL ROOM COMMITTEE, DEVELOPING THE CONTROL ROOM REVIEW PROCEDURE AND CHECKLIST. IN JANUARY 1981 THE FERMI 2 CONTROL ROOM WAS REVIEWED BY THE OWNERS' GROUP TEAM COMPRISING REPRESENTATIVES FROM TWO OTHER UTILITIES, GENERAL ELECTRIC PEOPLE AND A HUMAN FACTORS EXPERT FROM MIT. TO ASSURE THE INDEPENDENCE OF THE OBSERVATIONS, DETROIT EDISON LIMITED ITS PARTICIPATION TO SUPPORTING FUNCTIONS ONLY. NO SIGNIFICANT DEVIATIONS FROM THE LATEST HUMAN FACTORS PRINCIPLES WERE FOUND.

THE NRC CONDUCTED A VERY THOROUGH AND DETAILED REVIEW WITH SIMILAR RESULTS. THE NUMEROUS SMALL HUMAN ENGINEERING DEVIATIONS THAT THE STAFF REQUESTED TO CORRECT BEFORE FUEL LOAD WILL BE COMPLETED.

THERE WERE FIVE OPEN ITEMS THAT COULD NOT BE EVALUATED BY THE NRC STAFF.

1. PERMANENT HVAC SYSTEM PERFORMANCE

2. PROCEDURES FOR PERMANENT PANEL MODIFICATIONS

3. CONTROL ROOM SIGNAL EVACUATION

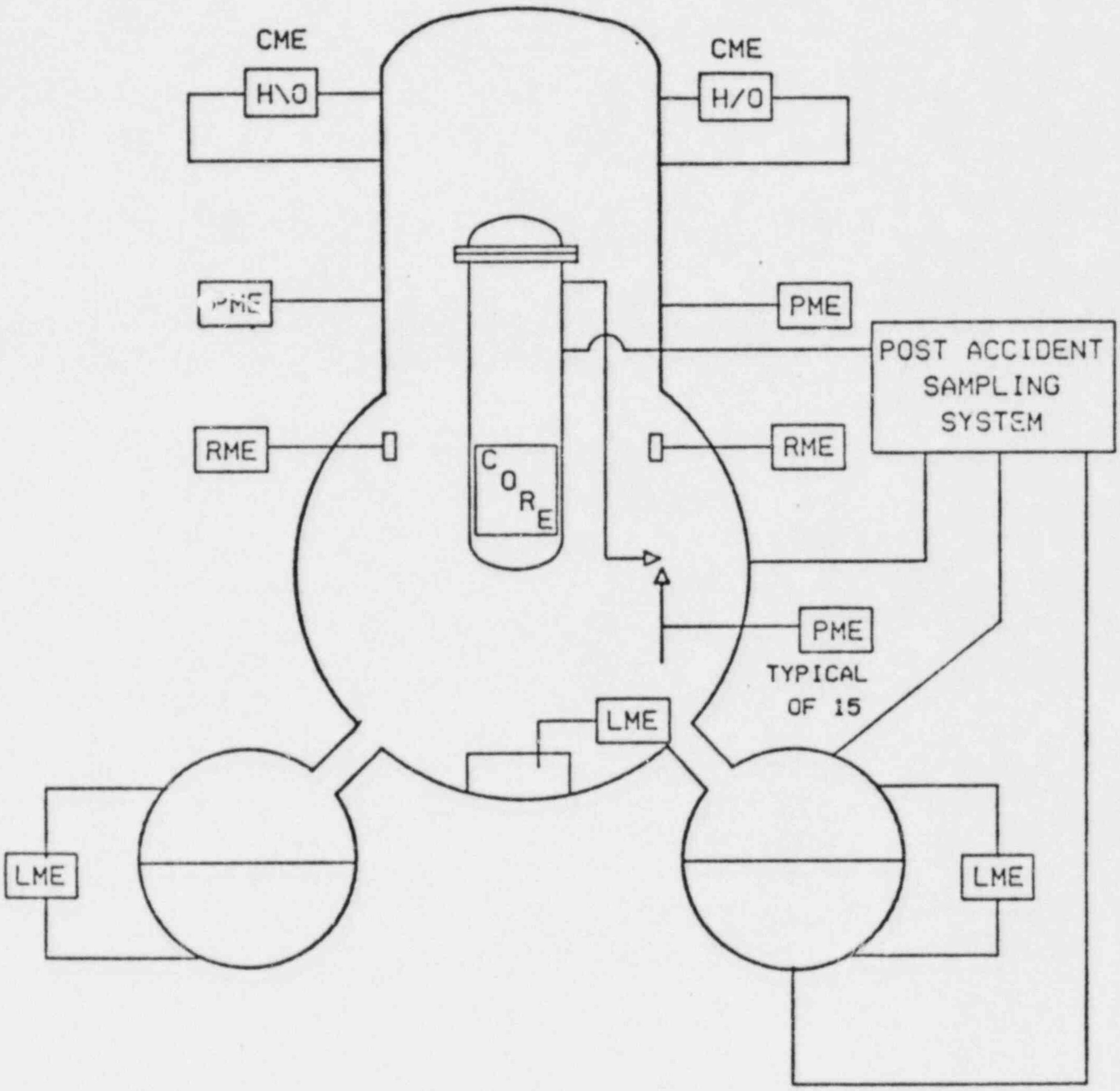
4. PROCESS COMPUTER SOFTWARE

5. COLOR CODING STANDARD

ITEMS 1, 2 AND 5 HAVE BEEN ADDRESSED AND REPORTS HAVE BEEN SUBMITTED TO THE NRC STAFF. ITEM 4 WILL BE SUBMITTED TO THE STAFF BEFORE FUEL LOAD. (IT IS A GE STANDARD PACKAGE, BUT THE PLANT SPECIFICS ARE NOT YET INCORPORATED.)

ITEM 3 IS BEING PROCURED, IT WILL BE TESTED OUT AND RESULTS WITNESSED BY THE NRC RESIDENT I&E INSPECTOR AND SUBMITTED TO THE STAFF BEFORE OCTOBER 15, 1981.

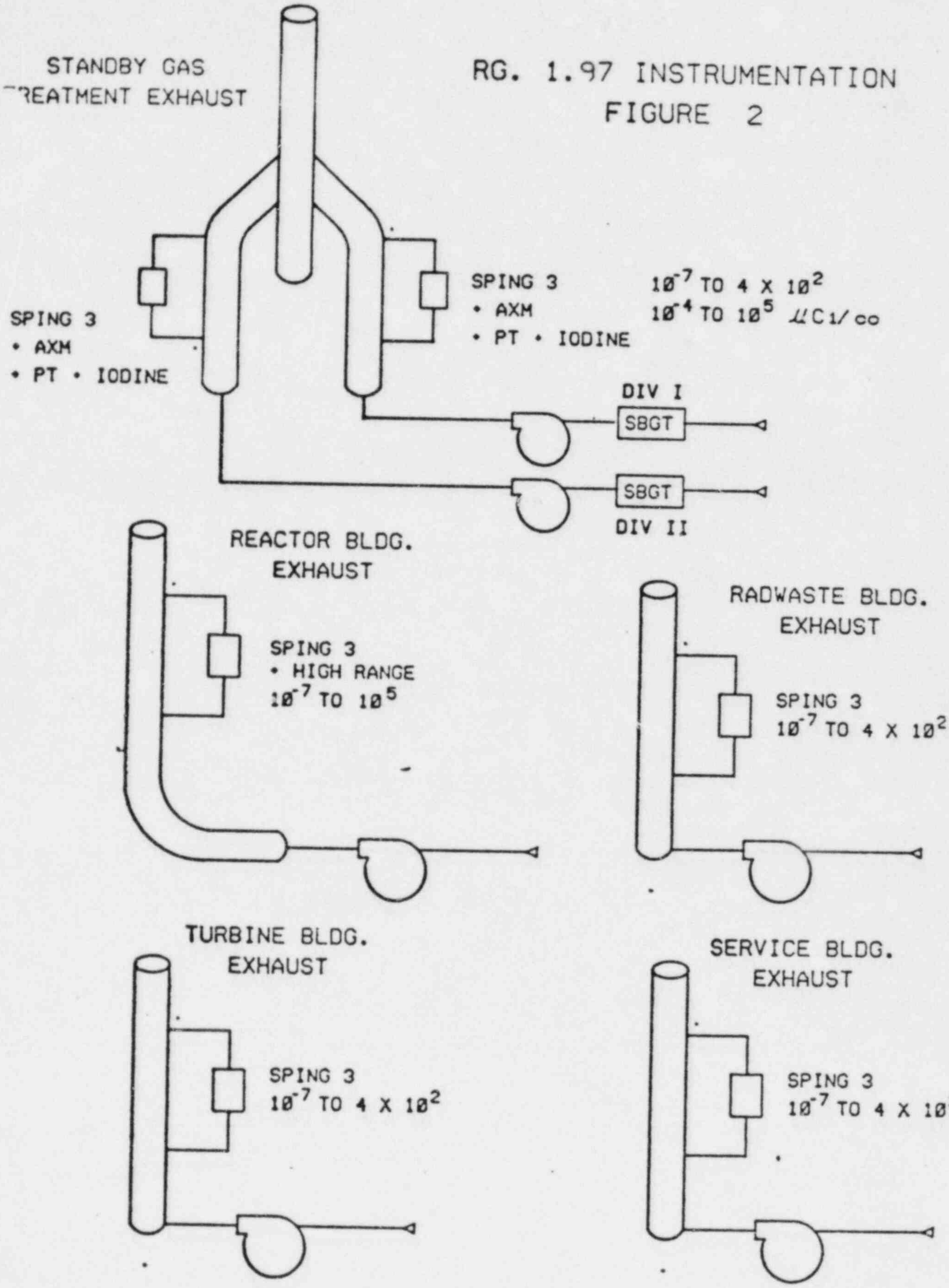
III.D INSTRUMENTATION TO FOLLOW THE COURSE OF A SERIOUS ACCIDENT

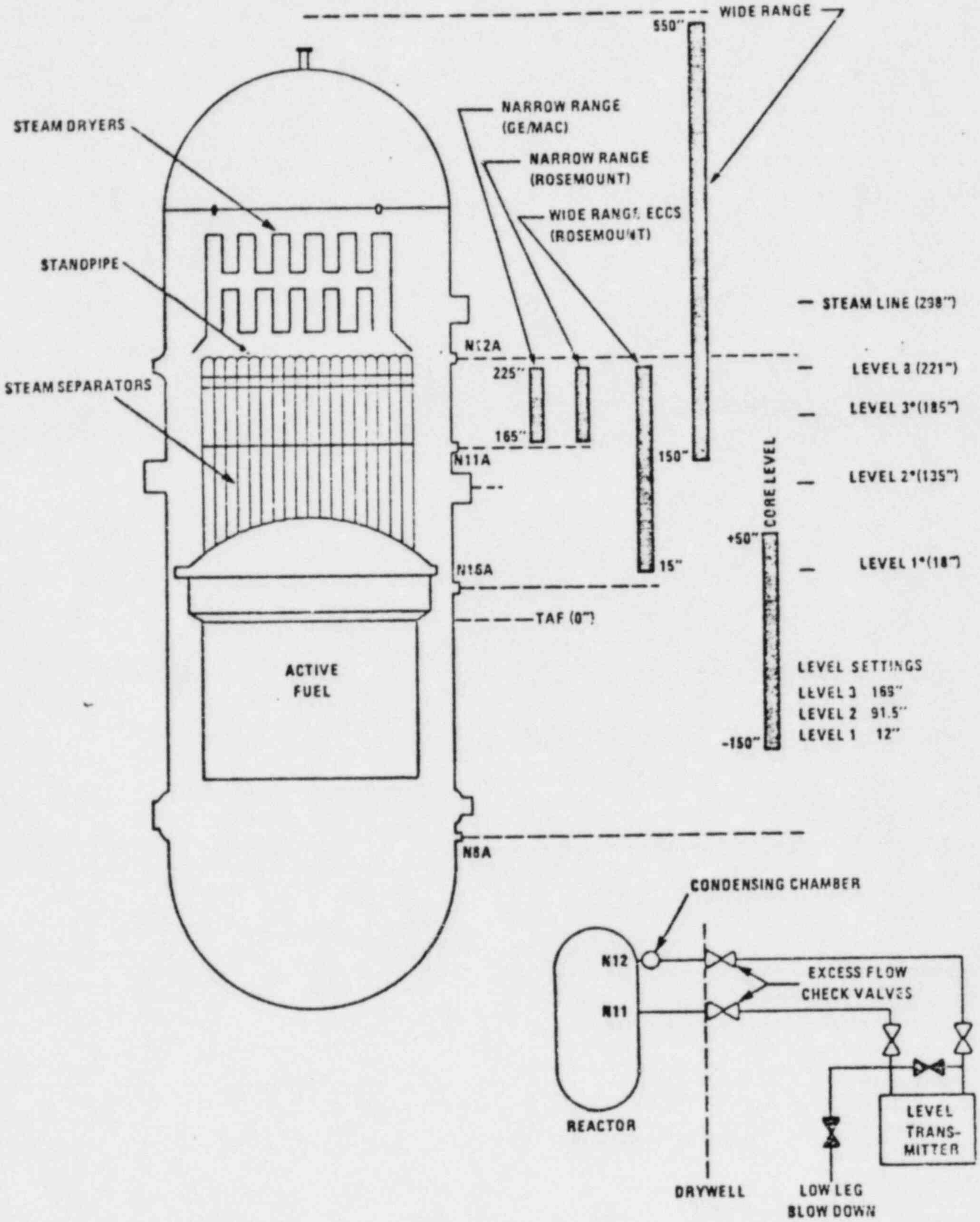


RG 1.97 INSTRUMENTATION
FIGURE 1

RG. 1.97 INSTRUMENTATION

FIGURE 2





III.D-3

FIGURE 2-7
FERMI 2 VESSEL LEVEL RANGES (REFERENCED TO TOP OF ACTIVE FUEL) AND DESIGN OF TYPICAL INDICATOR LINE

SUMMARY

- o BWR IS AN OPEN SYSTEM - RPV WATER LEVEL IS NOT AMBIGUOUS

- o MULTIPLE WATER INJECTION SYSTEMS AVAILABLE: FW, RCIC, HPCI, CS, LPCI, RHR SERVICE WATER CROSS-TIE

- o LEVEL MEASURED DIRECTLY IN VESSEL BY REDUNDANT, SINGLE-FAILURE PROOF, MULTIPLE RANGE SYSTEM

- o DIVERSE INDICATION OF POTENTIAL OR ACTUAL CLAD BREACH: LOW WATER LEVEL; FISSION PRODUCTS IN RX COOLANT/CONTAINMENT AIR/ SUPPRESSION POOL WATER; HIGH CONTAINMENT HYDROGEN: LOSS OF MAKEUP

- o EMERGENCY PROCEDURES ARE WRITTEN BASED ON PRESENT LEVEL INSTRUMENTATION -- OPERATOR ACTIONS WOULD NOT CHANGE IF THERMOCOUPLES WERE ADDED: WITH INDICATION OF TROUBLE, FLOOD THE CORE.

∴ INCORE THERMOCOUPLES ARE NOT NEEDED FOR BWR'S

REGULATORY GUIDE 1.97 REQUIREMENTS

FOLLOWING THE TMI EVENT, EDISON INTERNALLY INITIATED A REVIEW OF ACCIDENT PREVENTION AND DETECTION INSTRUMENTATION ALONG WITH THE POST ACCIDENT INSTRUMENTATION.

THE EDISON REVIEW INDICATED THE NEED FOR SEVERAL CHANGES WHICH WERE MADE TO THE PLANT SYSTEMS. A SHORT TIME LATER, REGULATORY GUIDE 1.97, REVISION 2, WAS ISSUED BY THE NRC AND CONCURRENTLY THE MORE SIGNIFICANT INSTRUMENT REQUIREMENTS WERE INCLUDED IN NUREG-0737.

EDISON BELIEVES THAT THE FERMI 2 PLANT, WITH FEW EXCEPTIONS, (NOTABLY THE INCORE THERMOCOUPLES), IS IN CLOSE CONFORMANCE WITH THE INTENT OF REGULATORY GUIDE 1.97, REVISION 2.

IN ORDER TO MORE COMPLETELY DESCRIBE THE NUMBER AND EXTENT OF CHANGES MADE TO THE FERMI 2 INSTRUMENTATION TO MEET THE REQUIREMENTS OF NUREG-0737 AND REGULATORY GUIDE 1.97, THE INSTRUMENTS AFFECTED ARE SHOWN ON THE ATTACHED SKETCHES AND LISTED BELOW:

- o NOBLE GAS EFFLUENT MONITORS
- o IODINE AND PARTICULATE MONITORS

- HIGH RANGE CONTAINMENT RADIATION MONITORS
- SUPPRESSION POOL LEVEL MONITORS
- DRYWELL PRESSURE MONITORS
- CONTAINMENT HYDROGEN AND OXYGENMONITORS
- SAFETY/RELIEF VALVE POSITION MONITORS
- POST ACCIDENT SAMPLING SYSTEM
- DRYWELL SUMP LEVEL MONITORS

EDISON PLANS TO WORK WITH THE BWR OWNERS' GROUP IN ADDRESSING THE ADDITIONAL REQUIREMENTS OF REGULATORY GUIDE 1.97 BEYOND NUREG-0737.

INADEQUATE CORE COOLING INSTRUMENTATION

AS A DIRECT RESULT OF THE TMI EVENTS, THE NRC STAFF DEFINED A BROAD OBJECTIVE IN ITEM II.F.2 OF NUREG-0737. THIS INADEQUATE CORE COOLING (ICC) INSTRUMENTATION REQUIREMENT WAS DIRECTED PRIMARILY TOWARD THE PRESSURIZED WATER REACTOR DESIGNS AS A RESULTS OF THE PROBLEMS FOUND IN THE MEASUREMENT OF CORE COOLING ADEQUACY UTILIZING PRESSURIZER LEVEL SYSTEM AND/OR THE CORE EXIT THERMOCOUPLES.

THE NEW REQUIREMENT SPECIFIES THAT AN UNAMBIGUOUS INDICATION OF ICC BE DEVELOPED. IN THE CASE OF A BWR DESIGN, THE WATER LEVEL INSTRUMENTATION CAN NEVER BE AMBIGUOUS. THE BWR WATER LEVEL INSTRUMENTATION PROVIDES AN ADVANCED INDICATION OF THE APPROACH TO ICC DUE TO THE USE OF INSTRUMENTS OF MULTIPLE OVERLAPPING RANGES. A COMPLETE ANALYSIS OF THE LEVEL MEASUREMENT SYSTEM OF A BWR SUCH AS FERMI 2 CAN BE FOUND IN NEDO-2470A. EDISON WAS REQUESTED BY THE STAFF TO SUPPLEMENT THIS INFORMATION BY LETTER AND ESTABLISH FERMI 2 SPECIFIC RANGES, INSTRUMENT TYPES AND OTHER PHYSICAL DESIGN INFORMATION. THE STAFF REVIEWED AND REPORTED ON THE APPLICABILITY OF THIS NEDO TO THE FERMI 2 LEVEL MEASUREMENT SYSTEM IN THE SER SECTION H.II.K.1.23.

IN ADDITION TO THE INSTRUMENTATION THE SYMPTOMATIC EMERGENCY PROCEDURES WHICH DEPEND PRIMARILY ON THE LEVEL SYSTEM ARE ESSENTIALLY COMPLETE AND HAVE BEEN APPROVED BY THE NRC.

INADEQUATE CORE COOLING INSTRUMENTATION

IMMEDIATELY FOLLOWING THE TMI EVENT, EDISON INDEPENDENTLY REVIEWED THE FERMI 2 LEVEL MEASUREMENT SYSTEM BECAUSE OF THE AMBIGUITY ASSOCIATED WITH THE PRESSURIZER LEVEL WHICH WAS OBSERVED AT TMI DURING THE ACCIDENT.

BASED ON THE RESULT OF OUR INTERNAL STUDY AND ANALYSIS, EDISON ENDORSED THE BWR OWNERS' GROUP POSITION THAT FERMI 2 DOES NOT NEED ANY ADDITIONAL INSTRUMENTATION TO PROVIDE UNAMBIGUOUS INDICATION OF ADEQUATE CORE COOLING.

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GE DESIGN CRITERIA FOR BWR STABILITY

TYPES OF STABILITY	DECAY RATIO ORIGINAL	DECAY RATIO PROPOSED
1. CHANNEL HYDRODYNAMIC STABILITY	$\leq .5$	≤ 1.0
2. REACTOR CORE STABILITY	$\leq .5$	≤ 1.0
3. TOTAL SYSTEM STABILITY	$\leq .25$	≤ 1.0

APPENDIX XI
GE REACTORS: DESIGN CRITERIA FOR BWR STABILITY

GE PLANTS (OPERATING AND NEW PLANTS)
DECAY RATIO AT THE EXTRAPOLATED ROD BLOCK
LINE INTERSECTION WITH THE NATURAL CIRCULATION LINE

TYPE OF REACTOR	PLANT	DECAY RATIO (RELOAD NO.)
BWR/3	QUAD CITIES-2	.42(1)
	DRESDEN 2	.37(1)
BWR/4	PEACH BOTTOM-2	.64(1)
	DUANE ARNOLD	.88(1)
	HATCH-1	.87(1)
	SHOREHAM	.58(NEW)
	FERMI	.63(NEW)
	SUSQUEHANNA	.68(NEW)
BWR/5	LaSALLE	.50(NEW)
	ZIMMER-1	.66(NEW)
BWR/6	GRAND GULF	.97(END OF LIFE)
	CLINTON	.98(END OF LIFE)

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*FOR NEWER CORE (FUEL) DESIGN, THE DECAY RATIO INCREASES AND APPROACHES TO 1.0 FOR THE CORE AT END OF LIFE CYCLE.

GE OPERATING PLANTS*

DECAY RATIO AT THE EXTRAPOLATED ROD BLOCK LINE INTERSECTION WITH THE NATURAL CIRCULATION LINE

	PLANT	DECAY RATIO (RELOAD NO.) EARLIER CYCLE	LATER CYCLE
1.	DRESDEN 2	.37(1)	.58(3)
2.	DRESDEN 3	.42(3)	.48(5)
3.	QUAD CITIES 2	.42(1)	.62(3)
4.	MONICELLO	.47(3)	.55(8)
5.	MILLSTONE-1	.45(3)	.70(6)
6.	NINE MILE POINT-1	.55(5)	.85(8)
7.	DUANE ARNOLD	.88(1)	.85(4)
8.	PILGRIM	.67(2)	.94(3)
9.	PEACH BOTTOM-2	.64(1)	.996(3)

---

*1. DECAY RATIO INCREASES AS OPERATION PROCEEDS INTO LATER CYCLE ⇒ LESS STABLE.

2. FOR MOST CASES, DECAY RATIO IS LARGER THAN .5.

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CONCLUSION

STABILITY FOR BWR DECREASES DUE TO RECENT FUEL DESIGN CHANGES:

- SMALLER ROD DIAMETER
- INCREASED GAP CONDUCTANCE AS A RESULT OF PRESSURIZATION

NRC POSITION

(1) PAST OPERATING EXPERIENCE, STABILITY TESTS, AND THE INHERENT THERMAL-HYDRAULIC CHARACTERISTICS OF LIGHT WATER REACTORS PROVIDE A BASIS FOR APPROVAL OF BWR STABILITY FOR NORMAL OPERATION AND ANTICIPATED TRANSIENTS.

(2) IN ORDER TO PROVIDE ADDITIONAL MARGIN TO STABILITY LIMITS, NATURAL CIRCULATION OPERATION IS PROHIBITED, UNLESS

(3) RESULTS OF NRC GENERIC EVALUATION, "THERMAL HYDRAULIC STABILITY" MAY PROVIDE A BASIS TO CHANGE THIS POSITION.

III.E. PLANT SEISMIC DESIGN

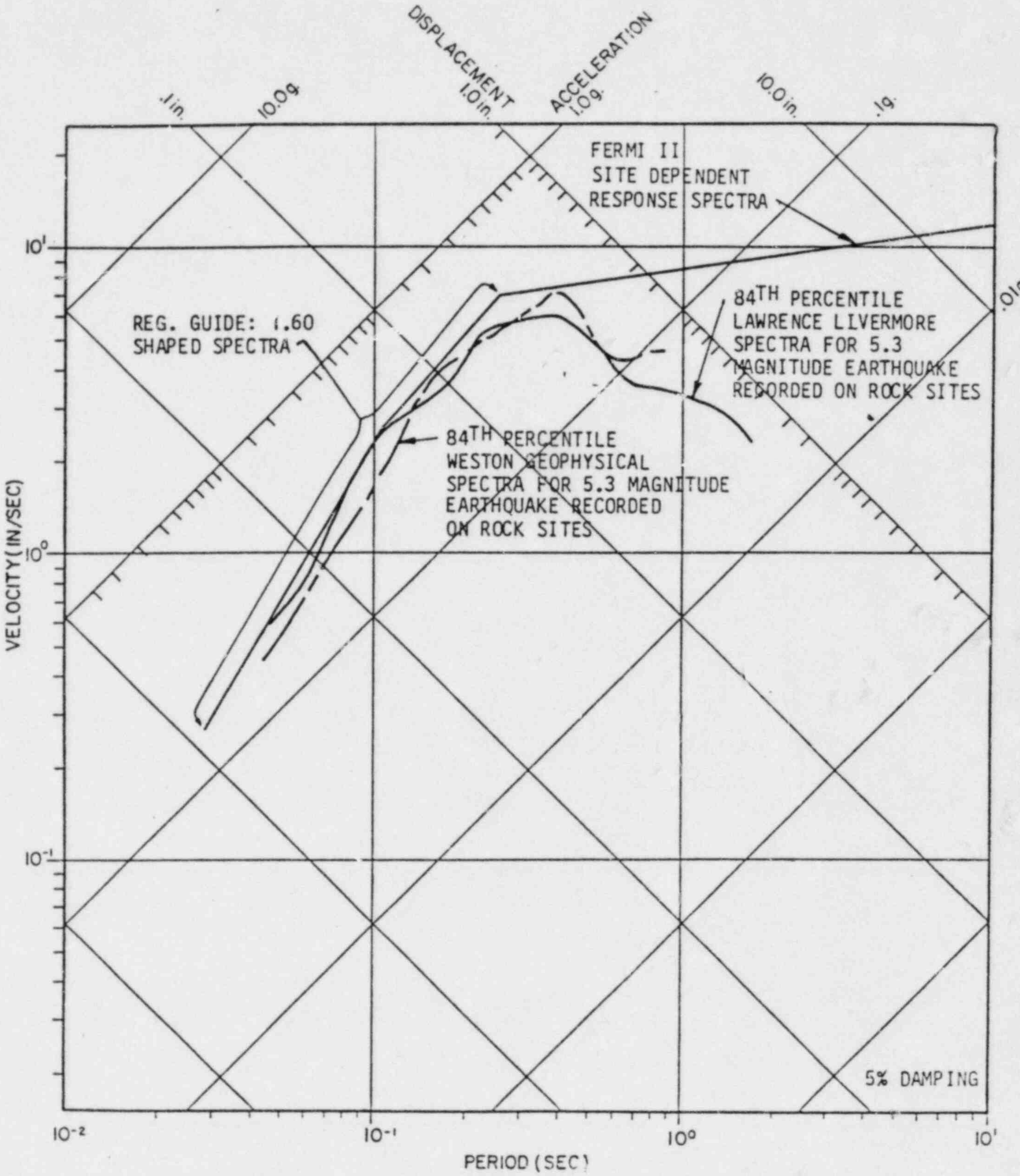
SUPPLEMENTARY SEISMIC EVALUATION

- o NRC Staff Request of March 12, 1981
  - Use R.G. 1.60 Spectra Shape Anchored at 19% G or
  - Site Specific for Rock Site at Magnitude 5.3 ± .5

- o Met with NRC on March 27, 1981 and Committed To:

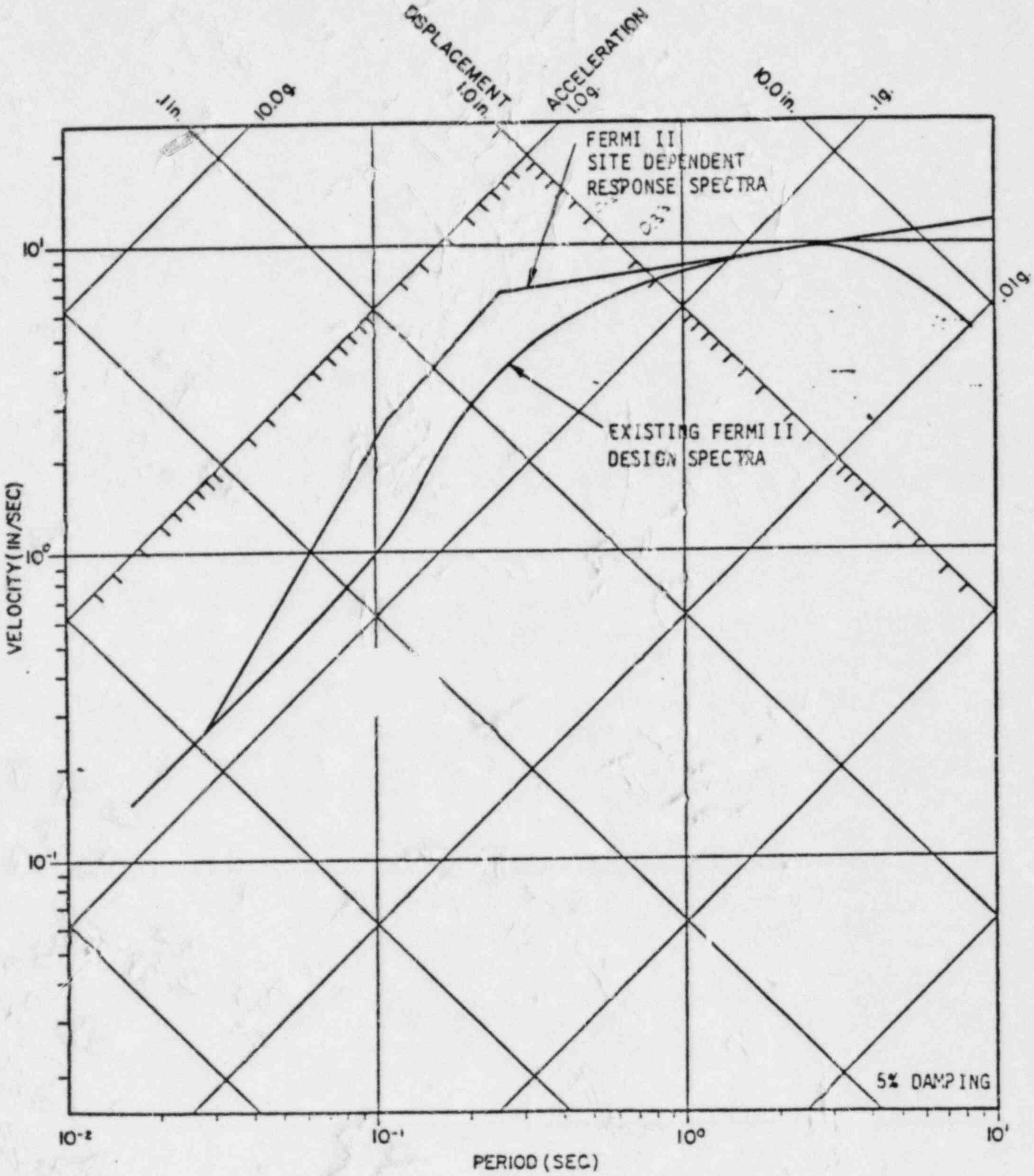
  1. Use Site Specific Spectrum
  2. Envelopes LLL-Spectrum (84th % For M = 5.3)
     Envelopes Weston Spectrum
  3. Has Basic R.G. 1.60 Shape
  4. Anchored at 15% G
  5. Modified for Far-Field Strong Motions
  6. Use Vertical Acceleration to be 2 x DBE (Floor (Response)
  7. No Change in OBE (100 - 300 Year Return Period)

F. E. Gregor
8/3/81



COMPARISON OF FERMI II SITE DEPENDENT RESPONSE SPECTRA WITH LAWRENCE LIVERMORE AND WESTON GEOPHYSICAL EASTERN U.S. ROCK SPECTRA FOR MAGNITUDE 5.3 EARTHQUAKES

FIGURE 2.1-1



COMPARISON OF FERMI II SITE DEPENDENT RESPONSE SPECTRA AND EXISTING FERMI II DESIGN SPECTRA

FIGURE 2.1-2

SARGENT & LUNDY ENGINEERS

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CALC NO. CALC NO 8 11 0-3
PROJECT FERMI
PROJECT NO. 8139-38
DAMPING 0.050
PAGE

REV 0

SLS(NI)-18
5-4-81

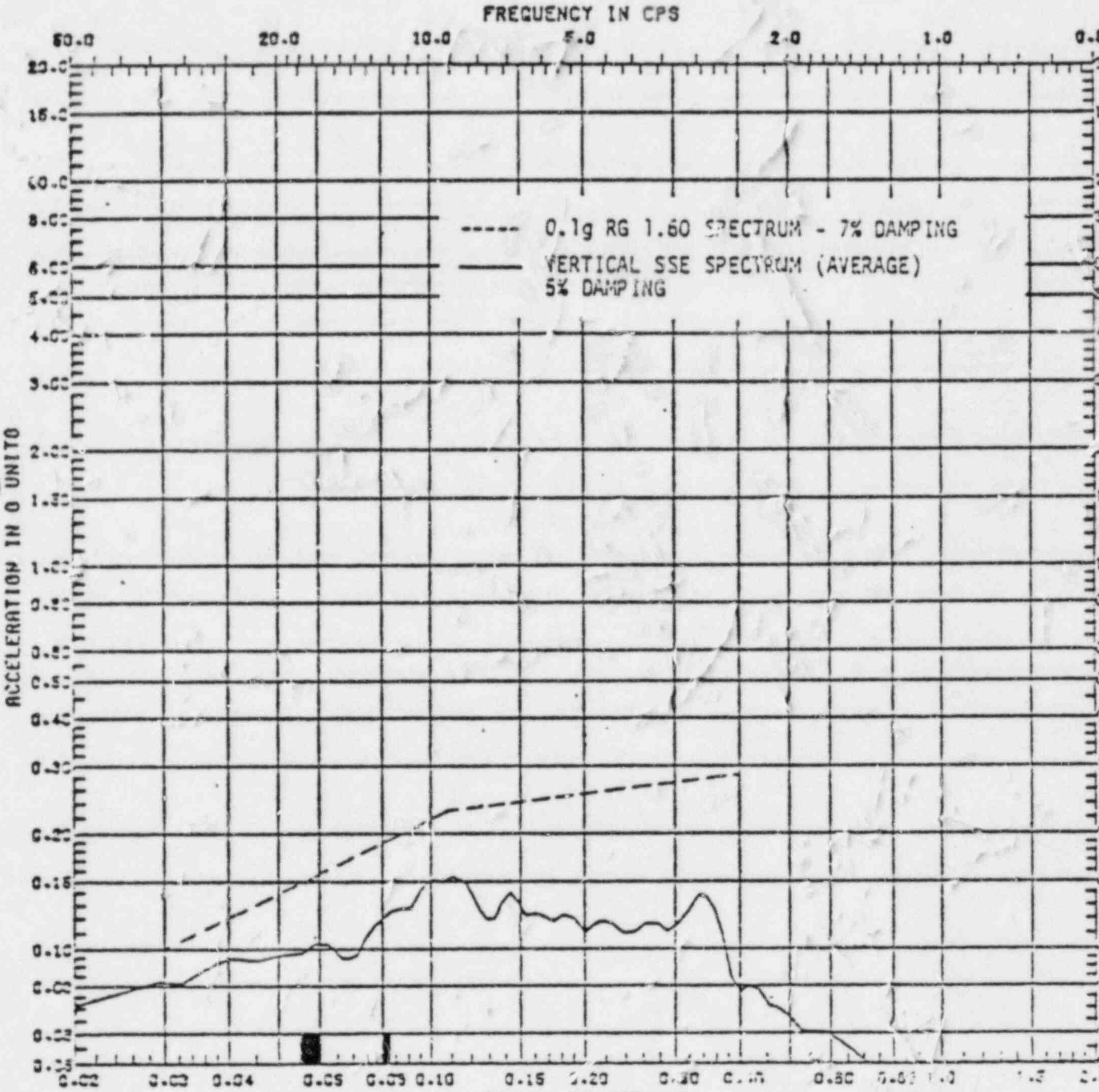

FIGURE 2.1-3

Comparison Of Regulatory Guide 1.60 and Average Time History Spectra

RESPONSE SPECTRUM
SPECTRA NO
NODE AVERAG
III.E-4
ELEVATION
1 DIRECTION
ANGLE
LOCATION ELCEN40.34.TAFT.OLYMPIA

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REEVALUATION

- Conducted Reevaluation:
  - Determine Shutdown scenario with LOPA
  - Identify Systems, Structures and Components
  - Not in Combination with LOCA
  - All Other Ground Rules per SRP
  - Completed Reassessment May 29, 1981, Report EF2-53,332
  - Final Report Docketed July 15, 1981 (Revision 1)

- Conclusions:
  - Plant is Capable of Safely Shutting Down
  - One Cable Tray Hanger Slightly Over Yield
  - 25 Items Require Further Evaluation, Requalification, Retesting or Replacement
  - Independent Assessment Using Ductility
  - Ductility Ratio of 2.0 will Reduce SSE Below DBE
  - NRR Audits on Structural and Mechanical Components

F.E.Gregor
7/20/81

PLANT SEISMIC DESIGN

ON MARCH 12, 1981 DETROIT EDISON WAS REQUESTED BY THE NRC STAFF TO REEVALUATE THE SEISMIC DESIGN OF THE FERMI-2 PLANT BASED ON:

A. USE OF A REGULATORY GUIDE 1.60 SPECTRUM SHAPE AND ANCHORED AT 19% G OR

B. DEVELOPMENT OF A SITE SPECIFIC GROUND RESPONSE SPECTRA REPRESENTATIVE OF EARTHQUAKE HISTORIES OF MAGNITUDE 5.3 $\pm$ .5 APPLICABLE TO ROCK SITES.

IN A MEETING WITH NRC STAFF ON MARCH 27, 1981, DETROIT EDISON COMMITTED TO CONDUCT A REEVALUATION OF THE PLANT'S CAPABILITY TO SAFELY SHUTDOWN, UTILIZING A QUASI SITE SPECIFIC SPECTRUM, DEVELOPED BY WESTON GEOPHYSICAL BASED ON A MAGNITUDE 5.3. $\pm$ 0.5 EARTHQUAKE. THE SHAPE OF THE SPECTRUM EVELOPES THE LAWRENCE LIVERMORE SPECTRUM AND FOLLOWS IN GENERAL TREND THE REGULATORY GUIDE 1.60 SHAPE. IN ADDITION, FAR FIELD STRONG MOTION EFFECTS HAVE BEEN INCLUDED. EVEN THOUGH THE ANCHOR POINT (ZPA) DID NOT CHANGE FROM THE SSE (0.15 g), THE GROUND ACCELERATION INCREASED BY AS MUCH AS 60% OVER THE PREDOMINANT BUILDING FREQUENCY RANGE. A COMPARISON OF THE SSE AND MODIFIED GROUND RESPONSE SPECTRA ARE SHOWN IN FIGURE 2.1-2.

ADEQUATE EARTHQUAKE HISTORIES ARE NOT AVAILABLE FOR DEVELOPMENT OF A SITE SPECIFIC VERTICAL GROUND RESPONSE SPECTRUM. THE SSE VERTICAL SPECTRUM IS BASED ON FOUR INDIVIDUAL TIME HISTORIES AND WAS COMPARED TO THE REGULATORY GUIDE 1.60 VERTICAL SPECTRUM SHAPE. THE COMPARISON (SHOWN ON FIGURE 2.1-3) DETERMINED THAT THE LATTER EXCEEDS THE SSE BY A FACTOR OF 1.6 IN THE DOMINANT STRUCTURAL FREQUENCIES.

DUE TO INHERENT VALLEYS IN THE FOUR TIME HISTORIES USED FOR THE VERTICAL SSE, WHICH TEND TO DRIVE DOWN THE AVERAGE SPECTRA, A FACTOR OF 2.0 WAS DETERMINED TO BE THE APPROPRIATE BASIS TO BE USED FOR THE REEVALUATION. THE MULTIPLIER WAS APPLIED DIRECTLY TO THE SSE FLOOR RESPONSE SPECTRA.

HISTORICAL RECORDS OF SEISMIC ACTIVITY IN THE CENTRAL REGION OF THE U.S. WERE USED TO ESTIMATE THE FREQUENCY OF OCCURRENCE OF A 0.08 g OBE GROUND ACCELERATION. ON THE BASIS OF THIS HISTORICAL ANALYSIS, IT WAS CONCLUDED THAT THE OBE RECURRENCE FREQUENCY AT THE FERMI 2 SITE IS, AS A MINIMUM, IN THE ORDER OF 100 TO 300 YEARS. THEREFORE, NO CHANGE IN THE OBE DEFINITION HAS OCCURRED.

BASED ON THE ABOVE CRITERIA, NEW SYNTHETIC N-S AND E-W TIME HISTORIES, MATCHING THE 7% DAMPED SITE SPECTRUM WERE DEVELOPED AND APPLIED SIMULTANEOUSLY TO OBTAIN ACCELERATION RESPONSE

TIME HISTORIES FOR THE STRUCTURAL MODES. THESE TIME HISTORIES IN TURN ARE USED AS INPUT MOTIONS TO THE GENERATION OF FLOOR RESPONSE SPECTRA. THE REACTOR VESSEL AND ITS INTERNALS ARE PART OF THE STRUCTURAL MODEL, GENERATING SHEAR LOADS AND MOMENTS USED IN THE COMPONENT STRESS EVALUATION.

REEVALUATION PROGRAM

THE PRINCIPAL STRUCTURES, SYSTEMS AND COMPONENTS REQUIRED FOR SAFE SHUTDOWN AND COOLDOWN WERE REEVALUATED BASED ON THE NEW EARTHQUAKE CRITERIA AND EMPLOYING DAMPING RATIOS AND LOAD COMBINATIONS CONSISTENT WITH CURRENT PRACTICE (I.E. REGULATORY GUIDE 1.60 AND SRP).

THE EVENT SCENARIO (TRANSIENT) WAS DEVELOPED TO IDENTIFY SHUTDOWN AND COOLDOWN EQUIPMENT REQUIRED TO FUNCTION, BUT NOT INCLUDING A SIMULTANEOUS LOCA. LOSS OF OFFSITE POWER (LOPA) WAS ASSUMED TO OCCUR. TABLE 5.2-1 OF OUR REPORT EF2-53332 LISTS THE PRINCIPAL AND AUXILIARY SYSTEMS REQUIRED FOR THIS EVENT. PROCESS AND FUNCTIONAL CONTROL DIAGRAMS WERE THEN USED TO IDENTIFY PARTICULAR EQUIPMENT AND COMPONENTS. OVER 200 INDIVIDUAL PIECES OF EQUIPMENT WERE SELECTED FOR REASSESSMENT, REPRESENTING MORE THAN 500 ITEMS. DETAILED RESULTS OF THE EVALUATION ARE PRESENTED IN A PRELIMINARY REPORT EF2-53,332 AND A FINAL REPORT, REVISION 1 OF EF2-53,332 WHICH WAS DOCKETED ON JULY 15,1981.

EVEN THOUGH THE REEVALUATION IDENTIFIED ABOUT 24 ITEMS THAT REQUIRE FURTHER ANALYTICAL REFINEMENT OR OTHER CORRECTIVE ACTIONS, THE REASSESSMENT CONCLUDED THAT THE PLANT IS CAPABLE OF SAFELY SHUTTING DOWN AND COOLING DOWN AND DEMONSTRATING THE MARGINS OF SAFETY AVAILABLE TO WITHSTAND A HIGHER MAGNITUDE EARTHQUAKE.

IN PARTICULAR THE PRIMARY SHUTDOWN AND COOLDOWN EQUIPMENT, INCLUDING THE PIPING SYSTEMS AND VALVES INHIBIT AMPLE MARGINS, EVEN WHEN COMPARED TO CODE ALLOWABLES.

III.F. RELIABILITY OF STATION ELECTRICAL POWER

III.F-1

A-141

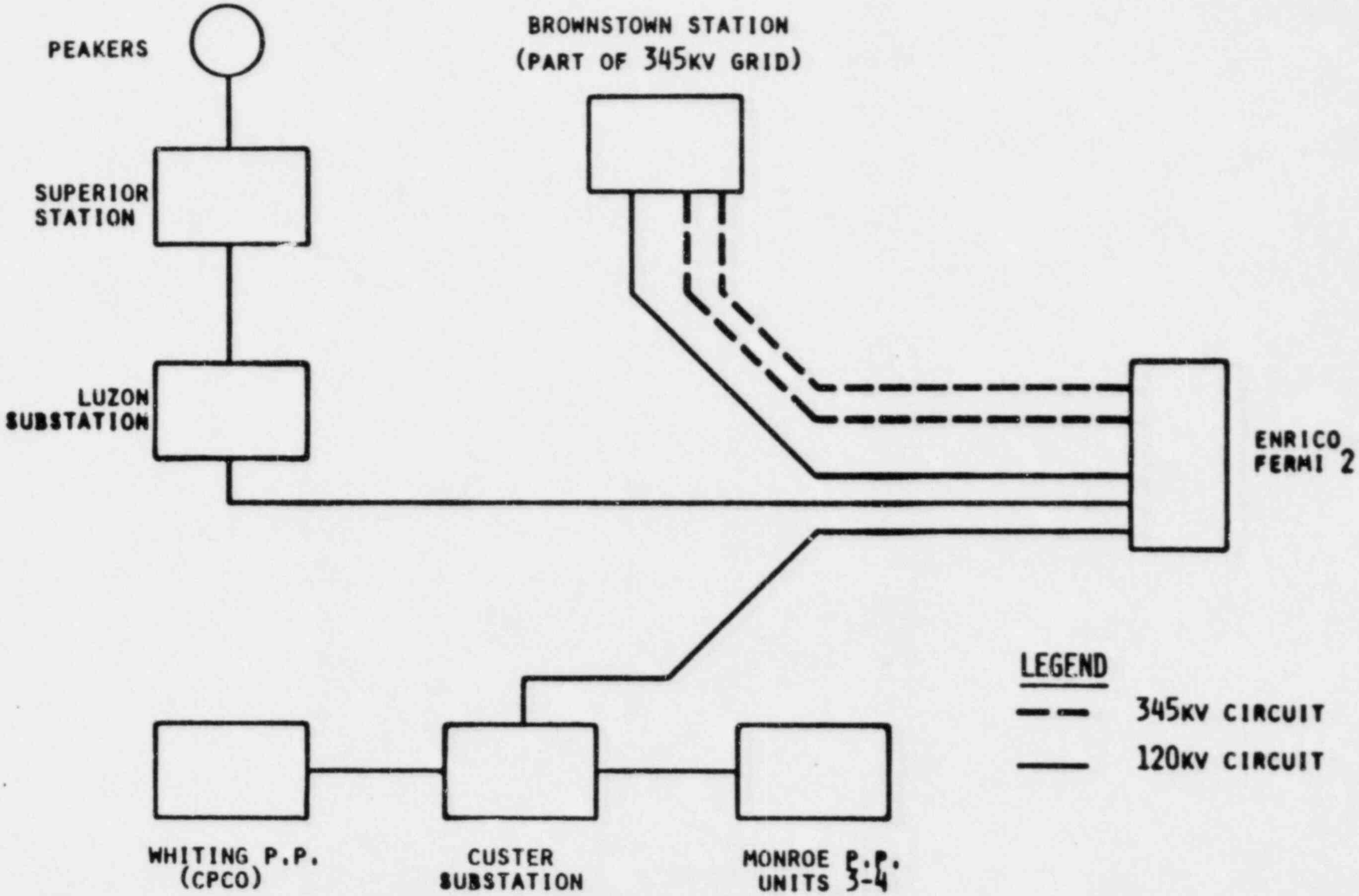


FIG. 1 ENRICO FERMI 2 OFFSITE POWER SOURCES

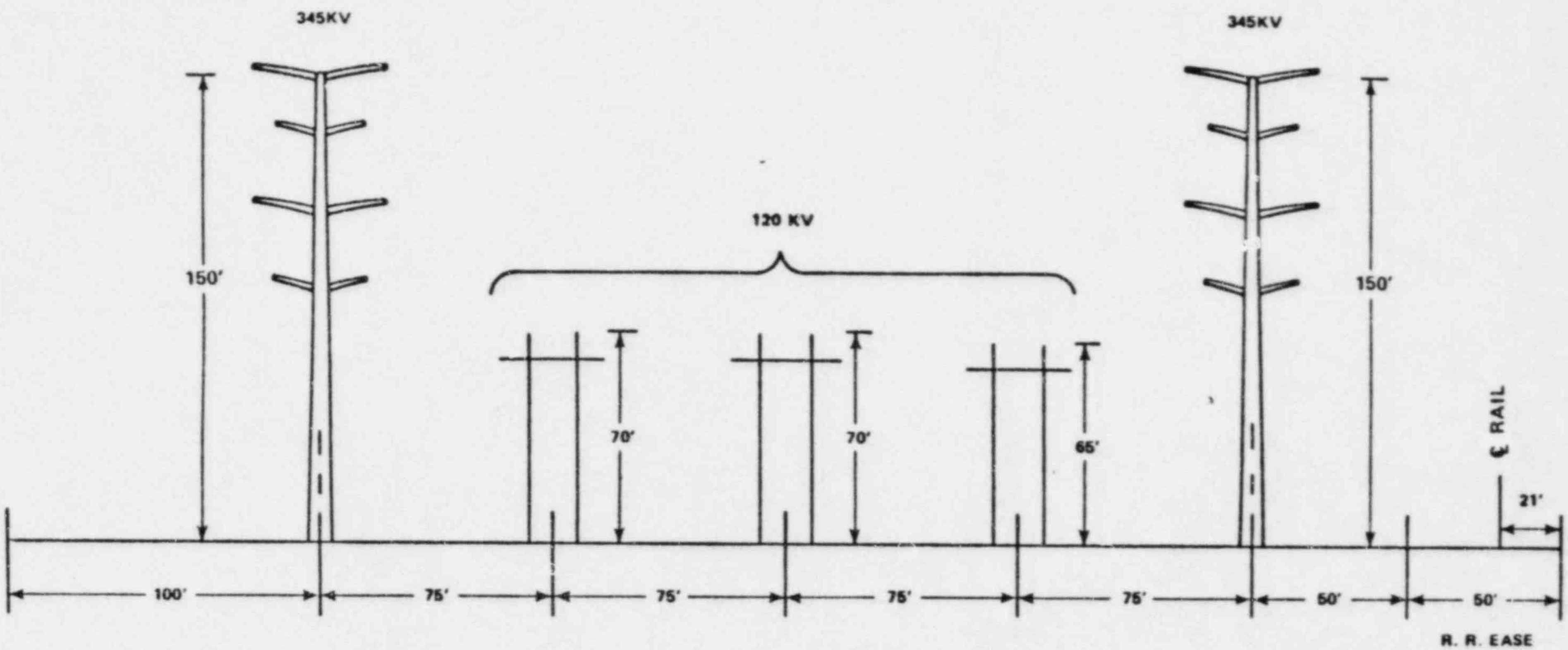


FIGURE 3
FERMI TRANSMISSION CORRIDOR
ELEVATION VIEW

III.F-3
A-143

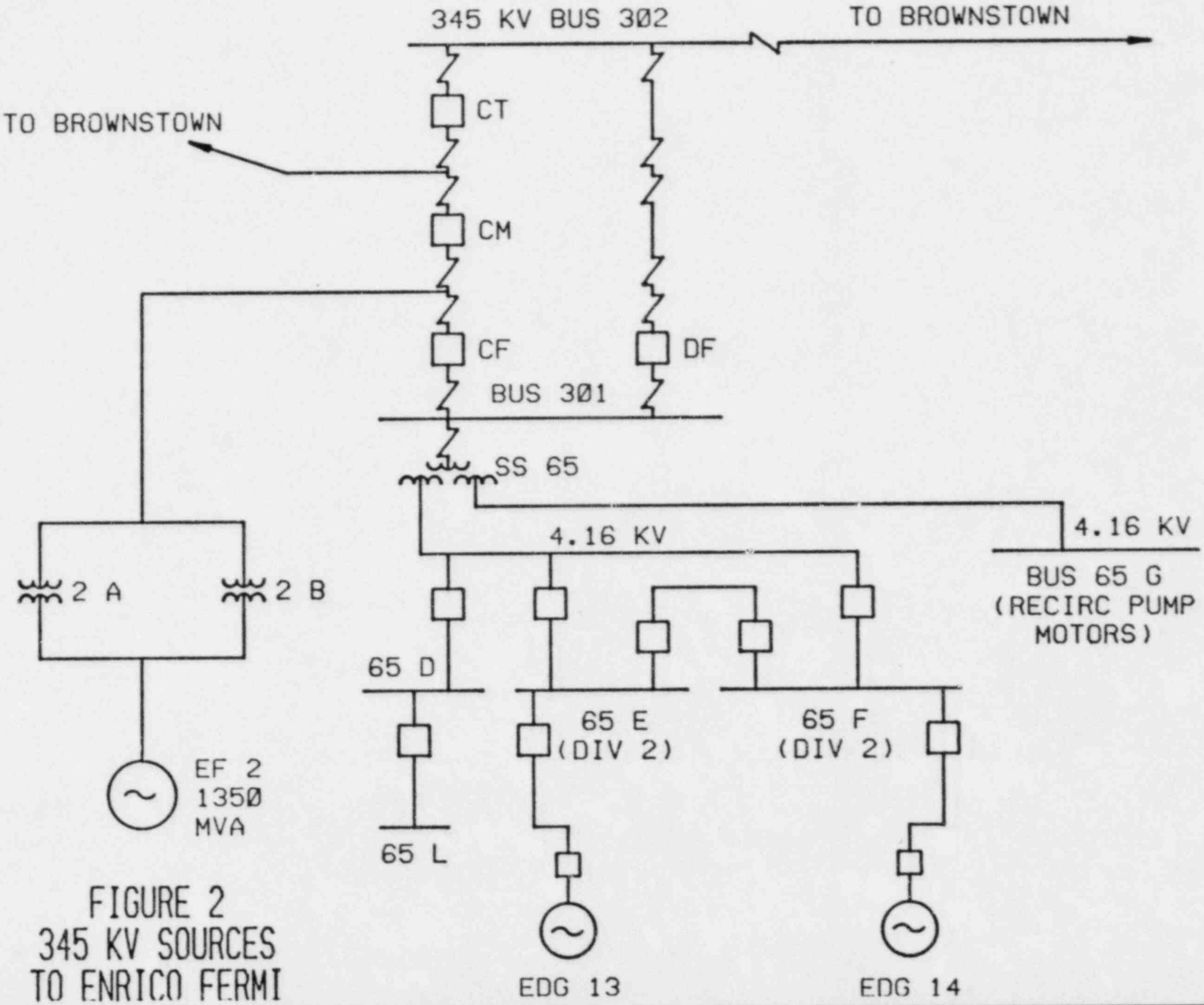


FIGURE 2
345 KV SOURCES
TO ENRICO FERMI

III.F-4
A-144

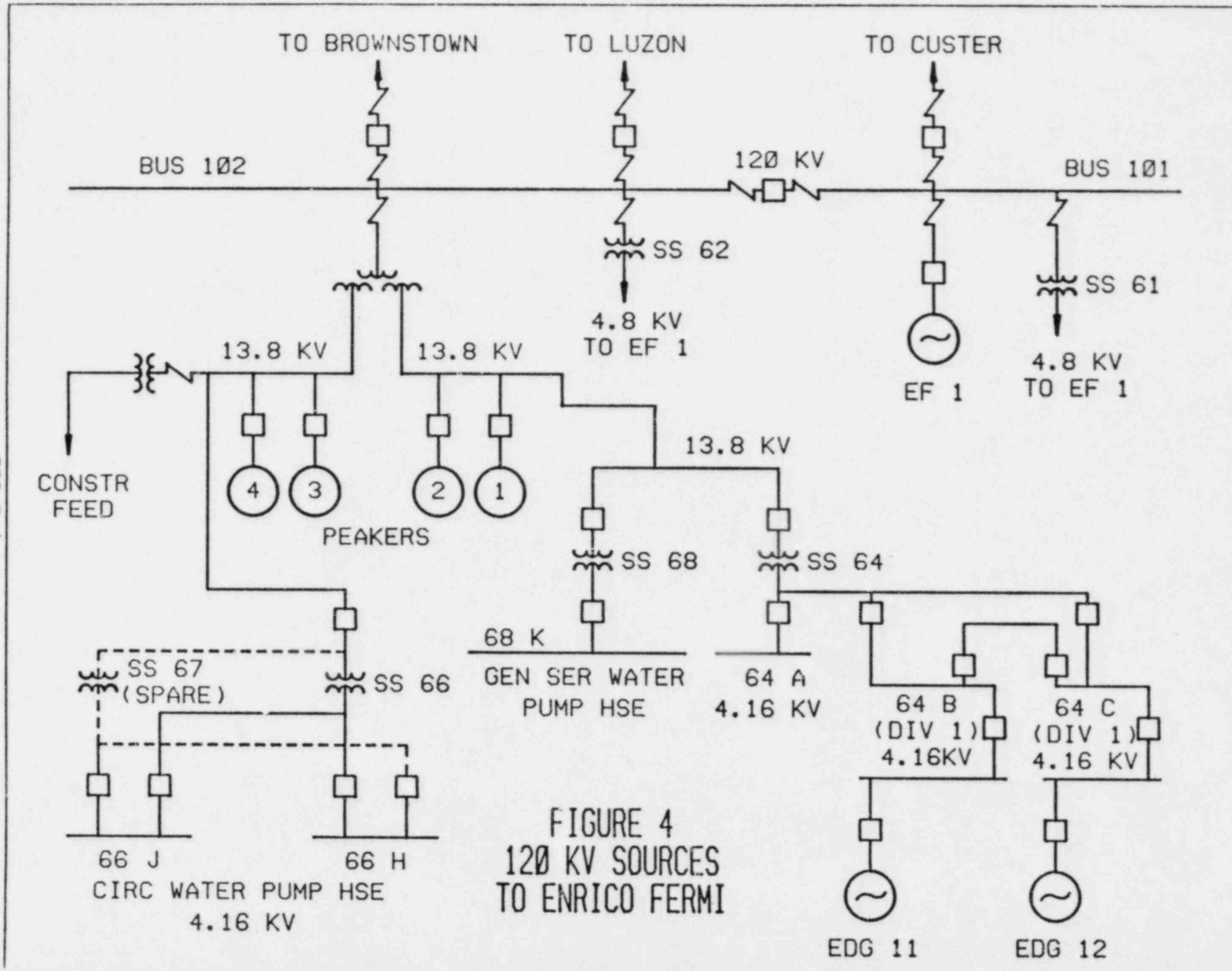


FIGURE 4
12Ø KV SOURCES
TO ENRICO FERMI

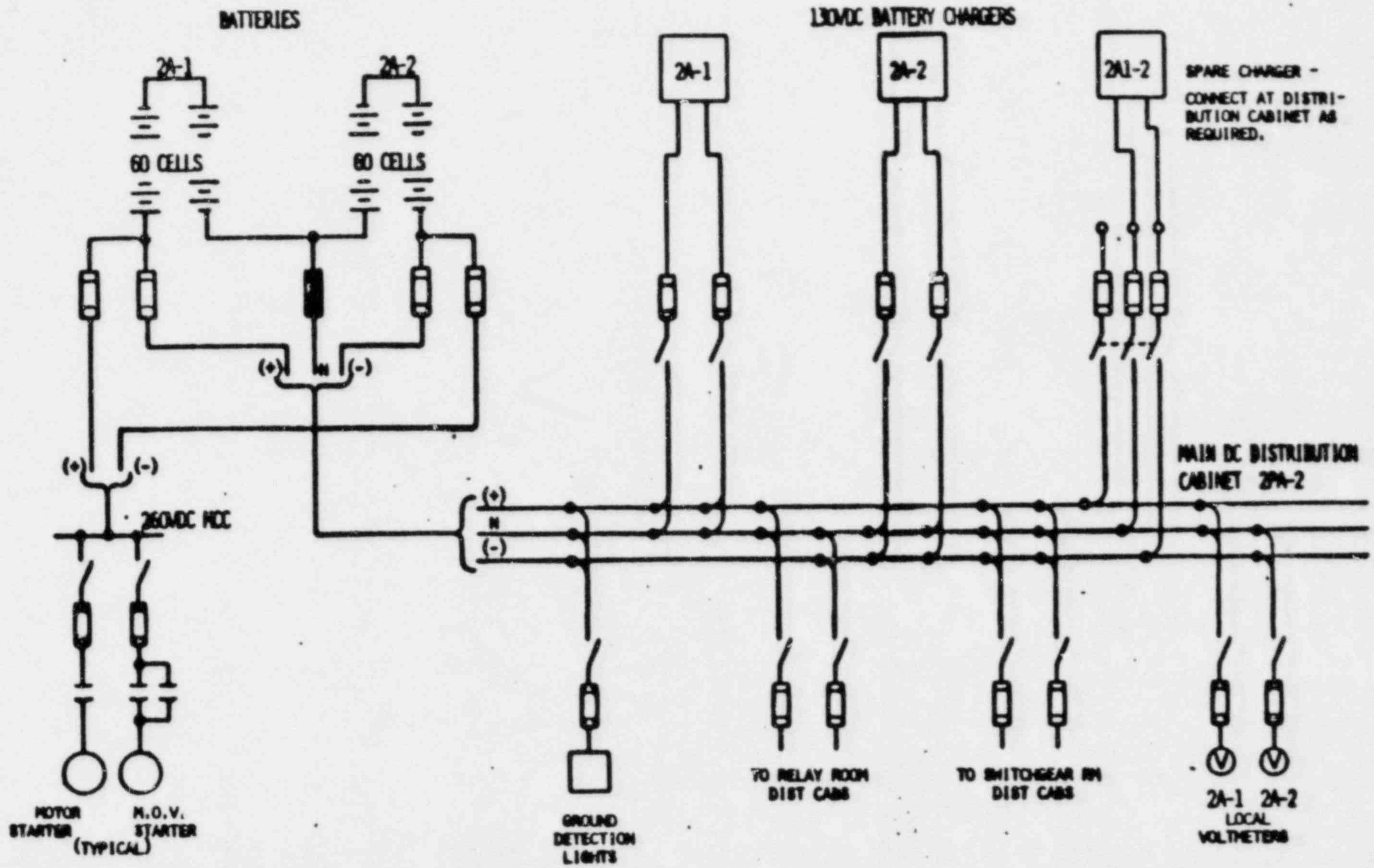


FIG. 5 ENRICO FERMI 2 260/130VDC DISTRIBUTION SYSTEM

RELIABILITY OF STATION ELECTRICAL POWER

1. OFFSITE SOURCES

THE FERMI 2 PLANT IS CONNECTED TO TWO SEPARATE OFFSITE POWER SYSTEMS - THE 120 kV AND 345 kV TRANSMISSION SYSTEMS - BY MULTI-CIRCUITED TIES TO INCREASE RELIABILITY. (SEE FIGURE 1).

TWO 345 kV CIRCUITS ON SEPARATE TOWERS AT OPPOSITE SIDES OF THE SAME CORRIDOR CONNECT THE PLANT SWITCHYARD TO BROWNSTOWN STATION. IT THERE LINKS WITH THE SERVICE AREA 345 kV NETWORK WHICH INCLUDES FREE FLOW EXCHANGE WITH CONSUMERS POWER COMPANY.

THERE IS NO UNIT AUXILIARY TRANSFORMER IN THE FERMI 2 DESIGN. GENERATOR OUTPUT IS DELIVERED DIRECTLY TO THE 345 kV SWITCHYARD BY TWO HALF CAPACITY STEP-UP TRANSFORMERS. (FIGURE 2) THE SWITCHYARD ALLOWS FLEXIBLE OPERATION BY EMPLOYING A TWO BUS, BREAKER AND A HALF SCHEME. PART OF THE NONSAFETY-RELATED PLANT LOAD AND ALL OF DIVISION II ARE FED DIRECTLY FROM THIS SWITCHYARD VIA AN AUXILIARY STEPDOWN TRANSFORMER.

THREE 120 kV CIRCUITS COMPRISE THE OTHER SYSTEM AVAILABLE TO THE PLANT. EACH CIRCUIT IS SEPARATELY SUPPORTED AND SHARES A COMMON CORRIDOR WITH THE 345 kV CIRCUITS FOR FIVE

MILES. THE CIRCUIT SUPPORTS ARE SPACED TO ALLOW AT LEAST ONE CIRCUIT FROM EACH SOURCE TO REMAIN INTACT UPON FAILURE OF ANY SUPPORT (FIGURE 3). THEREAFTER THE 120 kV CIRCUITS SPREAD OUT IN INDIVIDUAL CORRIDORS TO SEPARATE STATIONS WITH EXTENSIVE LINKING TO THE SYSTEM AT SEVERAL VOLTAGE LEVELS OVER A LARGE AREA. COMPREHENSIVE TRANSIENT STUDIES HAVE BEEN PERFORMED TO ASSURE CONTINUED SYSTEM STABILITY FOR LOSS OF ANY OR ALL LINES IN THE CORRIDOR.

THE LINES TERMINATE AT THE 120 kV SWITCHYARD, ARRANGED AS A RADIAL DOUBLE FEED BUS, ONE QUARTER MILE AWAY FROM THE OTHER SWITCHYARD (FIGURE 4) THERE IT IS TRANSFORMED TO 13.8 kV AND BROUGHT TO FERMI 2 ENTIRELY UNDERGROUND. NEAR THE SWITCHYARD, FOUR 18.8 MVA GAS TURBINE PEAKING UNITS ARE TIED INTO THE 13.8 kV FEED TO FERMI 2. THE 13.8 FEEDS A SYSTEM SERVICE STEPDOWN TRANSFORMER WHICH SUPPORTS DIVISION I AND PART OF THE NONSAFETY-RELATED LOAD.

2. AC POWER DISTRIBUTION

THE ENGINEERED SAFETY FEATURES SYSTEMS AT FERMI 2 ARE DIVIDED INTO TWO REDUNDANT AND INDEPENDENT DIVISIONS - DIVISIONS I & II. EACH DIVISION HAS AS ITS PREFERRED POWER SOURCE A SEPARATE OFFSITE SUPPLY.

FOUR 2850 kW EMERGENCY DIESEL GENERATORS (EDG) PROVIDE ON SITE AC POWER IN THE EVENT OF LOSS OF OFFSITE POWER. THERE ARE TWO EDG'S PER DIVISION WITH THE GROUP OF BUSES ASSOCIATED WITH EACH EDG (AND INDEPENDENT OF THE OTHER GROUPS) KNOWN AS A LOAD GROUP. WHILE BOTH LOAD GROUPS WITHIN A DIVISION ARE NECESSARY TO PROVIDE TRUE REDUNDANCY, NORMALLY OPEN INTRA-DIVISIONAL CROSS TIES ENABLE CRITICAL LOADS TO BE THROWN OVER IN THE EVENT ONE EDG IS NOT AVAILABLE. WITHIN A DIVISION, EMERGENCY CORE COOLING LOADS ARE SHARED BETWEEN LOAD GROUPS. PLANT AUXILIARY SERVICES SUCH AS AIR HANDLING, CLOSED LOOP COOLING, ETC., ARE CONFINED TO A PARTICULAR LOAD GROUP. INSTRUMENT POWER SUPPLY FEEDS CAN THROW OVER BETWEEN LOAD GROUPS. EDG AUXILIARY SERVICES ARE PROVIDED BY THEIR RESPECTIVE LOAD GROUPS.

EACH LOAD GROUP INCORPORATES RADIAL BUS ARRANGEMENT. TWO 4 kV BUSES WITHIN THE LOAD GROUP ARE TIED TOGETHER. ONE CONNECTS TO THE SYSTEM SERVICE POWER TRANSFORMER VIA CABLE BUS FOR THE OFFSITE SOURCE. THE OTHER CONNECTS TO THE EDG. LARGE LOADS ARE POWERED FROM THESE BUSES. EACH LOAD GROUP CONTAINS TWO 480 VOLT BUSES - ONE DERIVED FROM EACH OF THE 4 kV BUSES. SMALLER LOADS, INCLUDING THOSE DOWN TO 120 VOLTS, ARE FED FROM THESE BUSES OR THOSE THEREAFTER.

3. BUS LOSS CONSEQUENCES

THE DESIGN BASIS FOR FERMI 2 IS TO OPERATE ON THE PREFERRED POWER SOURCE WHEN AVAILABLE. WHEN NOT AVAILABLE, TWO LOAD

GROUPS WITHIN A DIVISION ARE REQUIRED. EACH DIVISION'S SAFETY-RELATED EQUIPMENT IS COMPLETELY REDUNDANT TO THE OTHER.

LOSS OF DIVISION I AC POWER SUPPLY WILL RESULT IN LOSS OF THE FOLLOWING EQUIPMENT:

A. RHR LOOPS A & C

B. CORE SPRAY LOOPS A & C

C. INBOARD ISOLATION VALVES

D. ONE REDUNDANT SYSTEM OF CONTROL & EQUIPMENT ROOM VENTILATION

E. ONE REDUNDANT SYSTEM OF STANDBY GAS TREATMENT

F. ONE REDUNDANT SYSTEM OF COMBUSTIBLE GAS CONTROL

LOSS OF DIVISION II POWER SUPPLY WILL RESULT IN LOSS OF THE FOLLOWING CRITICAL EQUIPMENT:

A. RHR LOOPS B & D

B. CORE SPRAY LOOPS B & D

C. ONE REDUNDANT SYSTEM OF CONTROL & EQUIPMENT ROOM VENTILATION

D. ONE REDUNDANT SYSTEM OF STANDBY GAS TREATMENT

E. ONE REDUNDANT SYSTEM OF COMBUSTIBLE GAS CONTROL

4. DC ONSITE POWER SYSTEM

THE DC ONSITE POWER SYSTEM FOR ENRICO FERMI UNIT 2 CONSISTS OF THE FOLLOWING FULLY INDEPENDENT BATTERY SYSTEMS:

A. DIVISION 1 260/130V BATTERY 2 PA (ESF LOADS ONLY)

B. DIVISION 2 260/130V BATTERY 2 PB (ESF LOADS ONLY)

C. BOP 260/130V BATTERY 2 PC (NON-ESF LOADS)

D. 48/24V BATTERY 21A (NON-ESF INSTRUMENT LOAD)

E. 48/24V BATTERY 21B (NON-ESF INSTRUMENT LOAD)

BATTERIES C, D AND E FEED ONLY NON-ESF LOADS SUCH AS MAIN TURBINE AUXILIARIES, THE ANNUCIATOR, AND NON-ESF CONTROL AND INSTRUMENTATION. LOSS OF ANY OF THESE SYSTEMS WILL NOT IMPAIR THE ABILITY TO SAFELY SHUTDOWN THE REACTOR.

A TYPICAL SYSTEM DIAGRAM OF AN ESF 260/130V CENTER TAPPED BATTERY DESIGN AT ENRICO FERMI 2 IS SHOWN IN FIGURE 5. BOTH POWER FOR MOTOR-OPERATED VALVES AND SMALL AUXILIARY PUMPS, AND CONTROL POWER REQUIRED BY THE DIVISION ARE SUPPLIED BY THE SAME BATTERY.

LOSS OF THE DIVISION 1 260/130V BATTERY WOULD RESULT IN A LOSS OF THE FOLLOWING ESF FUNCTIONS:

A. RCIC SYSTEM AND ITS AUXILIARIES

B. AUTOMATIC DEPRESSURIZATION UTILIZING THE SRV'S

C. CONTROL POWER TO DIVISION 1 ESF SWITCHGEAR AND EDG'S

D. CONTROL POWER TO THE DIVISION 1 ECCS SYSTEMS.

IF ACCIDENT CONDITIONS EXIST SIMULTANEOUSLY WITH THE LOSS OF THE DIVISION 1 BATTERY SYSTEM, REDUNDANT EQUIPMENT IN DIVISION 2 WOULD PERFORM THE REQUIRED SAFETY FUNCTIONS.

IF THE DIVISION 2 260/130V BATTERY SYSTEM WERE TO BE LOST DURING AN ACCIDENT, THE FOLLOWING ESF FUNCTIONS WOULD BE UNAVAILABLE:

A. HPCI SYSTEM AND ITS AUXILIARIES.

B. OUTBOARD ISOLATION VALVES.

C. CONTROL POWER TO DIVISION 2 ESF SWITCHGEAR AND EDG'S.

D. CONTROL POWER TO DIVISION 2 ECCS SYSTEMS.

THE DIVISION 1 BATTERY AND DIVISION 1 AC WOULD POWER REDUNDANT EQUIPMENT PERFORMING THE REQUIRED SAFETY FUNCTIONS.

5. ACTIONS FOR RESTORING OFFSITE AC POWER IN THE EVENT OF A LOSS OF THE GRID

DETROIT EDISON COMPANY HAS AN "EMERGENCY ELECTRICAL PROCEDURES OPERATING GUIDE" WHICH DESCRIBES THE PROCEDURES TO BE FOLLOWED BY THE ELECTRICAL SYSTEM SHIFT SENIOR SYSTEM SUPERVISOR IN ORDER TO MAINTAIN SYSTEM RELIABILITY FOR CASES OF A DEFICIENT BULK POWER SUPPLY. IN CASE OF A COMPLETE SYSTEM SHUTDOWN, THE GUIDE PROVIDES VARIOUS METHODS FOR RESTORING THE ELECTRICAL SYSTEM TO NORMAL IN AS SHORT A TIME AS POSSIBLE. ONE FEATURE OF THE GUIDE IS THAT EACH DETROIT EDISON GENERATING FACILITY WITH "BLACK START CAPABILITY" WILL INITIATE THEIR OWN RETURN TO NORMAL GENERATING MODE. THEN AUXILIARY POWER WILL BE RESTORED TO THE OTHER DETROIT EDISON GENERATING FACILITIES, INCLUDING FERMI 2 BY RE-ENERGIZING THE TRANSMISSION NETWORK

IN A SEQUENCE WHICH WOULD VARY BECAUSE OF THE AREA ENCOMPASSED BY THE BLACKOUT CONDITION. AFTER THIS IS ACCOMPLISHED, THE FULL TRANSMISSION NETWORK AND CUSTOMER LOADS WILL BE RESTORED IN SMALL INCREMENTS.

RESTORATION OF OFFSITE POWER TO FERMI 2 IS EXPECTED TO BE ACCOMPLISHED BY THE BLACK START OF TRENTON CHANNEL GENERATING FACILITY AND THE RESTORATION OF THE 120 kV NETWORK SUFFICIENTLY TO SUPPLY FERMI 2 VIA ONE OF THE 120 kV LINES INTO FERMI 1 SWITCHYARD. THE 345 kV OFFSITE SOURCE WOULD BE RESTORED BY RE-ENERGIZING THE 345 kV AT BROWNSTOWN STATION VIA ONE OF DETROIT EDISON GENERATING FACILITIES OR FROM OUR 345 kV INTERCONNECTING TIES TO THE NEIGHBORING UTILITY SYSTEM. THE TIME REQUIRED TO RESTORE THE FERMI 2 OFFSITE POWER SUPPLIES WILL VARY WITH THE NATURE OF THE GRID BLACKOUT AND THE EXTENT OF ANY ASSOCIATED DAMAGE TO THE TRANSMISSION LINES. IN ANY EVENT, THE TOTAL SYSTEM RESTORATION IS EXPECTED TO BE COMPLETED WITHIN A DAY. THE INDUSTRY EXPERIENCE IN RESTORING SYSTEMS HAS BEEN LESS THAN ONE DAY.

A LESS EXTENSIVE GRID DISTURBANCE COULD ISOLATE FERMI 2 WITH THE TOTAL LOSS OF OFFSITE POWER THROUGH THE UNLIKELY SIMULTANEOUS OR OVERLAPPING OUTAGES OF THE 120 kV STATIONS;

SIMULTANEOUS OR OVERLAPPING OUTAGES OF THE 120 kV STATIONS; BROWNSTOWN, CUSTER AND LUZON AND THE 345 kV FEEDS FROM BROWNSTOWN STATION OR BY THE LOSS OF THE FERMI 2 TRANSMISSION CORRIDOR. THE RESTORATION TIME OF ANY ONE LINE, FOR SUCH AN EVENT, WOULD VARY CONSIDERABLY WITH THE EXTENT OF THE DAMAGE ASSOCIATED WITH THE OUTAGE. AN INDICATION OF TIME TO RESTORE ANY ONE LINE IS THE AVERAGE RESTORATION TIME OF ALL DETROIT EDISON TRANSMISSION LINE LOCKOUT OUTAGES (EXCLUDING MOMENTARY OUTAGES WITH AUTOMATIC RECLOSING). THE AVERAGE RESTORATION TIME FOR A 120 kV LINE LOCKOUT OUTAGE IS 13 HOURS. THE AVERAGE RESTORATION TIME FOR A 345 kV LINE LOCKOUT OUTAGE IS 9.3 HOURS ON THE DETROIT EDISON TRANSMISSION SYSTEM.

FOR THOSE INCIDENTS IN WHICH SUBSTANTIAL DAMAGE IS DONE TO TRANSMISSION STRUCTURES, DETROIT EDISON HAS EMERGENCY PROCEDURES IN WHICH TEMPORARY STRUCTURES WOULD BE USED TO RESTORE THE TRANSMISSION LINE IN AS SHORT A TIME AS POSSIBLE. DETROIT EDISON CONSTRUCTION AND CONTRACT CREWS WOULD BE USED TO MAKE THE NECESSARY REPAIRS.

6. LOSS OF OFFSITE AC POWER DUE TO ONSITE EQUIPMENT FAILURES

DUE TO THE INDEPENDENCE OF ENRICO FERMI 2 OFFSITE POWER SOURCES AND THE METHOD OF ONSITE DISTRIBUTION OF POWER,

NO SINGLE EQUIPMENT FAILURE CAN CAUSE THE LOSS OF BOTH OFFSITE AC POWER SOURCES.

THE MOST CREDIBLE EVENT LIKELY TO CAUSE A LOSS OF ALL OFFSITE POWER IS A TORNADO THAT STRIKES THE COMMON TRANSMISSION CORRIDOR FOR THE 345 kV AND 120 kV LINES LEAVING THE SITE. SHOULD THIS HIGHLY IMPROBABLE EVENT OCCUR THE FOUR INDEPENDENT DIESEL GENERATORS WOULD AUTOMATICALLY START AND LOAD, RESTORING POWER TO THE ESF BUSES.

RESTORATION OF AN OFFSITE POWER SOURCE WOULD BE ACCOMPLISHED AS DESCRIBED IN PART 5.

7. STATION BLACKOUT

DETROIT EDISON DOES NOT BELIEVE THAT A COMPLETE LOSS OF AC POWER IS A CREDIBLE POSSIBILITY AT FERMI 2. COMBINING THE LOSS OF THE TWO INDEPENDENT OFFSITE SOURCES WITH THE FAILURE OF FOUR INDEPENDENT DIESEL GENERATORS GOES WELL BEYOND REQUIRED PROBABILITIES.

EVEN WITH SUCH INCREDIBLE EVENTS, A UNIQUE FEATURE OF THE FERMI 1 SWITCHYARD CAN SUPPLY ADEQUATE POWER TO ONE OF THE FERMI 2 ESF DIVISIONS. FOUR COMBUSTION TURBINE GENERATORS, (CTG) RATED 18.8 MVA EACH, USED FOR PEAKING PURPOSES, ARE LOCATED JUST OUTSIDE THE FERMI 1 120 kV SWITCHYARD. COMBUSTION TURBINE GENERATOR #1 (SEE FIGURE 4) IS EQUIPPED WITH

A BLACK START FEATURE. THIS UNIT IS CAPABLE OF STARTING AND ACCEPTING LOAD WITHIN TEN MINUTES. CTG #1 HAS ADEQUATE CAPACITY TO EASILY POWER DIVISION I ESF LOADS REQUIRED TO ENSURE SAFE SHUTDOWN OF THE REACTOR.

IN THE TIME PERIOD BETWEEN REALIZATION BY THE OPERATOR THAT NO EDG'S ARE AVAILABLE AND THE LOADING OF CTG #1, ADEQUATE ACTION CAN BE TAKEN UTILIZING SYSTEMS POWERED ONLY BY DC. EMERGENCY OPERATING PROCEDURES AT FERMI 2 COVER THE ACTIONS NECESSARY TO MAINTAIN REACTOR VESSEL WATER LEVEL USING THE STEAM-DRIVEN, DC-POWERED, HPCI AND RCIC SYSTEMS. IN ADDITION, ANY ACTIONS NECESSARY TO RESTORE ONSITE POWER, INCLUDING RESTART OF THE EMERGENCY DIESELS AND BLACK START OF CTG #1, WILL BE INCLUDED IN THE ABNORMAL OPERATING PROCEDURES.

III.G. STATUS OF MARK I CONTAINMENT MODIFICATIONS

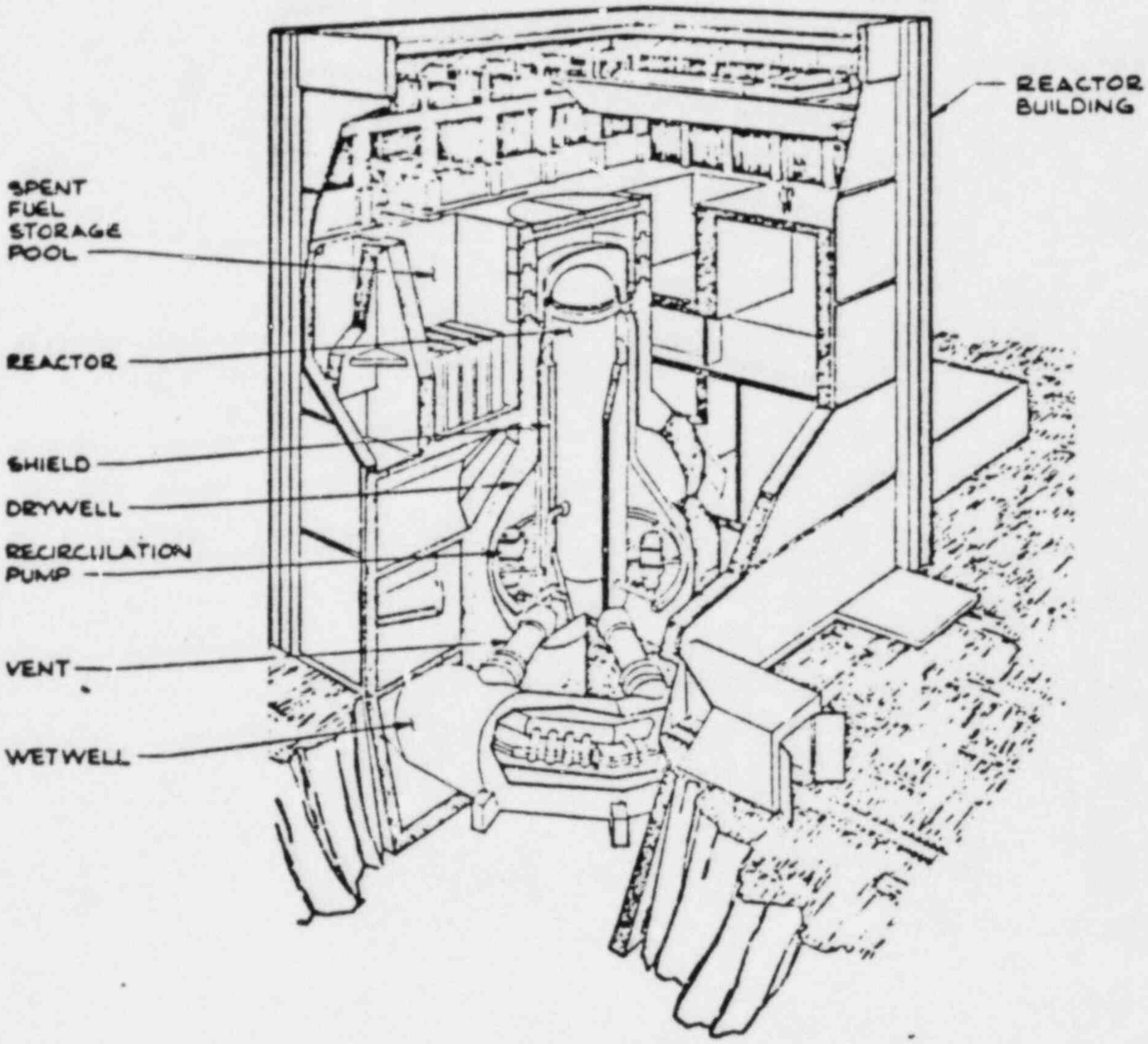


FIGURE 1

FERMI 2 CONTAINMENT SYSTEM

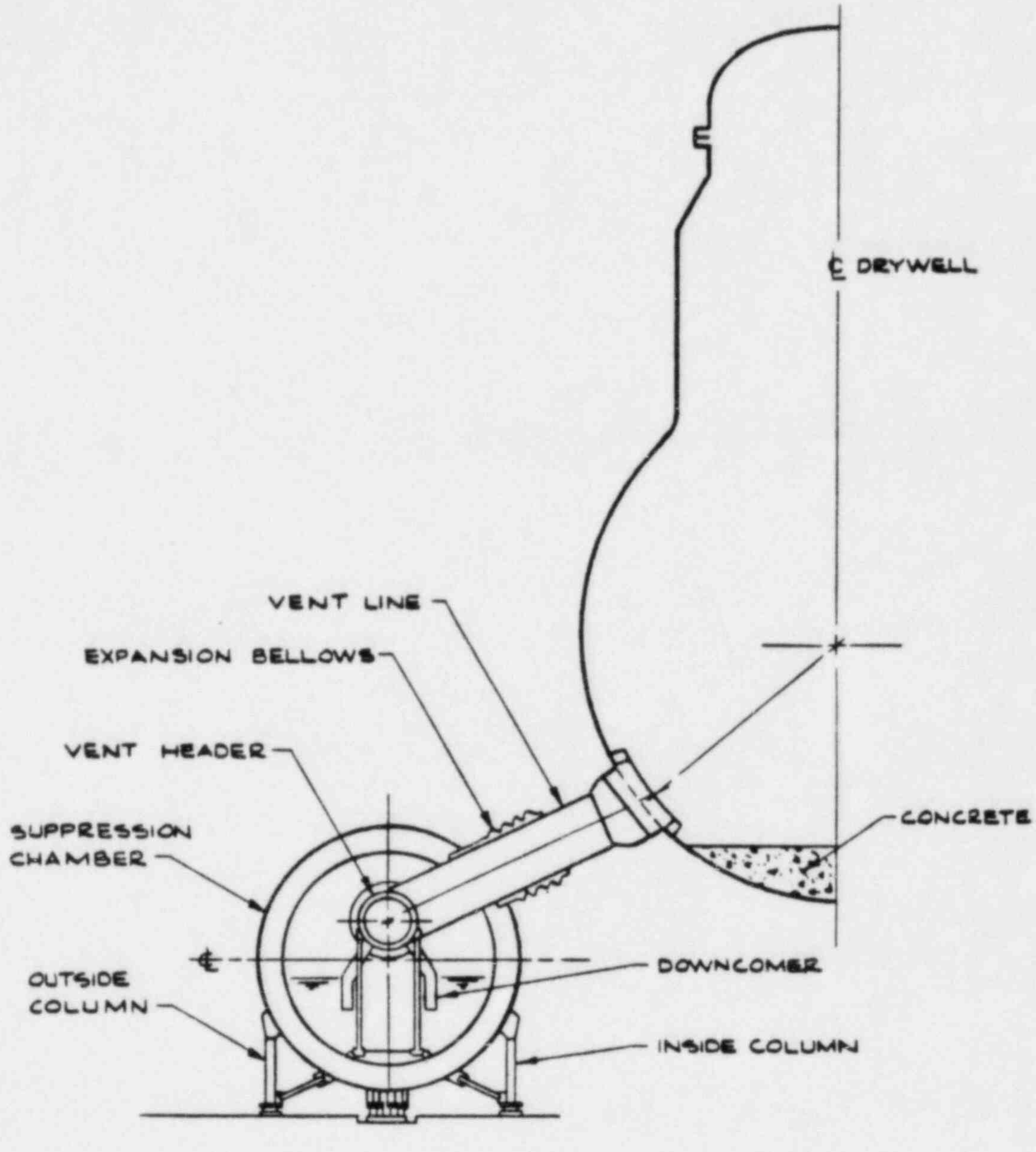


FIGURE 2

GENERAL ARRANGEMENT OF CONTAINMENT - SCHEMATIC

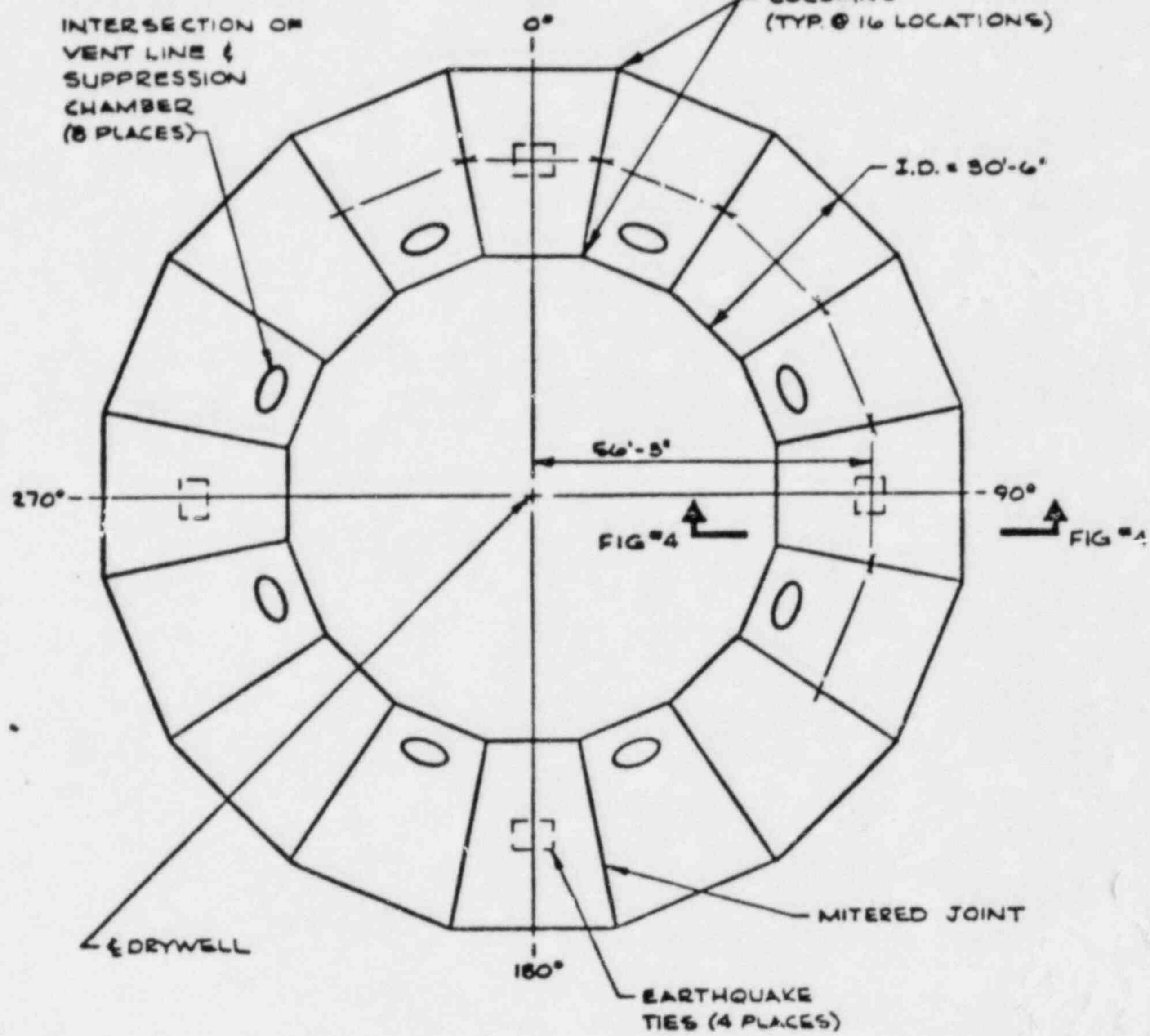


FIGURE 3

PLAN VIEW OF SUPPRESSION CHAMBER - SCHEMATIC

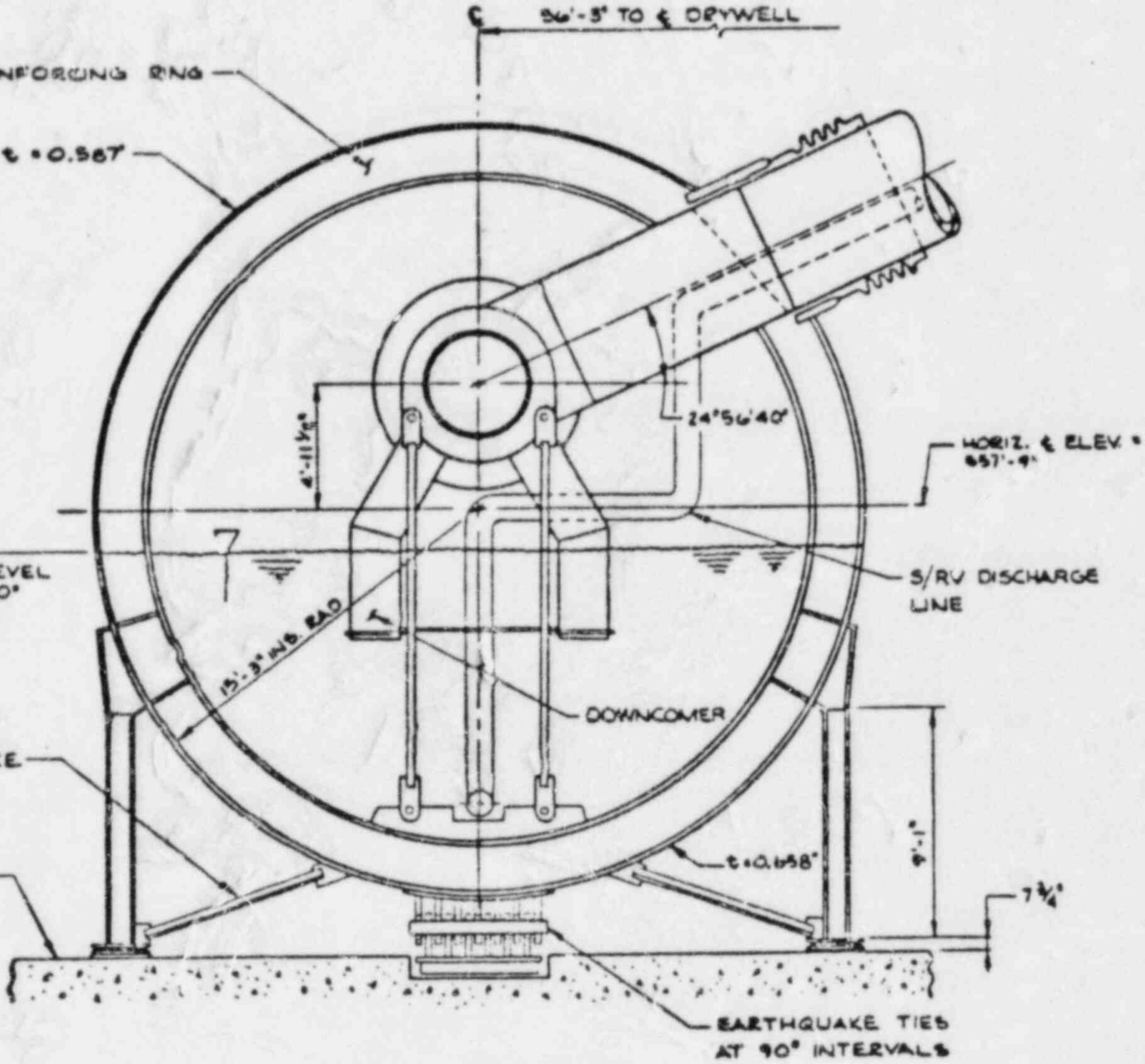


FIGURE 4

SECTION THROUGH SUPPRESSION CHAMBER

TABLE 1

COMPLETED MODIFICATIONS

CATEGORY		DESCRIPTION		APPROX. MOD. DATES
MAJOR	TORUS	RING GIRDER REINFORCEMENT		6/79
MAJOR	TORUS	COLUMN REINFORCEMENT		10/78
MAJOR	TORUS	COLUMN CONNECTION REINFORCEMENT		12/79
MINOR	VENT SYSTEM	DOWNCOMER SHORTENING		2/80
MAJOR	VENT SYSTEM	VENT HEADER/DOWNCOMER STIFFENING & BRACING		11/78
MAJOR	VENT SYSTEM	REINFORCED EXISTING VENT SYSTEM COLUMNS & CONNECTIONS		2/79
MAJOR	VENT SYSTEM	VENT HEADER DEFLECTOR		2/80
MAJOR	VENT SYSTEM	VENT LINE/VENT HEADER STIFFENING		6/79
MAJOR	VENT SYSTEM	REINFORCED VACUUM BREAKER TO VENT HEADER CONNECTION		7/79
MAJOR	INTERNAL STRUCTURES	MONORAIL	ADDITIONAL SUPPORTS	5/78
MAJOR	INTERNAL STRUCTURES	MONORAIL	STRENGTHEN EXISTING SUPPORTS	5/78
MINOR	INTERNAL STRUCTURES	MONORAIL	EXTENDING MONORAIL	5/78
MAJOR	INTERNAL STRUCTURES	CATWALK	ADDITIONAL SUPPORTS	8/78
MAJOR	INTERNAL STRUCTURES	CATWALK	GRATING (DELIVER TO SITE)	3/80
MAJOR	SRV PIPING	REROUTED PIPING IN WETWELL		4/80
MAJOR	SRV PIPING	ADDITIONAL WETWELL SUPPORTS		4/79
MAJOR	SRV PIPING	REINFORCED V. L. PENETRATION		11/78
MAJOR	SRV PIPING	ADDED QUENCHER/RAMSHEAD SUPPORTS		1/80
MINOR	TORUS ATTACHED PIPING	ADDED TORUS INTERNAL SUPPORTS		4/80

III.G-5

A-16.2

TABLE 2

CURRENT MODIFICATIONS

CATEGORY		DESCRIPTION	EST. COMPLETION	REMARKS
MAJOR	TORUS	MITERED JOINT SADDLES	2/82	FABRICATIONS AND FIELD WORK UNDERWAY
MAJOR		ADDITIONAL COLUMN ANCHOR BOLTS	4/82	FABRICATION AND FIELD WORK UNDERWAY
MAJOR	SRV PIPING	QUENCHER	9/82	MATERIAL ORDERED
MAJOR		QUENCHER SUPPORTS	9/82	MATERIAL ORDERED
MINOR	INTERNAL STRUCTURES	CATWALK GRATING INSTALLATION	9/82	

MARK I CONTAINMENT MODIFICATIONS

THE FERMI 2 PRIMARY CONTAINMENT IS A STEEL SHELL STRUCTURE DESIGNATED BY GENERAL ELECTRIC COMPANY AS A MARK I CONTAINMENT. THE PRIMARY CONTAINMENT CONSISTS OF A LIGHT-BULB-SHAPED DRYWELL AND A TORUS-SHAPED PRESSURE SUPPRESSION POOL (OR WET-WELL). THE ARRANGEMENT OF THE DRYWELL AND SUPPRESSION POOL IN THE REACTOR BUILDING IS SHOWN IN FIGURE 1.

THE FERMI 2 PRIMARY CONTAINMENT WAS DESIGNED, ERECTED AND N-STAMPED PER ASME SECTION III DURING THE EARLY 1970'S. THE SUPPRESSION POOL CHAMBER AND VENT SYSTEM ARE SHOWN IN FIGURES 2, 3, AND 4. THE DESIGN RULES FOR THE CONTAINMENT WERE IN ACCORDANCE WITH ASME SECTION III, 1967 EDITION THROUGH 1969 SUMMER ADDENDA, FOR CLASS 2 NUCLEAR PRESSURE VESSELS. THE DESIGN LOADS SPECIFIED FOR THE CONTAINMENT INCLUDED DEAD LOADS, LIVE LOADS, PRESSURE (ACCIDENT), EARTHQUAKE AND THERMAL LOADS. SINCE THAT TIME, ADDITIONAL LOADS ASSOCIATED WITH THE SAFETY/RELIEF VALVE (S/RV) DISCHARGE AND THE POSTULATED LOSS OF COOLANT ACCIDENT (LOCA) HAVE BEEN IDENTIFIED FOR THE SUPPRESSION POOL CHAMBER AND VENT SYSTEM. THE IDENTIFICATION OF THESE NEW LOADS PRESENTED A GENERIC OPEN ITEM FOR UTILITIES HAVING PLANTS WITH MARK I CONTAINMENTS. AS SUCH, UTILITIES OWNING MARK I CONTAINMENTS FORMED A GROUP TO IDENTIFY THE COURSES OF ACTION NEEDED TO RESOLVE THIS OPEN ITEM IN A TIMELY MANNER.

DETROIT EDISON HAS BEEN AN ACTIVE MEMBER OF THE MARK I OWNERS' GROUP. EDISON PARTICIPATED IN THE TECHNICAL REVIEW ADVISORY COMMITTEES FORMED TO ASSURE THE DEVELOPMENT OF THE GENERIC METHODS REQUIRED TO DEFINE THE SUPPRESSION POOL HYDRODYNAMIC LOADING EVENTS AND THE ASSOCIATED STRUCTURAL ASSESSMENT TECHNIQUES FOR THE MARK I CONFIGURATION. THE DEFINITION OF THE SUPPRESSION POOL HYDRODYNAMIC LOADS WAS PROVIDED TO THE UTILITIES, TO PERFORM PLANT UNIQUE ANALYSES (PUA), IN THE MARK I LOAD DEFINITION REPORT (GE DOCUMENT NO. NEDO-21888, DECEMBER 1978). THE NRC STAFF, IN NUREG-0661 (JULY, 1980), CONCLUDED THAT THE LOAD DEFINITION PROCEDURES UTILIZED BY THE MARK I OWNERS' GROUP, (AS MODIFIED BY THE STAFF'S REQUIREMENTS), PROVIDE CONSERVATIVE ESTIMATES OF THE LOADING CONDITIONS, AND THAT THE STRUCTURAL ACCEPTANCE CRITERIA ARE CONSISTENT WITH REQUIREMENTS OF THE APPLICABLE CODES AND STANDARDS.

THE FERMI 2 PLANT UNIQUE PROGRAM FOR THE MARK I CONTAINMENT SYSTEM PROVIDED AN EARLY AND PROMPT REASSESSMENT OF THE FERMI 2 CONTAINMENT DESIGN FOR THE SUPPRESSION POOL HYDRODYNAMIC LOADING CONDITIONS. A DETAILED EVALUATION ON AN INTERIM BASIS WAS PERFORMED BEFORE THE ISSUANCE OF THE MARK I LOAD DEFINITION REPORT AND NUREG-0661. THE DETAILS OF THE EVALUATION ARE CONTAINED IN THE INTERIM STRUCTURAL EVALUATION REPORT PREPARED BY NUCLEAR TECHNOLOGY, INC., DATED MAY 1978.

THE EVALUATION RESULTS IDENTIFIED THAT EXTENSIVE MODIFICATIONS WOULD BE REQUIRED TO RESTORE THE ORIGINALLY INTENDED MARGIN OF SAFETY IN THE CONTAINMENT DESIGN.

MOST OF THE MODIFICATIONS IDENTIFIED IN THE INTERIM STRUCTURAL EVALUATION HAVE BEEN INSTALLED. A LISTING OF THE COMPLETED MODIFICATIONS AND THEIR APPROXIMATE INSTALLATION DATES IS PROVIDED IN TABLE 1. THE REMAINING IDENTIFIED MODIFICATIONS AND THE SCHEDULED COMPLETION DATES HAVE BEEN LISTED IN TABLE 2. SELECTED PHOTOGRAPHS OF THE INSTALLED MODIFICATIONS HAVE ALSO BEEN INCLUDED IN THE HANDOUT MATERIAL. ALL CURRENTLY IDENTIFIED MODIFICATIONS WILL BE INSTALLED BEFORE THE SCHEDULED NOVEMBER, 1982 FUEL LOAD DATE.

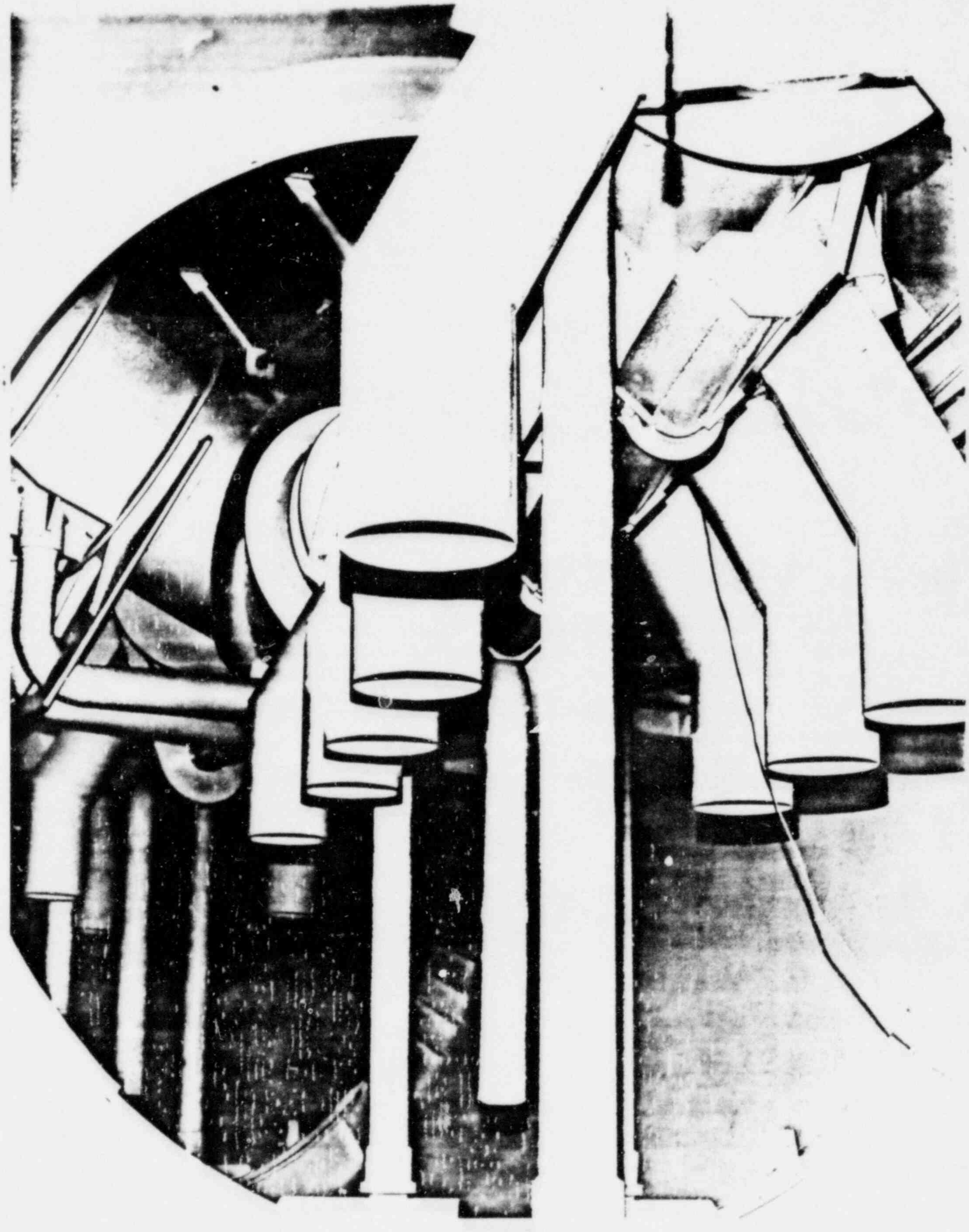
THE DESIGN BASIS FOR THE CURRENTLY IDENTIFIED MODIFICATIONS HAS RESULTED IN STRUCTURAL CAPACITIES THAT WILL MOST LIKELY BE ADEQUATE TO RESTORE THE ORIGINALLY INTENDED MARGIN OF SAFETY IN THE CONTAINMENT DESIGN. ON-GOING EVALUATIONS OF THESE DESIGNS AS THE GENERIC LOADS AND CRITERIA WERE FINALIZED, AND COMPARISONS WITH MODIFICATION DESIGNS BEING INSTALLED AT OTHER MARK I FACILITIES, HAVE IDENTIFIED THAT THE FERMI 2 CONTAINMENT MODIFICATION CAPACITIES WILL BE SUFFICIENT TO MEET THE LTP ACCEPTANCE CRITERIA. AS SUCH, EDISON ANTICIPATES THAT THE LONG TERM PROGRAM (LTP)-PUA WILL ONLY BE A CONFIRMATORY ANALYSIS. IT IS UNDERSTOOD, HOWEVER, THAT A CONSIDERABLE AMOUNT OF WORK STILL REMAINS

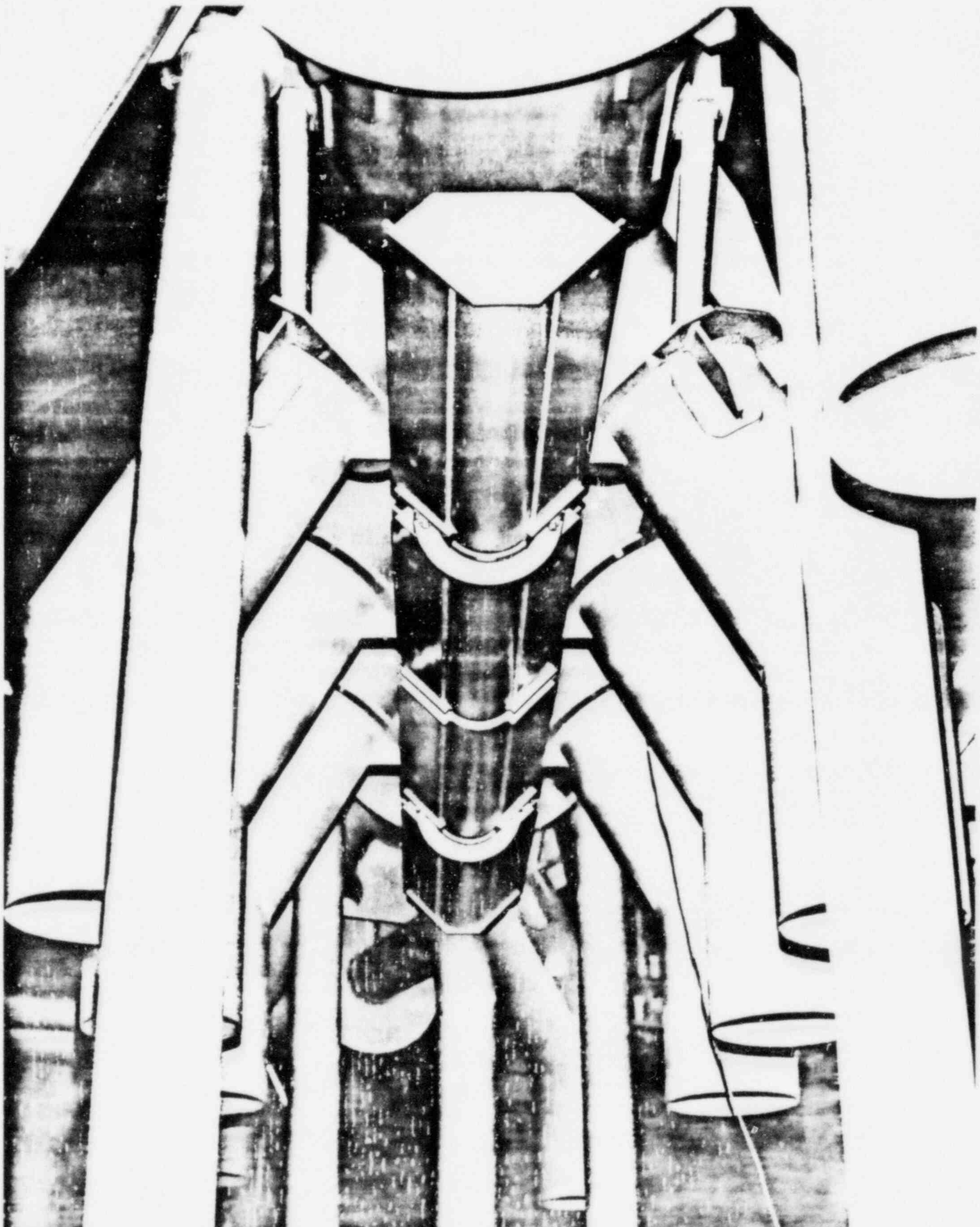
TO COMPLETE THE FERMI 2 PUA FOR THE LTP LOADS. EDISON IS PROCEEDING WITH MAXIMUM EFFORT TO COMPLETE THE LTP-PUA OF THE SUPPRESSION CHAMBER, VENT SYSTEM AND SUPPRESSION CHAMBER INTERNALS BY MAY, 1982. ANALYSIS OF THE TORUS ATTACHED PIPING WILL PROCEED AS SOON AS THE SUPPRESSION CHAMBER SHELL RESPONSE HAS BEEN PROPERLY CHARACTERIZED USING LTP CRITERIA. IN ADDITION, THE PREDICTED SUPPRESSION CHAMBER RESPONSE AND THE ATTACHED PIPING RESPONSE WILL BE VERIFIED BY IN-PLANT SAFETY/RELIEF VALVE CLEARING TESTS. ANY MODIFICATIONS TO THE TORUS ATTACHED PIPING REQUIRED BY THESE ANALYSES AND CONFIRMATORY TESTS WILL BE COMPLETED PRIOR TO RETURNING TO POWER AFTER THE FIRST REFUELING.

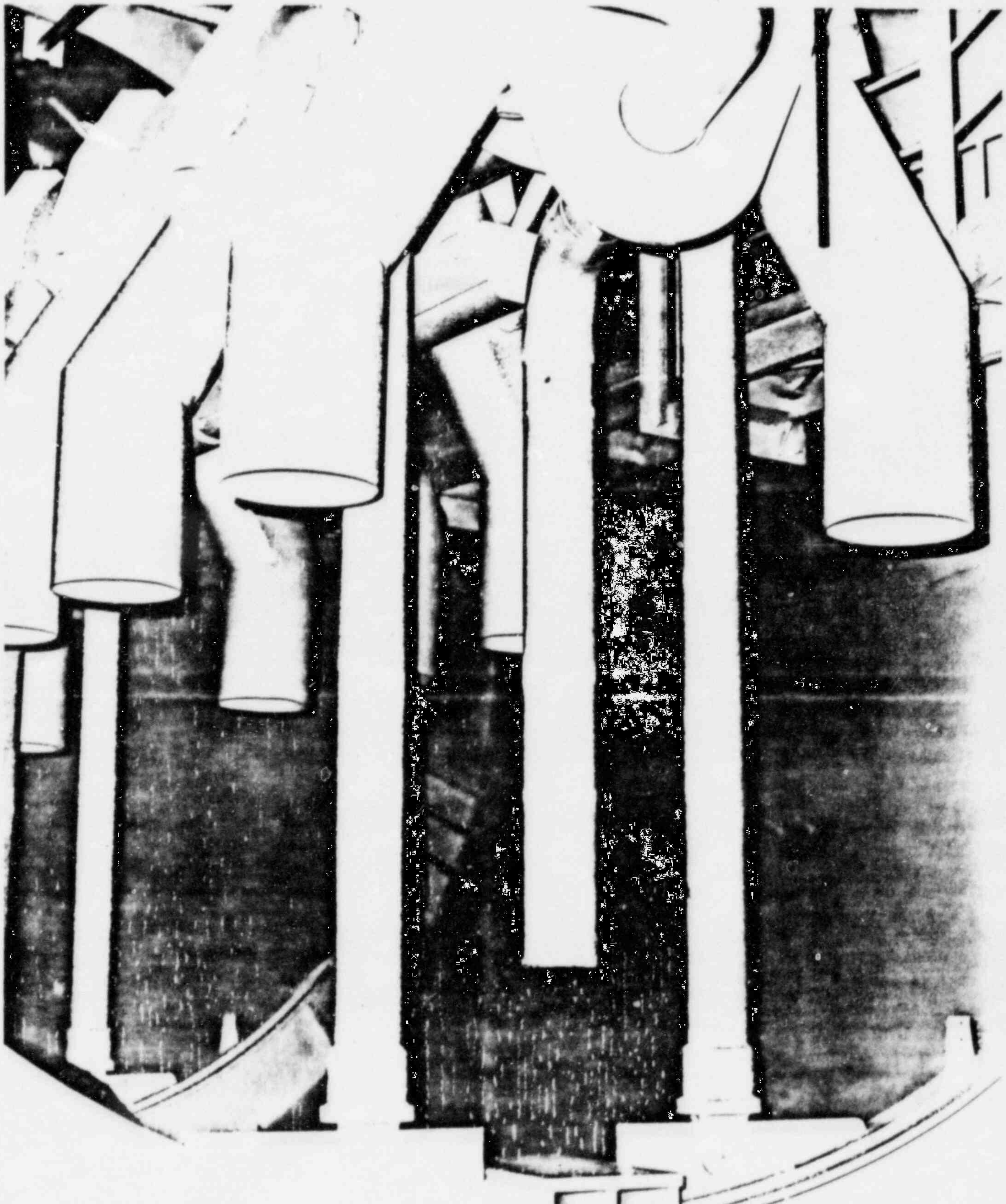
EDISON BELIEVES THAT THE FERMI 2 CONTAINMENT PROGRAM HAS ADDRESSED NUREG-0661 AND WILL LEAD TO A TIMELY RESOLUTION OF THE MARK I CONTAINMENT ISSUE. ACCORDINGLY, THE PLANT WILL FUNCTION SAFELY IN THE EVENT OF ALL POTENTIAL SAFETY/RELIEF VALVE DISCHARGE TRANSIENTS AND LOSS-OF-COOLANT ACCIDENTS.

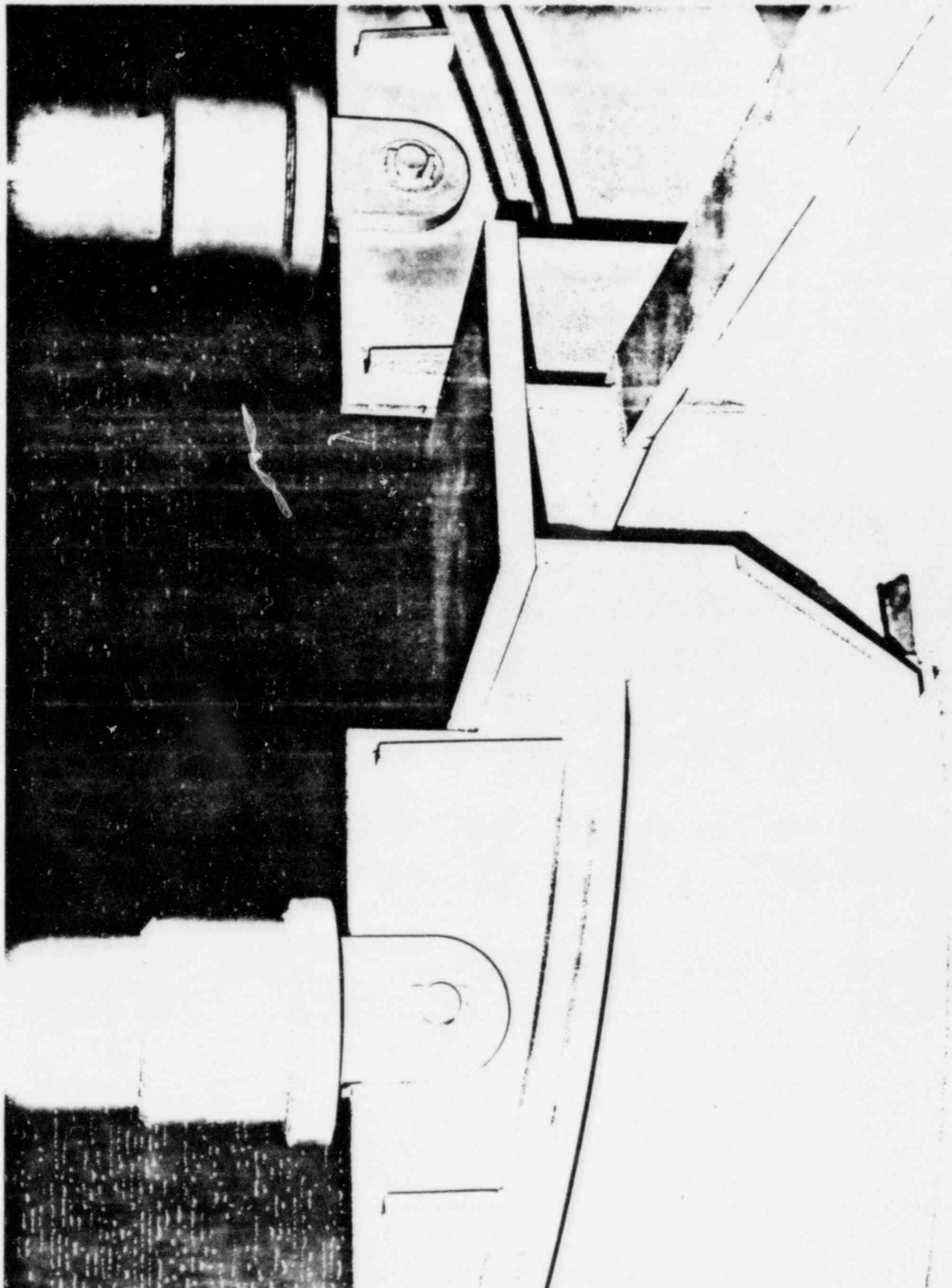
SEVEN PHOTOGRAPHS OF FERMI 2

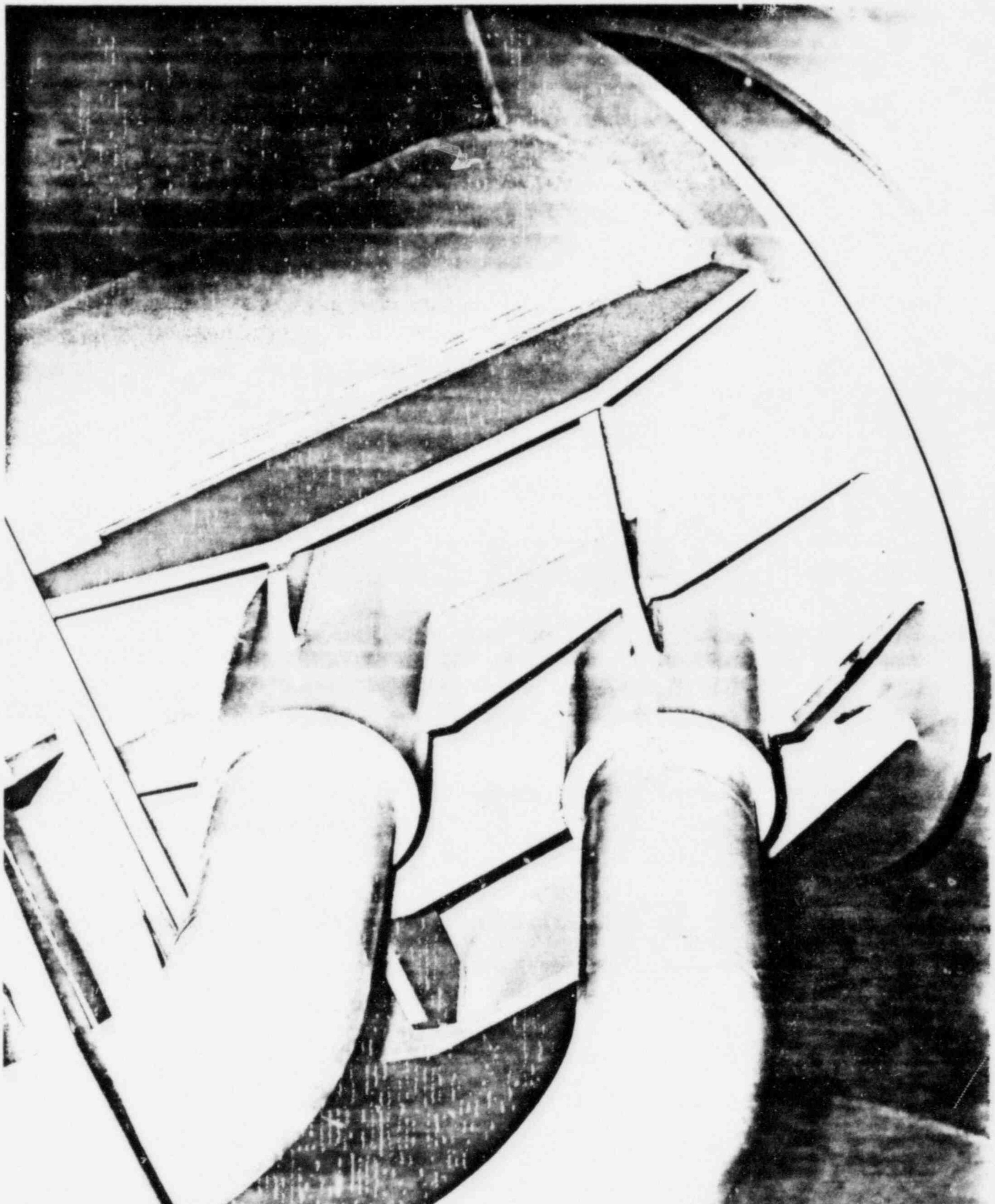
TORUS INTERNAL MODIFICATIONS

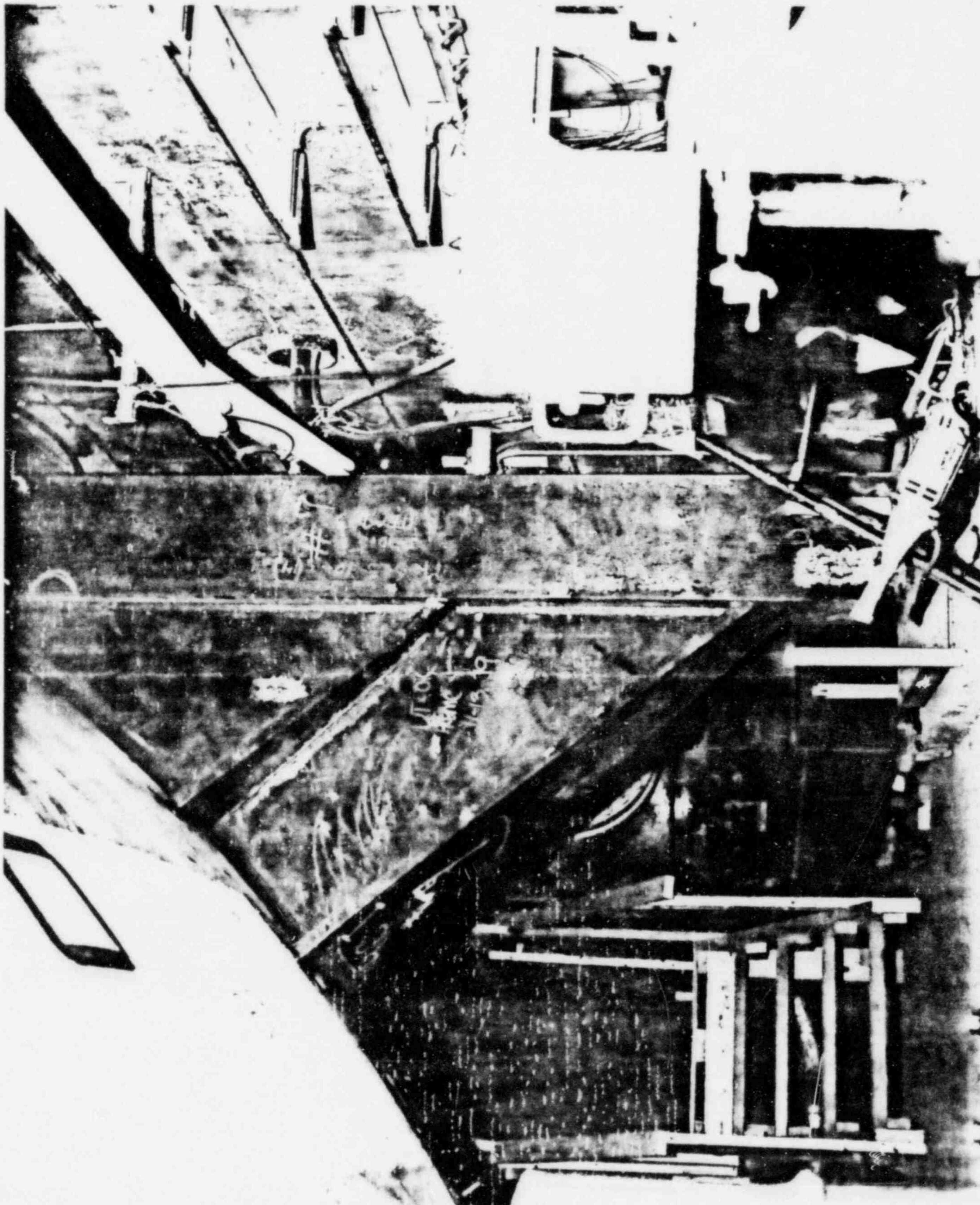


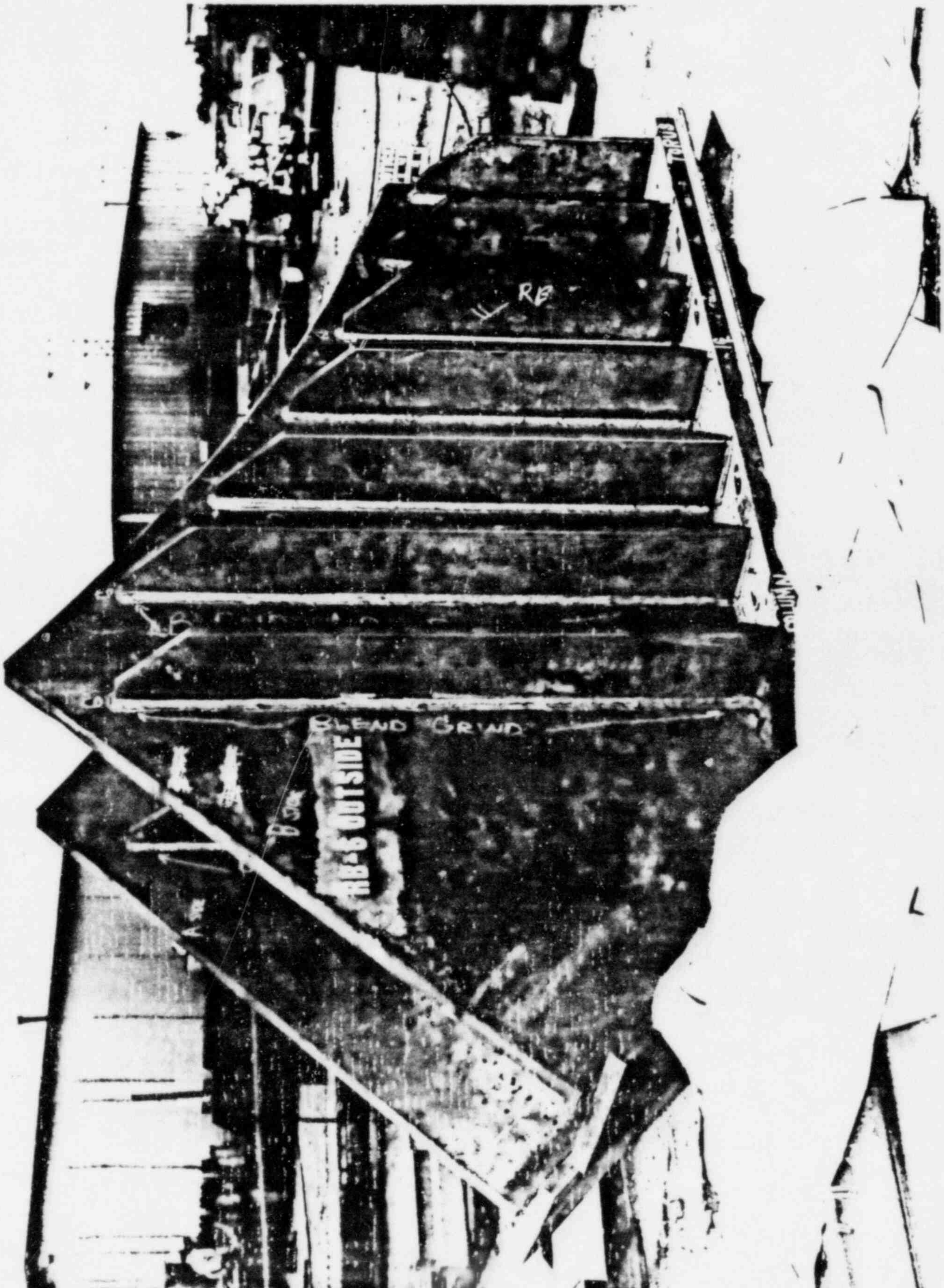











RB
BLEND GRIND
RB-6 OUTSIDE

ATWS PROCEDURES BRIEF

THERE IS AN ALTERNATIVE TO FULL ROD INSERTION FOR SHUTTING DOWN THE REACTOR -- STANDBY LIQUID CONTROL - TRADITIONAL - OPERATOR ACTION REQUIRED.

ENRICO FERMI OPERATIONS HAS BEEN PARTICIPATING ACTIVELY IN THE BWR OWNERS' GROUP SYMPTOMATIC EMERGENCY OPERATING PROCEDURE DEVELOPMENT SINCE MID-1979. EMERGENCY PROCEDURES FOR LEVEL CONTROL, CONTAINMENT CONTROL, AND REACTIVITY CONTROL SPECIFICALLY APPLICABLE TO FERMI ARE IN FINAL DRAFT. THEY HAVE BEEN THROUGH TWO REVIEW CYCLES INCLUDING COMMENTS BY THE NRC.

A TEAM OF TWO SHIFT SUPERVISORS, FIVE SUPERVISING OPERATORS, ONE GE AND ONE EDISON INSTRUCTOR PLUS EITHER MYSELF (E. P. GRIFFING) OR THE OPERATIONS ENGINEER AND HIS ASSISTANT AS MONITORS, HAVE BEEN REHEARSING EXECUTION OF THESE PROCEDURES IN OUR CONTROL ROOM SINCE LAST WEEK. WE WERE EVALUATED BY THE NRC AT THE BROWNS FERRY SIMULATOR IN CHATTANOOGA AND AT FERMI. OUR PROCEDURES HAVE BEEN APPROVED AS SATISFACTORY.

WE HAVE A NUCLEAR OPERATIONS DIRECTIVE ISSUED BY OUR VICE PRESIDENT TO THE NUCLEAR SHIFT SUPERVISOR WHICH ENJOINS THEM TO OPERATE THE PLANT WITH SAFETY FOREMOST.

IN SPITE OF, OR REGARDLESS OF, ANY DESIGN ANOMALIES KNOWN OR UNKNOWN, WE BELIEVE OUR OPERATORS WILL HAVE THE KNOWLEDGE TO RECOGNIZE THE NEED FOR A SCRAM AND IF IT SHOULD NOT COMPLETE, THEY WILL FOLLOW OUR PROCEDURES TO INITIATE STANDBY LIQUID CONTROL TO SHUT DOWN THE REACTOR.

SDV MODIFICATIONS

THE NRC HAS ISSUED THE OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD) REPORT ENTITLED, "SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM" AS DRAFT NUREG-0785. AN NRC REQUEST FOR A GENERIC AND PLANT SPECIFIC RESPONSE IN 45 AND 120 DAYS, RESPECTIVELY, WAS ATTACHED TO THE REPORT. DETROIT EDISON HAS FILED THE GENERIC RESPONSE AND REFERENCED THE GENERAL ELECTRIC REPORT NEDO-24342.

A GENERIC SAFETY EVALUATION REPORT (SER) FOR THE SCRAM DISCHARGE VOLUME PIPE BREAK IS SCHEDULED TO BE ISSUED IN LATE JULY BY THE NRC. DETROIT EDISON WILL REVIEW THE FERMI 2 DESIGN FOR COMPLIANCE WITH THE SER CRITERIA. WE EXPECT THAT THE SER WILL BE ISSUED AND OUR REVIEW COMPLETED FOR THE FERMI 2 PLANT SPECIFIC 120 DAY RESPONSE REQUIRED IN THE LETTER FROM ROBERT TEDESCO, DIVISION OF LICENSING.

III.H. ATWS

<u>SCRAM DISCHARGE VOLUME</u>

MICHELSON'S CONCERN

SDV PIPE BREAK REPORT

GE GENERIC RESPONSE

NRC GENERIC SER - AUGUST

FERMI 2 DESIGN

BROWNS FERRY 3

FERMI 2 SDV CONFIGURATION
SDV MODIFICATIONS

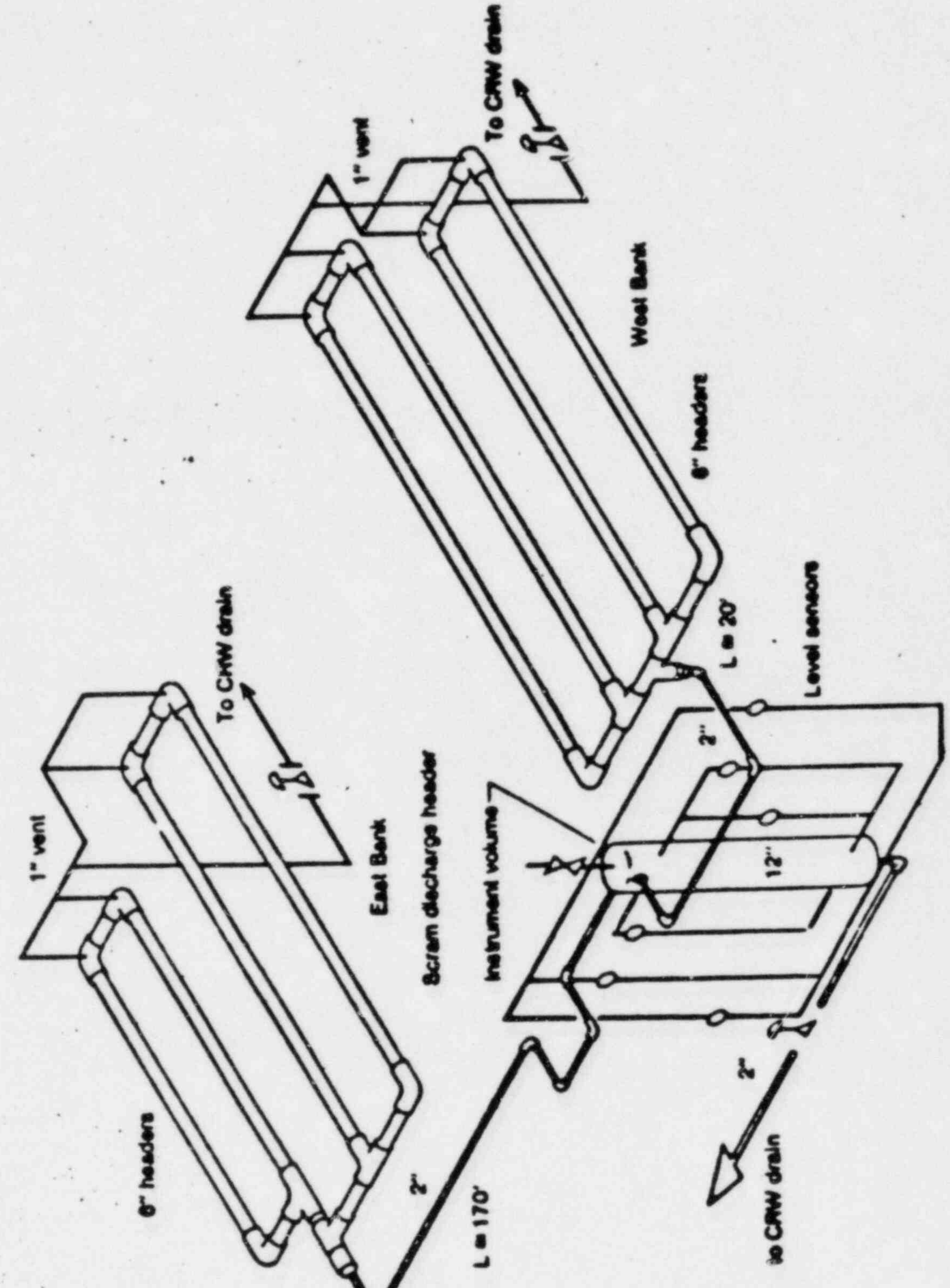
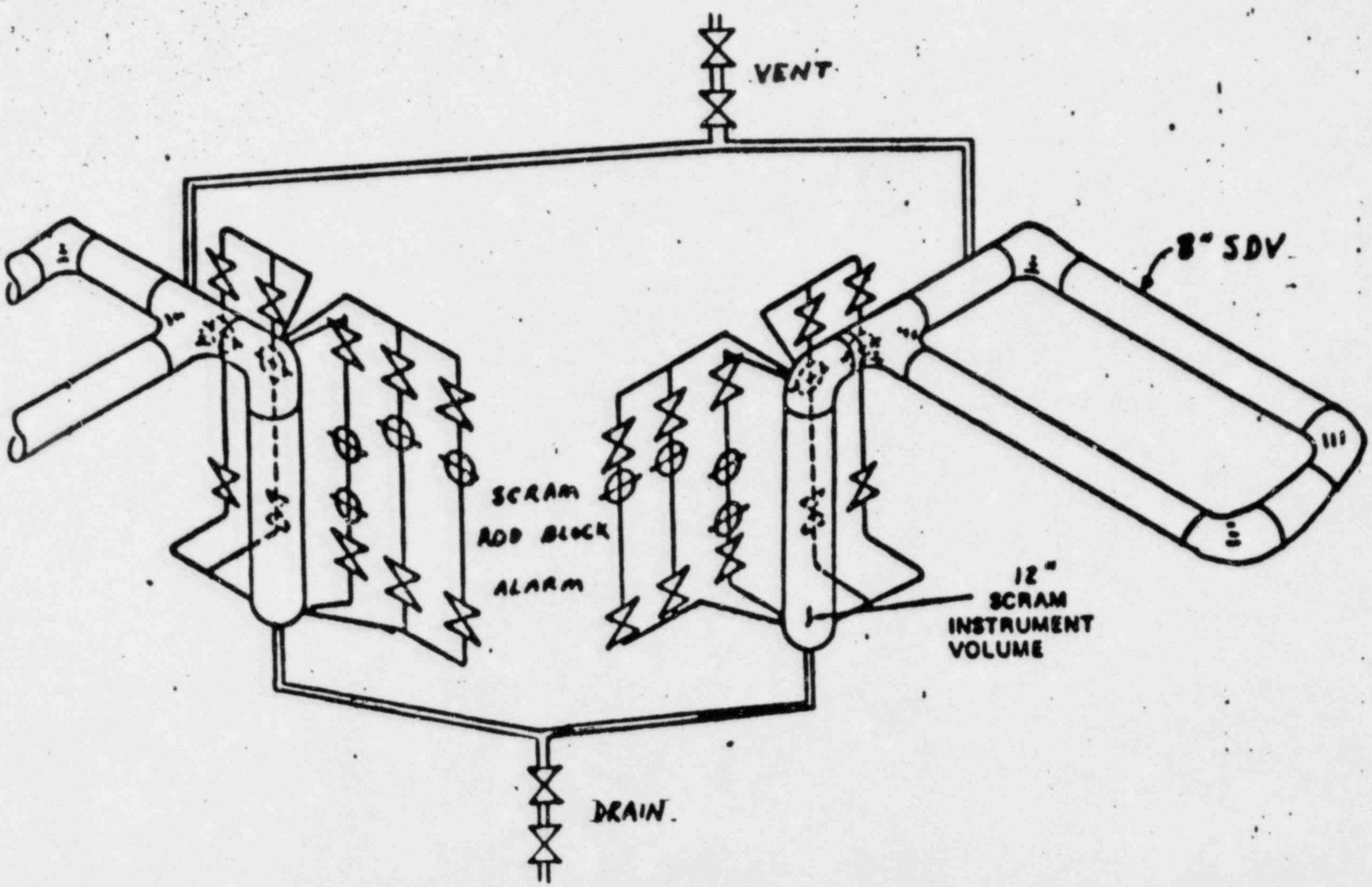


Figure 2. Browns Ferry SDV Equipment Layout (isometric view)

FERMI SDV CONFIGURATION



III.J. EMERGENCY PLANNING

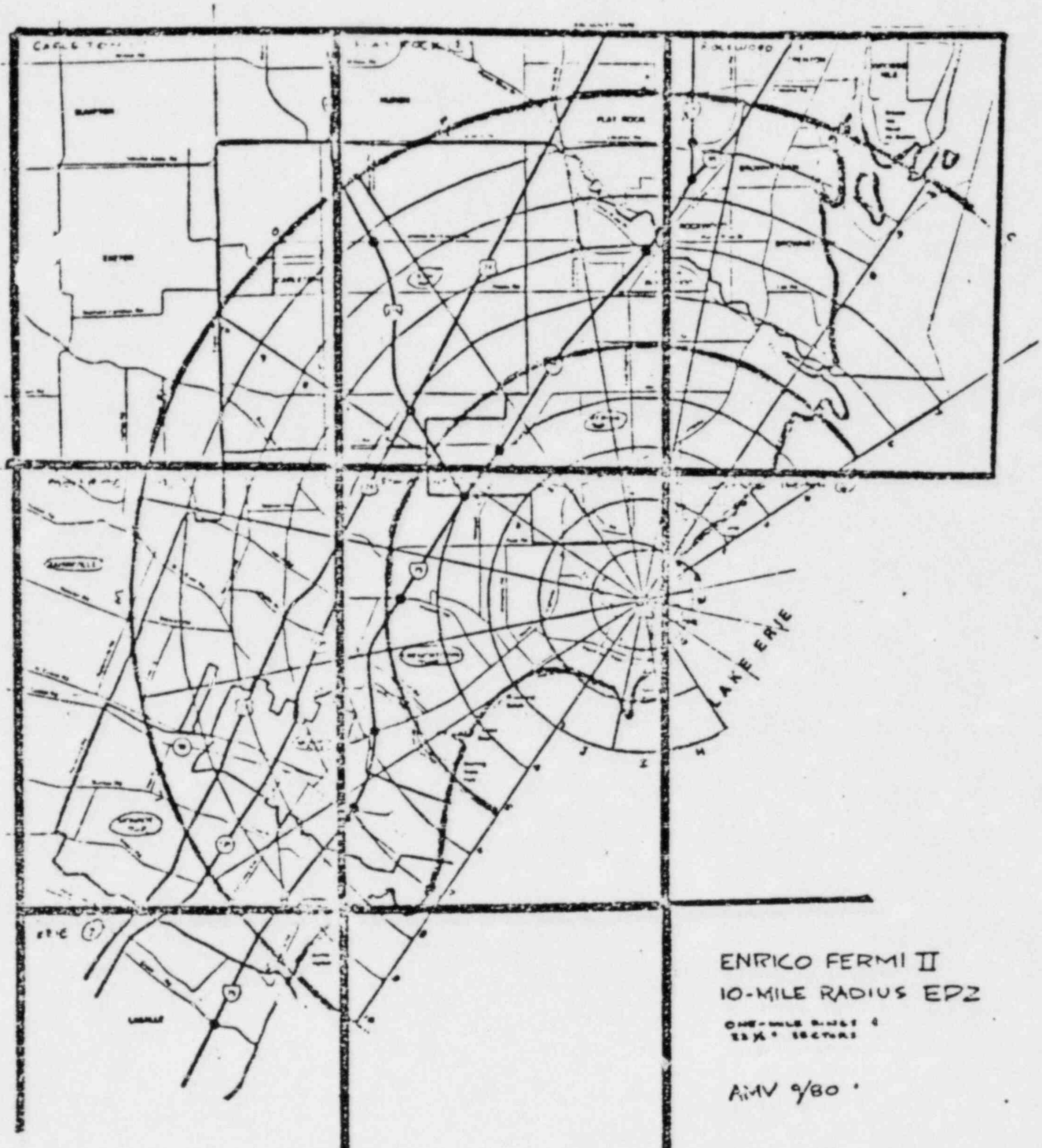
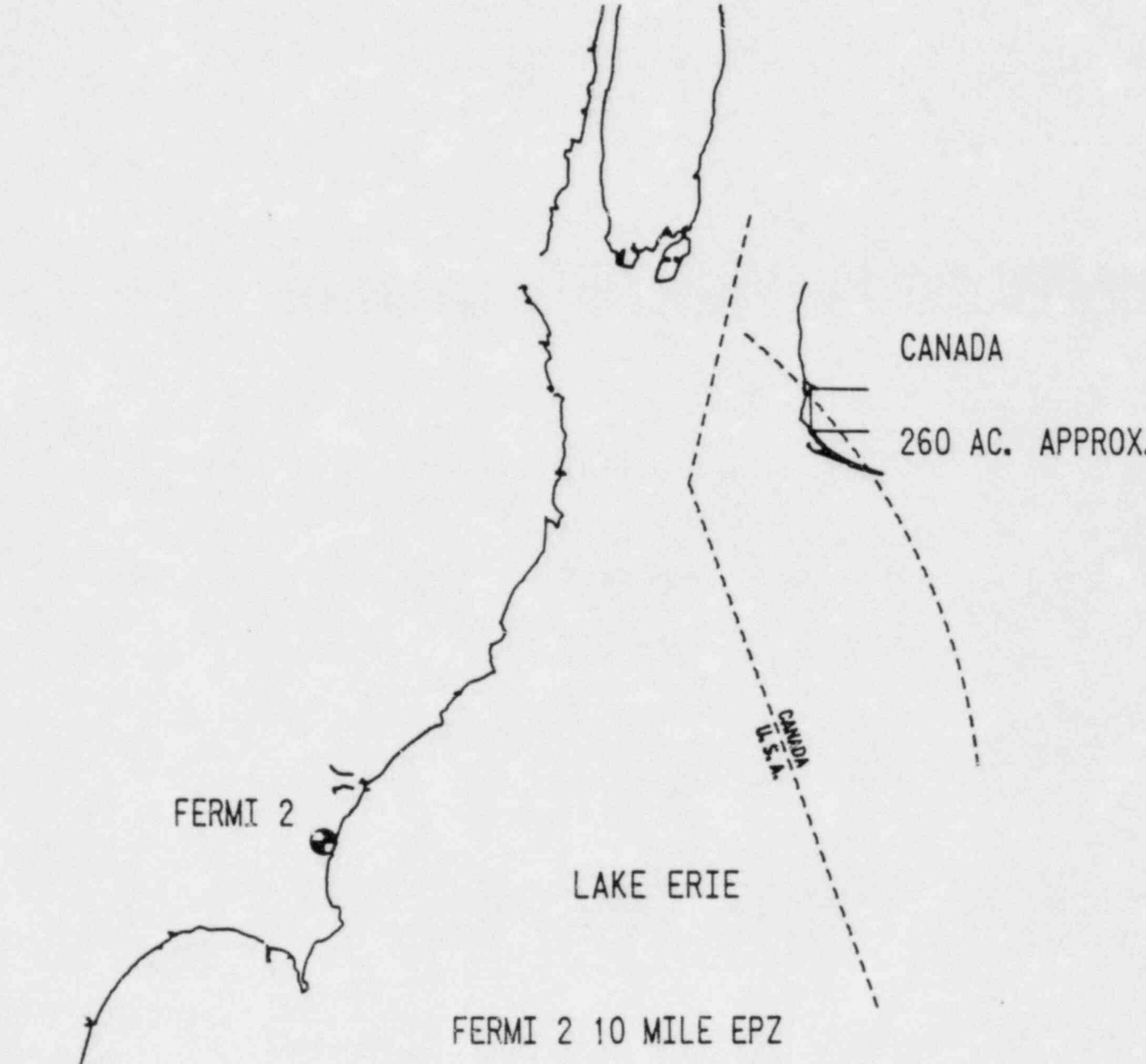


FIGURE 1



FERMI 2 10 MILE EPZ

FIGURE 1A

FIGURE 2

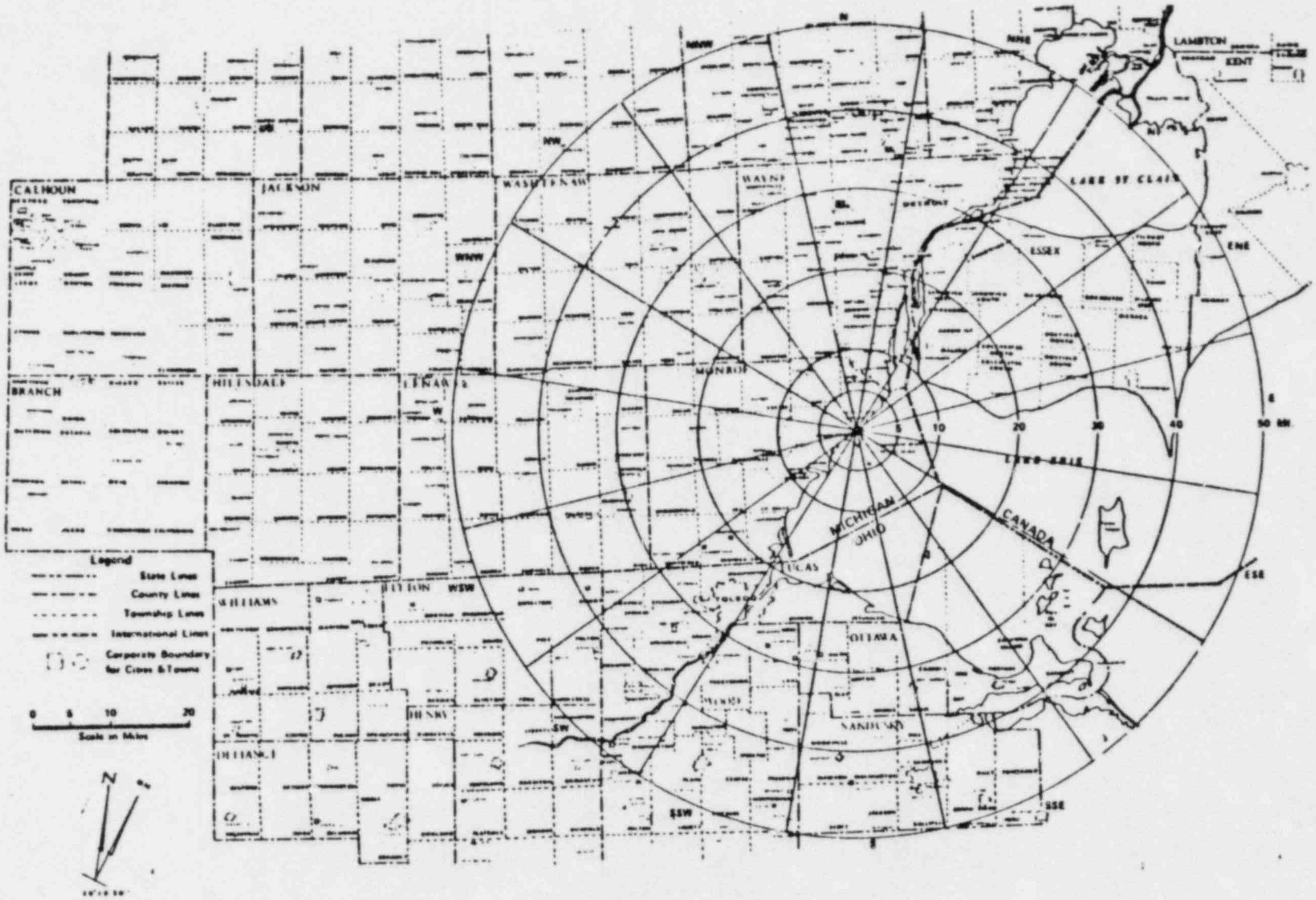
DISTRIBUTION OF PRELIMINARY 1980 POPULATION
PLUME EXPOSURE PATHWAY EMERGENCY PLANNING ZONE[(a)]

Sector (22 1/2°)	RING (One-Mile) 1	2	3	4	5	6	7	8	9	10	Total
N	29	263	177	79	197	230	873*	4,281*	4,297*	5,223*	15,657
NNE	0	102	12	90	81	377	1,191*	1,306**	945**	3,799**	7,903
NE	0	259	131	12	0	0	0**	--	--	--	402
ENE	0	--	--	--	--	--	--	--	--	--	--
E	0	--	--	--	--	--	--	--	--	--	--
ESE	0	--	--	--	--	--	--	--	--	--	--
SE	0	--	--	--	--	--	--	--	--	--	--
SSE	0	--	--	--	--	--	--	--	--	--	--
S	41	576	51	--	--	--	--	--	--	--	668
SSW	0	710	21	--	--	--	--	--	--	--	731
SW	0	208	9	--	117	0	0	0	65	871	1,270
WSW	0	24	846	2,236	1,779	979	4,404	9,277	13,725	6,089	39,359
W	0	58	29	165	600	974	978	1,176	678	685	5,343
WNW	0	18	31	52	109	1,924	474	188	608	612	4,016
NW	3	76	353	639	318	254	496	364	590	3,238	6,331
NNW	0	140	243	64	77	220	641	585	669	506*	3,145
Total	73	2,434	1,903	3,337	3,278	4,958	9,057	17,185	21,577	21,023	84,825

a. Legend:

--Area entirely in Lake Erie
**Area entirely in Wayne County
*Area includes both Wayne and Monroe Counties
Area entirely in Monroe County

A-186



ENRICO FERMI ATOMIC POWER PLANT, UNIT 2
50 MILE EMERGENCY PLANNING ZONE

FIGURE 3

IGURE 4

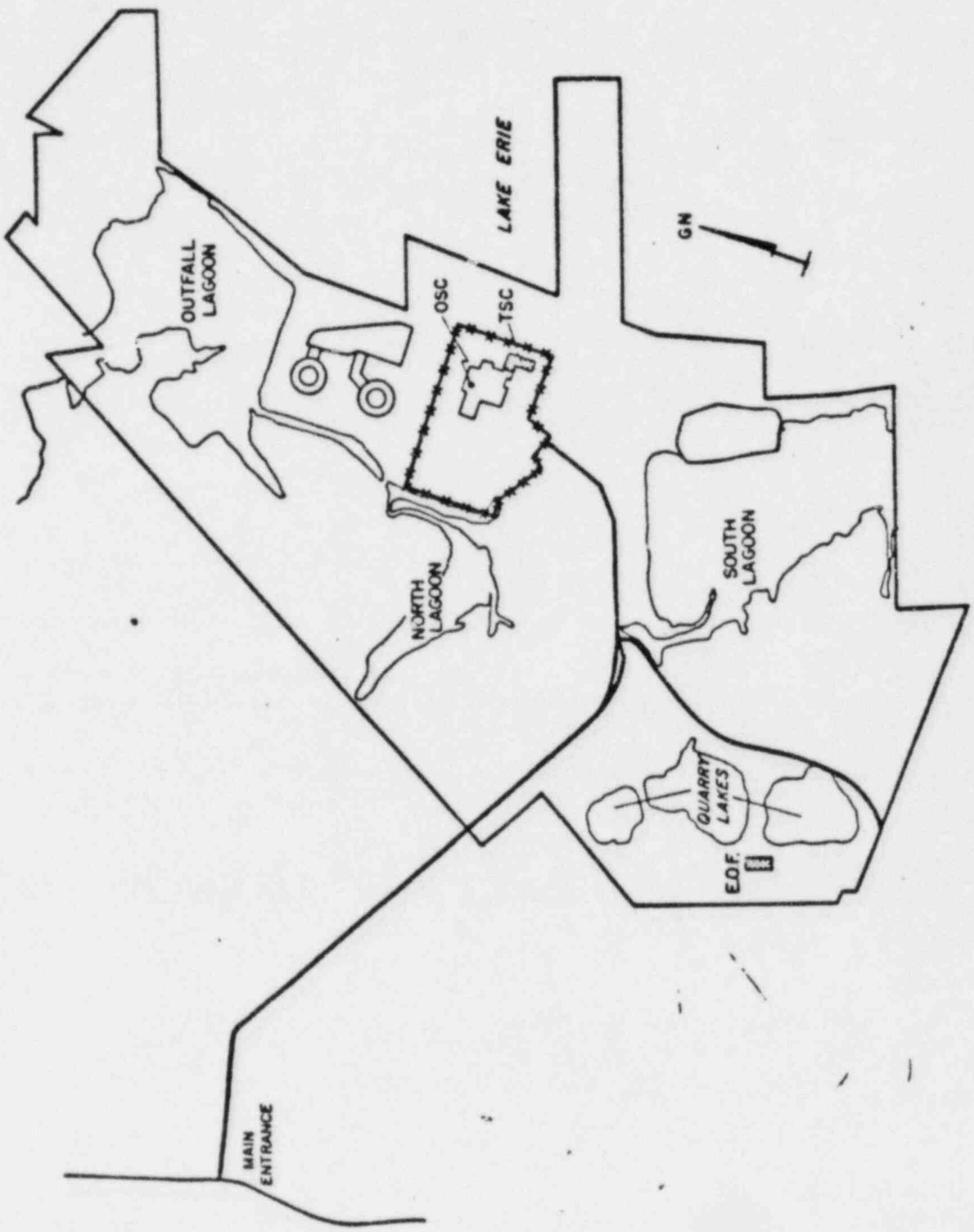
DISTRIBUTION OF PRELIMINARY 1980 POPULATION IN 10- to 50-MILE AREA AROUND FERMI 2

Dir	Distance (Miles) 10-15	15-20	20-25	25-30	30-35	35-40	40-45	45-50	10-50
N	10,355	95,468	199,447	331,196	369,786	316,228	128,588	183,949	1,635,017
NNE	24,993	75,873	87,272	527,834	585,704	440,902	251,639	112,020	2,106,237
NE	7,613	3,567	4,583	5,600	10,828	8,049	549	0	40,789
ENE	1,319	5,641	4,583	11,964	6,617	7,631	8,227	8,672	54,654
E	0	610	1,146	5,910	2,814	14,480	3,207	0	28,167
ESE	0	0	0	0	1,704	1,145	0	0	2,849
SE	0	0	0	401	1,853	4,860	11,622	36,051	54,787
SSE	0	0	0	1,052	6,398	10,255	7,664	14,256	39,625
S	0	0	3,693	2,875	9,116	6,539	26,539	8,591	57,353
SSW	0	3,004	36,423	71,520	16,369	6,211	14,170	24,353	172,050
SW	3,858	7,150	99,338	219,699	65,633	12,945	12,852	10,700	432,175
WSW	4,482	2,233	6,325	3,206	6,582	3,482	3,526	5,942	35,778
W	4,521	1,119	5,804	5,418	2,246	25,456	23,140	6,747	74,451
WNW	3,351	2,844	9,076	8,195	6,514	4,564	4,558	7,938	47,040
NW	0	7,398	23,921	74,264	104,923	11,262	21,742	16,059	259,570
NNW	5,273	14,272	68,913	51,444	56,144	21,463	27,752	41,318	286,579
Total	65,765	219,179	550,524	1,320,578	1,253,231	895,472	545,776	476,596	5,327,121

FIGURE 5

TENTATIVE SCHEDULE FOR FERMI 2 RADIOLOGICAL EMERGENCY RESPONSE PLAN (RERP)

• State of Michigan Plan Submitted to FEMA, Region V	February, 1981
• Fermi 2 RERP Submitted to NRC (10CFR50)	March 31, 1981
• Responses to NRC Comments	August 30, 1981
• FEMA, Region V recommendations to FEMA headquarters on Michigan plan	July, 1981
• Michigan submit to FEMA, Region V for approval • Michigan State Plan • Monroe County • Jurisdictions within Wayne Co. in 10 mile EPZ	November, 1981
• Fermi 2 full scale exercise and NRC appraisal	February, 1982 (first 2 weeks)



PLOT PLAN

FIGURE 6

FUNCTIONAL RESPONSIBILITIES

CONTROL ROOM

- DIAGNOSE THE ABNORMAL CONDITIONS
- PERFORM CORRECTIVE ACTIONS
- MITIGATE THE ABNORMAL CONDITIONS
- MANAGE PLANT OPERATIONS
- MANAGE EMERGENCY RESPONSE
- INFORM FEDERAL, STATE, AND LOCAL OFFICIALS
- RECOMMEND PUBLIC PROTECTIVE MEASURES TO STATE & LOCAL OFFICIALS
- RESTORE THE PLANT TO A SAFE CONDITION
- RECOVER FROM THE ABNORMAL CONDITIONS

TECHNICAL SUPPORT CENTER

- PROVIDE PLANT MANAGEMENT AND TECHNICAL SUPPORT TO PLANT OPERATIONS PERSONNEL DURING EMERGENCY CONDITIONS
- RELIEVE THE REACTOR OPERATORS OF PERIPHERAL DUTIES AND COMMUNICATIONS NOT DIRECTLY RELATED TO REACTOR SYSTEM MANIPULATIONS
- PREVENT CONGESTION IN THE CONTROL ROOM
- PERFORM EOF FUNCTIONS FOR THE ALERT EMERGENCY CLASS AND FOR THE SITE AREA EMERGENCY CLASS AND GENERAL EMERGENCY CLASS UNTIL THE EOF IS FUNCTIONAL

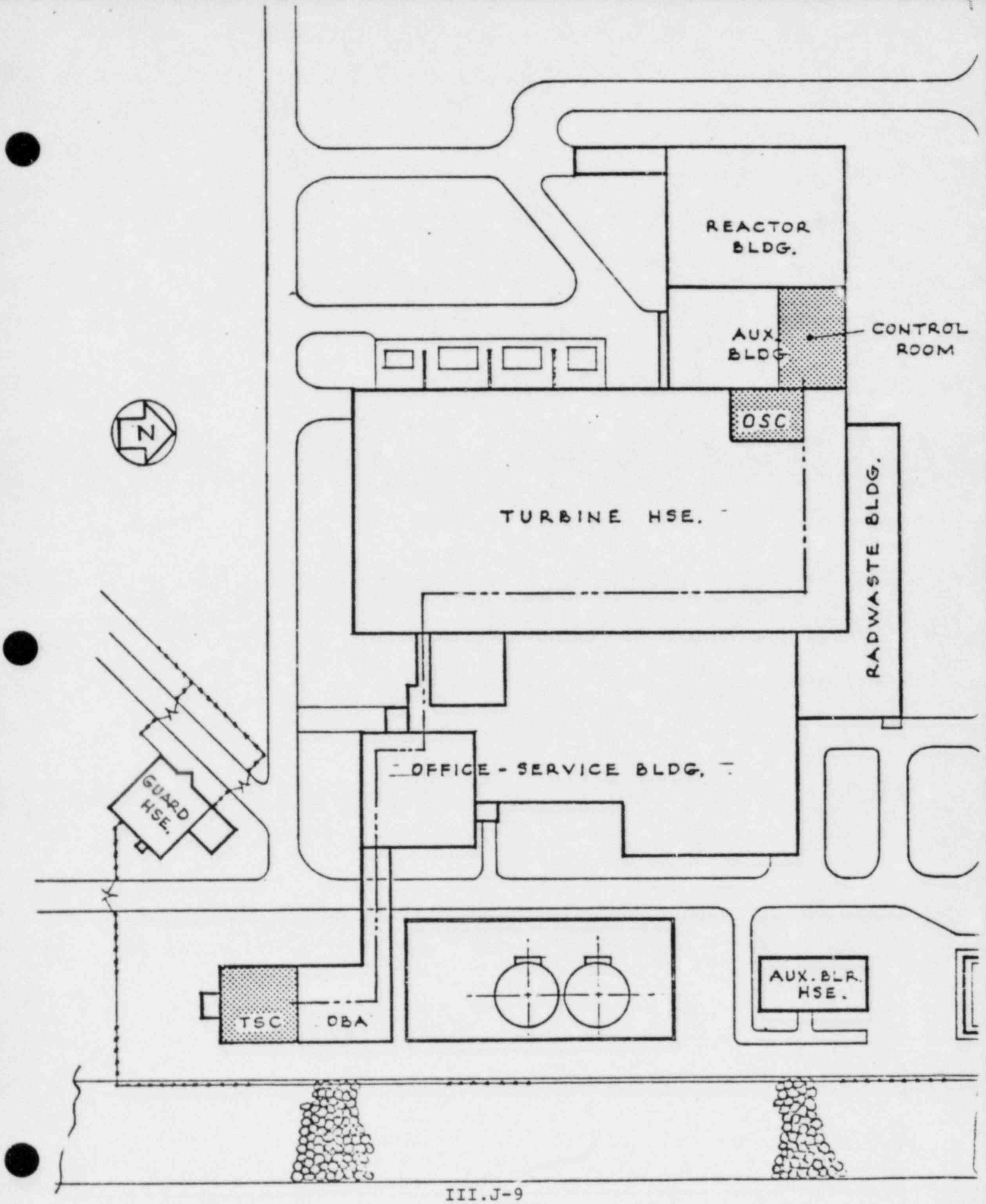
OPERATIONAL SUPPORT CENTER

- PROVIDE A LOCATION WHERE PLANT LOGISTIC SUPPORT CAN BE COORDINATED DURING AN EMERGENCY

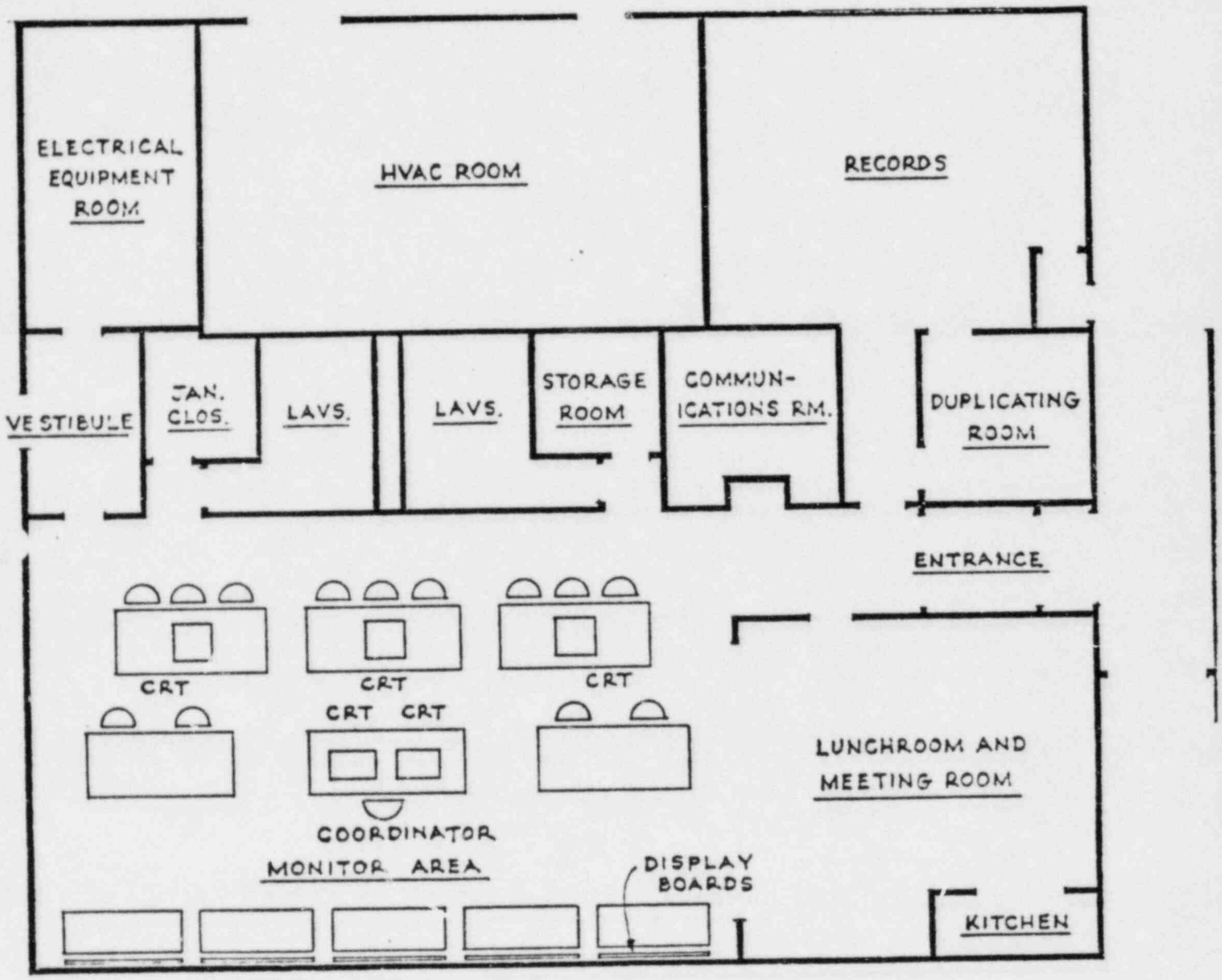
EMERGENCY OPERATIONS FACILITY

- MANAGEMENT OF OVERALL LICENSEE EMERGENCY RESPONSE
- COORDINATION OF RADIOLOGICAL AND ENVIRONMENTAL ASSESSMENT
- DETERMINATION OF RECOMMENDED PUBLIC PROTECTIVE ACTIONS
- COORDINATION OF EMERGENCY RESPONSE ACTIVITIES WITH FEDERAL, STATE, AND LOCAL AGENCIES

FIGURE 7



III.J-9
FIGURE 8



FLOOR PLAN

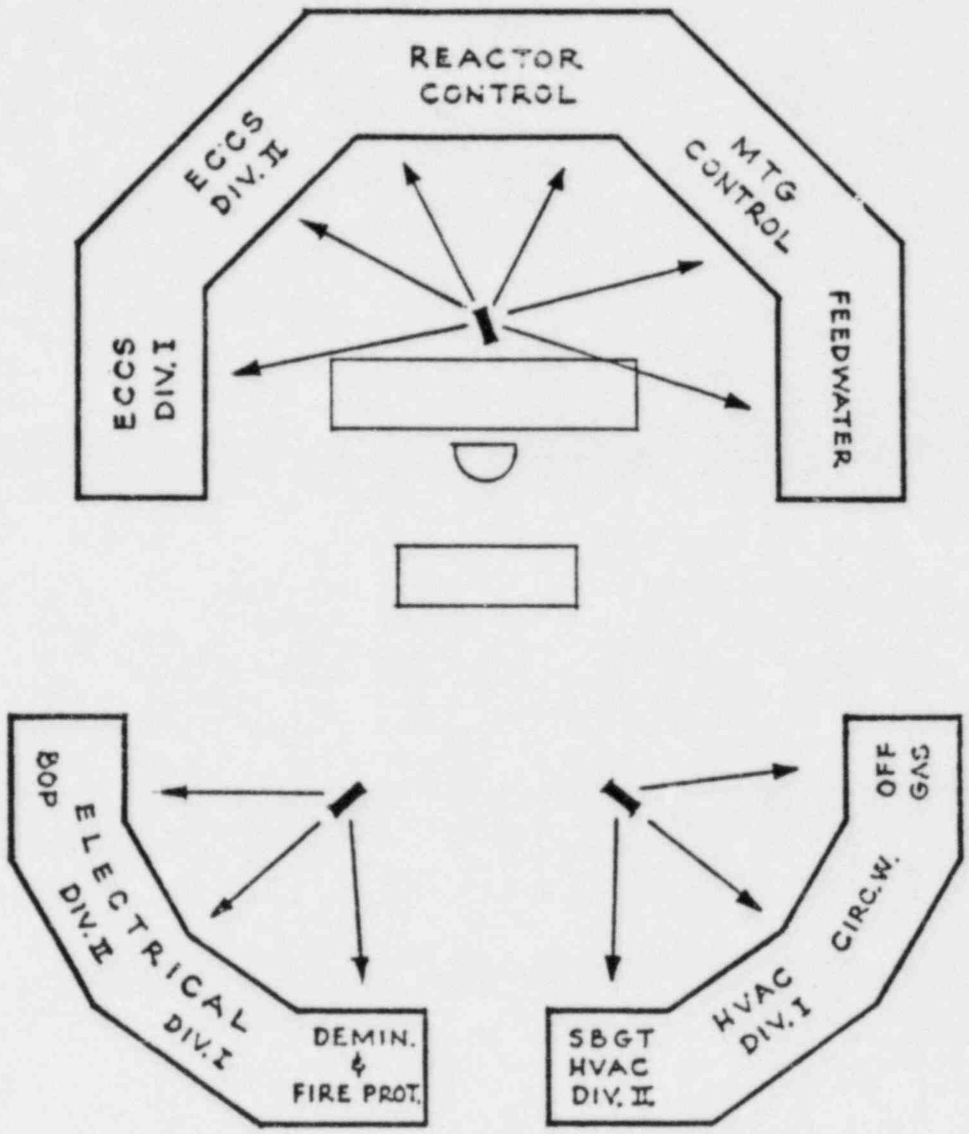
TECHNICAL SUPPORT CENTER

FIGURE 9

FIGURE 10

EMERGENCY RESPONSE INFORMATION SYSTEM

- PROVIDES INFORMATION TO THE CONTROL ROOM, TECHNICAL SUPPORT CENTER, AND EMERGENCY OPERATIONS FACILITY
- COMPUTER-BASED SYSTEM WITH CRTs AND PRINTERS
  - SUPPLEMENTED WITH CLOSED-CIRCUIT TV FROM EXISTING CONTROL ROOM DISPLAYS
- INFORMATION AVAILABLE
  - SAFETY PARAMETER DISPLAY SYSTEM
  - PLANT PARAMETER DATA TO DETERMINE STEADY STATE AND DYNAMIC BEHAVIOR PRIOR TO AND THROUGHOUT THE COURSE OF AN ACCIDENT
  - HISTORICAL DATA RETENTION/RETRIEVAL
  - METEOROLOGICAL DATA
  - DOSE ASSESSMENT AND PROJECTION
  - EFFLUENT MONITOR DATA



CONTROL ROOM

CCTV CAMERA LOCATIONS

FIGURE 11

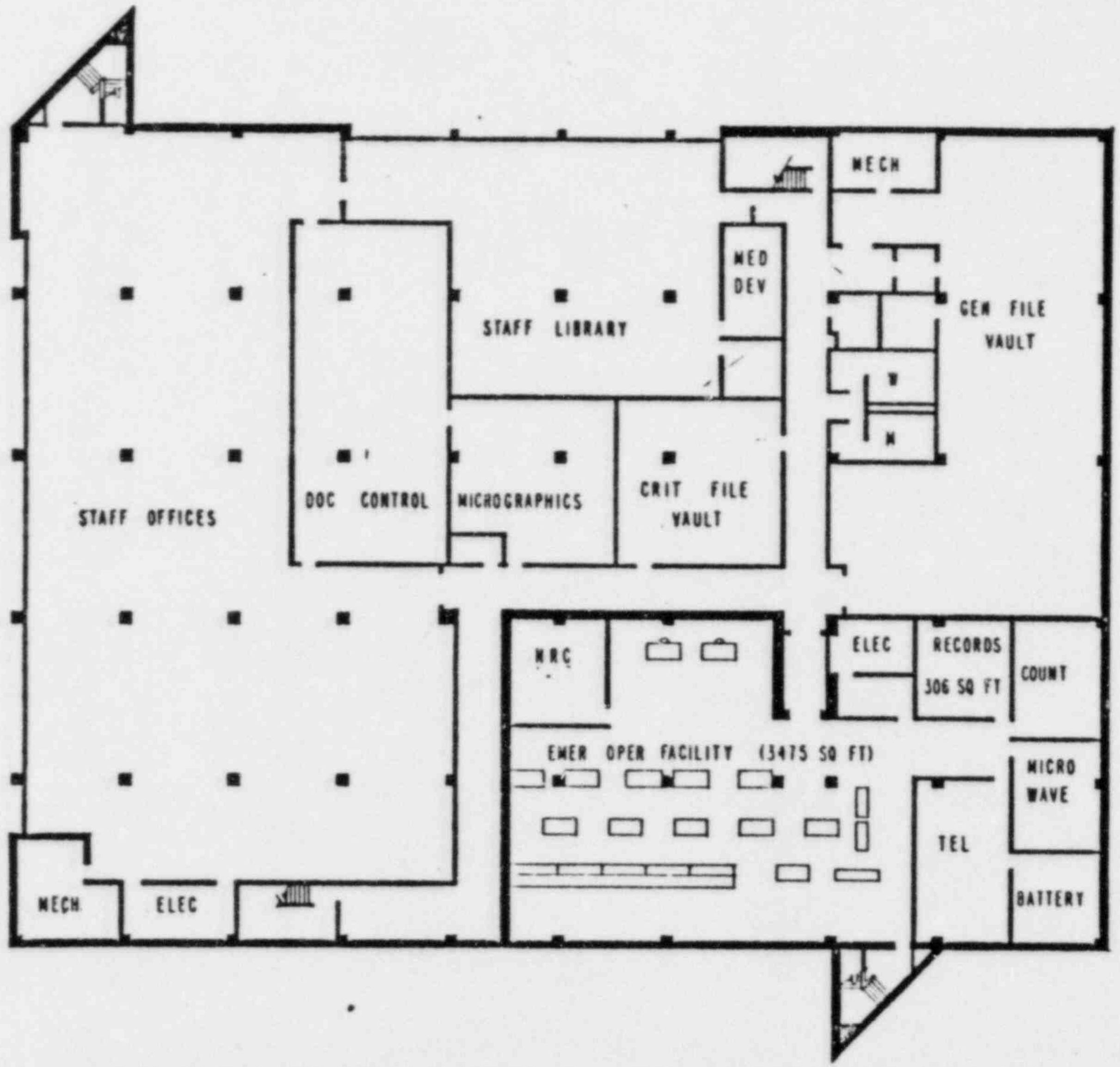
III.J-12

FIGURE 12

SAFETY PARAMETER DISPLAY SYSTEM

- BASED ON BWR OWNERS GROUP DISPLAYS
  - SUPPORTS EMERGENCY PROCEDURE ENTRY

    PARAMETERS INDICATING
    - CORE COOLING
    - FUEL INTEGRITY
    - REACTIVITY
    - REACTOR COOLANT SYSTEM INTEGRITY
    - CONTAINMENT INTEGRITY
    - RADIATION RELEASED TO ENVIRONMENT
  - APPROXIMATELY 30 VARIABLES ARE REQUIRED TO SHOW THE ABOVE PARAMETER SET. SOME ARE REPRESENTED BY SEVERAL INPUTS
- VALUES OF THE PARAMETERS ARE VALIDATED BY A SECOND VARIABLE, OR BY CALCULATED VARIABLES
- SYSTEM IS HIGH QUALITY BUT NOT SEISMIC
- DISPLAY IN CONTROL ROOM AND TSC



EMERGENCY OPERATIONS FACILITY

FIGURE 13

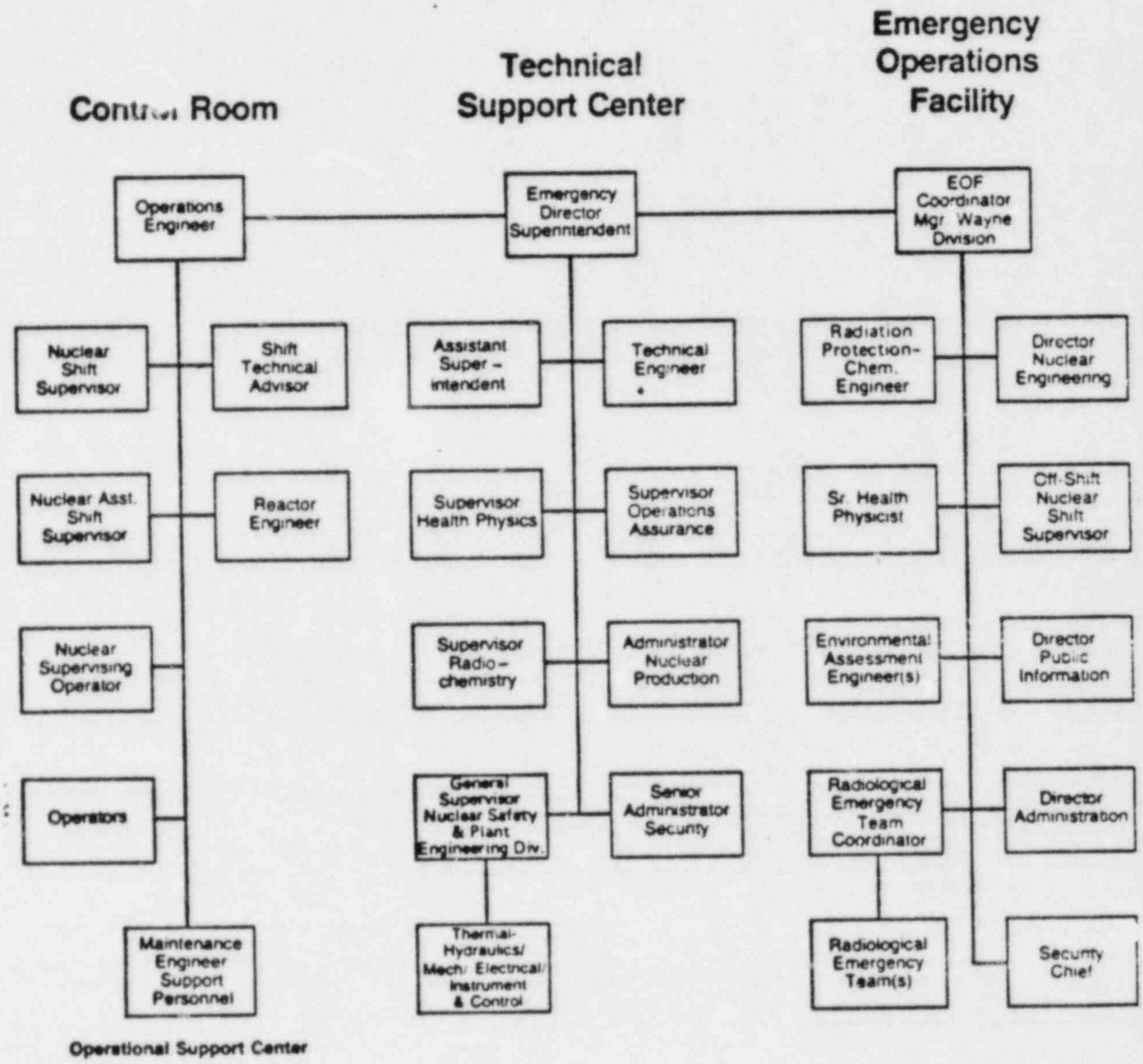
III.J-14

A-197

FIGURE 14

TRANSFER OF EMERGENCY RESPONSE FUNCTIONS FROM THE CONTROL ROOM TO THE EMERGENCY RESPONSE FACILITIES

EMERGENCY RESPONSE FUNCTIONS	EMERGENCY CLASS			
	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
SUPERVISION OF REACTOR OPERATIONS AND MANIPULATION OF CONTROLS	CR	CR	CR	CR
MANAGEMENT OF PLANT OPERATIONS	CR	TSC	TSC	TSC
TECHNICAL SUPPORT TO REACTOR OPERATIONS	CR	TSC	TSC	TSC
MANAGEMENT OF CORPORATE EMERGENCY RESPONSE RESOURCES	CR	TSC	EOF	EOF
RADIOLOGICAL EFFLUENT AND ENVIRONS MONITORING, ASSESSMENT, AND DOSE PROJECTIONS	CR	TSC	EOF	EOF
INFORM FEDERAL, STATE, AND LOCAL EMERGENCY RESPONSE ORGANIZATIONS AND MAKE RECOMMENDATIONS FOR PUBLIC PROTECTIVE ACTIONS	CR	TSC	EOF	EOF
EVENT MONITORING BY NRC REGIONAL EMERGENCY RESPONSE TEAM		TSC	TSC,EOF	TSC,EOF
MANAGEMENT OF RECOVERY OPERATIONS	CR	TSC	EOF	EOF
TECHNICAL SUPPORT OF RECOVERY OPERATIONS	CR	TSC	TSC	TSC



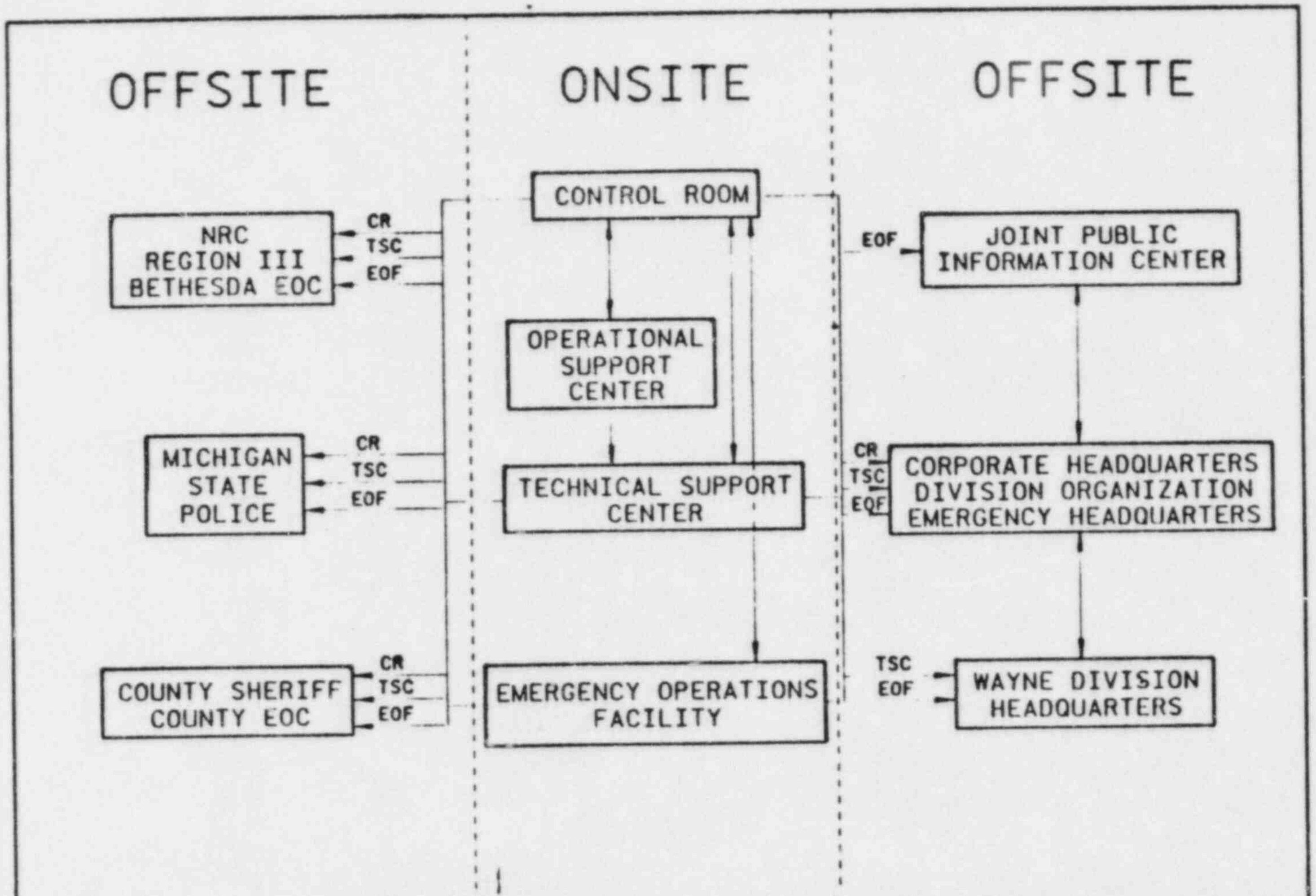
Control Room
Technical Support Center
Emergency Operations Facility
Operations Engineer
Emergency Director Superintendent
EOF Coordinator Mgr. Wayne Division
Nuclear Shift Supervisor
Shift Technical Advisor
Nuclear Asst. Shift Supervisor
Reactor Engineer
Nuclear Supervising Operator
Operators
Maintenance Engineer Support Personnel
Operational Support Center
Assistant Super-intendent
Technical Engineer
Supervisor Health Physics
Supervisor Operations Assurance
Supervisor Radio-chemistry
Administrator Nuclear Production
General Supervisor Nuclear Safety & Plant Engineering Div.
Senior Administrator Security
Thermal-Hydraulics/ Mech/ Electrical/ Instrument & Control
Radiation Protection-Chem. Engineer
Director Nuclear Engineering
Sr. Health Physicist
Off-Shift Nuclear Shift Supervisor
Environmental Assessment Engineer(s)
Director Public Information
Radiological Emergency Team Coordinator
Director Administration
Radiological Emergency Team(s)
Security Chief

FIGURE 15

FIGURE 16

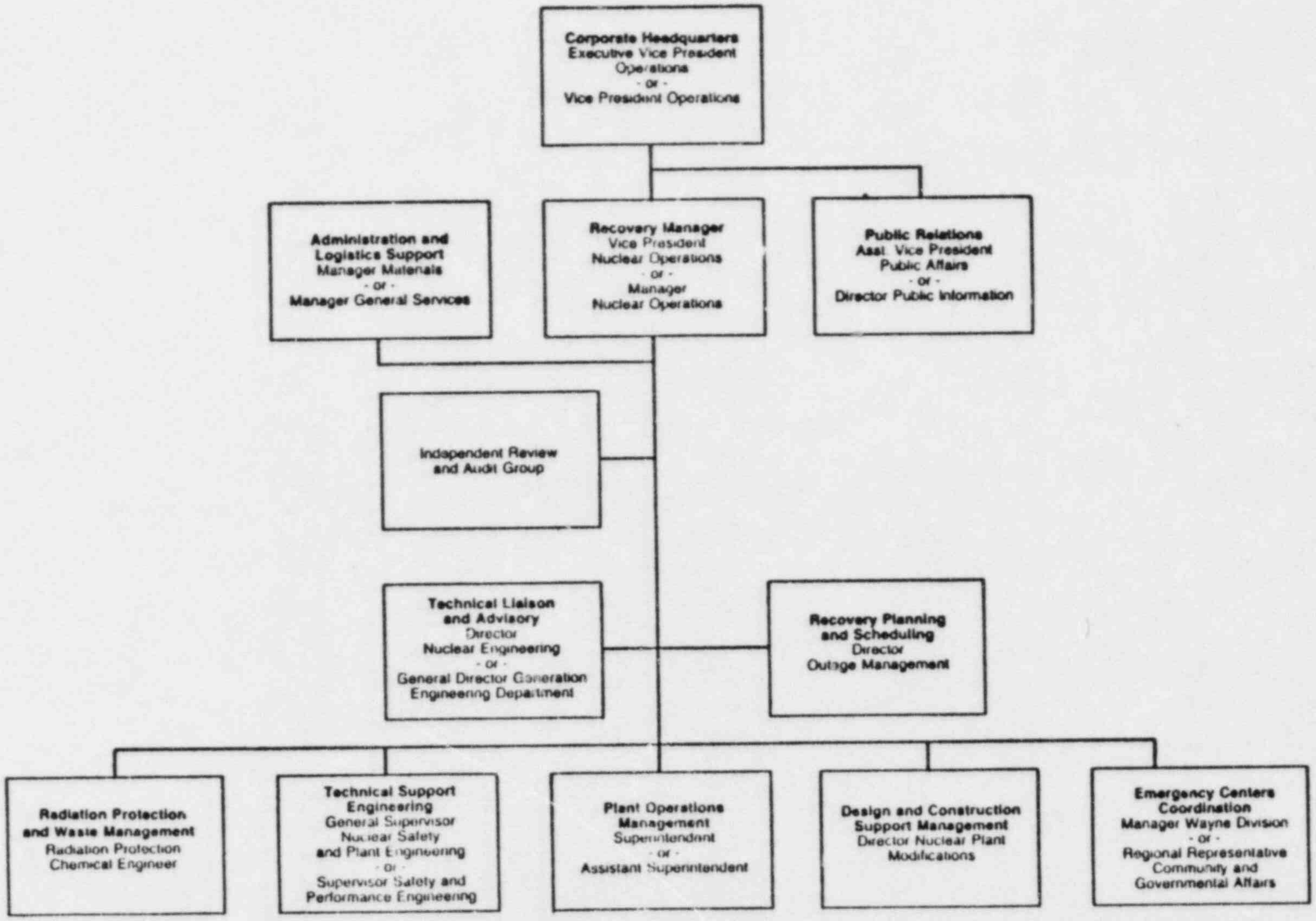
INFORMATION AVAILABLE IN EMERGENCY RESPONSE FACILITIES

	ISC	EOF
TECHNICAL DATA		
• PLANT SYSTEM VARIABLES	X	
• SAFETY PARAMETER DISPLAY SYSTEM	X	
• CLOSED CIRCUIT TV	X	
• IN-PLANT RADIOLOGICAL VARIABLES	X	X
• METEOROLOGICAL INFORMATION	X	X
• OFFISTE RADIOLOGICAL INFORMATION	X	X
RECORDS		
• PLANT TECHNICAL SPECIFICATIONS	X	X
• PLANT OPERATING PROCEDURES	X	X
• FINAL SAFETY ANALYSIS REPORT	X	X
• PLANT OPERATING RECORDS	X	X
• PLANT OPERATIONS REACTOR SAFETY COMMITTEE RECORDS AND REPORTS	X	X
• UP-TO-DATE RECORDS RELATED TO LICENSEE, STATE, AND LOCAL EMERGENCY RESPONSE PLANS	X	X
• OFFSITE POPULATION DISTRIBUTION DATA	X	X
• EVACUATION PLANS	X	X
• ENVIRONS RADIOLOGICAL MONITORING RECORDS	X	X
• LICENSEE EMPLOYEE RADIATION EXPOSURE HISTORIES	X	X
• CURRENT, AS-BUILT DRAWINGS, SCHEMATICS, AND DIAGRAMS	X	X
• CONDITIONS OF PLANT STRUCTURES AND SYSTEMS DOWN TO THE COMPONENT LEVEL		
• IN-PLANT LOCATIONS OF THESE SYSTEMS		



RADIOLOCICAL EMERGENCY RESPONSE COMMUNICATIONS

FIGURE 17



RECOVERY ORGANIZATION

FIGURE 18

EMERGENCY PLANNING

ROLE OF FEMA, STATE, AND LOCAL AGENCIES

THE RADIOLOGICAL EMERGENCY RESPONSE PLAN (RERP) FOR FERMI 2 WAS UPGRADED TO MEET THE REQUIREMENTS OF 10 CFR 50 AND NUREG-0654/ FEMA-REP-1 AND FILED WITH THE NRC ON MARCH 31, 1981. NRC COMMENTS ON THE PLAN WERE REVIEWED WITH THE STAFF ON JUNE 30 AND RESPONSES ARE DUE FROM EDISON AUGUST 30. A REVISED RERP AND PROCEDURES WILL BE SUBMITTED TO THE NRC JUST PRIOR TO THE FULL-SCALE EXERCISE IN FEBRUARY, 1982.

THE FERMI 2 RERP INCLUDES A PLUME EXPOSURE PATHWAY EMERGENCY PLANNING ZONE(EPZ) EXTENDING TO ABOUT 10 MILES AND AN INGESTION PATHWAY EPZ TO 50 MILES. THE EPZ FOR THE PLUME EXPOSURE INCLUDES ALL AREAS WITHIN 10 MILES THAT LIE IN MONROE COUNTY AND A SMALL PORTION OF THE SOUTHERN TIP OF WAYNE COUNTY, MICHIGAN (FIGURE 1). DETAILED MAPPING AND MEASUREMENT INDICATES THAT APPROXIMATELY 260 ACRES OF THE LAND MASS OF CANADA LIES WITHIN THE 10-MILE EPZ (FIGURE 1A). FIGURE 2 IS A BREAKDOWN OF THE U.S. POPULATION BY DIRECTION AND DISTANCE WITHIN THE 10-MILE EPZ.

FIGURE 3 REPRESENTS THE 50-MILE EPZ AND INCLUDES PORTIONS OF MICHIGAN, OHIO, AND THE PROVINCE OF ONTARIO, CANADA.

FIGURE 4 REPRESENTS THE POPULATION, INCLUDING THE U.S. AND CANADA WITHIN THE 50-MILE ZONE.

THE EMERGENCY SERVICES DIVISION OF THE MICHIGAN STATE POLICE IS THE RESPONSIBLE PARTY IN MICHIGAN FOR ALL EMERGENCIES, INCLUDING THOSE AT NUCLEAR POWER PLANTS. AT THE PRESENT TIME, SINCE THE PROVINCE OF ONTARIO, CANADA IS IN THE 50-MILE EPZ, NOTIFICATION OF ANY EMERGENCIES AT FERMI 2 WILL BE MADE BY THE STATE POLICE EMERGENCY SERVICES DI[illegible]TOR.

THE U. S. COAST GUARD IS RESPONSIBLE FOR THE WATERS WITHIN THE U. S. THAT ARE IN THE 10-MILE EPZ. AGREEMENTS ARE IN EXISTENCE BETWEEN THE CANADIAN AND U. S. COAST GUARDS THAT ARE BEING INVESTIGATED WITH RESPECT TO NOTIFICATION ON CANADIAN WATERS THAT LIE IN THE FERMI 2 EPZ.

THE MICHIGAN STATE PLAN, PREPARED BY THE EMERGENCY SERVICES DIVISION OF THE STATE POLICE, WAS SUBMITTED TO FEMA, REGION V FOR APPROVAL IN FEBRUARY, 1981. FEMA WILL BE SUBMITTING ITS FINDINGS ON THE MICHIGAN PLAN FOR THE OPERATING PLANTS TO FEMA HEADQUARTERS STARTING IN JULY, 1981. ONCE MICHIGAN IS COMPLETED, FEMA INTENDS TO PROCEED WITH THE OHIO STATE PLAN.

THE FERMI 2 FULL-SCALE EXERCISE AND NRC APPRAISAL PROGRAM ARE SCHEDULED FOR THE FIRST TWO WEEKS IN FEBRUARY 1982,

SUBJECT TO APPROVAL BY ALL PARTIES INVOLVED (FIGURE 5). THE MICHIGAN STATE POLICE ARE WORKING TOWARD THIS TARGET DATE AND ARE SCHEDULED TO SUBMIT THE PLANS FOR MONROE COUNTY AND JURISDICTIONS WITHIN WAYNE COUNTY IN NOVEMBER, 1981. FEMA HAS BEEN CONTACTED AND IS PRESENTLY REVIEWING THIS SCHEDULE.

FOLLOWING THE FULL-SCALE EXERCISE AND APPRAISAL, A SCHEDULE WILL BE DETERMINED FOR IMPLEMENTATION OF ANY UNRESOLVED ISSUES IDENTIFIED BY NRC OR FEMA BY FUEL LOAD OR FULL POWER OPERATION.

EMERGENCY SUPPORT FACILITIES

THE CONTROL ROOM (CR), TECHNICAL SUPPORT CENTER (TSC), OPERATIONAL SUPPORT CENTER (OSC), AND EMERGENCY OPERATIONS FACILITY (EOF) ARE THE FOUR FACILITIES THAT WILL BE USED TO RESPOND TO AN EMERGENCY AT FERMI 2. FIGURE 6 IS A PLOT PLAN OF THE SITE SHOWING THE LOCATIONS OF EACH FACILITY. THE EMERGENCY SUPPORT FACILITIES WHICH ADDRESS THE REQUIREMENTS OF NUREG-0737 AND 0696 ARE DESCRIBED IN APPENDIX H TO THE FSAR.

CONTROL ROOM (CR)

THE CONTROL ROOM (CR) PROVIDES A CENTRALIZED LOCATION FOR DAY-TO-DAY PLANT OPERATIONS. IN THE EVENT OF AN EMERGENCY, IT PROVIDES THE INITIAL ONSITE CENTER OF EMERGENCY CONTROL. CR PERSONNEL EVALUATE AND EFFECT CONTROL OVER THE INITIAL ASPECTS OF AN EMERGENCY, INITIATE ACTIONS NECESSARY FOR COPING WITH THE INITIAL PHASES OF AN EMERGENCY UNTIL THE SUPPORT CENTERS ARE ACTIVATED, AND INFORM THE FEDERAL, STATE, AND LOCAL OFFICIALS (FIGURE 7).

TECHNICAL SUPPORT CENTER (TSC)

FUNCTIONALLY, ONCE ACTIVATED THE TSC PROVIDES INFORMATION OF PLANT STATUS FOR USE BY TECHNICAL AND MANAGEMENT PERSONNEL

IN SUPPORT OF THE COMMAND AND CONTROL FUNCTIONS CARRIED OUT IN THE CONTROL ROOM. IT ALSO FUNCTIONS AS THE PRIMARY INFORMATION/COMMUNICATIONS SOURCE TO THE NRC, THE OSC, AND THE EOF. ADDITIONALLY, THE FUNCTIONS OF THE EOF ARE PERFORMED IN THE TSC UNTIL SUCH TIME AS THE EOF IS ACTIVITED (FIGURE 7).

THE TSC IS A 5000 SQUARE FOOT HARDENED FACILITY ON THE GROUND FLOOR OF A TWO-STORY OFFICE SERVICE BUILDING LOCATED WITHIN THE SECURITY PERIMETER OF THE FERMI 2 PLANT AND IS AN EASY 4-MINUTE WALK INSIDE FROM THE CONTROL ROOM THROUGH THE TURBINE BUILDING AS SHOWN IN FIGURE 8. THE TSC IS DESIGNED TO ACCOMMODATE 25 PERSONS REPRESENTING DETROIT EDISON AND THE NRC (FIGURE 9). THE OPEN DESIGN WITH MOVABLE PARTITIONS AND RAISED COMPUTER FLOOR PROVIDES MAXIMUM FLEXIBILITY FOR THE PERSONNEL AND CRT POSITIONING.

THE HEART OF THE TSC IS A COMPUTER-BASED INFORMATION SYSTEM WITH CRTs AND PRINTERS (FIGURE 10). PART OF THE PERMANENT DATA REQUIREMENTS ARE PROVIDED BY CLOSED CIRCUIT TV (CCTV) POSITIONED AS SHOWN IN FIGURE 11 ON EXISTING CONTROL ROOM DISPLAYS. THE CCTV ALSO PROVIDES THE INTERIM DATA INFORMATION SOURCE UNTIL THE COMPUTER-BASED SYSTEM IS AVAILABLE. THE SAFETY PARAMETER DISPLAY SYSTEM (SPDS) IS AN IMPORTANT FUNCTION ON THE COMPUTER-BASED SYSTEM AND INCLUDES THE FEATURES SHOWN IN FIGURE 12.

OPERATIONAL SUPPORT CENTER (OSC)

THE OSC (FIGURE 7 AND 8) IS A DESIGNATED AREA AT THE NORTH END OF THE THIRD FLOOR OF THE TURBINE BUILDING AND PROVIDES AN ASSEMBLY POINT FOR SHIFT SUPPORT PERSONNEL FOR ASSIGNMENT OF DUTIES IN SUPPORT OF EMERGENCY OPERATIONS. PERSONNEL SUCH AS INSTRUMENT TECHNICIANS, ENGINEERS, MECHANICS, ELECTRICIANS, RADIATION/HEALTH PHYSICS TECHNICIANS, EQUIPMENT OPERATORS, ETC., ARE DISPATCHED FROM THIS AREA.

EMERGENCY OPERATIONS FACILITY (EOF)

THE EOF IS A COMMAND POST FOR THE OVERALL MANAGEMENT OF THE EMERGENCY RESPONSE WITH OFFSITE ORGANIZATIONS, THE COORDINATION OF RADIOLOGICAL AND ENVIRONMENTAL ASSESSMENTS, THE DETERMINATION OF RECOMMENDED PROTECTIVE ACTIONS FOR THE PUBLIC, AND MANAGEMENT OF RECOVERY OPERATION (FIGURE 7). THE EOF (FIGURE 13), DESIGNED TO HANDLE 40 PERSONS, IS IN THE BASEMENT OF THE NUCLEAR OPERATIONS CENTER, AND WILL BE LOCATED APPROXIMATELY THREE-FOURTHS (3/4) OF A MILE SOUTH WEST OF THE PLANT ON OWNER-CONTROLLED PROPERTY (FIGURE 6).

THE EOF FUNCTIONS TO PROVIDE ASSISTANCE IN THE DECISION MAKING PROCESS TO PROTECT THE PUBLIC HEALTH AND SAFETY AND

TO CONTROL RADIOLOGICAL EMERGENCY MONITORING TEAMS AND FACILITIES ONSITE AND OFFSITE. RADIOLOGICAL AND METEOROLOGICAL DATA AND ADEQUATE PLANT SYSTEM INFORMATION ARE PROVIDED TO PERFORM THESE FUNCTIONS. THE EOF IS NORMALLY THE FOCAL POINT FOR THE RECEIPT AND ANALYSIS OF ALL FIELD MONITORING DATA AND THE COORDINATION OF SAMPLE MEDIA.

ACTIVATION OF EMERGENCY SUPPORT FACILITIES

FIGURE 14 SHOWS THE TRANSFER OF EMERGENCY RESPONSE FUNCTIONS FROM THE CONTROL ROOM TO THE EMERGENCY SUPPORT FACILITIES ACCORDING TO EMERGENCY CLASSIFICATION. AT THE ALERT LEVEL, THE TSC AND OSC ARE ACTIVATED. FOR A SITE AREA AND/OR GENERAL EMERGENCY, THE EOF IS FULLY ACTIVATED.

WHEN ALL SUPPORT FACILITIES ARE ACTIVATED, THE ORGANIZATION SHOWN IN FIGURE 15 IS AVAILABLE TO PERFORM THE FUNCTIONS REQUIRED FROM TECHNICAL SUPPORT TO ADMINISTRATIVE AND LOGISTICS SUPPORT. TO EFFECTIVELY CARRY OUT THESE DUTIES, THE INFORMATION SHOWN ON FIGURE 16 IS AVAILABLE IN THE TSC AND EOF.

SINCE THE CR, TSC AND EOF ARE THE PRIMARY COMMAND AND CONTROL CENTERS FOR BOTH ONSITE AND OFFSITE ORGANIZATIONS, A COMMUNICATIONS NETWORK OF DEDICATED AND DIRECT-DIAL TELEPHONE LINES ARE AVAILABLE WITH RADIO AND MICROWAVE BACKUP (FIGURE 17).

AT SUCH TIME AS THE RECOVERY MANAGER, WHO IS THE VICE PRESIDENT OF NUCLEAR OPERATIONS, DETERMINES THE PLANT IS IN THE RECOVERY PHASE, THE ORGANIZATION SHOWN IN FIGURE 18 BECOMES EFFECTIVE. THIS ORGANIZATION OPERATES FROM THE EOF AND PROVIDES THE TECHNICAL, ADMINISTRATIVE, AND LOGISTICS SUPPORT NECESSARY TO RECOVER FROM THE EMERGENCY.

III.I. HYDROGEN CONTROL

Not Done

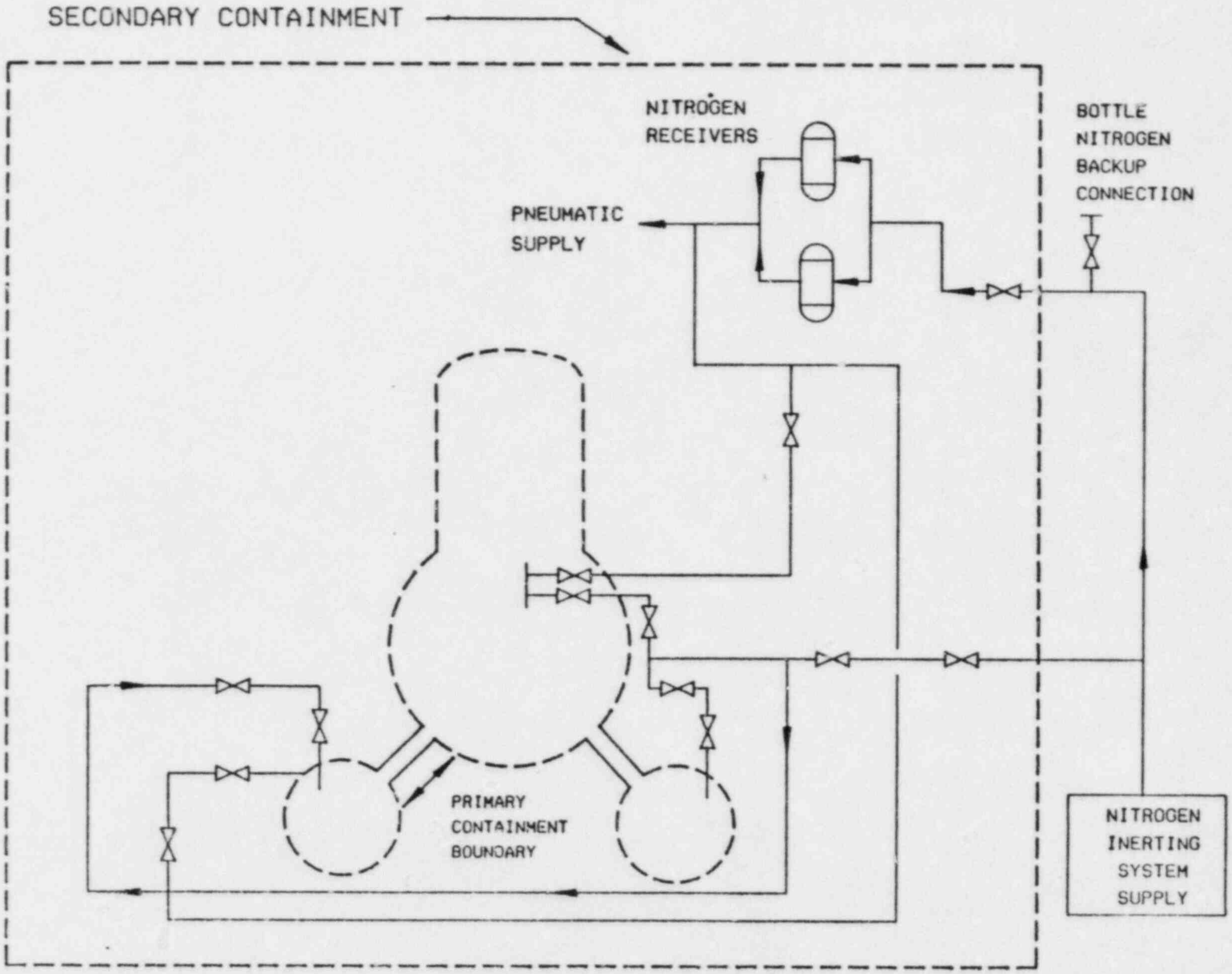
HYDROGEN CONTROL

NITROGEN INERTING SYSTEM

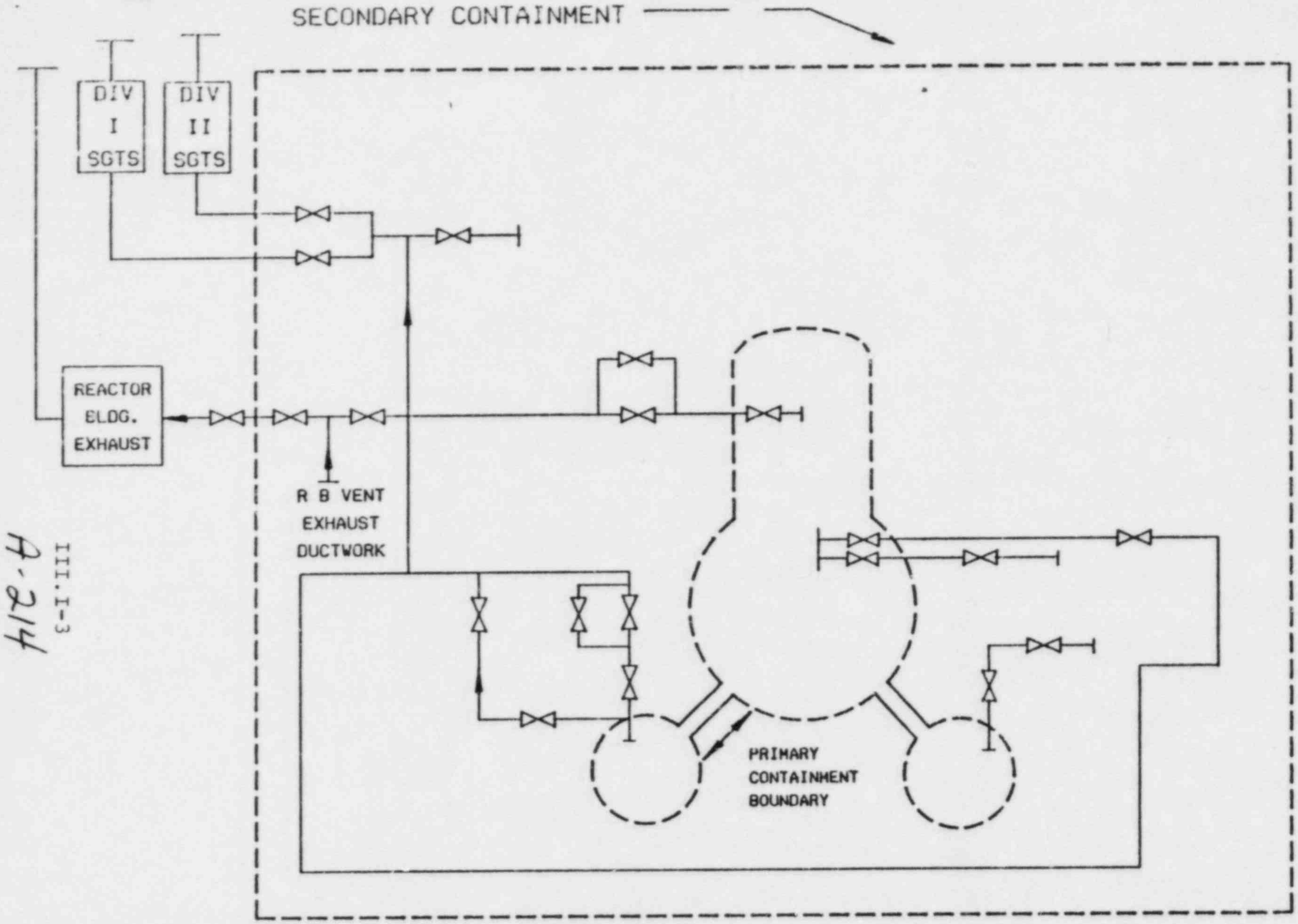
HYDROGEN/OXYGEN MONITORING

THERMAL RECOMBINERS

PURGE SYSTEM

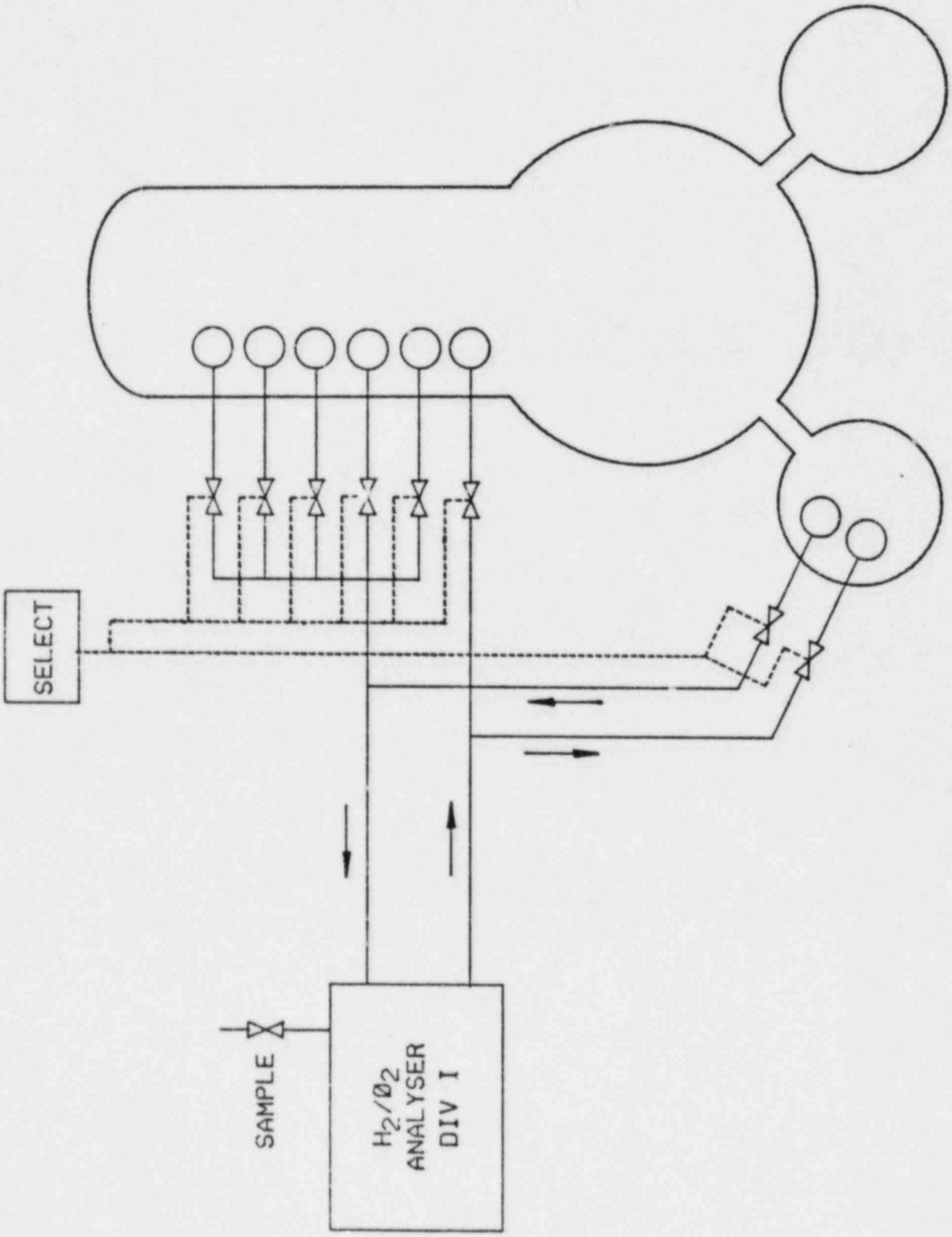


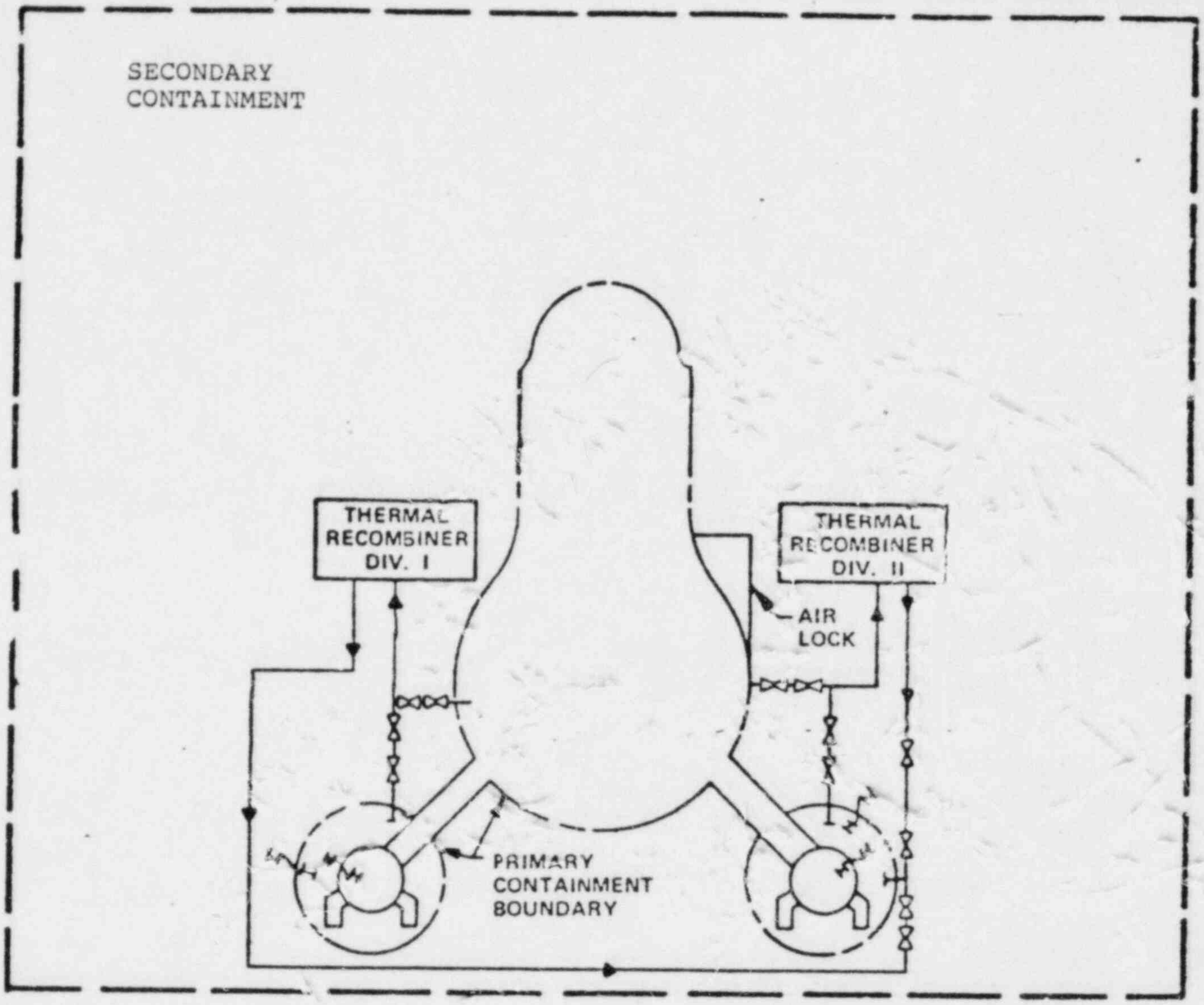
FERMI 2 NITROGEN INERTING SYSTEM



FERMI 2 PURGE SYSTEM

HYDROGEN / OXYGEN MONITORING





FERMI 2 THERMAL RECOMBINERS

HYDROGEN CONTROL

A COMPOSITE OF SYSTEMS IS USED TO DETECT AND CONTROL THE OXYGEN AND HYDROGEN CONCENTRATIONS IN PRIMARY CONTAINMENT. FOLLOWING A LOCA, HYDROGEN GAS COULD BE GENERATED AS A RESULT OF THE METAL-WATER REACTION AND BOTH HYDROGEN AND OXYGEN WOULD BE GENERATED AS A RESULT OF RADIOLYTIC DECOMPOSITION OF RECIRCULATING COOLANT. THE CORROSION OF CONTAINMENT MATERIALS WAS EVALUATED AND DOES NOT REPRESENT A SIGNIFICANT HYDROGEN SOURCE. THE CONTROL OF COMBUSTIBLE GAS MIXTURES IN THE PRIMARY CONTAINMENT IS ASSURED BY THE FOLLOWING PLANT SYSTEMS:

HYDROGEN/OXYGEN SAMPLING

THERMAL HYDROGEN RECOMBINERS

INERTING SYSTEM

PURGE SYSTEM

HYDROGEN/OXYGEN SAMPLING

THE HYDROGEN AND OXYGEN CONCENTRATIONS ARE CONTINUOUSLY MONITORED AND DISPLAYED IN THE CONTROL ROOM. THERE ARE

TWO SEPARATE SAMPLING SYSTEMS, EACH IS REDUNDANT AND INDEPENDENT. THE SAMPLING SYSTEMS ARE SEISMICALLY AND ENVIRONMENTALLY QUALIFIED ENGINEERED SAFETY FEATURE SYSTEMS. TO ASSURE REPRESENTATIVE SAMPLING, MULTIPLE PORTS ALLOW GAS TO BE DRAWN INTO THE MONITORING SYSTEM FROM SEVERAL LOCATIONS IN THE CONTAINMENT. AN ALARM INITIATES WHEN THE OXYGEN CONCENTRATION REACHES A PRESET LIMIT.

THE POST-LOCA ON-LINE SAMPLING SYSTEM SERVES AS A BACKUP FOR HYDROGEN/OXYGEN MONITORING.

THERMAL RECOMBINERS

REDUNDANT AND INDEPENDENT THERMAL RECOMBINERS ARE INSTALLED AT FERMI 2 TO ENSURE THAT A COMBUSTIBLE GAS MIXTURE DOES NOT BUILD UP AND IMPAIR THE CONTAINMENT INTEGRITY. EACH RECOMBINER IS INDIVIDUALLY CAPABLE OF LIMITING THE AMOUNT OF OXYGEN IN THE CONTAINMENT ATMOSPHERE TO LESS THAN THE COMBUSTIBLE CONCENTRATION IN CONFORMANCE TO REGULATORY GUIDE 1.7. THESE COMBUSTIBLE GAS CONTROL SYSTEMS (CGCS) CONFORM TO 10 CFR 100 SECTION 50.44 GENERIC DESIGN CRITRIA 41 AND BRANCH TECHNICAL POSITION CSB 6-2.

THE CGCS IS AN ENGINEERED SAFETY FEATURE SYSTEM AND IS SEISMICALLY AND ENVIRONMENTALLY QUALIFIED. THE RECOMBINERS ARE LOCATED IN THE REACTOR BUILDING OUTSIDE OF PRIMARY CONTAINMENT. THE SYSTEM OPERATES TO CONTROL THE MINORITY CONSTITUENT, OXYGEN. THE PROCESS FLOW IS 150 SCFM WITH AN INLET FLOW OF 60 SCFM FOR CONTAINMENT ATMOSPHERE CONTAINING FIVE VOLUME PERCENT OXYGEN.

A BLOWER DRAWS THE CONTAINMENT ATMOSPHERE FROM EITHER THE DRYWELL OR TORUS THROUGH DEDICATED PENETRATIONS. THE HYDROGEN AND OXYGEN RECOMBINATION TAKES PLACE IN THE SKID REACTION CHAMBER AT AN ELEVATED TEMPERATURE. AFTER THE REACTION, THE RESULTANT STEAM IS COOLED AND CONDENSED WITH THE RESULTING WATER AND ANY REMAINING GAS RETURNED TO THE TORUS.

THIS RECOMBINER SYSTEM IS AN INTEGRAL PACKAGE PRODUCED BY ATOMICS INTERNATIONAL. EACH SYSTEM INCLUDES THE SKID MOUNTED HYDROGEN RECOMBINER, A LOCAL POWER CABINET AND A REMOTE CONTROL CABINET IN THE RELAY ROOM. ALL EQUIPMENT NECESSARY TO START A COMBUSTIBLE GAS CONTROL SYSTEM IS LOCATED ON THE MAIN CONTROL ROOM PANEL.

NITROGEN INERTING SYSTEM

THE FUNCTION OF THE NITROGEN INERTING SYSTEM (NIS) IS TO PROVIDE AND MAINTAIN A NITROGEN ATMOSPHERE INSIDE THE PRIMARY

CONTAINMENT, AND TO PROVIDE PRESSURIZED NITROGEN FOR PNEUMATIC SERVICE INSIDE THE PRIMARY CONTAINMENT AND DISTRIBUTION THROUGHOUT THE PLANT. THE PRIMARY CONTAINMENT WILL BE INERTED AND CONTROLLED TO LESS THAN 4% OXYGEN. EVEN IF LARGE QUANTITIES OF HYDROGEN ARE GENERATED FOLLOWING A LOCA THE INERTED CONTAINMENT WILL HAVE INSUFFICIENT OXYGEN TO SUPPORT COMBUSTION OF HYDROGEN.

THE INERTING SYSTEM IS NOT A SAFETY-RELATED SYSTEM AND IS NOT DESIGNED TO MEET SEISMIC AND OTHER CRITERIA EXCEPT WHERE PENETRATION AND ISOLATION IS CONCERNED. THE CLOSURE OF CONTAINMENT ISOLATION VALVES AND OTHER SELECTED FEED VALVES IN THE NIS WILL OCCUR FOR LOW REACTOR WATER LEVEL-2, HIGH DRYWELL PRESSURE OR HIGH RADIATION IN THE REACTOR BUILDING EXHAUST. THE ISOLATION SIGNALS AND VALVE ACTUATORS CONFORM TO THE REQUIREMENTS OF BRANCH TECHNICAL POSITION CSB 6-4.

THE INERTING SYSTEM IS COMPOSED OF LARGE LINES TO THE TORUS AND DRYWELL VALVES SIZED 20 AND 24 INCHES, RESPECTIVELY. THIS SYSTEM IS HIGH FLOW AT LOW PRESSURE AS PROVIDED THROUGH A STEAM VAPORIZER FED FROM A NITROGEN STORAGE TANK. THE INERT SYSTEM IS USED INFREQUENTLY WITH ISOLATION VALVE OPERATION LIMITED TO 90 HOURS PER YEAR FOR CONDITIONS OTHER THAN COLD SHUTDOWN OR REFUELING PER CSB 6-4.

A PRESSURE CONTROLLED NITROGEN MAKE-UP SYSTEM IS PROVIDED AT FERMI 2 FOR ON-LINE MAKE-UP OF NITROGEN DUE TO LEAKAGE. THE DRYWELL IS KEPT SLIGHTLY POSITIVE RELATIVE TO SECONDARY CONTAINMENT. THE MAKE-UP SYSTEM IS COMPRISED OF 1-1/2 INCH LINES TO THE DRYWELL AND TORUS FROM THE PRESSURIZED NITROGEN SYSTEM. THE MAKE-UP SYSTEM IS LOW FLOW AT HIGH PRESSURE AND IS FED THROUGH AN ELECTRIC HEATER FED FROM THE NITROGEN STORAGE TANK. THE ISOLATION VALVES IN THIS SYSTEM COMPLY WITH CSB 6-4.

PURGE SYSTEM

CONTAINMENT PURGE CAPABILITY IS PROVIDED TO EVACUATE THE CONTAINMENT ATMOSPHERE AND TO FUNCTION AS BACK-UP HYDROGEN CONTROL. THE PURGE SYSTEM IS NOT A SAFETY-RELATED SYSTEM AND IS NOT SEISMICALLY QUALIFIED WITH THE EXCEPTION OF THE PENETRATIONS AND ISOLATION VALVES.

THE DRYWELL PURGE INLET AND VENT OUTLET LINES ARE 20 INCHES IN DIAMETER WHILE THE TORUS INLET AND OUTLET VALVES ARE 24 INCH IN DIAMETER. THESE LINES ARE USED TO EVACUATE THE CONTAINMENT ATMOSPHERE DURING THE PURGE (DE-INERT) OPERATION. THE PURGE SYSTEM CONNECTS TO EITHER THE REACTOR BUILDING EXHAUST OR THE STANDBY GAS TREATMENT SYSTEM. THE USE OF THESE ISOLATION VALVES IN THE PURGE OPERATION IS LIMITED TO 90 HOURS PER YEAR DURING CONDITIONS OTHER THAN COLD SHUT-DOWN OR REFUELING PER CSB 6-4.

AN ON-LINE PURGE SYSTEM IS INCLUDED AT FERMI 2. THIS IS A 1-1/2 INCH DIAMETER SYSTEM USED TO VENT CONTAINMENT ATMOSPHERE TO MAINTAIN PRESSURE CONTROL RELATIVE TO THE SECONDARY CONTAINMENT.

BOTH THE LARGE PURGE SYSTEM AND SMALL ON-LINE SYSTEM COMPLY WITH CSB 6-4. THIS INCLUDES THE FIVE-SECOND VALVE CLOSURE, DEBRIS SCREENS, 90-HOUR LIMIT AND DIVERSE ISOLATION (I.E., LOW REACTOR WATER LEVEL-2, HIGH DRYWELL PRESSURE AND HIGH RADIATION IN THE REACTOR BUILDING EXHAUST).

<u>SUMMARY</u>

THE FERMI 2 DESIGN HAS INCORPORATED SEVERAL HYDROGEN CONTROL MEASURES. THE LIMITING OF A COMBUSTIBLE GAS MIXTURE WITH AN INERTED CONTAINMENT AND POST LOCA HYDROGEN CONTROL WITH DEDICATED THERMAL RECOMBINERS ARE THE KEY FEATURES.

F001F017

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

June 15, 1981

M. Plesset, Chairman
Fluid Dynamics Subcommittee

NRC MEMO IN RESPONSE TO YOUR REQUEST FOR INFORMATION ON THE MARK II CONTAINMENT LATERAL LOAD SPECIFICATION

Attached is a copy of a memo from K. Kniel responding to questions you posed on the development of the lateral load specification for the Mark II containment downcomers. I have also attached a copy of a memo from NRC to Dr. Chau, Mark II Owners Group Chairman. The Chau memo requests response to questions raised by the Fluid Dynamics Subcommittee at our April 28-29, 1981 meeting.

Per your instructions, arrangements have been made to have the above questions discussed by the Owners' Group/NRC at the August full Committee meeting in conjunction with the Shorham plant review.

Attachment: As Stated

cc: ACRS Members
ACRS Technical Staff

ATTACHMENT I

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

June 4, 1981

MEMORANDUM FOR: Raymond Fraley, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology, NRR

SUBJECT: MARK II CONTAINMENT LATERAL LOAD SPECIFICATION

This memorandum is in response to several questions raised by Dr. Plesset, Chairman, Subcommittee on Fluid Dynamics, (following the April 28, 1981 Subcommittee Meeting) through Paul Boenert to Cliff Anderson regarding the long term program lateral load specification. The two questions are:

1. What is the basis for fatigue analysis of Mark II downcomers?

2. Was the impulsive nature of the downcomer lateral load considered by the staff?

Regarding the first question, the design controlling load for a Mark II downcomer is an impulsive chugging load* applied one time to the downcomer. Fatigue calculations are performed using the maximum impulsive load (i.e., 30,000 lb_f) for several hundred cycles. Fatigue calculations performed for the lead Mark II plants indicate that only a small fraction of the fatigue life would be used during a LOCA. The staff has requested that the Mark II owners provide the details of the fatigue calculation (see Enclosure 1).

With regard to the second question, the most significant Mark II downcomer lateral load is an impulsive load. This is true of both the original lead plant static equivalent load (See Appendix C.3 of NUREG-0487) and the long term program dynamic load. The dynamic load proposed by the Mark II owners is represented by half sine waves of duration ranging from 3 to 6 ms. for high and low intensities, respectively. The maximum load amplitude ranges from 10,000 lb_f to 30,000 lb_f and may be considered to be uniformly distributed over 1 to 4 feet of the downcomer end. The

*Note: This is in contrast to the downcomer design controlling load for a Mark I containment. For Mark I's, the controlling load is a harmonic condensation oscillation load. The impulsive chugging load for Mark I's is a secondary load. Thus, for Mark I containments, fatigue considerations are important.

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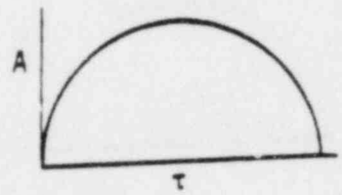
mathematical expression for the transient load function can be represented by:

$$F(t) = A \sin \frac{\pi t}{\tau}\text{, Lateral Load } (lb_f)$$

where: $A = 10^4\ lb_f$ = maximum amplitude with τ (period) = 6 msec

and $A = 3 \times 10$- lb_f with τ = 3 msec



Sine Function

This load is considered in combination with LOCA induced submerged structure drag loads, safety relief valve loads and seismic loads. This load combination is evaluated for a faulted condition.

As discussed in Enclosure 2, the staff and our consultants are not convinced that the 30,000 lb_f dynamic load is conservative for single vent calculations. We are considering a dynamic load criterion which is approximately double the Mark II owners load specification (i.e., 65,000 lb_f, 3 msec period). The basis for this specification will be provided in the staff NUREG report documenting our evaluation of the Mark II long term program. We have discussed this increase in the single vent load with the Mark II owners. It appears that most of their plants can accommodate this load increase without any hardware modifications. We do not see a need to modify the Mark II owners' multivent load or fatigue calculations (i.e., 30,000 lb_f is acceptable for fatigue calculations).

I hope that this information has been responsive to your questions. Should you have any questions about this memo, contact Cliff Anderson, extension 29424.

Karl Kniel

Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology
Office of Nuclear Reactor Regulation

Enclosures:
1. ACRS Questions to Mark II Owners Group
2. NRC Consultants evaluation of Mark II OG Lateral Load

cc: w/enclosures
M. Plesset, ACRS
J. Ebersole, ACRS
W. Mathis, ACRS
D. Ward, ACRS
P. Boehner, ACRS (6)
F. Schroeder
P. Norian
C. Anderson
W. Butler

Enclosure 1

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

May 20, 1981

Dr. H. Chau, Chairman
Mark II Owners Group
Long Island Lighting Company
175 E. Old Country Road
Hicksville, New York 11801

Dear Dr. Chau:

Two questions were raised by the ACRS at the April 28, 29, 1981 Fluid Dynamics Subcommittee meeting. These questions are enclosed.

We are asking that you provide the staff with your response within 30 days of receipt of this letter in order that we may respond to the ACRS. Contact Clifford Anderson, (301) 492-9424 should you have any questions about this request.

Sincerely,

Karl Kniel

Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology

Enclosure:
Questions

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Enclosure

ACRS INFORMATION REQUEST

MARK II POOL DYNAMIC LOADS

The ACRS questions relate to potential pool by-pass from stuck open wetwell-to-drywell vacuum breakers and by-pass thru ruptured main vent downcomers.

1. The Mark II containment wetwell/drywell vacuum breakers may be called upon to function repeatedly during intermittent steam condensation phenomena. Failure of a vacuum breaker to close during this time could result in pool bypass, thus jeopardizing the integrity of the containment. Provide valve service information for the range of wetwell/drywell vacuum breaker operation under pool dynamic loading. This information should be sufficient to describe the opening and closing characteristics of these valves during intermittent steam condensation.

2. The steam chugging loads proposed as design loads by the Mark II Owner's Group for containment evaluation were developed to represent limiting conditions during a postulated loss-of-coolant accident. As such, they are applied one time to the downcomers. The Mark II Owner's have indicated that chugging induced fatigue loads are insignificant due to the large reduction in loads with time and the relatively low number of fatigue cycles. Additional information is needed to evaluate this position. Provide the following information for a typical plant. For a range of break sizes, estimate the number of "Pool Chug" cycles. This analysis should be extended sufficiently in time to include operation of the emergency core cooling system and the associated vessel steaming rates. If there exists a vent flow threshold below which the chugging loads are insignificant, this should be clearly identified in the analysis. The information should be provided for both the chugging downcomer lateral loads and the pool chugging loads.

The following pages A-229 thru A-247 has been deleted containing proprietary information.

DELETION - 3

C. Anderson
C. Coebrect
H-1016

LONG ISLAND LIGHTING COMPANY

175 EAST OLD COUNTRY ROAD · HICKSVILLE, NEW YORK 11801

Direct Dial Number 516/733-4698

MK II-084-LIC

July 8, 1981

Mr. Karl Kniel, Chief
Generic Issues Branch
Division of Safety Technology
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: ACRS Information Request
Mark II Pool Dynamic Loads

Dear Mr. Kniel:

The enclosed information is provided in response to your letter of May 20, 1981 requesting Mark II Owners Group comments on two questions raised by the ACRS at the April 28-29, 1981 Fluid Dynamics Subcommittee meeting. The ACRS questions relate to potential pool bypass from stuck open wetwell-to-drywell vacuum breakers and bypass through ruptured main vent downcomers. Our enclosed response addresses both the questions on main vent vacuum breaker performance and main vent downcomer fatigue evaluation from LOCA chugging events.

In addition, the Staff requested that we comment on containment fatigue analysis. The Mark II Owners felt further evaluation of fatigue on the containment is not necessary based on the following:

1. The low number of equivalent full strength stress cycles obtained for the pool loads using very conservative assumptions as compared to the allowable number of stress cycles before design stress reduction is required.

2. The stresses on the containment induced by chugging are relatively low and have no significant effect on fatigue.

3. The further reduction possible by taking advantage of a mass flux threshold (which means that chugging loads only occur over a fraction of the entire duration of the SBA).

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ATTACHMEN

We trust that the enclosed information has been responsive to your request.

Very truly yours,

H. Chau
Chairman
Mark II Containment Owners Group

HC/mvf

Enclosure

xc: Mark II Owners Group
Mr. W. M. Davis (GE)
Mr. C. Anderson (NRC)

bxc: D. J. Binder
B. R. McCaffrey
J. E. Metcalf (S&W)

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ACRS INFORMATION REQUEST
MARK II POOL DYNAMIC LOADS

Question 1

The Mark II containment wetwell/drywell vacuum breakers may be called upon to function repeatedly during intermittent steam condensation phenomena. Failure of a vacuum breaker to close during this time could result in pool bypass, thus jeopardizing the integrity of the containment. Provide valve service information for the range of wetwell/drywell vacuum breaker operation under pool dynamic loading. This information should be sufficient to describe the opening and closing characteristics of these valves during intermittent steam condensation.

Answer 1

The scenario postulated by the ACRS at the April 28 and 29, 1981 fluid dynamics subcommittee meeting requires the following occurrences:

(1) LOCA

(2) During chugging, steam bubble collapse causes underpressure in the vent of sufficient strength to open the check valves in the vacuum breaker assembly (i.e., cycling of the valve occurs).

(3) Cycling of the vacuum breaker continues for some period of time.

(4) A vacuum breaker assembly is damaged from this continued cycling and both swing check valves in the assembly fail in the open position.

(5) A small portion of the steam being discharged from the drywell to the suppression pool bypasses the pool and begins to increase overall containment pressure.

(6) If plant operator actions are not adequate to mitigate the pool bypass, steam flow continues so that containment integrity is jeopardized.

In order to comply with single failure criteria, each Mark II vacuum breaker assembly is comprised of two swing check valves in series. The normal position for both check valves is closed. Valve discs are held closed by spring actuated lever arms. Each valve would open under a negative differential pressure (drywell to wetwell)

Answer 1 (Cont'd.)

of less than 0.5 psid. The valves are self-actuated by the differential pressure across the valve ports. Closing of the valves, by gravity and action of the spring, occurs after relief of the negative pressure differential. Position indication is provided for each valve, utilizing contact probes on the valve disc base wired to indicator lights in the control room. Air cylinders are provided for remote operation of each valve from the control room. The vacuum breaker asembly is a safety-related component and, as such, is included in the on-going equipment requalification program required of all Licensees. This includes dynamic qualification to the building response from the dynamic Mark II loads, but does not include aerodynamic loading as is postulated by the given scenario.

Two of the Mark II plants (Zimmer and LaSalle) have vacuum breakers located in dedicated drywell floor or containment penetrations. As such, they do not experience the direct effects of downcomer steam condensation loading and are not expected to experience valve cycling.

Four of the Mark II plants (Susquehanna, Shoreham, Hanford and Limerick) have vacuum breakers located on their downcomer pipes. A check has been made with the valve manufacturer to determine if any information exists which would define the opening and closing characteristics of these valves during intermittent steam condensation. Such information does not presently exist. A program to determine the valve loading and valve capability during intermittent steam condensation is being developed. This issue will be addressed as part of the equipment qualification programs for these plants.

The remaining two plants (Nine Mile Point 2 and Bailly) are reviewing their designs to determine the best approach to preclude vacuum breaker cycling.

ACRS INFORMATION REQUEST
MARK II POOL DYNAMIC LOADS

Question 2

The steam chugging loads proposed as design loads by the Mark II Owners Group for containment evaluation were developed to represent limiting conditions during a postulated loss-of-coolant accident. As such, they are applied one time to the downcomers. The Mark II Owners have indicated that chugging induced fatigue loads are insignificant due to the large reduction in loads with time and the relatively low number of fatigue cycles. Additional information is needed to evaluate this position. Provide the following information for a typical plant. For a range of break sizes, estimate the number of "Pool Chug" cycles. This analysis should be extended sufficiently in time to include operation of the emergency core cooling system and the associated vessel steaming rates. If there exists a vent flow threshold below which the chugging loads are insignificant, this should be clearly identified in the analysis. The information should be provided for both the chugging downcomer lateral loads and the pool chugging loads.

Answer 2

The Mark II Owners Group has been addressing the concern of fatigue on the Downcomers and Main Steam vent discharge lines (SRVDLs). Recently, particular attention has been directed at the fatigue effect of chugging related loads.

A full ASME Section III Class 1 fatigue evaluation is performed on the downcomers and the SRVDLs. All significant loading conditions including pool dynamic loads (chugging, CO, lateral loads, SRV, etc.) seismic, pressure, thermal expansion and thermal transient stresses are considered. Load combinations for IBA, SBA, DBA, and normal/upset plant operations are evaluated based on the events conservative defined in the DFFR, Rev. 3 charts.

Results of extensive tests have shown that all chugs do not occur at a maximum amplitude over the entire chugging duration; hence, a procedure was developed for fatigue evaluation in which an equivalent number of full amplitude load cycles is obtained and used to determin an equivalent number of stress cycles. This is similar to the procedure used in Appendix N of ASME BPVC Section III to determine the equivalent number of full strength stress reversal cycles per earthquake. This equivalent number, which would give the same fatigue effect as the actual chugs with varying pressures, can be obtained from the "Equivalent Occurrence Factor" (EOF).

Answer 2 (Cont'd.)

$$EOF = \frac{\sum_i [N_i * (\frac{P_i}{P_{max}})^{4.3}]}{\sum_i N_i}$$

where: P_i = Individual chug peak overpressure

N_i = Number of occurrences at pressure P_i

P_{max} = Maximum peak over pressure

4.3 = Fatigue exponent given in Appendix II of ASME BPVC Section III

Furthermore, preliminary investigations on a spectrum of break sizes have indicated the existence of low mass flux threshold between 0.2 and 0.3 lbm/ft^2-sec below which the chugging downcomer lateral loads and the pool chugging loads are insignificant. As a consequence, the effective chugging load cycles will occur for only a fraction of their recommended DFFR duration. Typical results of analysis on a representative plant for a range of small break sizes have shown that significant chugging will occur for less than one-quarter of the entire specified SBA duration.

Nevertheless, assuming no low mass flux threshold for the chugging load distribution data considered,

$$EOF = 0.02$$

hence, the equivalent number of full load chugs for the SBA is 200.

In a similar fashion, operating with a family of typical loading time histories, the equivalent number of stress cycles per chugging event can be determined. This can be done using either the actual systems response or, conservatively, from operating on a family of single degree of freedom oscillators. This was done for a sample chug and the number of equivalent stress cycles per chug was found to be 3. Based on this, the total number of chugging stress cycles is 600.

Similar calculations were performed to assess the fatigue impact of the other significant pool related loads. For the case of lateral loads, the EOF was determined to be 0.04. This resulted in a reduction similar to that found for the chugging load.

Answer 2 (Cont'd.)

An earlier sample computation to assess the general impact of this approach for a typical Mark II Plant was performed using conservative and arbitrarily assumed values for full amplitude chug cycles and stress cycles of 1000 and 7000, respectively. These numbers of cycles are conservative when compared to expected values by more than an order of the magnitude. For this sample computation the fatigue usage factor on the main vent is less than 0.55 (including lateral load fatigue effects) and on the SRVDL was less than 0.45.

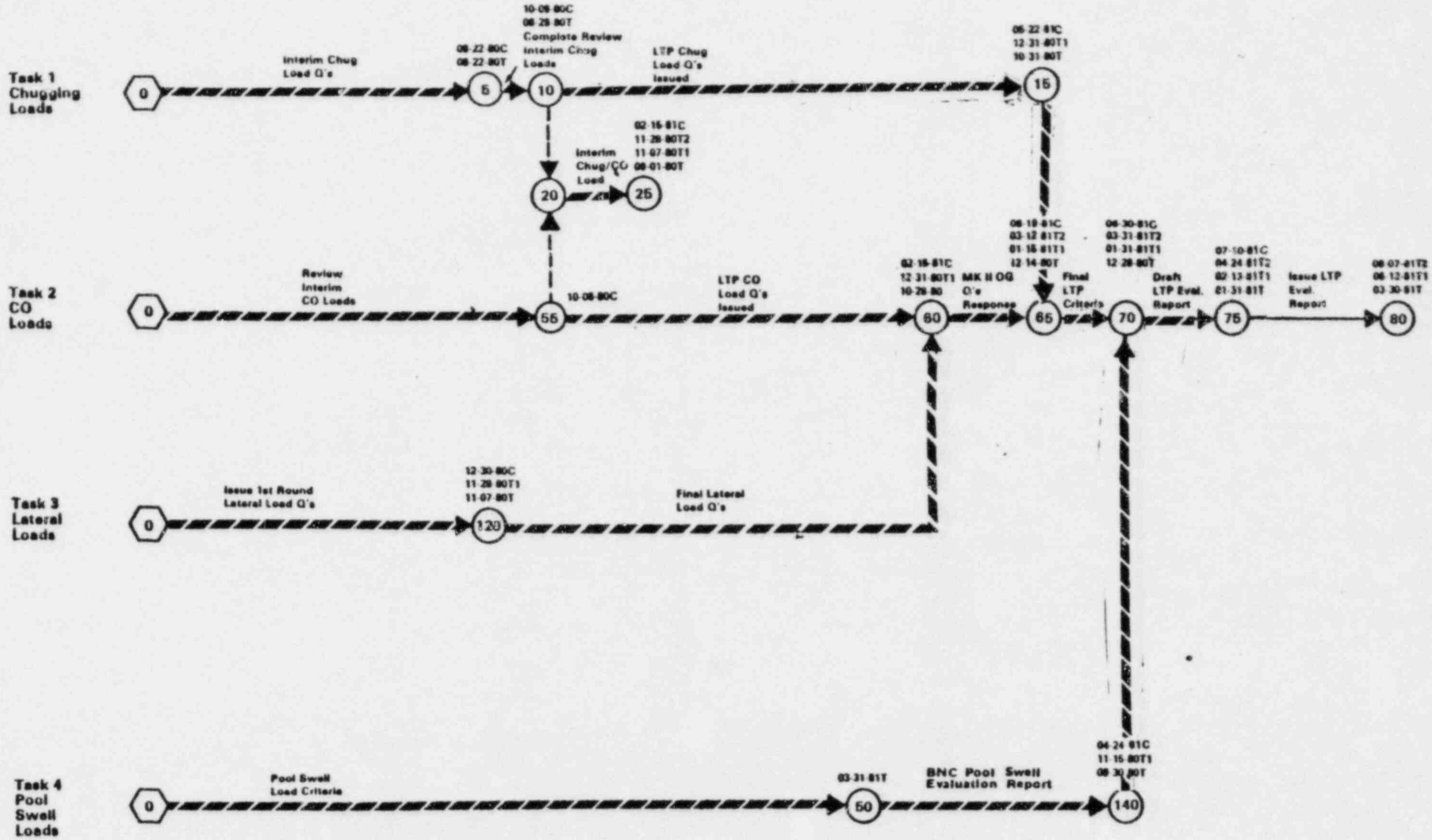
Even with a conservative load amplitude from tests at higher mass fluxes used in combination with a very conservative number of chugging cycles for fatigue evaluation, the fatigue usage factors are well within the acceptable limits.

SIGNIFICANT A-8 MARK II POOL DYNAMIC LOADS PROGRAM MILESTONES

05/75	Formation of Mark II Owners Group
10/78	NRC Issued NUREG-0487, Mark II Lead Plant Loads
09/80	NRC Issued NUREG-0487, Supplement 1, Alternate Lead Plant Loads
02/81	NRC Issued NUREG-0487, Supplement 2, Interim CO/Chugging Loads - Lead Plants
04/81	ACRS Fluid Dynamics Subcommittee Meeting, Mark II Long Term Program Loads
08/81	Scheduled Release of NUREG-0808, Mark II Long Term Program Loads

-- End USI A-8 Program --

MARK II LONG TERM PROGRAM
REVISED SCHEDULE (A-8)



A-256

MARK II POOL DYNAMIC LOADS SUPPORTING TEST PROGRAMS

LEAD PLANT PROGRAM

- 4T FULL SCALE, SINGLE CELL, STEAM, 46 TESTS
  (POOL SWELL AND STEAM LOADS)
- MKIII PSTF .6 SCALE, SINGLE CELL, AIR
  (SMALL STRUCTURE IMPACT LOADS)
- GKM FULL SCALE, SINGLE CELL, STEAM
  (VENT LATERAL LOADS)
- EPRI 1/13 SCALE, 90° SECTOR, AIR
  (3D POOL SWELL LOADS)

MARK II POOL DYNAMIC LOADS SUPPORTING TEST PROGRAMS

LONG TERM PROGRAM

- 4TCO Full Scale, Single Cell, Steam, 28 Tests
  (CO & Chugging Loads, Diaphragm Reverse Pressure Load)

- GKM II M Full Scale, Single Cell, Steam, 22 Tests
  (CO & Chugging Loads)

- JAERI Full Scale, 20^{o} Sector, Steam 18+ Tests
  (Multivent Steam Loads)

- Karlstein 1/2-Scale, 1-6 Vents, Steam
  (Multivent Lateral Loads)

- GKM II Full Scale, Single Cell, Steam, 57 Tests
  (Single Vent Lateral Loads)

- Creare 1/16, 1/10, 1/6, 1/4, 5/12 Scale, 1-19 Vents, Steam,
  750 Tests
  (Steam Loads)

MARK II LONG TERM PROGRAM LOADS SUMMARY
POOL SWELL LOADS

- Vent Clearing Submerged Boundary Loads
- Air Bubble Pressure
- Pool Velocity
- Pool Acceleration
- Pool Elevation
- Diaphragm Floor Reverse Pressure
  (New Load 5.5 psid)
- Wetwell Air Space Pressure

MARK II LONG TERM PROGRAM LOADS SUMMARY
STEAM CONDENSATION AND CHUGGING LOADS

Downcomer Lateral Load - Dynamic Loads

- Single Vent
  (Half Sine Wave, A=65 klbf, τ= 3 ms)

- Multiple Vent
  Based on Single Vent Load Multivent Reduction Factor

Submerged Boundary Load

- Condensation Oscillation Loads
  (Bounding 4TCO Pressure Histories)

- Chugging Loads
  (10 4TCO Derived Sources, 50 ms Source Desynchronization, IWEGS/MARS Acoustic Model of Plant)

MARK II LEAD PLANT
LATERAL LOAD SPECIFICATION

SINGLE VENT LOAD

8.8 KLBF - STATIC EQUIVALENT LOAD
(GKM & 4T DATA)

MULTIVENT LOAD

8.8 KLBF x M - STATIC EQUIVALENT LOAD
M (# VENTS) - MAGNITUDE/DIRECTION RANDOM
(GKM & KARLSTEIN DATA)

MARK II LONG TERM PROGRAM
LATERAL LOAD SPECIFICATIONS

SINGLE VENT LOAD - DYNAMIC LOAD

- IMPULSIVE LOAD
  - HALF SINE WAVE
  - $A = 10 \rightarrow 30$ KLBF
  - $\tau = 6 \rightarrow 3$ MS
  - APPLICATION $1 \rightarrow 4$ FT VENT END

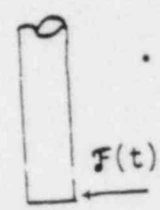
- FATIGUE LOAD
  - SINGLE VENT IMPULSIVE LOAD (30 KLBF, 3 MS)
  - ~ 400 LOAD CYCLES, ~800 STRESS CYCLES

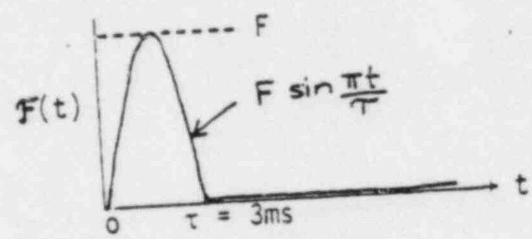
MULTIVENT LOAD - DYNAMIC LOAD

- $F = MA \sin(\pi T/\tau)$
- $A = (50 - 20\ \tau/3)$ KLBF FOR $3 \leq \tau \leq 6$ MS
- M (# VENTS) - MAGNITUDE/DIRECTION RANDOM INPHASE APPLICATION

DEFINITION OF TIP DYNAMIC LOAD F

WHICH APPEARS IN WHAT FOLLOWS:





MARK II OWNERS GROUP

SINGLE VENT LATERAL LOAD BASIS

-- "Comfortable Bound" of 4T Data

-- "Bound" of Related Foreign Test Data

PRELIMINARY STAFF EVALUATION

MARK II OWNER GROUP SINGLE VENT LOAD

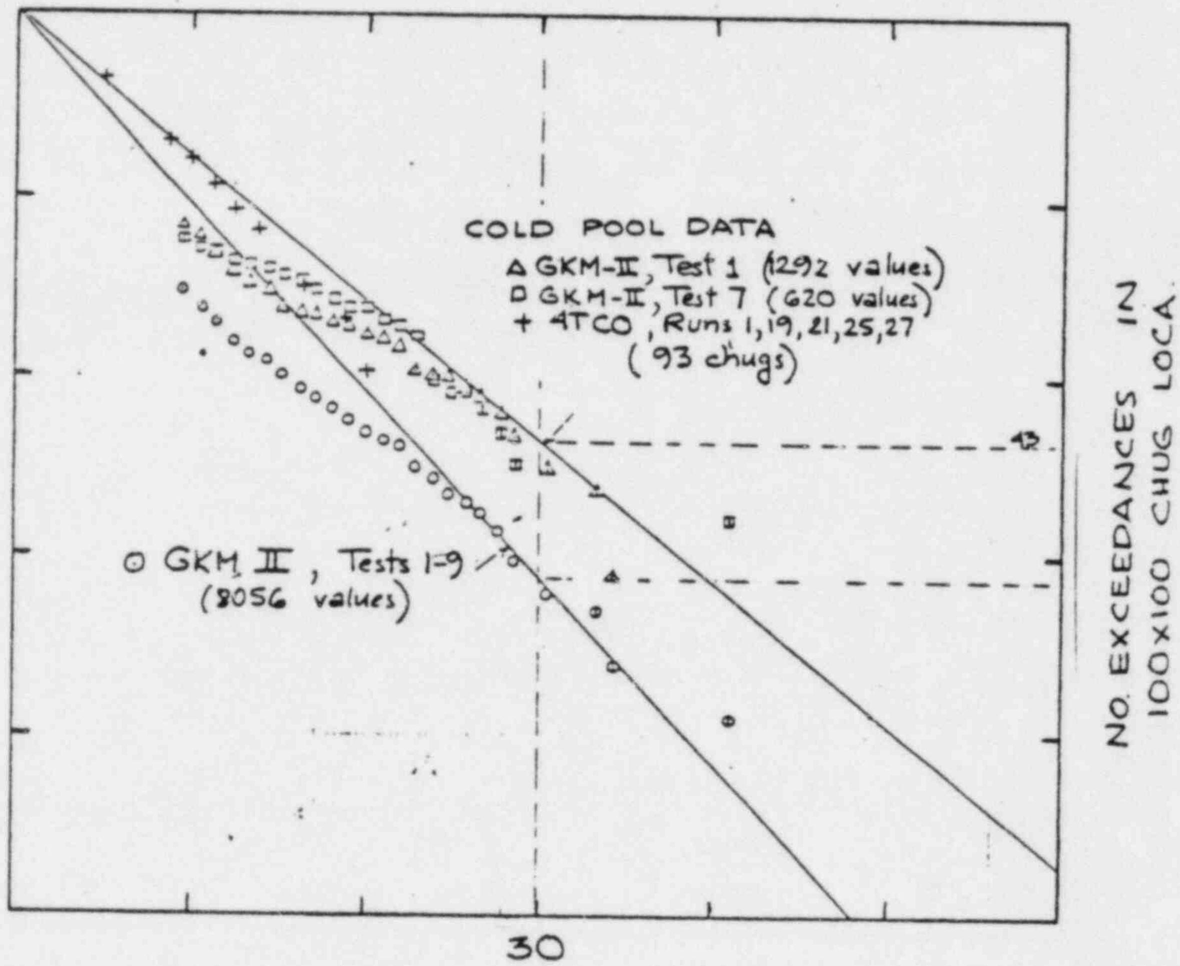
-- 30 KLBF "BOUND" NOT COMFORTABLE FOR ALL RELEVANT DATA

-- APPLICABLE FOREIGN DATA EXCEED THE 30 KLBF "BOUND"

-- 4T DATA BASE (600 CHUGS) SMALL COMPARED TO ANTICIPATED LOCA CHUGS (100 CHUGS x 100 VENTS)

-- 4T QUESTIONABLE DATA BASE

-- STATISTICAL EVALUATION OF RELATED DATA REQUIRED (GKM-II, 4TCO & GKMIIM)



TIP DYNAMIC LOAD AMPLITUDE
F (klbf) @ 3ms.

RELATION BETWEEN LOAD AND NO. OF EXCEEDANCES IN 100x100 LOCA.

NO. EXCEEDANCES, N	LOAD, $\bar{F}$, klbf	
	"COLD POOL DATA"	GMK-II, TESTS 1-9
10		
1	extrapolation	
10^{-1}	63	extrapolation
10^{-2}		

NRC CONCLUSIONS
MARK II SINGLE VENT
LATERAL LOAD SPECIFICATION

-- The Proposed SV Load (30klbf, 3ms) is Unacceptable (>100 Exceedances - 250 Chug x 100 Vent LOCA)

-- The Load Should be Based on Limiting Cold Pool - Low Mass Flux Data Subset from Applicable Data Base

-- The Proposed Load Should be Amended to Include a High Intensity Lateral Load with Pexc/LOCA < 1.0

-- NRC Acceptance Criterion
A = 65klbf, τ = 3ms
GKM-II Data Base
  57 Tests Total
    9 Tests (Detailed Data)
    2 Tests (Cold Pool, Low Mass Flux)
Pexc/LOCA ~0.1 Cold Pool Data
  ~0.01 9 Test Data

SINGLE VENT LATERAL LOAD CONSERVATISMS

-- HIGH AMPLITUDE, SHORT PERIOD LOAD

LATERAL LOAD IS IMPULSIVE
$\tau <$ 3 MS FOR A = 65 KLBF

-- LOAD COMBINATION

PEXC/LOCA < 0.1 FOR
LATERAL LOAD
+ SUMBERGED STRUCTURE DRAG-CHUG LOAD
+ SRV
+ SEISMIC

-- LOW PROBABILITY EXTRAPOLATION

A = 65 KLBF FOR EXPONENTIAL DISTRIBUTION (GKM-II)
NORMAL OR LOG-NORMAL PROVIDE GOOD FIT FOR OTHER DATA

The following pages A-270 thru 271 has been deleted containing proprietary information.

DELETION 3

FATIGUE EVALUATION OF SRV AND DOWNCOMER LINES IN THE WETWELL

- EVALUATION RESULTED FROM NRC/MEB CONCERNS EXPRESSED IN AUGUST 1979
- EVALUATION ALSO ADDRESSES RECENT ACRS QUESTIONS

BASIS FOR FATIGUE EVALUATION

- ASME CLASS 1 RULES USED FOR EVALUATION. (ASME BPVC SECTION III, NB-3600)

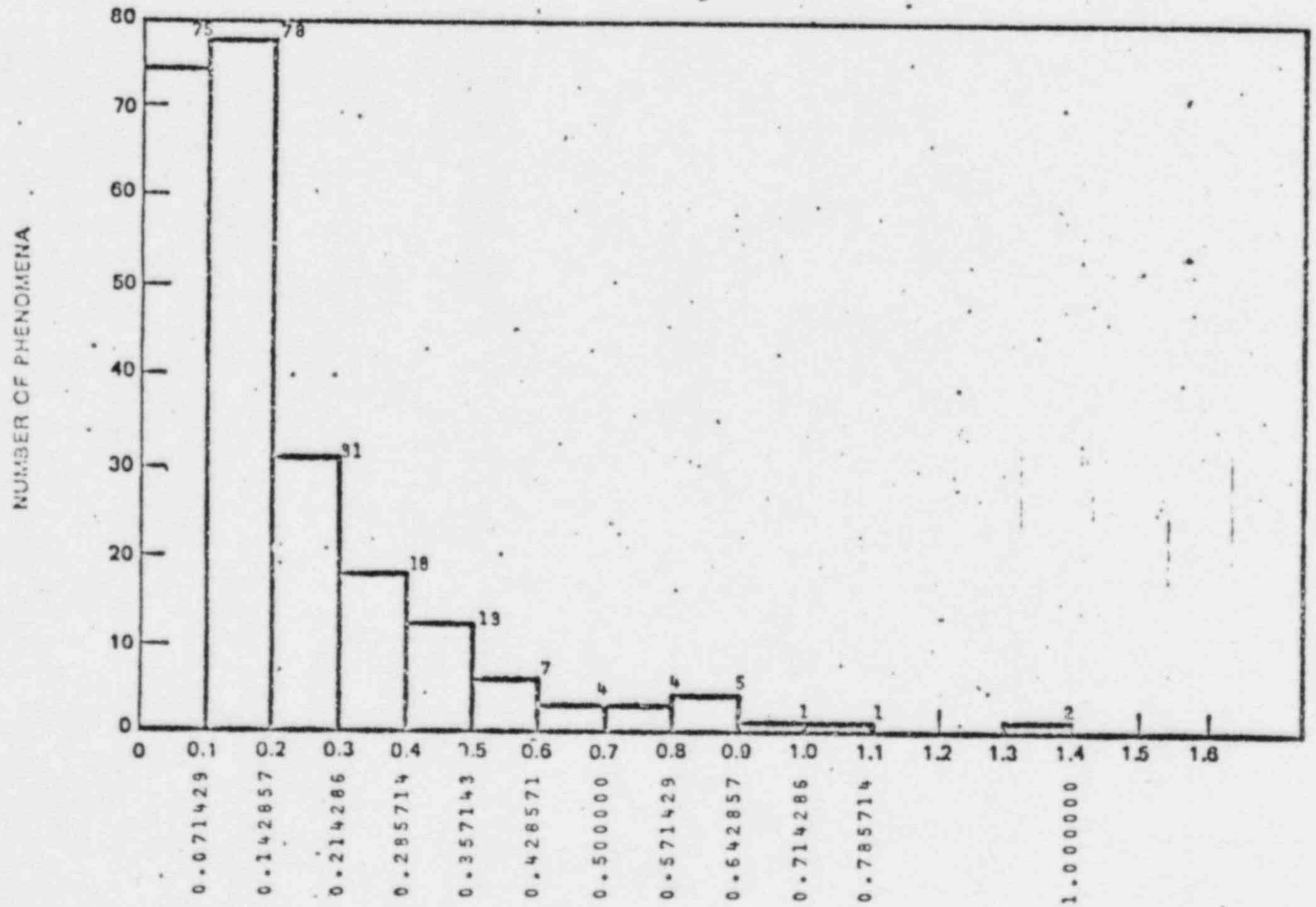
- EVALUATION INCLUDED:
  - WEIGHT
  - PRESSURE
  - SEISMIC
  - SRV POOL LOADS & BUILDING RESPONSE
  - CO POOL LOADS & BUILDING RESPONSE
  - CHUGGING POOL LOADS & BUILDING RESPONSE
  - DOWNCOMER LATERAL LOADS

- ALL SIGNIFICANT THERMAL TRANSIENT STRESSES ARE INCLUDED.

EVALUATION TECHNIQUES

- MORE ACCURATE LOADING SCENARIOS (BASED ON DFFR BAR CHARTS) WERE DEVELOPED
- TIME PHASING OF LOADS/RESPONSES WAS CONSIDERED
- EQUIVALENT NUMBER OF FATIGUE CYCLES WERE DETERMINED FOR DYNAMIC LOADS
- ACTUAL PRESSURES AND TEMPERATURES WERE USED

$$EOF = \frac{\sum_i \left[N_i * \left(\frac{P_i}{P_{max}} \right) \right]}{\sum_i N_i} \qquad 4.3$$



Frequency Distribution of Pressure Amplitudes

TYPICAL NUMBER OF STRESS CYCLES

EVENT	EOF	STRESS CYCLES/OCCURRENCES (1)	TOTAL STRESS CYCLES (1)
CHUGGING	0.02	3	600 (2)
LATERAL LOADS	0.04	2	800 (2)
CONDENSATION OSCILLATION	1.0	5	175

(1) BOTH THE STRESS CYCLES PER OCCURRENCE AND THE TOTAL NUMBER OF STRESS CYCLES ARE TYPICAL VALUES.

(2) THESE NUMBERS ARE CONSERVATIVE AND WOULD BE REDUCED BY AS MUCH AS A FACTOR OF 10 BY CONSIDERING A MASS FLUX THRESHOLD.

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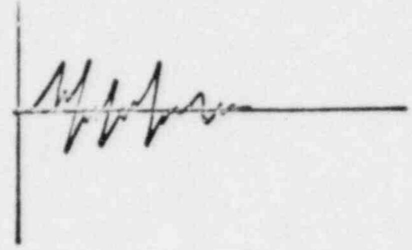
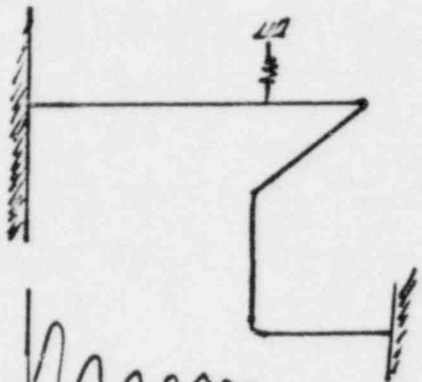
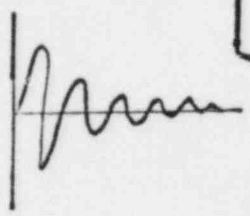
DETERMINATION OF EQUIVALENT STRESS CYCLES



LOAD TIME HISTORY



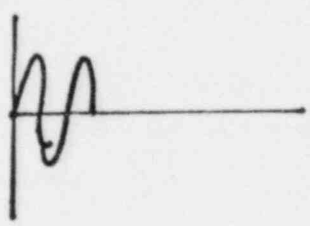
APPLY TO MODEL



RESPONSE TIME HISTORY

$$EOF = \frac{\sum N_i \left(\frac{\sigma_i}{\sigma_{max}}\right)^{4.3}}{\sum N_i}$$

CALCULATE STRESS CYCLE EOF



EQUIVALENT STRESS CYCLES WITH SAME FATIGUE DAMAGE

SBA LOAD COMBINATION

$$SSE + SRV_{ARS} + SRV_{BUBBLE} + CHUG_{ARS} + CHUG_{BUBBLE} + T + \Delta T$$

SSE : SAFE SHUTDOWN EARTHQUAKE

SRV : SAFETY RELIEF VALVE EFFECTS

(ARS : AMPLIFIED RESPONSE SPECTRA)
(BUBBLE : SUBMERGED STRUCTURE LOADS)

CHUG : CHUGGING LOADS

T : THERMAL EXPANSION & ANCHOR MOVEMENT

ΔT : THERMAL TRANSIENT

SUMMARY OF RESULTS

- BASED ON A PRELIMINARY, CONSERVATIVE NUMBER OF CHUGGING STRESS CYCLES (7000 vs. 600) IN ADDITION TO ALL OTHER LOADS ACCEPTABLE USAGE FACTORS WERE OBTAINED

  DOWNCOMER $u = 0.55 < 1.0$

- THESE COULD HAVE BEEN REDUCED EVEN FURTHER BY USING MORE ACCURATE DEFINITIONS OF THE NUMBER OF STRESS CYCLES AND BY FACTORING IN THE MASS FLUX THRESHOLD.

- INDICATES EXISTING DESIGN PROCEDURES ARE GOOD.

MARK II WETWELL TO DRYWELL VACUUM BREAKER CYCLING DURING CHUGGING

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- ACCIDENT SCENARIO
- VACUUM BREAKER FUNCTION
- VACUUM BREAKER DESIGN AND LOCATION
- PLANT OPERATIONS FOR NORMAL AND ACCIDENT CONDITIONS
- VACUUM BREAKER QUALIFICATION
- CONCLUSIONS

DMO 8/07/81

ACCIDENT SCENARIO

- LOCA EVENT OCCURS
- CHUGGING PHASE OF BLOWDOWN IS REACHED
- PRESSURE OSCILLATIONS IN DOWNCOME CAUSE VACUUM BREAKERS TO OPEN AND CLOSE
- CYCLE CONTINUES FOR DURATION OF CHUGGING
- SBA EVENT IS CONTROLLING

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VACUUM BREAKER FUNCTION

- EQUALIZE POST LOCA WETWELL AND DRYWELL PRESSURE
- DRYWELL REDUCED PRESSURE CAUSED BY
  - ECCS FLOW OUT BREAK
  - DRYWELL SPRAY

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- WHY ARE WE CONCERNED WITH BREAKERS FAILING OPEN FROM CYCLING DUE TO CHUGGING?
- PRESSURE SUPPRESSION SYSTEM MAY BE BYPASSED BY BLOWDOWN STEAM
- CONTAINMENT OVERPRESSURIZATION MAY RESULT

VACUUM BREAKER DESIGN

- SIMPLE SWING CHECK VALVE TYPE
- TWO VALVES MOUNTED IN SERIES FOR SINGLE FAILURE PROTECTION
- SET PRESSURE MAINTAINED TO PREVENT ACTUATION DURING NORMAL PRESSURE FLUCTUATION (APPROX. 0.5 PSID)
- SIZED FOR FLOW CAPACITY NECESSARY TO MAINTAIN WETWELL TO DRYWELL ΔP ACROSS DIAPHRAGM SLAB BELOW DESIGN ALLOWABLE
- REDUNDANT VALVE DISC POSITION INDICATION
- PNEUMATIC ACTUATORS FOR REMOTE MANUAL ACTUATION FOR TEST PURPOSES

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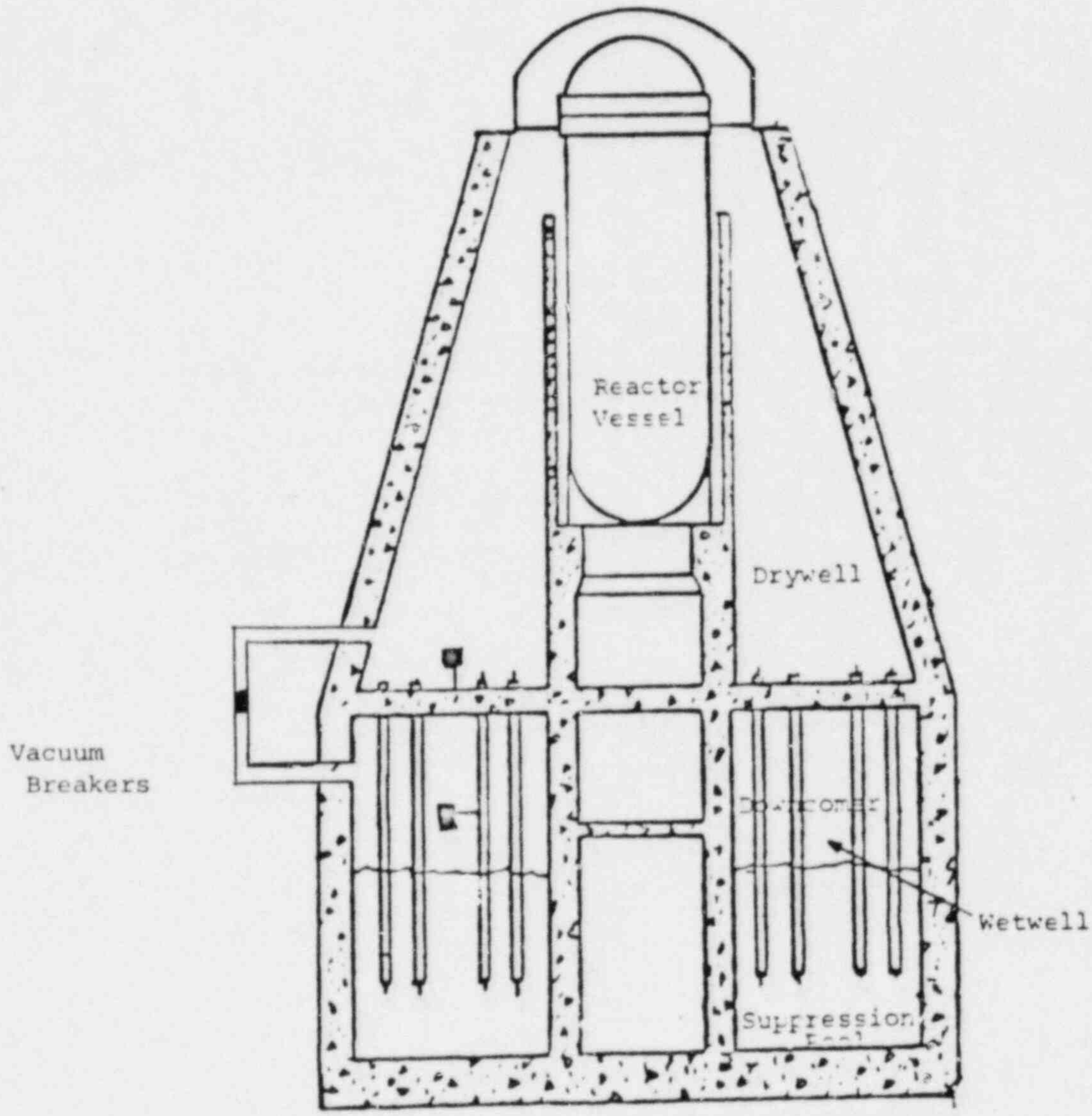
MARK II WETWELL TO DRYWELL
VACUUM BREAKER LOCATION



PLANT OPERATIONS FOR NORMAL AND ACCIDENT CONDITIONS

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- NORMAL OPERATION
  - POSITION INDICATION ALARMS IN CONTROL ROOM
  - INDICATOR LIGHTS ON PANEL IN REACTOR ENCLOSURE FOR OPEN AND CLOSE POSITION
  - VALVES CYCLED MONTHLY BY HAND SWITCHES ON PANEL IN REACTOR ENCLOSURE FOR OPERABILITY ASSURANCE
  - TECH SPEC WILL REQUIRE OPERATOR ACTION TO CYCLE VALVE IF STUCK OPEN

- ACCIDENT CONDITIONS
  - EMERGENCY PROCEDURES WILL REQUIRE OPERATOR ACTION TO INITIATE DRYWELL AND WETWELL SPRAY TO REDUCE CONTAINMENT PRESSURE
  - EMERGENCY PROCEDURES WILL PROVIDE OPTION FOR OPERATOR ACTION TO INITIATE ADS TO REDUCE RPV PRESSURE

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VACUUM BREAKER QUALIFICATION

- DYNAMIC LOADS TO BE CONSIDERED
  - BUILDING RESPONSE
    - SEISMIC
    - SRV DISCHARGE
    - CO
    - CHUGGING
  - SUBMERGED STRUCTURE
    - SRV DISCHARGE
    - CO
    - CHUGGING
    - CHUGGING LATERAL LOADS
  - DYNAMIC CYCLIC LOADS
- DEFINITION OF CYCLIC LOADS
  - ANALYSIS
    - CALCULATE MAXIMUM IMPACT VELOCITY
    - CALCULATE COMPONENT STRESSES
  - EXPERIMENTAL RESULTS
    - VERIFY VALVE CAPABILITY TO WITHSTAND LOADS

CONCLUSIONS

- FOR VACUUM BREAKER CYCLING CONSIDERATIONS, SMALL BREAK ACCIDENT WOULD BE CONTROLLING DUE TO LONGER DURATION OF CHUGGING
- VACUUM BREAKERS LOCATED ON DOWNCOMERS ARE MORE SUSCEPTIBLE TO CYCLING DURING CHUGGING
- FOR VACUUM BREAKERS LOCATED ON DOWNCOMERS IT IS EXTREMELY UNLIKELY THAT TWO CHECK VALVES IN SERIES WILL FAIL OPEN FOR BYPASS LEAKAGE TO OCCUR
- IN THE EVENT OF BYPASS LEAKAGE, OPERATOR WILL TAKE ACTION TO REDUCE CONTAINMENT PRESSURE BY INITIATING DRYWELL AND WET-WELL SPRAY AND REDUCE RPV PRESSURE BY INITIATING ADS
- VACUUM BREAKERS WILL BE QUALIFIED FOR CYCLIC LOADS AND OTHER DYNAMIC LOADS

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

TAB 5.1

EXEMPTION (b) 5
8/6/81-GYoung/bgs

256TH ACRS MEETING
SUSQUEHANNA OPERATING LICENSE REVIEW
AUGUST 7, 1981

- PROJECT STATUS REPORT -

Purpose:

The purpose of the meeting is to review the application of the Pennsylvania Power & Light Co. for a license to operate the Susquehanna Steam Electric Station, Units 1 and 2. The Susquehanna Subcommittee met on July 23, 1981 to review the OL application. Mr. Ray toured the facility on July 2, 1981.

Background:

Pertinent facts concerning the Susquehanna Project include:

Location

The site for the station consists of 1522 acres on the west bank of the Susquehanna River in Salem Township, Luzerne County, Penn. It is located 12 miles northwest of Hazelton and 15 miles southwest of Wilkes-Barre, the nearest cities having populations in excess of 25,000.

Plant

The NSSS consists of a GE BWR/4 design housed in a Mark II type containment building. The design power level of the reactor is 3293 MWt and the design gross electric output is 1134 MWe. The attached Table 1.1 from the SER compares the principal design features of Susquehanna with similar facilities.

Previous ACRS Review:

The ACRS reviewed Susquehanna for a CP license in March and April 1972. A copy of the Committee's CP letter is attached. Susquehanna is similar in design to the LaSalle and Zimmer Plants which the ACRS reviewed for an OL in April 1981 and March 1979, respectively.

Subcommittee Review:

Minutes of the July 23, 1981 Subcommittee meeting are included in this Meeting Folder Section. The Subcommittee had no major reservations concerning the Project and NRC review.

Highlights of Review Topics:

The Tentative Schedule of Presentations for the meeting is attached. TMI-related topics will be the focus of the meeting: organization and management, operator training, control room design, and decay heat removal capability. Other topics/plant features of interest include:

Mark II Containment:

The Susquehanna Mark II containment has been the subject of considerable analysis and testing to assure that accident loads do not exceed design loads. This subject will be discussed as a generic item and then briefly the specifics of the Susquehanna review. It appears that Susquehanna has done an admirable job in regard to this subject.

Open Items:

The SSER lists 14 open items. All of these items will be reviewed at the meeting. The items are:

1. Turbine Missiles*
2. Environmental Qualification of Electrical Equipment
3. Steam Bypass of the Suppression Pool
4. Additional Justification Required for T-Quencher Loads*
5. Review of Submerged Dragloads
6. IE Bulletin 79-27 and 80-06**
7. Fire Review of Alternate Safe Shutdown System
8. Modification of Automatic Depressurization System Logic
9. Provide Common Reference Level for Vessel Level Instrumentation*
10. Upgrade Emergency Preparedness
11. Upgrade Emergency Support Facilities
12. Long-Term Emergency Support Facilities
13. Heavy Loads Generic Letter*
14. Scram Discharge Volume Generic Letter

* These items have been closed since the Subcommittee meeting.
** 80-06 has been closed since the Subcommittee meeting.

Other Issues:

The only area of contention between the NRC Staff and PP&L is over the requirement for incore thermocouples. This is not an open item because the NRC Staff plans to require incore thermocouples as a license condition. PP&L does not agree that the benefit of such a costly addition to the plant has been demonstrated and they are being placed in a difficult positon of being committed to a design feature which has not been snown to be acceptable from a safety viewpoint.

Attachments:
As stated

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 13, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

Dear Dr. Schlesinger:

At its 144th meeting, April 6-8, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application from the Pennsylvania Power and Light Company for a permit to construct the Susquehanna Steam Electric Station, Units 1 and 2. The project was previously considered at a Subcommittee meeting at the Station site on March 24, 1972. During the review the Committee had the benefit of discussions with representatives and consultants of the applicant, the General Electric Company, the Bechtel Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Susquehanna Station will be located in Pennsylvania on a 1522 acre site on the west bank of the Susquehanna River approximately 12 miles northwest of Hazleton and 15 miles southwest of Wilkes-Barre, the nearest cities having populations in excess of 25,000. The low population zone radius is 3.0 miles within which the 1970 population was about 2,400 and the projected 2020 population about 4,000. The exclusion zone has a minimum radius of 1,800 feet and is separated from the river on the east by U. S. Route 11 and a single-track line of the Erie-Lackawanna Railroad. The principal facilities are located approximately 3,000 feet from the bank of the river at a grade elevation of about 170 feet above the bank.

The Susquehanna Station will utilize two General Electric boiling water reactors, each to be operated at a power level of 3293 MWt with waste heat rejected to the atmosphere by two natural-draft cooling towers. The primary containment is of the over-under pressure suppression type similar to those previously reviewed for Zimmer, Limerick, and Shoreham. The reactors are of the 1967 General Electric product line and similar to those of other facilities now under construction, particularly Browns Ferry 1, 2, and 3 and Peach Bottom Units 2 and 3.

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The applicant does not currently own all portions of the proposed site south of the reactors and within the exclusion radius. Similarly, mineral rights within the exclusion radius are not yet owned by the applicant. Procedures are being initiated to obtain ownership of the needed properties, and the applicant has stated that no construction will begin until this has been accomplished.

The applicant's criteria for protecting low pressure piping from overpressure include interlocks to prevent residual heat removal (RHR) system valves from opening unless the reactor coolant system pressure is below the RHR system design pressure. Although the applicant will design these interlocks to meet the requirements of IEEE 279-1971, the Committee recommends that diverse pressure sensors also be employed to provide greater assurance of performance of this important function.

The Susquehanna Station is the second plant for which the relief valve augmented bypass (REVAB) system is proposed. This system allows a full-load rejection without a reactor scram even though the turbine bypass capacity is only 25% of full-power steam flow. REVAB utilizes rapid-response pressure relief valves discharging into the suppression pool and rapid reactor power reduction to avoid reaching scram setpoints. As this system provides an additional signal causing opening in the primary system coolant boundary, the Committee believes that attention should be given to the possibility of valves remaining open following REVAB action.

The Committee believes that the main steam lines up to and including the turbine stop valves, and all branch lines 2-1/2 inches and larger up to their first valve, should be dynamically analyzed to ensure structural integrity during a design basis earthquake. A sealing system designed to standards applicable to engineered safety features should be provided to minimize leakage through the main steam line isolation valves. These matters should be resolved in a manner satisfactory to the Regulatory Staff.

The applicant has studied design features to make tolerable the consequences of failure to scram during anticipated transients, and has concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Susquehanna reactors. However, further evaluation of the sufficiency of the approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the reactors.

The techniques for analysis of the control rod drop accident are being revised by the General Electric Company. The adequacy of the revised model and the acceptability of the results should be established in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed of the resolution of this matter.

Current analysis indicates acceptably low peak clad temperatures following a postulated loss-of-coolant accident. A research program, which was recently begun under the auspices of the General Electric Company and the USAEC, should provide more detailed knowledge of the flow and heat transfer processes during the first stages of such postulated accidents. More detailed analytical studies, particularly as they relate to the time to critical heat flux and the level swell process, should also be performed during construction of the plant. These studies should be reviewed by the Regulatory Staff. The Committee wishes to be kept informed.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Susquehanna Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the Susquehanna Steam Electric Station, Units 1 and 2, can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

C P Siess

C. P. Siess
Chairman

References

List Attached

References

1. Pennsylvania Power and Light Company letter dated 4/1/71 transmitting their Application for Licenses for the Susquehanna Steam Electric Station together with an Environmental Report and Vols. 1 through 6, Preliminary Safety Analysis Report

2. Amendments 1 and 3 through 7 to the Application

3. Pennsylvania Power and Light Company letter dated 4/3/72

TABLE 1.1

COMPARISON OF PRINCIPAL DESIGN FEATURES OF SUSQUEHANNA AND SIMILAR FACILITIES

Design Feature	Susquehanna	La Salle	Zimmer	Hatch Unit 2
Rated thermal power, megawatts	3293	3293	2436	2436
Gross electrical output, megawatts	1134	1122	883	822
Main steam flow rate, pounds per hour	13,480,000	14,166,000	10,477,000	10,470,000
Total reactor core flow rate, pounds per hour	100,000,000	106,500,000	78,500,000	77,000,000
System pressure, nominal in steam dome, pounds per square inches	1020	1020	1020	1020
Fuel lattice	8x8	8x8	8x8	8x8
Number of fuel assemblies	764	764	560	560
Number of fuel per fuel assembly	62	62	63	62
umber of control rods	185	185	137	137
Reactor vessel inside diameter, inches	251	251	218	218
Reactor vessel inside height, feet	72.9	72.9	69.3	69.3
Reactor vessel design pressure, pounds per square inch gauge	1250	1250	1250	1250
Reactor vessel wall thickness, inches	6.19	6.75	5.375	5.531
Number of recirculation loops	2	2	2	2

TABLE 1.1 (Continued)

Design Feature	Susquehanna	La Salle	Zimmer	Hatch Unit 2
Recirculation loop inside diameter, inches	28	24	20	28
Recirculation pump flow rate, gallons per minute	45,200	47,250	33,880	45,200
Number of jet pumps	20	20	20	20
Number of high pressure coolant injection pumps	1	1	1	1
Number of core spray loops	2	1	1	2
Number of low pressure coolant injection pumps	4	3	3	4
Maximum heat flux, British thermal units per square foot per hour	361,000	361,000	354,000	361,591
Average heat flux, British thermal units per square foot per hour	144,100	145,208	143,900	145,528
Maximum power per fuel rod length, kilowatts per foot	13.4	13.4	13.4	13.4
Maximum centerline fuel temperature, degrees Fahrenheit	3435	3325	3325	3435
Minimum critical power ratio	1.23	1.24	1.21	1.30
Total peaking factor	2.51	2.25	2.43	2.49

DATE ISSUED: 7/30/81

MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON SUSQUEHANNA
JULY 23, 1981
WASHINGTON, D.C.

The ACRS Subcommittee on Susquehanna held a meeting on July 23, 1981 in Room 1046, 1717 H St., N.W., Washington, D.C. The purpose of the meeting was to review and discuss the Pennsylvania Power and Light (PP&L) Company's request for an operating license and to determine if the license application was complete enough to be brought before the ACRS during the August Committee Meedting. Most of the meeting was open to the public except for a brief discussion of the security system at Susquehanna. Notice of this meeting was published in the Federal Register on Friday, July 17, 1981. A copy of this notice is included as Attachment A. A list of attendees for this meeting is included as Attachment B, the schedule for the meeting is included as Attachment C, and a list of reference material is included as Attachment D. A complete set of handouts has been included in the ACRS files. There were not written or oral statements from the public. The Designated Federal Employee for the meeting was Mr. John C. McKinley.

NRC Staff Status Report on OL Review

Mr. R. Stark, NRC Project Manager, presented the status and current schedule for completion of the 14 open items in the Susquehanna SER. Some of the 14 open items are likely to be resolved before the August

ACRS meeting. There are no major disagreements between the applicant and the NRC Staff over the resolution of any of the currently identified open items. However, the NRC plans to require the addition of incore thermocouples as part of the operating license and the licensee does not currently agree that the thermocouples are necessary for detection of inadequate core cooling. The thermocouple issue is still being discussed.

PP&L Presentation on Construction Schedule and OL Application

Mr. N. Curtis, Vice President of Pennsylvania Power & Light (PP&L) for Nuclear Engineering and Construction, presented the history and schedule for completion of Susquehanna. He indicated that the target construction completion date for Unit 1 is January 1982 and that construction is presently running six weeks behind schedule. The currently expected fuel load date is April 1982.

Mr. P. Hendrikson, PP&L Manager of Licensing, presented the status of the 14 open items from the licensee's viewpoint. He indicated that PP&L was working with the NRC on each of these items and that no significant disagreements currently exist in resolving any of these open items. The resolutions primarily involve documenting commitments and completing some analysis work.

PP&L Presentation on Management Structure and Technical Resources

Mr. B. Kenyon, PP&L Vice President of Nuclear Operations, presented the management structure and organization of the nuclear department at PP&L.

In response to a question, Mr. Kenyon stated that the President of PP&L has a master's degree in mechanical engineering, a master's degree in business administration, and a doctorate in jurisprudence. Mr. Kenyon indicated that much of the management structure recommended by NRC, following the accident at TMI, was already in place at PP&L prior to TMI. Currently, PP&L has three former plant superintendents in the nuclear department management structure. The nuclear department contains 732 personnel as of May 1981 with 395 personnel making up the plant staff. By the end of 1982, it is projected that a total of 881 personnel will be in the nuclear department of which 531 will be on the plant staff.

Dr. I. Catton asked if the Training Manager were on the Safety Review Board. Mr. Kenyon indicated that he was not but that PP&L was currently reconsidering whether the Training Manager should be on the Safety Review Board.

Dr. W. Kerr asked if a QA program exists for the fossil plants at PP&L. Mr. Curtis stated that no such program exists since the last fossil unit was completed in the early 1970's. In response to further questioning, Mr. Curtis stated that he could see no benefit if a QA program were added to a future fossil plant construction project.

Dr. D. Moeller questioned the 737 person-rem annual collective dose estimate for each unit at Susquehanna. He indicated that newer plants should be able to do better than older operating plants. Mr. S. Cantone, PP&L

Manager of Nuclear Support, responded that the maintaining of annual collective doses at the level of older plants was actually an improvement since new plants are required to do considerably more work in contaminated areas than older plants (e.g., Inservice Inspection, Inservice Testing, etc.

Mr. H. Keiser, PP&L Plant Superintendent, described the plant organization at Susquehanna. He stated that the security force was made up of PP&L personnel and 73% of the security force personnel had college degrees (most of the degrees were in the area of criminology). Dr. Kerr asked Mr. Keiser if he felt the Shift Technical Adviser (STA) was a useful position at Susquehanna. Mr. Keiser stated that, in his opinion, the STA was a useful position because only experienced, knowledgeable people are chosen as STAs. Furthermore, the STA is given a routine function in which the Shift Supervisor seeks assistance from him. Therefore, during an emergency the Shift Supervisor will naturally turn to the STA for advice and assistance.

PP&L Presentation on Training

Mr. G. Ward, PP&L Manager of Nuclear Training, presented the training philosophy and organization at Susquehanna. The training department is independent of the plant staff and reports directly to the Vice President of Nuclear Operations. The training department is responsible for training the licensed operators, non-licensed operators, mechanics, instrument technicians, health physicists, chemists, engineers, and subgroups of each of these job categories. The training is tailored to each subgroup and the course curriculum is based on the requirements identified by an organized and specialized committee. In response to several questions regarding licensed operator training, it was determined that the operators are taught to respond intelligently to unusual situations rather than relying completely on procedures

Mr. Keiser added to Mr. Ward's presentation by discussing the curriculum committee organization. He then discussed the Susquehanna simulator which is located at the plant site. The simulator has been in use now for approximately two years. It has been used to checkout procedures, uncover plant design problems, train the operators, and for other similar projects. Mr. Keiser then discussed some details concerning the operators training programs. Dr. Kerr asked how PP&L kept people interested with all the training and retraining required to be an operator. Mr. Kenyon responded to that question by saying that some of the operators are becoming a little fatigued by all the training and they are anxious to get on with the actual operation of the plant.

PP&L Presentation on Control Room and Remote Shutdown Panel Design

Mr. T. Crimmins, PP&L Manager of Nuclear Plant Engineering, briefly discussed the remote shutdown panel design at Susquehanna and the proposed resolution of this open item. He indicated that PP&L was proposing some permanent jumpering of interlocks, carefully controlled by procedures, to satisfy NRC concerns about diversity in the remote shutdown panel design. He stated that this design appears to be acceptable to the NRC. Mr. Stark responded that the NRC Staff is still reviewing this item.

Mr. S. Cantone, PP&L Manager of Nuclear Support, described the Advanced Control Room (ACR) design at Susquehanna. He explained the use of cathode ray tubes (CRTs), advanced graphics, and alphanumerics in the ACR which provides more information in a more easily accepted format for the operator. Additionally, all of the computer supplied information is available on hardwired meters in the control room in case of multiple computer failures. The post-TMI human factors review of the control room resulted in very few changes since a human factors approach to the design of the ACR was taken at Susquehanna in 1974.

Mr. T. Crimmins explained the status of PP&L compliance with Reg. Guide 1.97 and Inadequate Core Cooling (ICC) instrumentation requirements. He noted that 93% of the Reg. Guide 1.97 required variables are already measured with existing instrumentation. Approximately 55% of the required variables will need upgrading to completely comply with the Reg. Guide. On the subject of ICC instrumentation, Mr. Crimmins noted that the NRC has indicated that BWR incore thermocouples will be a license requirement for Susquehanna. PP&L finds this a difficult situation since they have not determined the acceptability of such a design feature. Mr. L. Phillips of the NRC Staff commented that incore themocouples are required by Reg. Guide 1.97.

PP&L Presentation on Emergency Planning

Mr. S. Cantone presented the PP&L program for emergency planning. He stated that a drill to test the plan is scheduled for mid-March 1982. Dr. Moeller asked if the warning system was seismically designed. Mr. Cantone replied that it was not.

PP&L Presentation on Station Electrical Power

Mr. N. Curtis, Mr. D. Cole, and Mr. H. Keiser of PP&L described the station electrical system. Mr. Ray asked if Susquehanna had priority for restoration of power from the system dispatcher. Mr. Curtis replied that Susquehanna has top priority due to its special needs relative to the fossil plants on the system. Dr. Kerr questioned the reliability of the 250 V, two-battery system design for Susquehanna. PP&L representatives indicated that they believed the design was reliable enough although they could not define the reliability quantitatively.

PP&L Presentation on Decay Heat Removal Capability

Mr. H. Keiser described the normal and degraded modes of decay heat removal. In response to some questions, he indicated that if all AC and DC power were lost, the decay heat removal systems would be unavailable since the steam driven pumps are controlled by DC powered valves and control systems.

PP&L Presentation on Environmental Qualification of Equipment

Mr. T. Crimmins discussed the equipment environmental qualification program. He stated that approximately 25% of Class 1E equipment has complete documentation of qualification. The remaining equipment is still being qualified by a documentation search, testing program, or replacement if necessary. The goal for completion of this program is June 1982 but difficulties in meeting this goal exist. Specifically, the NRC has continuously changed the scope and content of the program, the instrumentation vendors have not been especially responsive to requests for additional documentation, and test facilities are limited.

PP&L Presentation on Spent Fuel and Low-Level Waste Storage

Mr. H. Keiser presented the spent fuel and low-level waste storage capability for Susquehanna. He stated that each unit has spent fuel storage capacity for 10 years of operation assuming 12 month refueling cycles and maintaining complete off-loading capability for the reactor. Due to the uncertainity of shipping low-level wastes to burial sites, PP&L is planning to build an onsite storage building to handle up to eight years of low-level waste. Dr. Moeller asked what PP&L was doing to reduce the volume of low-level waste. Mr. Keiser responded that they had no plans for waste incinerators, however, they are working to reduce the waste volume by proper operation of equipment and design of systems.

PP&L Presentation on the Susquehanna Scram System

Mr. T. Crimmins discussed the BWR scram system concern identified by the NRC Office of Analysis and Evaluation of Operational Data (AEOD) and how PP&L has responded to that concern. Basically Mr. Crimmins pointed out that the AEOD conern was over a low-probability event at a BWR Mk. I designed plant (i.e. Browns Ferry). The Susquehanna BWR Mk. II design has significant improvements over the Mk. I in the area of better ECCS pump room isolation and flood protection as well as higher sump pump flow rates. Therefore, the AEOD concern is less of a problem for Susquehanna than it was for Browns Ferry. Dr. Catton asked if the Scram Discharge Volume (SDV) was part of the primary pressure boundary in consideration of material fracture toughness. PP&L representatives responded that the SDV did not meet the fracture toughness requirements of the primary pressure boundary.

PP&L Presentation on ATWS

Mr. T. Crimmins discussed the ATWS issue as it relates to Susquehanna. Dr. Catton and Dr. Lipinski asked if consideration was being given to neutron stability analysis during an ATWS event. Mr. Crimmins stated that such an analysis is being done relative to the ATWS issue. All other ATWS issues are being actively addressed by PP&L for Susquehanna.

PP&L Presentation on the Mk. II Containment Design

Mr. D. Roth of PP&L discussed the Mk. II containment program at the Susquehanna. They have gone further in their analytical work than GE Mk. II Owner's Group. Dr. Catton indicated that he had been following the PP&L work and he believed they had done a good job.

Mr. H. Keiser briefly discussed the containment hydrogen control systems at Susquehanna. The systems discussed were the containment air mixing system, hydrogen/oxygen monitoring system, hydrogen recombiners, containment hydrogen purge system, and the nitrogen inerting system.

PP&L Presentation on the Security Program

The meeting went into closed session to discuss the security system at Susquehanna. Mr. C. Sprunk of PP&L described some of the security features. The value and application of psychological tests were briefly discussed.

Subcommittee Conclusion and Future Meetings

The Subcommittee concluded that, based on the information they had heard, the Susquehanna application for an operating license was ready to be reviewed by the Full ACRS. Dr. Kerr outlined an agenda for the Full Committee meeting, scheduled for August 7, 1981 in Washington, D.C.

For additional details, a complete transcript of the meeting is available in the NRC Public Document Room, 1717 H St., N.W., Washington, D.C. 20555 or from Alderson Reporters, 300 7th St., S.W., Washington, D.C. (202) 554-2345.

PLANT DESCRIPTION AND SCHEDULE

I. SITE DESCRIPTION

A. LOCATION

THE SUSQUEHANNA SES IS A 1075 ACRE SITE LOCATED ON THE WEST BANK OF THE SUSQUEHANNA RIVER IN SALEM TOWNSHIP, LUZERNE COUNTY, PENNSYLVANIA. IT IS LOCATED 15 MILES NORTHWEST OF HAZLETON AND 20 MILES SOUTHWEST OF WILKES-BARRE, THE NEAREST CITIES WITH POPULATIONS IN EXCESS OF 25,000. IT IS FOUR MILES SOUTH OF SHICKSHINNY AND FIVE MILES NORTHEAST OF THE BOROUGH OF BERWICK.

THE TOPOGRAPHY IN THE SITE AREA RANGES FROM RELATIVELY FLAT FLOODPLAINS TO GENTLY ROLLING HILLS. ELEVATIONS RANGE FROM 500 FEET ON THE FLOODPLAIN TO 1,600 FEET ABOVE MEAN SEA LEVEL ON THE NORTHERN BOUNDARY.

THE MAIN STATION BUILDINGS ARE LOCATED ON A TERRACE ABOVE THE FLOODPLAIN, APPROXIMATELY 4,000 FEET WEST OF THE SUSQUEHANNA RIVER.

B. EXCLUSION AREA

THE EXCLUSION AREA DISTANCE IS 1800 FEET FROM THE PLANT COMMON RELEASE POINT. THE SITE PROPERTY OWNED BY PP&L (1075 ACRES) IS SIGNIFICANTLY LARGER THAN THE EXCLUSION AREA (235 ACRES). THE EXCLUSION AREA BOUNDARY AND THE SITE BOUNDARY ARE COINCIDENT FOR ABOUT 1350 FEET ALONG THE SOUTHERN PORTION OF THE SITE. PP&L OWNS THE EXCLUSION AREA

INCLUDING MINERAL RIGHTS EXCEPT FOR TOWNSHIP ROUTE T-419 AND THEREFORE HAS AUTHORITY TO DETERMINE ALL ACTIVITIES WITHIN IT. TOWNSHIP ROUTE T-419, A LOCAL ROAD, TRAVERSES THE EXCLUSION AREA AT ITS NORTHERN EXTREMITY. THIS IS THE ONLY AREA WITHIN THE EXCLUSION AREA WHERE ACTIVITIES UNRELATED TO THE PLANT WILL OCCUR. PP&L HAS ARRANGED WITH THE SALEM TOWNSHIP SUPERVISORS AND WITH THE PA STATE POLICE TO CONTROL TRAFFIC ON ROUTE T-419 IN THE EVENT OF AN EMERGENCY.

C. POPULATION DENSITY

THE LOW POPULATION ZONE (LPZ) HAS BEEN DEFINED AS A CIRCULAR AREA OF 3 MILE RADIUS, THE CENTER WHICH COINCIDES WITH THAT OF THE EXCLUSION AREA. THE ESTIMATED POPULATION IN THE LPZ IN 1980 WAS ABOUT 2700 PERSONS AND IS PROJECTED TO REACH ABOUT 3000 BY 2020 (THE PROJECTED END OF PLANT LIFE).

THERE ARE NO SCHOOLS, HOSPITALS, STATE OR MUNICIPAL PARKS WITHIN THE LPZ. THE STATION RECREATION AREA, WITH PEAK DAILY ATTENDANCE ESTIMATED TO BE 800 PERSONS IS WITHIN THE LPZ. LUZERNE OUTWEAR COMPANY, AN INDUSTRIAL FACILITY IS LOCATED ABOUT 1.25 MILES NORTH-NORTHWEST OF THE SITE (~486 PERSONS EMPLOYED). CAR-MAR INC., IS A FIRM LOCATED IN A PLANNED INDUSTRIAL PARK ABOUT 1.7 MILES SOUTHWEST OF THE PLANT. CAR-MAR EMPLOYES APPROXIMATELY 70 PEOPLE. NO OTHER FIRMS HAVE MOVED IN YET. THE BERWICK AREA INDUSTRIAL ASSOCIATION, BEACH HAVEN SITE, IS THE ONLY OTHER INDUSTRIAL FACILITY KNOWN TO BE WITHIN THE LPZ. OTHER TRANSIENT POPULATION WITHIN THE LPZ IS LOW.

THE LARGEST COMMUNITY WITHIN 10 MILES OF THE SITE IS BERWICK LOCATED ABOUT 5 MILES SOUTHWEST OF THE SITE, WHICH HAD A 1980 POPULATION OF 12,189 PERSONS. THE NEAREST DENSILY POPULATED CENTER WITH A POPULATION OF ABOUT 25,000 PERSONS, IS THE CITY OF HAZLETON, ABOUT 15 MILES SOUTHEAST, WHICH HAD 1980 POPULATION OF 27,318 PERSONS. PP&L HAS EXAMINED POPULATION TRENDS WITHIN 10 MILES OF THE SITE AND HAS CONCLUDED THAT IT IS UNLIKELY THAT A POPULATION CENTER, (WITHIN THE MEANING OF THE TERM IN 10 CFR PART 100) CLOSER TO THE SITE THAN HAZLETON WILL DEVELOP DURING THE PLANT LIFETIME. THE CITIES OF WILKES-BARRE AND SCRANTON, WITH 1980 POPULATIONS OF 51,551 AND 88,117 PERSONS, RESPECTIVELY, ARE LOCATED ABOUT 18 MILES and 35 MILES, RESPECTIVELY, NORTHEAST OF THE SITE. PP&L HAS ESTIMATED THAT THE 1980 POPULATION WITHIN 30 MILES OF THE SITE IS APPROXIMATELY 652,000 PERSONS. PP&L PROJECTS THAT THE POPULATION WITHIN 30 MILES WILL DECLINE IN A VALUE OF ABOUT 600,000 PERSONS BY THE YEAR 2020.

D. TRANSPORTATION ROUTES

THE SUSQUEHANNA RIVER FLOWS NORTH TO SOUTH ABOUT 4000 FEET EAST OF THE PLANT. NAVIGATION, EXCEPT FOR RECREATION BOATING, IS NEGLIGIBLE ALONG THIS STRETCH OF THE RIVER.

THERE ARE TWO RAILROAD LINES WITHIN 5 MILES OF THE PLANT. THE ERIE-LACKAWANNA LINE TRAVERSES THE FLOODPLAIN NEAR THE WEST BANK OF THE SUSQUEHANNA RIVER APPROXIMATELY 2900 FEET EAST OF THE CENTER OF THE EXCLUSION AREA. THIS LINE IS USED ONLY FOR PLANT ACCESS VIA A SPUR

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THE DELAWARE AND HUDSON LINE IS LOCATED ON THE EAST BANK OF THE SUSQUEHANNA RIVER ABOUT 1.25 MILES EAST OF THE PLANT. A PROFILE OF HAZARDOUS CHEMICALS SHIPPED ON THE RAILROAD WAS OBTAINED IN 1975 AND IT WAS DETERMINED THAT AMMONIA AND SULFUR DIOXIDE WERE BEING SHIPPED SUFFICIENTLY FREQUENTLY TO REQUIRE A DETAILED ANALYSIS. A PROBABILISTIC MODEL WHICH CONSIDERS RAILROADS ACCIDENT RATES, RAILCAR SHIPPING WEIGHT AND FREQUENCY OF SHIPMENTS, AS WELL AS DISTANCES OF VARIOUS TRACK SEGMENTS FROM THE PLANT, AND METEOROLOGICAL DISPERSION CONDITIONS WAS

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THERE ARE FOUR ROADS THAT PASS IN THE SITE VICINITY. THESE ARE:

1) SALEM TOWNSHIP ROAD T-419 WHICH PASSES TO THE NORTH ABOUT 1600 FEET FROM THE CENTER OF THE EXCLUSION AREA AND 500 FEET FROM VITAL PLANT STRUCTURES.

2) SALEM TOWNSHIP ROAD T-438 WHICH PASSES TO THE WEST ABOUT 2000 FEET FROM THE CENTER OF THE EXCLUSION AREA AND 1400 FEET FROM VITAL PLANT STRUCTURES.

3) SALEM TOWNSHIP ROAD T-456 WHICH PASSES TO THE SOUTH ABOUT 1800 FEET FROM THE CENTER OF THE EXCLUSION AREA AND 1600 FEET FROM VITAL PLANT STRUCTURES.

4) U.S. ROUTE 11 WHICH PASSES TO THE EAST ABOUT 2600 FEET FROM THE CENTER OF THE EXCLUSION AREA AND 2500 FEET FROM VITAL STRUCTURES.

PP&L HAS ANALYZED A PROPANE TRUCK ACCIDENT ON U.S. ROUTE 11 AND HAVE SHOWN THIS POSES NO HAZARD. TOXIC MATERIAL TRANSPORT AND HAZARDS ALONG THESE ROADS IS NOT EXPECTED ON THE BASIS THAT THROUGH TRUCK TRAFFIC IS EXPECTED TO USE INTERSTATE 80 AND 81 AND THAT LOCAL INDUSTRY IN THE BERWICK AREA DOES NOT INCLUDE INDUSTRY CLASSIFICATIONS WHICH ARE EXPECTED TO PRODUCE OR CONSUME QUANTITIES OF HAZARDOUS MATERIALS.

E. METEOROLOGY

EASTERN PA IS SUBJECTED TO THUNDERSTORM ACTIVITY AND THE EFFECT OF TROPICAL STORMS. FREEZING RAIN AND SNOW ARE COMMON WINTERTIME PHENOMENON. TORNADOES HAVE BEEN REPORTED IN THE SITE VICINITY. PP&L HAS DESIGNED PLANT STRUCTURES TO WITHSTAND SEVERE OR EXTREME CONDITIONS AT THE SITE IN ACCORDANCE WITH 10 CFR PART 50, APPENDIX A, GENERAL DESIGN CRITERIA 2. ONSITE METEOROLOGICAL MEASUREMENTS HAVE PROVIDED DATA FOR RELEASES OF RADIOACTIVE GAS IN ACCORDANCE WITH 10 CFR PART 100.10 AND 10 CFR PART 50 APPENDIX I.

F. HYDROLOGY

1. FLOOD POTENTIAL

THE ONLY NATURAL WATER BODY POSING A SIGNIFICANT POTENTIAL FLOOD HAZARD TO THE PLANT IS THE SUSQUEHANNA RIVER. EXCEPT FOR THE RIVER INTAKE AND DISCHARGE STRUCTURES, ALL MAJOR PLANT STRUCTURES ARE 4000 FEET FROM THE SUSQUEHANNA RIVER AT ELEVATION 670 FEET MSL OR HIGHER. THIS ELEVATION IS APPROXIMATELY 175 FEET ABOVE THE SUSQUEHANNA RIVER FLOODPLAIN, MORE THAN 150 FEET ABOVE THE HIGHEST RECORDED RIVER LEVEL AND OVER 120 FEET ABOVE THE CALCULATED PROBABLE MAXIMUM FLOOD ELEVATION.

THE LARGEST FLOOD OF RECORD ON THE SUSQUEHANNA RIVER NEAR THE SITE (1972 - TROPICAL STORM AGNES) YIELDED A WATER ELEVATION MORE THAN 150 FEET BELOW PLANT GRADE AND MORE THAN 8 FEET BELOW THE DESIGN FLOOD LEVEL OF THE RIVER INTAKE STRUCTURE.

2. ULTIMATE HEAT SINK (UHS)

THE UHS IS THE ONSITE SEISMIC CATEGORY 1, SPRAY POND. THE RIVER INTAKE PROVIDES WATER FOR NORMAL OPERATION DURING EXTREME HYDRAULIC EVENTS. THE SPRAY POND PROVIDES COOLING WATER FOR EMERGENCY SHUTDOWN AND FOR THE RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM DURING NORMAL SHUTDOWN.

THE SPRAY POND IS A KIDNEY SHAPED, CONCRETE LINED BASIN, THAT CONTAINS 25 MILLION GALLONS OF WATER WITH A SURFACE AREA OF 8 ACRES. THERE ARE 4 SPRAY NETWORKS CONTAINING A TOTAL OF 1056 SPRAY NOZZLES ON 264 "TREES". THE SPRAY POND WILL BE SUFFICIENT TO PROVIDE 30 DAYS OF COOLING WATER WITHOUT MAKEUP.

3. GROUNDWATER

NO PLANT USE OF GROUNDWATER DURING OPERATION OF THE PLANT IS ANTICIPATED. A POSTULATED FAILURE OF THE EVAPORATOR CONCENTRATE TANK, THE TANK OUTSIDE OF CONTAINMENT WHOSE FAILURE HAS THE GREATEST POTENTIAL TO RESULT IN HIGH OFFSITE RADIONUCLIDE CONCENTRATIONS; WAS ANALYZED TO ESTIMATE THE CONCENTRATION OF RADIOACTIVE CONTAMINANTS AT OFFSITE LOCATIONS. THE CONTENTS OF THE TANK WERE CONSERVATIVELY ASSUMED TO ENTER THE GROUNDWATER INSTANTANEOUSLY, AS A SLUG RELEASE. THE NUCLIDES WERE ASSUMED TO TRAVEL WITH A VELOCITY CONTROLLED BY THE GROUNDWATER VELOCITY AS MODIFIED BY CONSERVATIVELY ESTIMATED ION EXCHANGE CHARACTERISTICS.

THE GROUNDWATER GRADIENT IS TOWARD A BEDROCK VALLEY NORTH OF THE RADWASTE BUILDING AND THEN TOWARDS THE SUSQUEHANNA RIVER. IT IS CONSERVATIVELY ESTIMATED THE MINIMUM GROUNDWATER TRAVEL TIME TO BE 9.2 YEARS. FOR THOSE NUCLIDES THAT ARE AFFECTED BY ION EXCHANGE PROCESSES THE TRAVEL TIMES WOULD BE LONGER. THE CALCULATED CONCENTRATIONS OF ALL NUCLIDES WERE WELL BELOW THE

MAXIMUM PERMISSIBLE CONCENTRATIONS LISTED IN 10 CFR PART 20, APPENDIX B, TABLE II BEFORE MIXING WITH SUSQUEHANNA RIVER WATER.

G. GEOLOGY AND SEISMOLOGY

THE SEISMIC CATEGORY 1 STRUCTURES IN THE POWERBLOCK AREA ARE SUPPORTED ON FIRM, UNWEATHERED ROCK HAVING AN INTACT UNCONFINED COMPRESSIVE STRENGTH IN EXCESS OF 3,600 PSI AND A YOUNGS MODULUS IN EXCESS OF 3 X 10^6 PSI. THE STRUCTURAL LOADS WILL PRODUCE NO SIGNIFICANT TOTAL OR DIFFERENTIAL SETTLEMENT OF FOUNDATIONS SUPPORTED ON BEDROCK.

THE EMERGENCY SERVICE WATER PUMPHOUSE IS SUPPORTED ON GLACIAL SOILS AT A DEPTH OF ABOUT 50 FEET BELOW ORIGINAL GRADE. THE GROSS FOUNDATION LOADS ARE CALCULATED TO BE 2.8 KSF OR DEAD LOADS AND 0.3 KSF FOR LIVE LOAD. THE SPRAY POND AND SEISMIC CATEGORY I PIPELINE AND CONDUITS WILL NOT EXERT SIGNIFICANT STATIC LOAD ON SUBSURFACE SOILS; THUS NO SIGNIFICANT SETTLEMENT DUE TO STATIC LOADS IS EXPECTED.

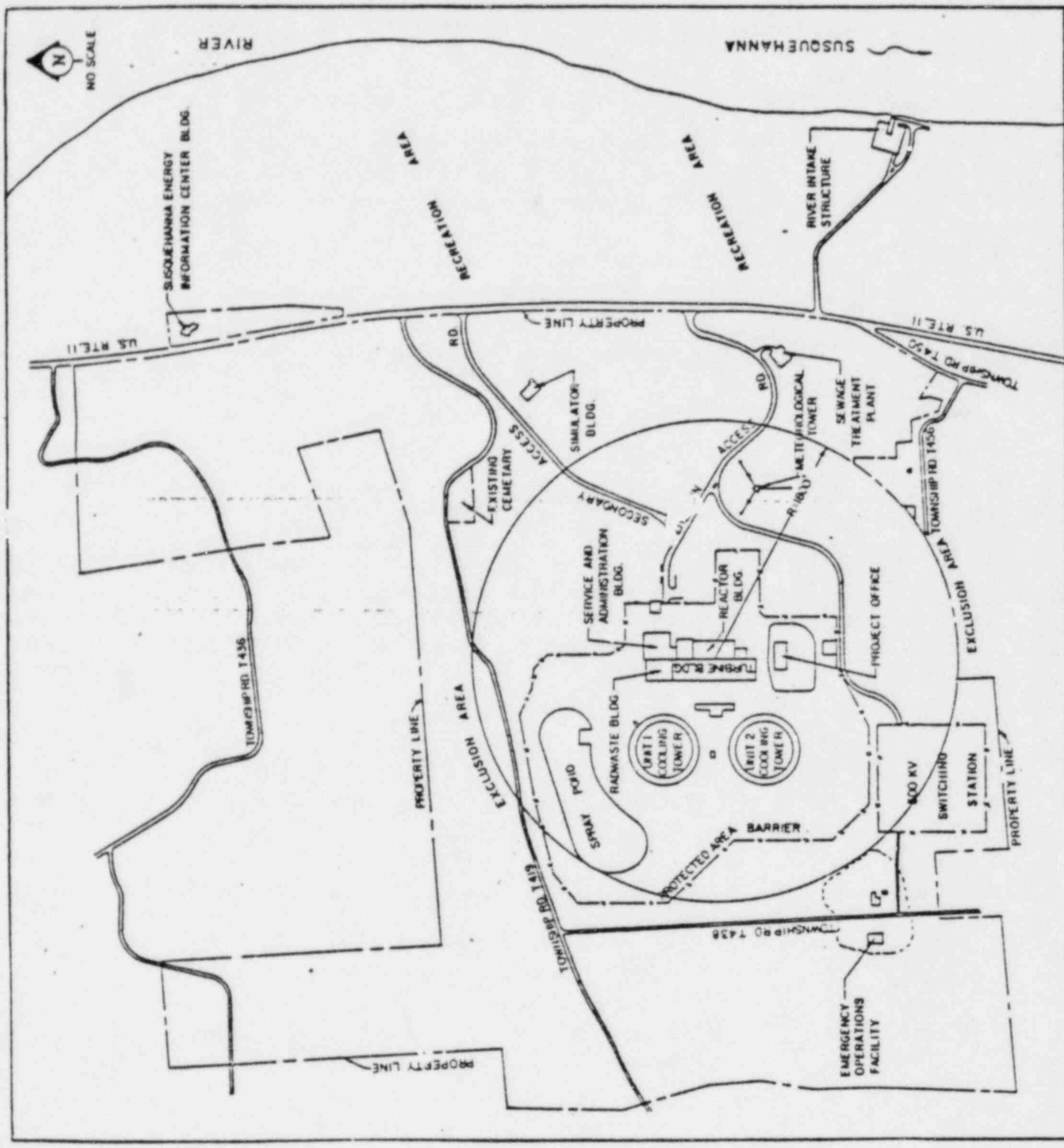
THE SAFE SHUTDOWN EARTHQUAKE DESIGN ACCELERATION FOR STRUCTURES SUPPORTED ON ROCK IS 0.10 G. TO ACCOMMODATE SOIL AMPLIFICATION OF SEISMIC MOTION, THIS SAFE SHUTDOWN EARTHQUAKE VALUE HAS BEEN INCREASED TO 0.15 G FOR SOIL-SUPPORTED STRUCTURES. THE CORRESPONDING VALUES FOR THE OPERATING BASES EARTHQUAKE ARE 0.05 G AND 0.08 G.

THE MAXIMUM SETTLEMENT UNDER THE SPRAY POND DURING A SAFE SHUTDOWN EARTHQUAKE WAS CALCULATED TO BE 1.2 INCHES. THE MAXIMUM SETTLEMENT UNDER THE EMERGENCY SERVICE WATER PUMPHOUSE WAS CALCULATED TO BE 1.0

INCH. THE MAXIMUM DIFFERENTIAL SETTLEMENT ACROSS THE SPRAY POND WILL BE EQUAL TO THE MAXIMUM TOTAL SETTLEMENT BECAUSE PART OF THE SPRAY POND IS CUT INTO ROCK. THE MAXIMUM SETTLEMENT OF THE SEISMIC CATEGORY I PIPELINES AND CONDUITS IS EXPECTED TO BE LESS THAN 1.0 INCH BECAUSE THEY ARE SUPPORTED ON A LESSER DEPTH OF GLACIAL SOIL THAN THE SPRAY POND. CONSERVATIVE ASSUMPTIONS WERE USED IN THESE CALCULATIONS AND ACTUAL SETTLEMENTS AS A RESULT OF DYNAMIC LOADING ARE EXPECTED TO BE LESS. ALL PIPING AND STRUCTURES WERE DESIGNED FOR THIS SETTLEMENT.

II. COMPARISON OF PRINCIPAL DESIGN FEATURES

MANY FEATURES OF THE DESIGN OF SUSQUEHANNA ARE SIMILAR TO OTHER PLANTS UNDER CONSTRUCTION (E.G. LASALLE, ZIMMER) OR IN OPERATION (E.G. HATCH 2). A LISTING OF COMPARABLE PRINCIPAL PARAMETERS AND FEATURES OF SUSQUEHANNA, LASALLE, ZIMMER AND HATCH 2 IS ATTACHED.



N
NO SCALE
SUSQUEHANNA RIVER
SUSQUEHANNA ENERGY INFORMATION CENTER BLDG.
RECREATION AREA
RECREATION AREA
RIVER INTAKE STRUCTURE
U.S. RTE.11
U.S. RTE.11
PROPERTY LINE
TOWNSHIP RD T450
SEWAGE TREATMENT PLANT
METEOROLOGICAL TOWER
SIMULATOR BLDG.
SECONDARY ACCESS RD.
ACCESS RD.
EXISTING CEMETARY
TOWNSHIP RD T456
SERVICE AND ADMINISTRATION BLDG.
REACTOR BLDG.
TURBINE BLDG.
PROJECT OFFICE
EXCLUSION AREA
EXCLUSION AREA
RADWASTE BLDG
UNIT 1 COOLING TOWER
UNIT 2 COOLING TOWER
SPRAY POND
PROTECTED AREA BARRIER
500 KV SWITCHING STATION
PROPERTY LINE
PROPERTY LINE
TOWNSHIP RD T436
TOWNSHIP RD T419
TOWNSHIP RD T438
EMERGENCY OPERATIONS FACILITY
PROPERTY LINE

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A REPRINT FROM PENNSYLVANIA POWER & LIGHT COMPANY'S profile
A Statistical Review and Discussion of Current Issues

- The Economic Impact of Susquehanna Nuclear Generation
- Renewable Energy Resources and Load Management

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July 1981

The Economic Impact of Susquehanna Nuclear Generation

The company currently has two nuclear generating units under construction at its Susquehanna plant. This article presents a conceptual review of the long-term economic impact of the Susquehanna generating units on the cost of providing electric service to customers. The units are expected to operate into the early part of the next century, so it is important that their impact be viewed over an extended period of time, rather than making a short-term assessment.

The company expects that the benefits of generation from nuclear units with lower fuel costs will substantially offset Susquehanna's operating and investment-related costs. This conclusion is developed in the following discussion of the general nature and trend of the operating and investment-related costs and the anticipated fuel-related savings.

In the first part of this discussion, the types of costs expected and the fuel-related savings anticipated from nuclear generation are shown in a series of charts accompanied by narrative. The second section of the article presents key assumptions and provides a more in-depth discussion of certain of the costs and savings. The trends shown in the first section are dependent on the assumptions set forth in the second section. While the company believes the assumptions used in this discussion are reasonable based on information available at this time, changes in these assumptions could alter the results presented in the charts and text.

Susquehanna Costs and Savings

The fundamental principle of utility ratemaking involves setting customers' rates at a level that enables the utility to recover the costs of providing service. These costs include operation and maintenance expenses, taxes, recovery of the cost of property investment through depreciation, and a fair rate of return to investors for money provided to finance construction of facilities. Recovery of the total cost of providing electric service is generally referred to as the "cost of service principle."

Certain costs are expected to increase when the Susquehanna units are placed in service. These costs fall into two general categories:

- Capital-related costs—depreciation, return on investment and taxes
- Operating costs—wages, material, payments to contractors, etc. to operate and maintain the units

However, in the long run, these increased costs are expected to be offset by lower fuel costs in two ways. First, the fuel cost of electricity used by PP&L's customers will be less with the Susquehanna units. Second, the company will be able to sell more energy from its coal and oil-fired stations to other utilities with the savings from the sales passed on to PP&L's customers.

Chart 1
Susquehanna
Annual Capital-Related Costs

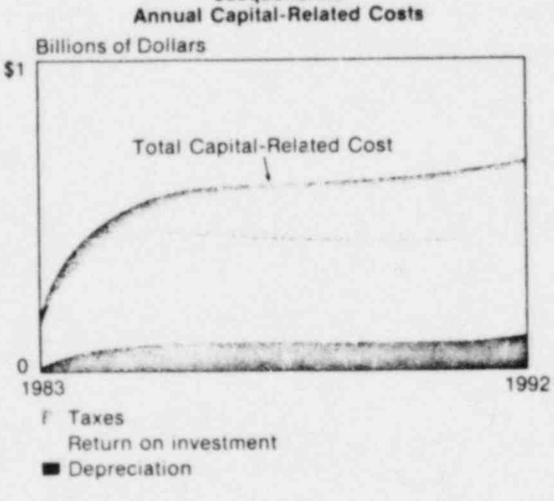
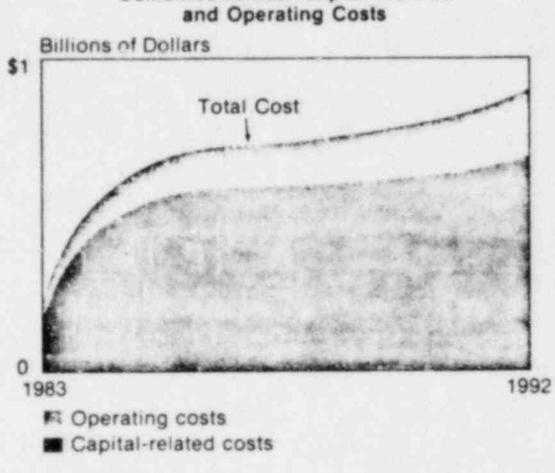


Chart 2
Susquehanna
Combined Annual Capital-Related and Operating Costs



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Capital-Related and Operating Costs

The trend of the projected capital-related costs, based on the company's $3.15 billion share of the estimated in-service cost of the plant and an average level of capital additions after the units are placed in service, is shown in Chart 1.

As shown in the chart, depreciation expense is a fairly constant amount after both units are in service, increasing slightly due to anticipated capital additions to the plant. For a fixed dollar investment in utility plant, the return on investment declines over the life of the facility as the initial investment is reduced through depreciation. After the Susquehanna units are placed in service, the return component shown in Chart 1 is about the same each year because the anticipated capital additions offset the depreciation in determining the net investment. The tax component gradually increases during most of the period shown. Income taxes compose the major portion of this category and generally track the return on the equity investment in the plant. Income taxes are influenced further by tax depreciation, which permits income tax reductions in the early years of the plant life. After both units are in service, the total annual capital-related cost is expected to increase gradually for the first 10 years of operation.

Operating costs represent the personnel, materials and other expenditures necessary to keep the units running and to perform necessary maintenance work. Operating costs are expected to increase, principally reflecting escalating prices for wages, material and supplies. The actual rate of cost escalation experienced will be a major factor in the operating costs ultimately incurred.

Chart 2 projects the combined capital-related and operating costs. The total of these two types of costs represents the annual revenues that would be required from customers if there were no fuel-related savings. The general trend shown on this chart is a gradually increasing total cost of service for these two categories of expenses.

Savings in Energy Costs

Part of the benefit to PP&L's customers from the low-cost nuclear generation of the Susquehanna units is shown in the next two charts. Chart 3 shows the estimated fuel cost to serve customers if the Susquehanna units were **not** placed in service. Chart 4 projects the expected fuel cost to serve customers **with** the Susquehanna units operating as planned. In this case, electricity for PP&L's customers that would otherwise be produced by higher-cost coal and oil generation will be provided by lower-cost nuclear generation. A comparison of these charts illustrates the expected fuel-cost savings for PP&L's customers due to operation of the Susquehanna units.

In addition to the fuel-cost savings, customers are expected to benefit from the sale of energy to other

Chart 3
Annual Fuel Cost to Serve Customers Without Susquehanna

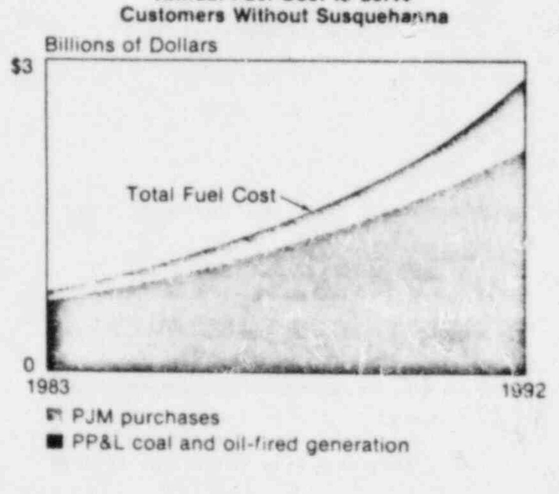
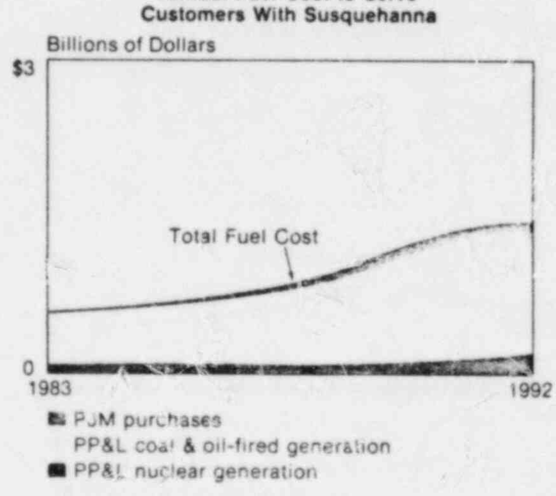


Chart 4
Annual Fuel Cost to Serve Customers With Susquehanna



utilities. Generation from PP&L's coal and oil-fired stations is expected to replace more expensive oil-fired generation of other companies within the Pennsylvania-New Jersey-Maryland Interconnection (PJM). With the PJM split savings pricing arrangement, 50 percent of the total fuel-cost savings from these sales will be realized by PP&L's customers in the form of lower energy costs.

The total savings in energy costs to PP&L's customers because of Susquehanna includes the direct fuel savings from using the lowest cost nuclear and fossil units to serve PP&L's load and 50 percent of the savings from additional interchange power sales. The anticipated total savings in energy costs is shown in Chart 5.

Net Revenues Required From Customers

The net revenues required from PP&L's customers due to operation of the Susquehanna units can be shown by comparing the total capital-related and operating costs with the savings in energy costs expected to be realized. Chart 6 shows that the combined costs related to Susquehanna are expected to be greater than the savings during the early years of operation. After that, the savings in energy costs are projected to be greater than the capital-related and operating costs, resulting in lower revenue requirements from customers than would have been required if the Susquehanna units were not in service. Changes in the key assumptions discussed in the second part of this article would alter the net revenue requirements shown in this chart.

The company cannot, at this time, predict the ultimate impact on customers' bills of placing the Susquehanna units in service. However, another way of looking at the economic impact of the Susquehanna units is to show the net cost or net savings reflected in Chart 6 on the basis of projected cents per kilowatt-hour of energy sales.

The annual capital-related and operating costs, minus the savings in energy costs shown in Chart 6, represents the net cost or net savings attributable to the Susquehanna plant. Dividing this amount by the expected energy sales gives the annual net cost or net savings per kilowatt-hour of sales. Chart 7 shows the annual net cost or net savings in cents per kilowatt-hour of estimated energy sales.

The chart shows that about five years after the units are placed in service, nuclear generation will result in savings to customers compared with what it would have cost to meet customers' energy needs with more expensive fossil-fueled plants and purchases of electricity from other utilities.

The cumulative net revenue requirement for PP&L's customers is shown in Chart 8. The cumulative annual savings have been deducted from the cumulative annual costs to determine the total-to-date net revenue requirement. The operation of the Susque-

Chart 5
Total Savings in Energy Costs for Customers With Susquehanna

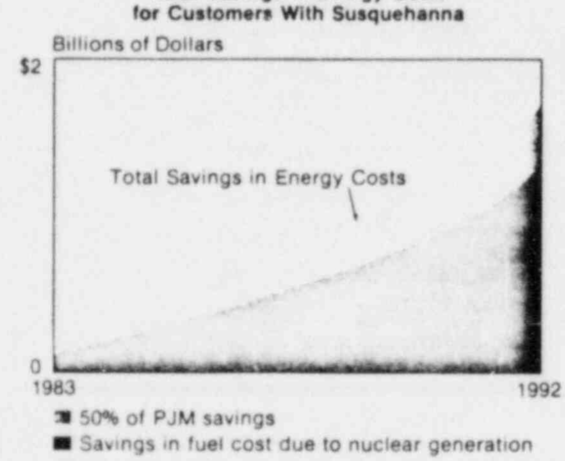
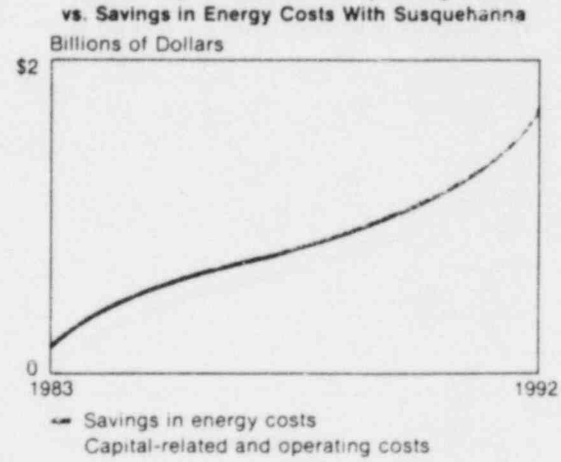


Chart 6
Annual Capital-Related and Operating Costs vs. Savings in Energy Costs With Susquehanna



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hanna units will require additional revenues for the early years of operation. After this, however, the savings in energy costs from the nuclear generation are expected to result in a lower revenue requirement than would otherwise occur. In less than a decade, the company estimates that savings in energy costs realized from nuclear generation will more than offset the total Susquehanna capital-related and operating costs incurred during that time period.

When the cumulative net cost associated with the Susquehanna units becomes zero, customers will have been charged the same amount for electric service with the Susquehanna units in service that they would have been charged if the units had not been built. After that time, the total amount charged customers for electric service is expected to be lower than it would have been if customers' energy needs had been met with existing fossil-fueled plants and additional interchange purchases.

Key Assumptions and Related Discussion

When making projections for an extended period of time, certain assumptions about future conditions must be made. The key assumptions used for this presentation are identified in this section. Significant changes in assumptions obviously would alter the costs and savings trends presented in the first section of this article. Also presented in this section is a more in-depth discussion of certain of the cost components and energy cost savings.

The capital-related and operating costs and savings in energy costs shown in this article reflect the company's agreement to sell 6.6 percent of its share of the capacity and energy of the Susquehanna units to another utility from the in-service dates to 1991.

Susquehanna Construction

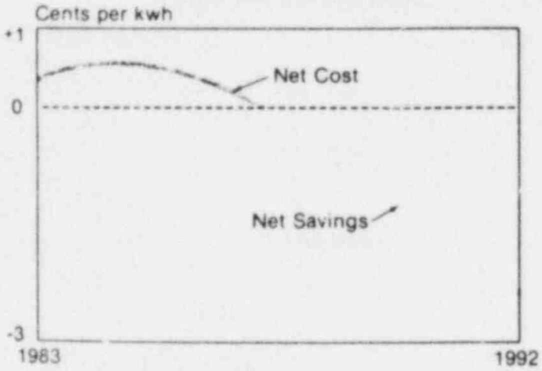
The following costs and in-service dates of the Susquehanna units were used in preparing this article:

- Total cost of the two units—$3.5 billion; the company's 90 percent share of the cost being $3.15 billion
- In-service dates of the second quarter of 1983 for Unit 1 and the second quarter of 1984 for Unit 2
- Capital additions to the Susquehanna plant of approximately $50 million a year were assumed to occur after the units are placed in service

Capital-Related Costs

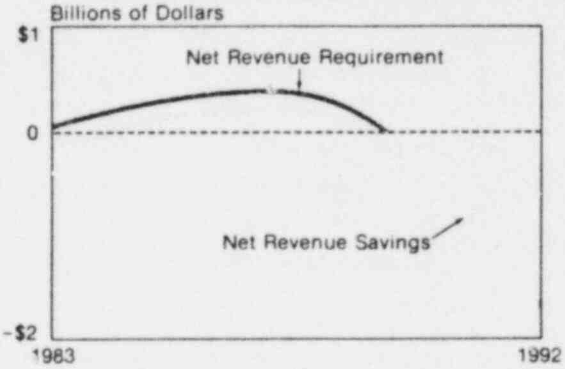
Essentially, capital-related costs are the revenues required to: (1) recover the cost of the plant through depreciation; (2) compensate investors for the use of the money used to finance the construction of the plant—called return on investment; and (3) pay income and other taxes.

Chart 7
Net Cost or Savings Per Kilowatt-hour of Sales With Susquehanna



- Net cost per Kwh of sales
Net savings per Kwh of sales

Chart 8
Cumulative Net Revenue Requirement or Savings for Customers With Susquehanna



— Cumulative net revenue requirement
Cumulative net revenue savings

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The capital-related costs of plant investment were based on the following assumptions:

- Book depreciation—straight line method
  —35-year life for Unit 1
  —34-year life for Unit 2
- Tax depreciation —16-year tax life using double declining balance method and flow through of income tax reductions
- Rate of Return —approximately a 12 percent cost of capital
- Taxes —tax laws and rates in effect at Jan. 1, 1981
  —investment tax credit amortized over the book life of the plant

Book Depreciation—represents recovery of the original cost of the plant investment over the life of the facility. For a fixed dollar investment, the annual depreciation cost remains the same each year.

Return on Investment—represents the amount required to compensate investors for the money they provided to finance construction of the units. Interest and dividends must be paid as a return to bond and stockholders.

Taxes—included in this cost category are federal income taxes and the state income, capital stock, utility realty and gross receipts taxes. In its January 1981 rate decision, the Pennsylvania Public Utility Commission disallowed normalization of state and federal income taxes on timing differences related to tax depreciation, so all income tax reductions resulting from accelerated tax depreciation flow through to customers.

Operating Costs

The operating costs category of expense includes such items as:

- Wages and employee benefits
- Material and supplies
- Work performed by outside contractors
- Rentals
- Insurance

Also included in this category of expense are:

- Carrying charges on nuclear fuel in the reactor
- Plant decommissioning costs

Excluded from this category is nuclear fuel expense, which is included as energy cost.

Operation of the Susquehanna units will require personnel, replacement parts, maintenance work by specialized outside contractors, rental of equipment and insurance. These costs are expected to increase because of inflation during the life of the units. Reflected in the operating costs set forth in this article is an overall annual cost escalation rate of 9 percent.

Certain costs unique to nuclear generating units are also included in the general category of operating costs. When nuclear fuel is placed in the reactor, it remains there for a period of time. The cost of the portion of nuclear fuel not yet used to generate electricity is treated similarly to utility plant for rate-making purposes. A return on invested capital used to finance the investment in nuclear fuel in the reactor and related taxes are included in the operating costs component.

At the end of the useful lives of the Susquehanna units, certain expenditures will be required in order to retire the units. This process is called decommissioning the plant. Decommissioning will require end-of-life decontamination, disposal of radioactive materials and dismantling of the plant. While the eventual cost to retire a nuclear generating facility is uncertain at this time, the company has studied the required work and has included in operating costs an estimated annual charge to customers over the life of the plant to pay for decommissioning the radioactive portion of the Susquehanna plant.

Energy Costs

Major factors that will affect the economic impact of the Susquehanna units on customers' energy costs are PP&L's generating capacity by fuel type, the operating performance of generating units, the relative costs of different types of fuels and the relationship of PP&L's generating units to those of other companies in the power pool.

Fuel Expense

The following schedule shows the fuel cost per kilowatt-hour generated by PP&L's coal and oil-fired steam stations for the past five years. Also shown is a representative industry fuel cost per kilowatt-hour generated by nuclear stations during the past five years.

Fuel Cost per Kilowatt-hour Generated
Steam Electric Stations
(Cents per Kwh)

	Coal	Oil	Nuclear
1976	1.09¢	2.14¢	.29¢
1977	1.13	2.30	.34
1978	1.26	2.23	.36
1979	1.30	3.20	.37
1980	1.40	4.55	.45

The projected fuel costs used in this article assume that nuclear fuel costs per kilowatt-hour will continue to be substantially lower than coal or oil-fired generation. Likewise, the fuel cost of coal-fired generation is expected to remain cheaper than oil-fired generation. Long-range fuel costs used in this article reflect annual escalation rates of about 12 percent for coal and oil and 8 percent for nuclear.

When nuclear fuel is removed from the reactor, it must be stored temporarily and then moved to a permanent storage facility or reprocessed. There are currently no commercially operating facilities in the United States for reprocessing spent nuclear fuel. However, the Department of Energy has proposed a policy under which the federal government would provide off-site storage of spent fuel. The projected nuclear fuel expense for Susquehanna includes estimated off-site storage fees.

PJM Interconnection Operations

PP&L is one of 11 electric utilities that make up the Pennsylvania-New Jersey-Maryland (PJM) Interconnection. Operation of generating units of the 11 utilities in PJM is based generally on a coordinated economic basis. From a computerized power control center, the lowest-cost generating units of member companies are brought on-line as additional energy is needed throughout PJM, without regard to individual company requirements. This provides economies for the customers of all member companies.

Since, in most instances, the lowest-cost generating units are operating throughout PJM without regard to individual system requirements, in any given hour, some companies will be generating less than the requirements of their own customers and some will be generating more. Therefore, in that hour, some companies automatically become interchange purchasers and others become sellers.

The price of these interchange transactions is set midway between what it costs the seller to produce the energy (principally fuel costs) and what it would have cost the buyer to produce it. The savings are split equally between the buyer and seller.

As an example, if at a particular point in time, it costs company A, 2 cents to produce a kilowatt-hour and company B, 4 cents if it were to use its own generating units, the selling price would be 3 cents, or a splitting of the 2 cent difference in cost. Both benefit—company A receives 1 cent more than it cost to produce the electricity sold, and company B pays 1 cent less than it would have cost to generate the electricity. The split-savings pricing arrangement provides equal benefits to both the selling and purchasing company's customers.

Economics of Generating Capacity

After the Susquehanna units are placed in service, PP&L will have more generating capacity than needed to meet customers' peak load. A vital point to consider when reviewing the composition of installed capacity and reserves is the economics of operating the various types of generating capacity. The most expensive units to operate, based on fuel costs and operating characteristics, are combustion turbines, followed by oil-fired steam, coal and then nuclear.

When the Susquehanna units are placed in service, PP&L will use the low-cost nuclear units and the cheapest coal-fired units to meet most of its customers' energy needs. The more expensive coal-fired units and the oil-fired units generally will be available to provide energy to other PJM companies.

To meet the energy requirements of customers within PJM, oil-fired steam generation is expected to be utilized for at least the next 10 to 15 years. With the composition of PJM generating capacity, PP&L expects that its coal-fired generating units will operate at high usage rates since they are cheaper to operate than PJM oil-fired units. Additionally, PP&L's oil-fired units are expected to continue to operate at about their present rates, since they are more efficient than most of the oil-fired units in PJM.

Except for outage time required for refueling and maintenance, the Susquehanna units are expected to generate electricity around the clock. The annual rates of usage (capacity factor) of the Susquehanna units assumed in this article range from 55 percent to 65 percent during the first four years of operation and average 70 percent for the fifth year and thereafter.

Serving PP&L's customers principally with low-cost nuclear and coal-fired units, combined with the sale of energy to other PJM companies, will result in substantial energy cost savings for PP&L's customers as shown in this article.

Renewable Energy Resources and Load Management

In today's complex society, energy requirements no longer can be viewed separately from other important social issues, such as protecting the environment, improving our economic health and maintaining national security.

There is a growing awareness that there is no one way to solve our energy problems—no one source of energy to meet all our needs. The choice clearly is not one source of energy before another, nor an emphasis on energy supply without comparable management of end-use demand. Rather, it is a recognition of all available energy sources, their advantages and disadvantages, their practicality in different applications and our ability to manage their end use.

At PP&L, we believe it is our responsibility to explore all the options objectively and measure their applicability in meeting the needs of our customers. We recognize that trade-offs in energy policy between legitimate, yet competing, objectives will be necessary. Moreover, we believe a balanced energy program, based on both supply and demand, provides the best course for our customers and for the American people.

PP&L has a history of commitment to a balanced energy program. This commitment has taken many forms, including the development of a diverse fuel supply for generating electricity, initiatives into conservation even before the OPEC oil embargo staggered our economy, and extensive research activities with renewable and alternative forms of energy.

PP&L now is taking further steps to provide the best balance of energy priorities with new initiatives into renewable energy resource development and customer financial incentives for load management.

Pioneer Rate Offered

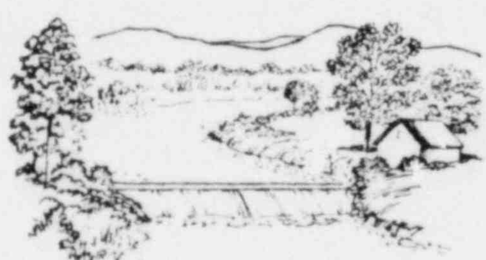
Renewable energy resources can play a vital part in our country's energy future. The challenge is to tap these resources limited not by quantity, but only by imagination. As an innovative step in utility rate-making, PP&L is now offering a "pioneer" rate to purchase electricity from those enterprising individuals, municipalities and businesses that lead the way in the use of renewable resources.

PP&L's plan is to purchase excess electricity generated by these renewable energy resources at a price designed to encourage their development. In this way, all of our customers may benefit—both from the delay in the need for new, large generating facilities and the accelerated development of these important energy sources. In addition, generating electricity by burning certain waste materials could help alleviate the growing solid waste disposal problem.

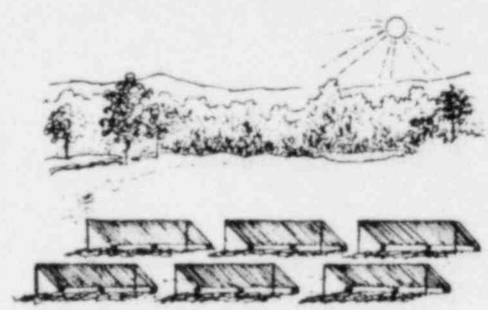
The pioneer rate for electricity generated with renewable energy resources will pay customers 6 cents per kilowatt-hour or PP&L's annual average interchange rate with other power companies, whichever is greater. The company's current interchange rate is about 4 cents per kilowatt-hour. Since it takes time to develop renewable energy projects, this pioneer rate will be available for projects installed prior to 1990. After that time, PP&L will purchase electricity from "pioneer" projects already in service for at least the 1989 rate. The company will continue to evaluate the equitability of this rate as energy costs change in the future.

Renewable energy resources include small-scale hydroelectric (5,000 kilowatts or less), wind, solar, trash-burning, biomass and others. This pioneer rate schedule does not apply to traditional sources of fuel such as oil, gas or coal. Co-generation of electricity with these fossil fuel sources is an additional aspect of a balanced energy program and will be covered by a rate separate from the pioneer schedule for renewable energy.

This pioneer rate is not for everybody. Not all renewable energy projects are economical at this time. PP&L is offering this rate however, because of our desire to encourage the development of renewable sources. As with all pioneering ventures, the path is uncertain. We believe that by offering a favorable rate for electricity from renewable energy projects, PP&L can contribute to broadening the base of knowledge about these potentially unlimited resources. From our research over the years, we expect to be able to advise our customers about the advantages and disadvantages of various projects.

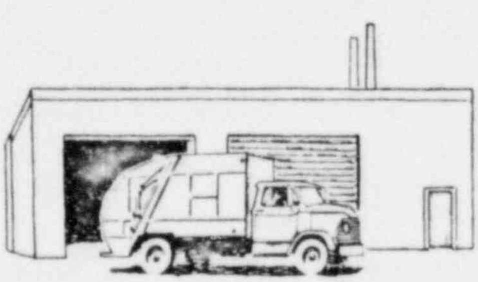


Hydroelectric

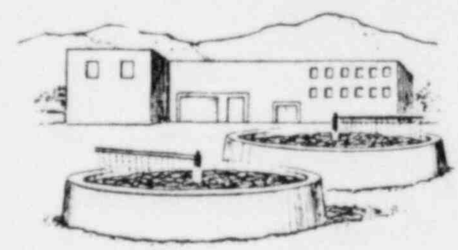


Solar

Wind



Trash-burning



Biomass

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Currently, the company is spending almost $5 million a year for research and development. Projects under way include the design and construction of passive solar homes, solar photovoltaic cells that generate electricity directly from sunlight, generation of electricity with wind and many others.

As a part of PP&L's encouragement of renewable energy resources, the company will consider becoming a partner or joint owner in those projects with the most potential. If we believe that PP&L's professional and technical resources can help assure the viability of a project, we will work with the developers to bring renewable resource applications on line.

Based on preliminary estimates, the pioneer rate could stimulate the development of renewable energy facilities to produce from 400 million to 450 million kilowatt-hours of electricity annually by 1990. This would be enough electricity to supply about 47,000 average homes.

The impact this proposal has on the charges to our customers is difficult to assess with any degree of certainty since it is contingent on many factors beyond our control. This impact is expected to range from $10 million to $20 million over the next 10 years. However, recognizing that a number of circumstances could vary, the impact could be insignificant or as much as $70 million.

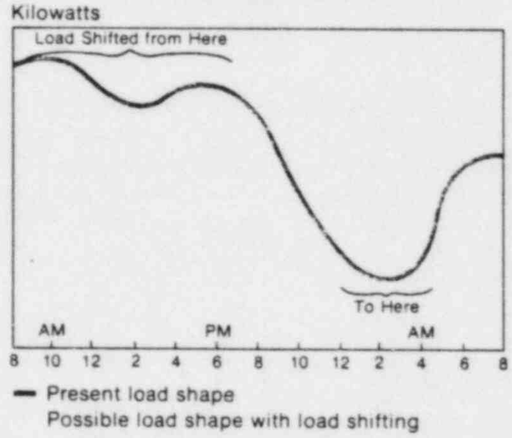
Initiatives to Control Peak Load Growth

In addition to PP&L's commitment to renewable energy resource development, the company is taking major steps to give customers financial incentives to shift their use of electricity away from periods of peak demand. This type of load management is necessary to delay the need for new generating units after the Susquehanna plant comes into service. Utilities have to plan facilities to meet periods of peak demand by customers, as well as overall use. By shifting demand from traditional periods of peak load, we can make more efficient use of our system. This, in turn, will hold down the need for new facilities and keep costs down.

With today's soaring construction costs and staggering interest rates, building any new facilities is extremely costly. We also are faced with siting difficulties, capital shortages, long lead times and licensing hurdles that make the addition of new power plants a difficult way of meeting customer demand for electricity.

Our present capability to meet customer peak demand, while maintaining adequate reserves for maintenance and possible equipment breakdowns, can be extended by customers shifting their use of electricity to off-peak periods. This "load shifting" can help hold down future electric costs.

Daily Load Shape



While PP&L already has been encouraging customers to shift their use of electricity in this manner, we recently have provided further financial incentives for load shifting. As a part of the rate increase granted in late January, the company requested and received permission to offer experimental time-of-use rates for certain customers. These rates reflect the differences in the actual cost of providing electricity during periods of peak demand and during periods when use of electricity is lower. Consequently, the new rates to customers will be less during off-peak hours and greater during on-peak hours.

This experimental program will help us determine customer willingness to shift energy use to off-peak hours and the extent to which this load shifting may contribute to a reduced rate of peak load growth. Also included in this program are time-of-use rates for water heating and off-peak storage for space heating.

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This involves using water or some other medium to store heat energy produced during off-peak hours for use during on-peak hours.

Direct Load Control Being Studied

Another method of load shifting being developed by the company is direct control of customer load. Various methods to control customers' use in appliances such as water heaters are being actively studied. These methods of load shifting will offer customers financial incentives to participate voluntarily in such programs. These methods include communications links by way of power lines, radio transmitters or telephone lines. Other benefits, in addition to load shifting, could include the possibility of automatic meter reading, remote monitoring of energy consumption and automatic distribution switching.

These programs are designed to deal with the challenge of meeting customer needs for electricity by focusing attention on the demand side of the energy equation, rather than the traditional supply side. This is in recognition of the changing energy world in which we live. Today, balancing our energy system requires a constant effort at both supplying energy needs and moderating those same needs so they can be met economically and in a socially acceptable manner.

PP&L is reaching into the future with programs to encourage the development of renewable energy resources, while maintaining a firm commitment to balancing today's energy needs through load management and efficient operation of generating plants that use a diversity of fuel sources. Not only do we want to ensure that there are no institutional barriers on our part to development of renewable energy projects, PP&L wants to be a prime motivator in bringing these new technologies into widespread use. It is this type of balanced approach that will help our customers, our company and our country meet the energy challenge before us.

To receive a free copy of the
complete PP&L Profile write to:

Pennsylvania Power & Light Co.
Information Center, TW-4
2 N. 9th St.
Allentown, PA 18101

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101 • 215 / 770-5151

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MARK II CONTAINMENT SUMMARY

ISSUE:

ORIGINAL DESIGN OF MARK II PLANTS DID NOT DIRECTLY CONSIDER THE SUPPRESSION POOL HYDRODYNAMIC LOADS DUE TO SAFETY RELIEF VALVE DISCHARGE AND POSTULATED LOCA EVENTS.

POSITION:

THE SUSQUEHANNA PLANT DESIGN HAS BEEN UPGRADED TO ACCOMODATE VERY CONSERVATIVE SRV AND LOCA LOAD SPECIFICATIONS.

JUSTIFICATION:

PP&L'S MARK II CONTAINMENT PROGRAM EXTENDS BEYOND GENERIC MARK II OWNERS GROUP EFFORT. LOAD SPECIFICATIONS ARE BASED ON FULL SCALE TEST DATA. EXTENSIVE INDUSTRY AND NRC REVIEW OF ENTIRE PROGRAM HAS CONFIRMED CONSERVATISM OF METHODOLOGIES. FINAL PLANT ASSESSMENT UNDERWAY AND WILL BE COMPLETED BY FUEL LOAD.

- SSES DESIGN HAS BEEN UPGRADED TO ACCOMODATE VERY CONSERVATIVE SRV AND LOCA LOADS

- SUSQUEHANNA DESIGN ASSESSMENT UTILIZES BOTH MARK II GENERIC LOADS AND PLANT UNIQUE LOADS

- PLANT UNIQUE LOADS AND PLANT ASSESSMENT ARE DOCUMENTED IN SUSQUEHANNA DESIGN ASSESSMENT REPORT

- REVIEW DOCUMENTED IN SECTION 6.2.1.8 OF SER AND SUPPLEMENT # 1 TO SER

- PLANT ASSESSMENT UTILIZING THESE VERY CONSERVATIVE LOADS IS PROCEEDING

- THROUGHOUT THIS EFFORT WE HAVE FOUND OURSELVES IN A UNIQUE SCHEDULE PROBLEM
    - NOT CLASSIFIED AS A LEAD PLANT
    - NOT A LONG TERM PLANT
- BECAUSE OF THIS WE HAVE HAD TO AGGRESSIVELY ATTACK THE PROBLEM ON OUR OWN
    - T - QUENCHER DEVELOPMENT
    - GKM II M TEST

- T-QUENCHER DEVELOPMENT AND TEST PROGRAM

  - IN 1977 WE INITIATED A PROGRAM TO DESIGN A SPECIFIC QUENCHER DEVICE FOR USE ON SUSQUEHANNA

  - A FULL SCALE TEST PROTOTYPICAL OF SUSQUEHANNA WAS PERFORMED TO VERIFY THIS DESIGN

  - THIS QUENCHER DESIGN IS NOW BEING USED BY SIX OF THE SEVEN OTHER MARK II PLANTS

  - NRC STAFF ACCEPTABILITY

- GKM II M TEST PROGRAM

  - THESE TESTS WERE FULL SCALE SINGLE CELL LOCA TEST PERFORMED IN A PROTOTYPICAL TEST FACILITY AND UNDER PROTOTYPICAL CONDITIONS

  - THESE TESTS HAVE PROVIDED US WITH AN EXTENSIVE DATA BASE FOR SPECIFICATION OF A VERY CONSERVATIVE LOCA STEAM CONDENSATION LOAD

  - THIS LOAD HAS BEEN ADOPTED AS OUR DESIGN BASIS LOCA LOAD FOR PLANT ASSESSMENT

- UPGRADING OF PLANT DESIGN TO INCLUDE HYDRODYNAMIC LOADS HAS RESULTED IN SIGNIFICANT CHANGES

  - ADDITIONS AND MODIFICATION TO CONTAINMENT CONCRETE REINFORCING BARS

  - RE-ROUTING OF SRV LINES

  - INSTALLATION OF T-QUENCHERS ON SRV LINES

  - RE-DESIGN AND REPLACEMENT OF DOWNCOMER BRACING SYSTEM

  - UPGRADED SUPPRESSION POOL TEMPERATURE MONITORING SYSTEM

  - REMOVAL OF MAJOR EQUIPMENT FROM POOL SWELL ZONE IN WETWELL

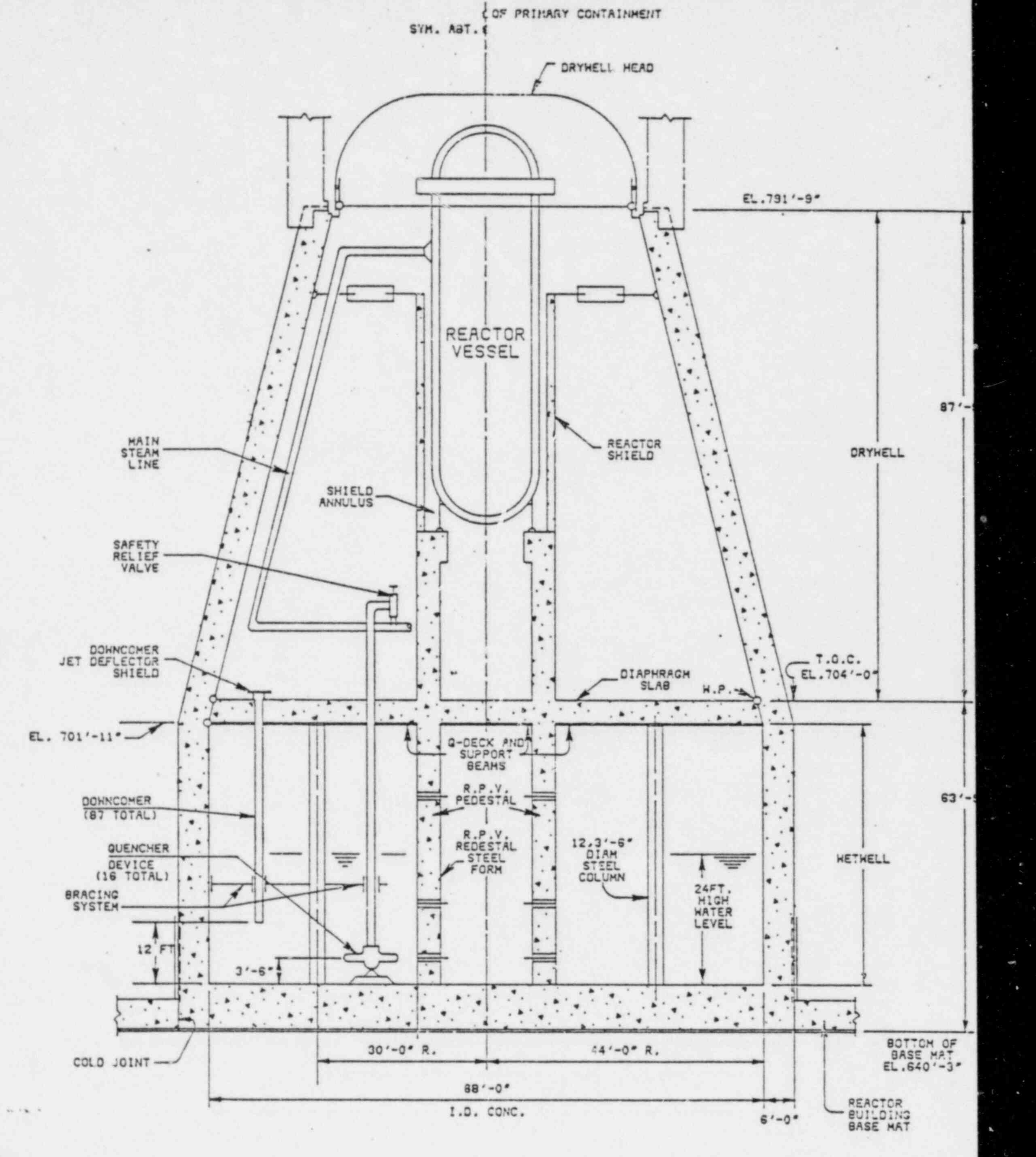
  - RE-DESIGN AND MODIFICATION OF A LARGE NUMBER OF CONTAINMENT AND REACTOR BUILDING PIPING SYSTEMS

CONCLUSIONS:

- SUPPRESSION POOL HYDRODYNAMIC LOADS HAVE BEEN THOROUGHLY INVESTIGATED BY MARK II OWNERS GROUP AND PP&L OVER THE LAST 6 YEARS

- SUSQUEHANNA HAS BEEN DESIGNED TO ACCOMODATE VERY CONSERVATIVE LOADS AND LOAD COMBINATIONS BASED ON A WIDE RANGE OF EXPERIMENTAL DATA AND ANALYTICAL APPROACHES

- BECAUSE OF THESE CONSERVATISMS AND RESULTING PLANT MODIFICATIONS THE PLANT WILL FUNCTION SAFELY IN THE EVENT OF ALL POTENTIAL SAFETY RELIEF VALVE DISCHARGES AND LOCA'S



SUSQUEHANNA STEAM ELECTRIC STATION
CROSS SECTION OF CONTAINMENT

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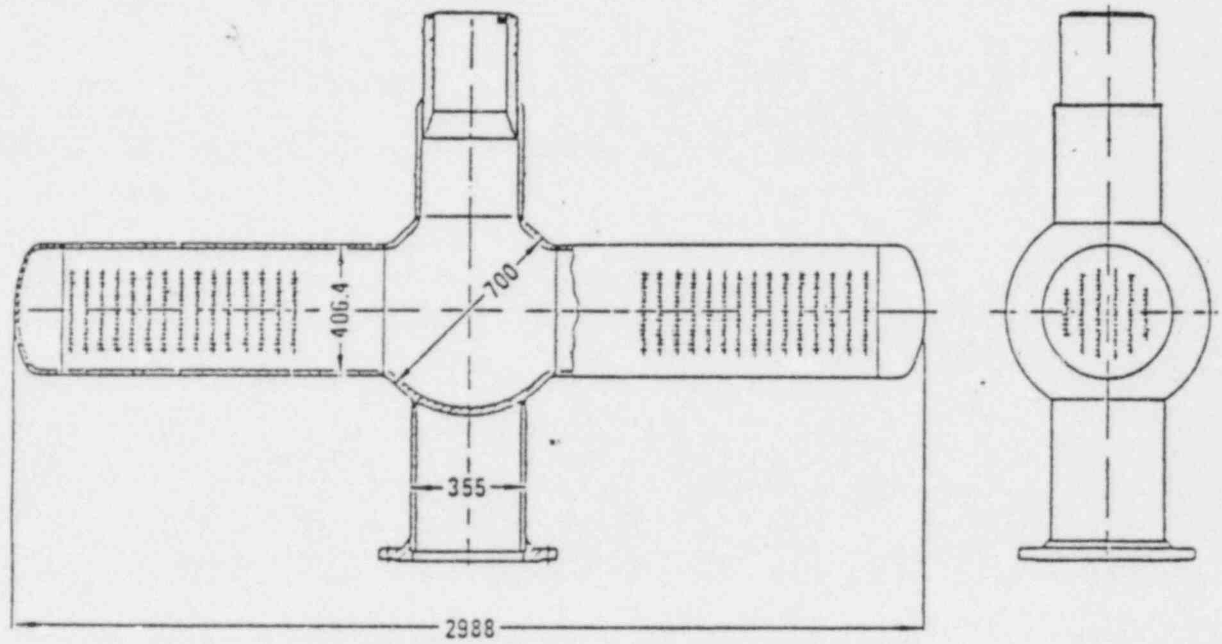
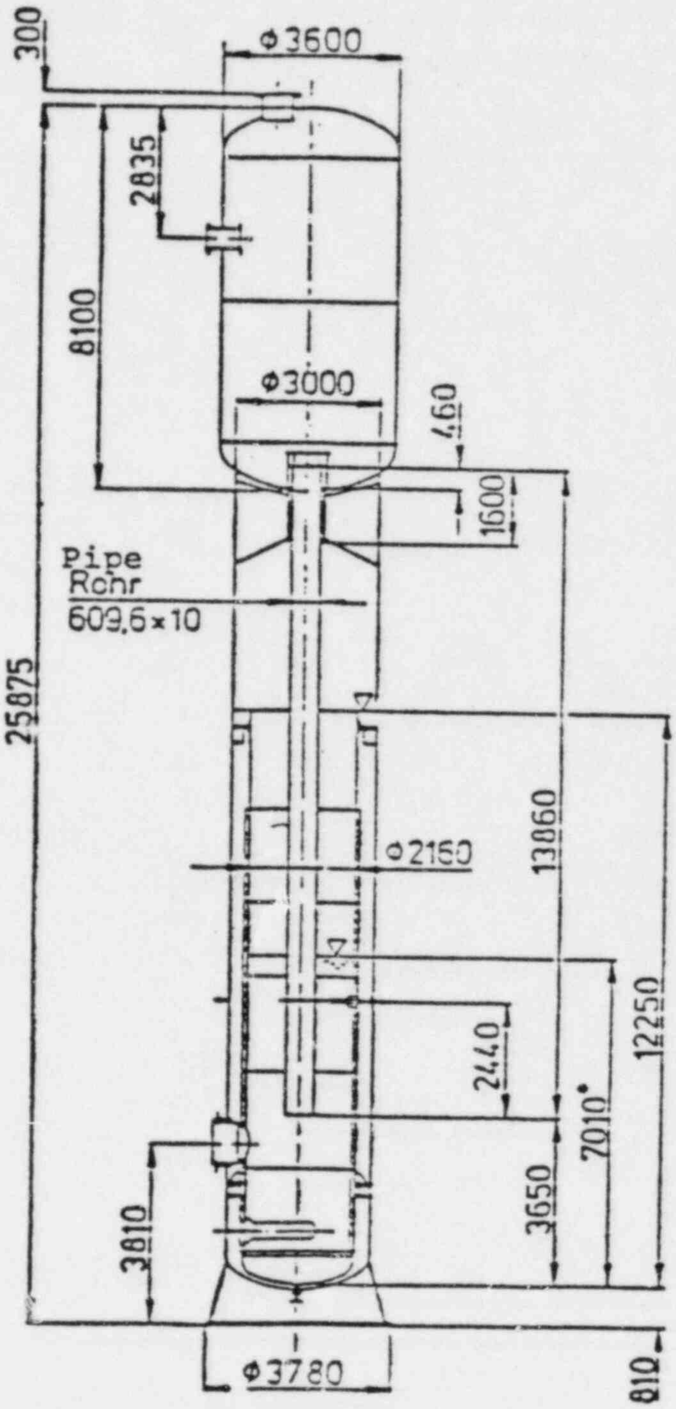


NOTE: All dimensions in mm.

SSES QUENCHER



GKM II-M-Condensation Tests

Test Tank

*Normal Water Level

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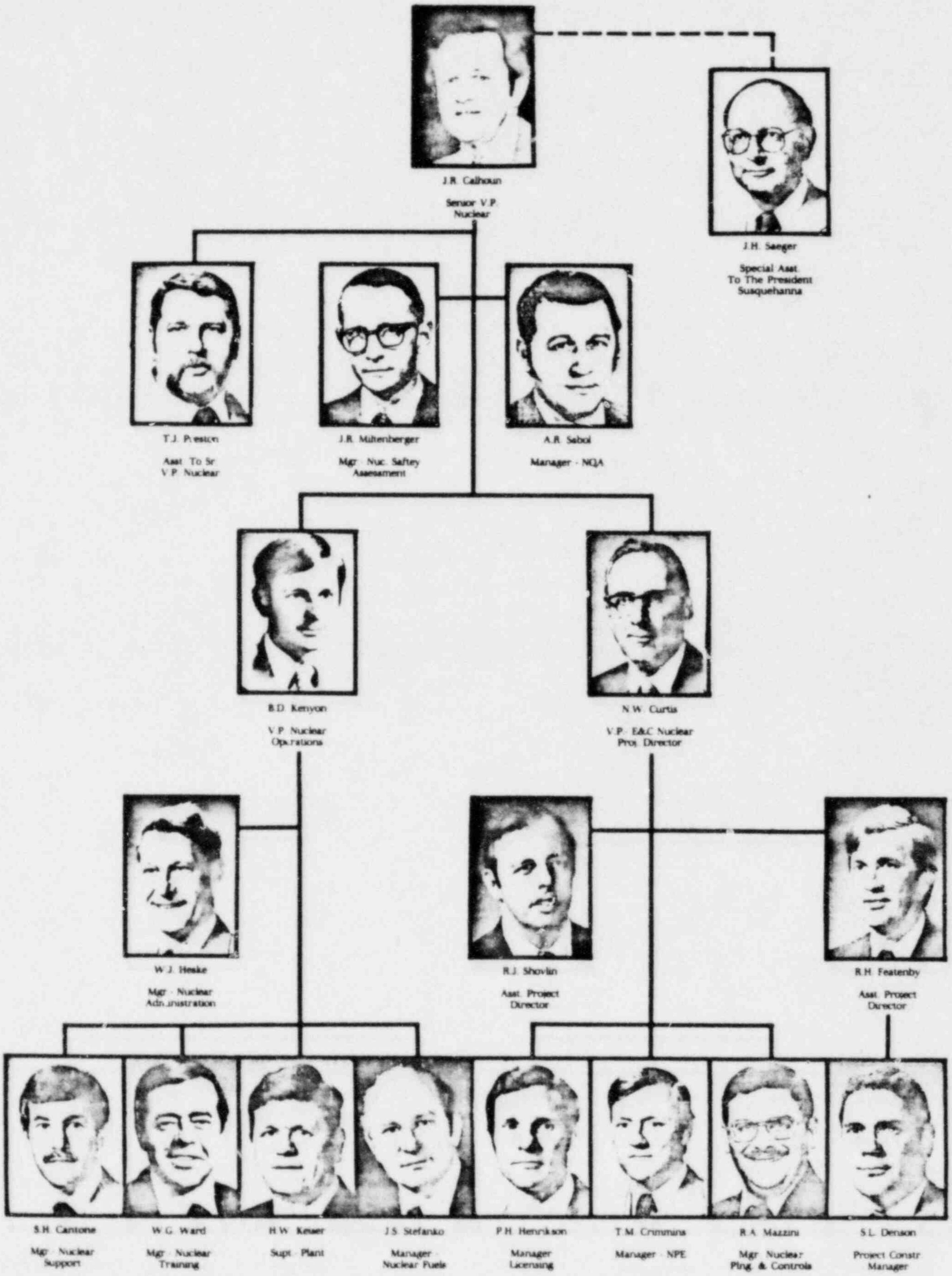
PP&L

Pennsylvania Power & Light Company

Nuclear Department Organization

Nuclear Department Organization



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J. R. Calhoun

Senior Vice President Nuclear

Summary of Position Accountability:
Accountable for providing managerial leadership and strategic direction for the timely and safe construction and operation of the nuclear-powered Susquehanna Steam Electric Station, and for a contribution to long-range growth by effectively participating with top management in formulating corporate goals and objectives.

Address: Senior Vice President - Nuclear
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4194

Resume: Jack R. Calhoun

Education: Tennessee Technological University
BS Electrical Engineering

Additional Misc. Courses: Oak Ridge School of Reactor Technology

General: Member - American Nuclear Society
Past National Chairman, Reactor Operation Division, American Nuclear Society

Consultant to Argonne Universities Association for reactor operations associated with the Experimental Breeder Reactor, Idaho Falls, Idaho

Past Advisor to Nuclear Engineering Department, Pennsylvania State University

Experience:
1980-Present Pennsylvania Power & Light Company
Senior Vice President - Nuclear

1949-1980 Tennessee Valley Authority

1979-1980 Director of Nuclear Power
1977-1979 Assistant Director of Power Production
1971-1977 Chief, Nuclear Generation Branch
1968-1971 Power Plant Superintendent, Browns Ferry Nuclear Plant (Three-1098-MW GE Reactors)
1964-1968 Assistant Chief, Power Plant Maintenance Branch, Division of Power Production

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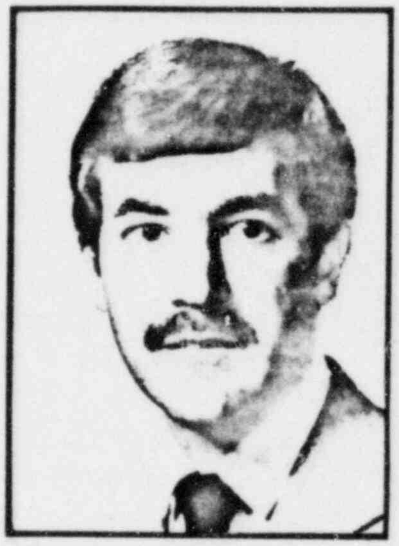
1960-1964 Experimental Gas-Cooled Reactor Operations Superintendent and Assistant Project Manager,)(6 months training on Berkely Nuclear Power Station, Bristol, England)
1958-1960 Assistant Power Plant Superintendent, Shawnee Steam Plant (10-150 MW units)
1954-1958 Electrical Maintenance Supervisor, Johnsonville Steam Plant (6-135 MW units)
1949-1954 Student Generating Plant Operator - Operator, Watts Bar Steam Plant (4-60 MW units)

U.S. Navy

1943-1945 USS Oklahoma City (Cruiser) - Electrical Officer
1941-1943 USS Saratoga (Aircraft Carrier) - Main Propulsion Electrical Officer
1938-1941 USS Dale (Destroyer) - Electrician's Mate



S. H. Cantone
Manager — Nuclear Support

Summary of Position Accountability:
Accountable for a variety of nuclear support services including: special projects and studies for the Vice President-Nuclear Operations, development of major milestone schedules for planned outages, emergency planning, environmental monitoring, health-physics policies and programs, low-level radioactive waste disposal, and availability improvement program.

Address: Manager - Nuclear Support
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4379

Resume: Steven H. Cantone

Education: Stevens Institute of Technology
Bachelor of Engineering

Additional Misc. Courses: Alexander Hamilton Institute Modern Business Program

Experience:

1979-Present Pennsylvania Power & Light Company
Manager - Nuclear Support

1976-1979 Power Authority of the State of New York

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Superintendent of Power - IP3NPP

1963-1976 Consolidated Edison Co. of New York, Inc.

1974-1976 Chief Operations Engineer - Indian Point
1972-1974 Operations Engineer - Indian Point 3
1963-1972 Various Positions of Successively Increasing Responsibility

T. M. Crimmins
Manager - NPE

Summary of Position Accountability:
Accountable for the Company's Nuclear Plant Engineering management to insure that the plant is designed in accordance with all applicable safety and quality standards and modifications designed to meet new regulatory requirements or operational needs.

Address: Manager - Nuclear Plant Engineering
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-5174

Resume: Thomas M. Crimmins, Jr.

Education: College of the Holy Cross, Worcester, MA
BS Physics
New Jersey Institute of Technology, Newark, NJ
MS Engineering Management

Additional Misc. Courses: U.S. Navy Officers Nuclear Power School
U.S. Navy Officers Submarine School

General: Past Advisor to the Electric Power Research Institute, Nuclear Safety and Analysis Department

Member of the Oyster Creek Nuclear Generating Station, General Office Review Board (Offsite Nuclear Safety Committee), 1971-1980

Three Mile Island Unit 1, General Office Review Board (Offsite Nuclear Safety Committee), 1980-1981

Lecturer, Massachusetts Institute of Technology, Nuclear Reactor Safety Summer Session, 1979 to Present

Qualified as Engineering Officer of the Watch (equivalent to Senior Reactor Operator) on two naval nuclear propulsion plants

Experience:

1981-Present Pennsylvania Power & Light Company
Manager - Nuclear Plant Engineering

1970-1981 General Public Utilities System Companies

1980-1981 GPU Nuclear Company, Manager of Engineering Projects
1977-1980 Jersey Central Power & Light Company, Manager - Generation Engineering
1972-1977 GPU Service Corporation, Nuclear Safety and Licensing Manager
1970-1972 GPU Service Corporation, Nuclear Safety and Licensing Engineer

1965-1970 United States Navy Nuclear Power Submarine Program

1968-1970 Newport News Shipbuilding and Drydock Company, Nuclear Submarine Overhaul and Refueling
1966-1968 USS Daniel Webster (SSBN 626) Engineering Division Officer on Nuclear Propulsion Plant
1965-1966 U.S. Navy Nuclear Training

N. W. Curtis

Vice President - E & C-Nuclear, Proj. Director

Summary of Position Accountability:

Accountable for ensuring that design, construction and modifications are completed in a timely manner and conform with engineering drawings and specifications and licensing requirements of the Nuclear Regulatory Commission.

Address: Vice President - Engineering and Construction - Nuclear
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-5381

A-3422

Resume: Norman W. Curtis

Education: University of Maine
B.S. Engineering Physics

Additional Misc. Courses: Executive Program - Columbia University, Graduate School of Business Administration

General: Registered Professional Engineer - Pennsylvania

Charter Member - American Nuclear Society

Member - Institute of Electrical and Electronics Engineers

Past Member - Edison Electric Institute Construction Committee

Experience:

Pennsylvania Power & Light Company

1980-Present V.P. - Engineering & Construction - Nuclear
1974-1980 V.P. - Engineering & Construction
1973-1974 Manager - Engineering & Construction
1972-1973 Construction Manager
1970-1972 Manager - Power Supply
1965-1970 Superintendent - System Operation
1954-1965 Project Engineer, Senior Project Engineer -Atomic Power Department
1950-1954 Helper, Wireman, Foreman - Construction, Construction Department

S. L. Denson

Project Constr. Manager

Summary of Position Accountability:

Accountable for monitoring and auditing site activities on a selective basis, administering site related portions of the Bechtel contract, and construction-site management.

Address: Project Construction Manager - Susquehanna
Susquehanna Steam Electric Station
P.O. Box 467
Berwick, PA 18603
(717) 542-2181, Ext. 2800

A-343

Resume: Stephen L. Denson

Education: United States Naval Academy
B.S. Naval Science

Additional Misc. Courses: U.S. Naval Nuclear Power School & Prototype

U.S. Naval Submarine School

Pennsylvania State University Extension, Management Courses

General: Registered Professional Engineer - Pennsylvania

Member - National Society of Professional Engineers
Pennsylvania Society of Professional Engineers

Past President - Susquehanna Chapter of Pennsylvania Society of Professional Engineers

Experience:
1969-Present Pennsylvania Power & Light Company

1979-Present Project Construction Manager - Susquehanna
1978-1979 Assistant Project Construction Manager - Susquehanna
1975-1978 Construction Starting & Testing Engineer
1972-1975 Senior Project Engineer - Construction
1971-1972 Project Engineer - Construction
1969-1971 Engineer - Mechanical Design

1962-1969 United States Navy

1966-1969 Instructor and Division Director, U.S. Naval Nuclear Power School
1964-1969 USS Andrew Jackon (SSBN 619), Engineering Division Officer
1962-1964 Submarine and Nuclear Power Schools

3432

R. H. Featenby
Assistant Project Director

Summary of Position Accountability:
Accountable during the construction phase for integrating site related activities into the overall project plan including activities related to construction, engineering, startup, operations, and Bechtel and General Electric site management.

Address: Assistant Project Director - Susquehanna - Site
Susquehanna Steam Electric Station
P.O. Box 467
Berwick, PA 18603
(717) 542-2181, ext. 2900

Resume: Robert H. Featenby

Education: Lehigh University, Bethlehem, PA
B.S. Electrical Engineering

Additional Misc. Courses: Project Management Seminar, American Management Research International

Senior Project Management, American Management Association

General: Registered Professional Engineer, Commonwealth of Pennsylvania
Member - Edison Electric Institute, Construction Committee
Chairman, Project Management Subcommittee

Member - Pennsylvania Society of Professional Engineers
Member - National Society of Professional Engineers
Member - Susquehanna Chapter Pennsylvania Society of Professional Engineers

Experience:
1969-Present Pennsylvania Power & Light Company

1979-Present Assistant Project Director - Susquehanna - Site
1978-1979 Project Construction Manager - Susquehanna
1978 Project Manager - Generation Projects
1977-1978 Assistant Project Director - Anthracite Generation Project
1976-1977 Project Manager - Martins Creek
1971-1976 Construction Project Supervisor - Martins Creek
1970-1971 Construction Project Engineer - Martins Creek
1969-1970 Lines and Substation Engineer - Substations

1966-1969 Ebasco Services Incorporated

1966-1969 Resident Office Engineer, Millstone Nuclear Power Station, Unit 1, Waterford, CT

1966 United Engineers and Constructors, Inc.

1966 Electrical Supervisor, Sioux Power Station, Portage de Sioux, MO

1962-1966 Pennsylvania Power & Light Company

1962-1964 Lines and Substation Engineer
1964-1966 Construction Project Engineer, Brunner Island Unit 2

P. H. Henrikson
Manager Licensing

Summary of Position Accountability:
Accountable for tracking progress and preparing documents which identify activities necessary to obtain an operating license and for managing those licensing activities performed by the Nuclear Licensing Group.

Address: Manager - Nuclear Licensing
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4172

Resume: Philip H. Henrikson

Education: B.S. in Chemistry, University of Nevada
M.S. in Chemical Engineering (minor in Nuclear Engineering), University of Idaho
Juris Doctorate, Lincoln University

Additional Misc. Courses: U.S. Navy Submarine Officer Nuclear Training Program

General: Registered Professional Engineer - Nuclear

Member-American Nuclear Society

Member-American Bar Association

A 3442

Experience:

1980-Present	Manager - Nuclear Licensing, Pennsylvania Power & Light Company
1973-1980	General Electric - Served in Safety and Licensing Operation of the Nuclear Energy Division. Program Manager for GE Involvement in NRC's Systematic Evaluation Program.
1971-1973	Bechtel - Nuclear Systems generic group. Later was Assistant Supervisor for Rancho Seco Nuclear Power Station.
1966-1970	U.S. Naval Officer
1964-1966	Laboratory Assistant, Desert Research Institute, Atmospheric Physics Division

W. J. Heske

Manager - Nuclear Administration

Summary of Position Accountability:
Accountable for records management document control; financial administration and budget coordination; nuclear personnel, policies, programs, and procedures; and publication of a departmental newsletter.

Address: Manager - Nuclear Administration
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4161

Resume: William J. Heske

Education:
U.S. Naval Academy
B.S. Engineering
Auburn University
M.S. Political Science

Additional Misc. Courses:
Basic Personnel Officers Course - USAF
Minuteman Combat Crew Training - USAF
D.G., Air Command Staff College - USAF
D.G., Systems Analysis & Operations Research - U.S. Army
D.G., Industrial College of the Armed Forces

Experience:

1979-Present Pennsylvania Power & Light Company

1980-Present Manager - Nuclear Administration
1979-1980 Assistant to Vice President Engineering & Construction

1957-1979 United States Air Force

1976-1979 Deputy Commander for Maintenance and Director Logistics F.E. Warren AFB
1975-1976 Student - Industrial College of the Armed Forces, U.S. Army
1972-1975 Chief - Future Missile Systems Analysis, Strategic Air Command
1971-1972 Student - Air Command & Staff College
1968-1971 Chief Missile, Space, and Weapons Director Officer Manning
1965-1968 Chief Missile Manning Branch Strategic Air Command
1962-1965 Minuteman Crew Commander, Malmstrom AFB
1957-1962 Personnel & Administrative Officer

H. W. Keiser

Superintendent - Plant

Summary of Position Accountability:
Accountable for the safe and efficient startup, operation and maintenance of the Susquehanna units in accordance with license requirements.

Address: Superintendent of Plant - Susquehanna
Susquehanna Steam Electric Station
P.O. Box 467
Berwick, PA 18603
(717) 542-2181, ext. 220

Resume: Harold William Keiser

Education: 1972 - University of Illinois, Urbana, Illinois
Bachelor of Science Degree in Metallurgical Engineering

1973 - University of Illinois, Urbana, Illionois
Master of Science Degree in Nuclear Engineering

A-3452

General: Licenses and Certificates:

Senior Reactor Operator license, University of Illinois, Triga Reactor

Senior Reactor Operator license, Palisades Nuclear Plant

Experience:

1980-Present	Pennsylvania Power & Light Company Superintendent of Plant - SSES
1979-1980	Consumers Power Company, Operations/Maintenance Superintendent, Palisades Nuclear Plant
1976-1979	Operations Superintendent, Palisades Nuclear Plant
1976-1976	Senior Engineer, Nuclear Production Department
1973-1976	Senior Engineer, Palisades Nuclear Plant
1973-1975	General Engineer, Palisades Nuclear Plant
1968-1973	University of Illinois
1961-1968	U.S. Navy

B. D. Kenyon

Vice President Nuclear Operations

Summary of Position Accountability:

Accountable for the safe and efficient startup, operation and maintenance of the Susquehanna units, plus support services including: nuclear fuel procurement, analysis, and disposition; nuclear training; records management; nuclear personnel policies, programs, and procedure; environmental monitoring; emergency planning; and health physics policies and programs.

Address: Vice President - Nuclear Operations
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4378

Resume: Bruce D. Kenyon

Education: Miami University
B.A. Mathematics

Additional Misc. Courses: U.S. Navy Submarine Officer Nuclear Training Program

Senior Operator qualification at the Navy SIC Nuclear Prototype (CE PWR)

Senior Operator qualification on USS George Washington (SSBN 598) (W PWR)

Senior Operator qualification at the Navy DIG Nuclear Prototype (GE PWR)

NRC Senior Operators License on Millstone Unit 1 (GE BWR)

NRC Senior Operators License on Millstone Unit 2 (CE PWR)

General: Member-Edison Electric Institute Nuclear Operations Committee

Past Member-Edison Electric Institute Construction Committee

Member-Atomic Industrial Forum Design, Construction, and Operations Committee

Member-Institute of Nuclear Power Operations Analysis and Special Projects Division - Industry Review Committee

Past Member-Electric Power Research Institute Engineering and Operations Committee

Experience:

1976-Present Pennsylvania Power & Light Company

1980-Present Vice President - Nuclear Operations
1979-1980 Assistant Vice President - Nuclear
1978-1979 Construction Manager
1976-1978 Manager - Nuclear Support

1970-1976 Northeast Utilities - served successively as Startup Engineer, Startup Supervisor, Operations Supervisor, and Unit Superintendent on Millstone Unit 2.

1965-1970 U.S. Navy - Division Officer on the USS George Washington (SSBN 598). Also served as Leading Engineering Watch Officer at the DIG Prototype.

R. A. Mazzini

Manager - Nuclear Planning & Controls

Summary of Position Accountability:

Accountable for coordinating all planning, scheduling and cost control activities required for plant construction, modification and operation.

Address: Manager - Nuclear Planning & Controls
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4125

Resume: Richard A. Mazzini

Education: Villanova University
Bachelor of Electrical Engineering

Columbia University
M.S. Nuclear Engineering

Additional Misc. Courses: Columbia University - Post graduate courses in Nuclear Engineering

General Electric BWR Design Course

General Electric Simulator Training (Dresden)

General: Registered Professional Engineer - Pennsylvania (Nuclear)

Certified Cost Engineer

Experience: Pennsylvania Power & Light Company

1980-Present Manager - Nuclear Planning & Controls
1977-1980 Supervisor - Project Cost & Schedule
1975-1977 Resident Contractor Liason Engineer
1973-1975 Senior Project Engineer - Nuclear
1970-1973 Project Engineer - Nuclear
1967-1970 Engineer - Atomic Power Division

J. R. Miltenberger

Manager - Nuclear Safety Assessment

Summary of Position Accountability:

Accountable for developing and implementing programs to assess the safety/environmental aspects of PP&L's nuclear related activities.

Address: Manager - Nuclear Safety Assessment
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4127

Resume: James R. Miltenberger

Education: Pennsylvania State University
B.S. Civil Engineering

Rensselaer Polytechnic Institute
M.S. Management

Additional Misc. Courses: U.S. Navy Submarine Pre-Command Nuclear Power Training

S-5-W Reactor Design Course, Bettis Atomic Power Laboratory

U.S. Navy Officer's Nuclear Power School

U.S. Navy Officer's Submarine School

General: Wrote several papers on nuclear submarine technology and research and development

Chaired the Naval Sea System Command Submarine Fire Protection Committee

Qualified for command of nuclear submarine.

Qualified as Engineer Officer of a nuclear submarine.

Qualified as Engineering Officer of the Watch (equivalent to Senior Reactor Operator) on three naval nuclear propulsion plants.

Experience:
1981-Present Pennsylvania Power and Light Company - Manager - Nuclear Safety Assessment.

A-347a

1979-1981 Columbia Research Corporation - Director ASW Systems

1958-1979 U.S. Navy

1977-1979 Executive Assistant to the Deputy Commander for Submarines
1975-1977 Program Manager - R&D for Attack Submarine Project
1972-1975 Commanding Officer U.S.S. Sam Rayburn (SSBN 635)
1970-1972 Executive Officer U.S.S. Abraham Lincoln (SSBN 602)
1969-1970 Student Rensselaer Polytechnic Institute
1967-1969 Staff Nuclear Power Training Unit Windsor - for last four months served as Commanding Officer
1965-1967 Engineer Officer U.S.S. James K. Polk (SSBN 645)
1963-1965 Engineering Division Officer U.S.S. James Monroe (SSBN 622)
1958-1963 Served on a destroyer and a diesel submarine - attended submarine school and nuclear power training

T. J. Preston
Assistant to Senior Vice President Nuclear

Summary of Position Accountability:
Provides liaison between the Nuclear Department and the Financial Department in developing and implementing systems to account for post-commercial construction, operation, maintenance, and co-ownership activities of the Nuclear Department.

Address: Assistant to Senior Vice President - Nuclear
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-4654

Resume: Thomas J. Preston

Education: B.B.A. in Management, St. Francis College
M.B.A. in Accounting, Pace University

General: Past member and Vice Chairman of the Pennsylvania Electric Association's General Accounting Committee

Experience:
1972-Present Pennsylvania Power & Light Company

Present Assistant to Senior Vice President - Nuclear
1980-1981 Assistant to Executive Vice President - Financial
1973-1980 Chief - General Accounting
1972-1973 Senior Accountant - Auditing

1967-1972 Arthur Andersen & Company

1970-1972 Senior Analyst - Administrative Services Division
1967-1970 Senior Accountant - Audit Division

1964-1967 Manufacturers Hanover Trust Company - Income Tax Accountant

A. R. Sabol

Manager - Nuclear Quality Assurance

Summary of Position Accountability:

Accountable for the administration, development, and coordination of a Quality Assurance Progam which complies with Government regulatory requirements as they apply to the construction and operation of the Susquehanna Steam Electric Station.

Address: Manager - Nuclear Quality Assurance
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-5930

Resume: Andrew R. Sabol

Education: Purdue University
B. S. Mechanical Engineering

Additional Misc. Courses: Undergraduate and graduate courses in Business Administration-Pennsylvania State University

General: Professional Engineer Registration - Quality Engineering - California
Member-American Society for Nondestructive Testing

Member-American Society for Testing and Materials

A-3482

Experience:

1974-Present	Pennsylvania Power & Light Company Manager - Nuclear Quality Assurance
1971-1974	Gilbert Associates, Inc., Quality Assurance Division, Quality Assurance Program Manager
1967-1971	The Pennsylvania State University, Manager - Industrial Reference
1960-1967	U.S. Atomic Energy Commission **1963-1967** Reactor Production Engineer, Pittsburgh Naval Reactors Office **1961-1963** Nuclear Production Engineer, Pittsburgh Naval Reactors Office **1960-1961** General Engineer, Hartford Area Office Connecticut Aircraft Nuclear Propulsion Project
1956-1960	Curtis - Wright Corporation, Nuclear Science and Engineering Department, Research Engineer
1954-1956	Bethlehem Steel Corporation, Technical Assistant to Superintendent of Hot Forge
1949-1950	Department of the Air Force, Far Eastern Area Material Command, Japan, Supervising - Engineering Drafting

J.H. Saeger

Special Assistant to the President Susquehanna

Summary of Position Accountability:
Accountable for keeping the public informed on nuclear issues and concerns and Susquehanna plans impacting the surrounding local communities, especially those related to emergency planning, and for answering public questions relative to company plans and operations at Susquehanna on behalf of the President.

Address:

Special Assistant to the President -
Susquehanna Community Representative
Pennsylvania Power & Light Company
1009 Fowler Avenue
Berwick, PA 18603
(215) 770-4382

Resume: John H. Saeger

Education: Lafayette College
B.S. Electrical Engineering

Additional Misc. Courses: Power Systems Engineering Course, General Electric Co.

Electric Utility Engineering Conference, Westinghouse Electric Corporation

General: Registered Professional Engineer in Pennsylvania

Member - Institute of Electrical and Electronic Engineers

Member - Power Engineering Society

Member - National Society Professional Engineers

Member - Edison Electric Institute, Electrical System and Equipment Committee

Past Chairman - Electric Power Research Institute, Overhead Transmission Line Advisory Task Force

Experience:
1960-Present Pennsylvania Power & Light Company

1981-Present Special Assistant to the President-Susquehanna Community Representative
1978-1981 Manager-Bulk Power Engineering
1977-1978 Acting Manager-Bulk Power Engineering
1976-1977 Manager-Transmission Engineering
1975-1976 Project Manger-Energy Park Development
1973-1975 Manager-Transmission Engineering
1970-1973 Transmission Line Engineer
1968-1970 Senior Project Engineer-Transmission
1964-1968 Project Engineer-Transmission
1960-1961 Student Trainee

A-3492

R. J. Shovlin
Assistant Project Director

Summary of Position Accountability:

Accountable for providing direction to home-office related project activities, including activities related to engineering, licensing, planning and controls, and administration of the Bechtel and General Electric contracts.

Address: Assistant Project Director - Allentown
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-5709

Resume: Robert J. Shovlin

Education: Lehigh University
B.S. Electric Engineering

Additional Misc. Courses: General Electric Company Power Systems Engineering Course

Pennsylvania Power & Light Company - Nuclear Power Seminar

Graduate Courses - Lehigh University M.S. in Electrical Engineering

General: Registered Professional Engineer, State of Pennsylvania

Member-Atomic Industrial Forum Design, Construction and Operations Committee

Past Member Institute of Electrical and Electronics Engineers

Experience:
1962-Present Pennsylvania Power & Light Company

1976-Present Project Manager/Assistant Project Director-Allentown - Susquehanna Project
1973-1976 Manager - Martins Creek Project
1971-1973 Electrical Research and Development Engineer
1965-1971 Senior Project Engineer - Relay and Control Engineering
1964-1965 Project Engineer - Substation Engineering Department
1962-1964 Engineer - Distribution Engineering Department

J. S. Stefanko

Manager - Nuclear Fuels

Summary of Position Accountability:

Accountable for the management of the nuclear fuel cycle from acquisition of nuclear fuel and related goods and services to the ultimate disposition of spent fuel.

Address: Manager - Nuclear Fuels
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA 18101
(215) 770-5927

Resume: Jerome S. Stefanko

Education: University of Dayton
B.S. Physics

Additional Misc. Courses: Graduate studies in Physics, Mathematics and Nuclear Engineering

General: Member-Edison Electric Institute - Nuclear Fuel Cycle Committee

Member-Atomic Industrial Forum - Policy Committee on Uranium Mining and Milling

Member-American Nuclear Society

Author of six papers to National Technical Societies

Holder of two patents for Nuclear Reactor Startup and Controls Systems (under USAF sponsorship)

Experience:

1976-Present Pennsylvania Power & Light Company, Manager - Nuclear Fuels

1971-1976 Westinghouse Electric Corporation, Applications Engineer-Nuclear Fuels Marking Division

1963-1971 Astronuclear Laboratory of Westinghouse Electric Corporation, Senior Nuclear Design Engineer

1960-1963 Allis-Chalmers' Nuclear Division, Core Physicist

A-3502

W. G. Ward
Manager - Nuclear Training

Summary of Position Accountability:
Accountable for providing and/or coordinating all essential training activities for nuclear personnel, including the operation of a simulator training facility to insure all department personnel are effectively trained in compliance with NRC and corporate requirements.

Address: Manager - Nuclear Training
Susquehanna Steam Electric Station
P.O. Box 467
Berwick, PA 18603
(717) 542-2140

Resume: William G. Ward

Education: 1973 Oklahoma State University, Doctor of Education

1970 Oklahoma State Unversity, Master of Science

1964 Oklahoma State University Bachelor of Science

1962 Sayre Junior College, Sayre, Oklahoma, Associate of Arts

Additional Misc. Courses: **Currently** pursuing a Master of Science in Special Education - twenty-one hours completed as of January 1, 1981.

General: Member American Vocational Association, Phi Delta Kappa, American Association of Community and Junior Colleges, United School Administrators, Society of Military Engineers, Kiwanis, American Technical Education Association, Kansas Technical Society.

Experience:

1981-Present Pennsylvania Power & Light Company, Manager - Nuclear Training

1974-1981 Pittsburg State University

1978-1981 Director of the Vocational Technical Institute at Pittsburg State University
1974-1978 Associate Professor, Department of Vocational and Technical Education, School of Technology and Applied Science, and Director of the Kansas Vocational Curriculum and Research Center, Pittsburg State University

1970-1974 Oklahoma State University

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1973-1974 Assistant Coordinator of Planning in the Division of Research, Planning, and Evaluation in the Oklahoma State Department of Vocational and Technical Educaton and as a Research Associate with the School of Occupational and Adult Education, College of Education
1970-1973 Research Assistant to the Head of the Division of Research, Planning and Evaluation of the Oklahoma State Department of Vocational and Technical Education

1966-1969 United States Marine Corps. Platoon/Company Commander

1964-1966 Trade and Industrial Carpentry Teacher

1958-1964 Construction work

P. D. Cargen

MANAGEMENT AND ORGANIZATION

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PENNSYLVANIA POWER & LIGHT

NUCLEAR ORGANIZATION & STAFFING

- COMPLIANCE WITH NUREG 0731
- ENHANCEMENTS
- MANAGEMENT PHILOSOPHY

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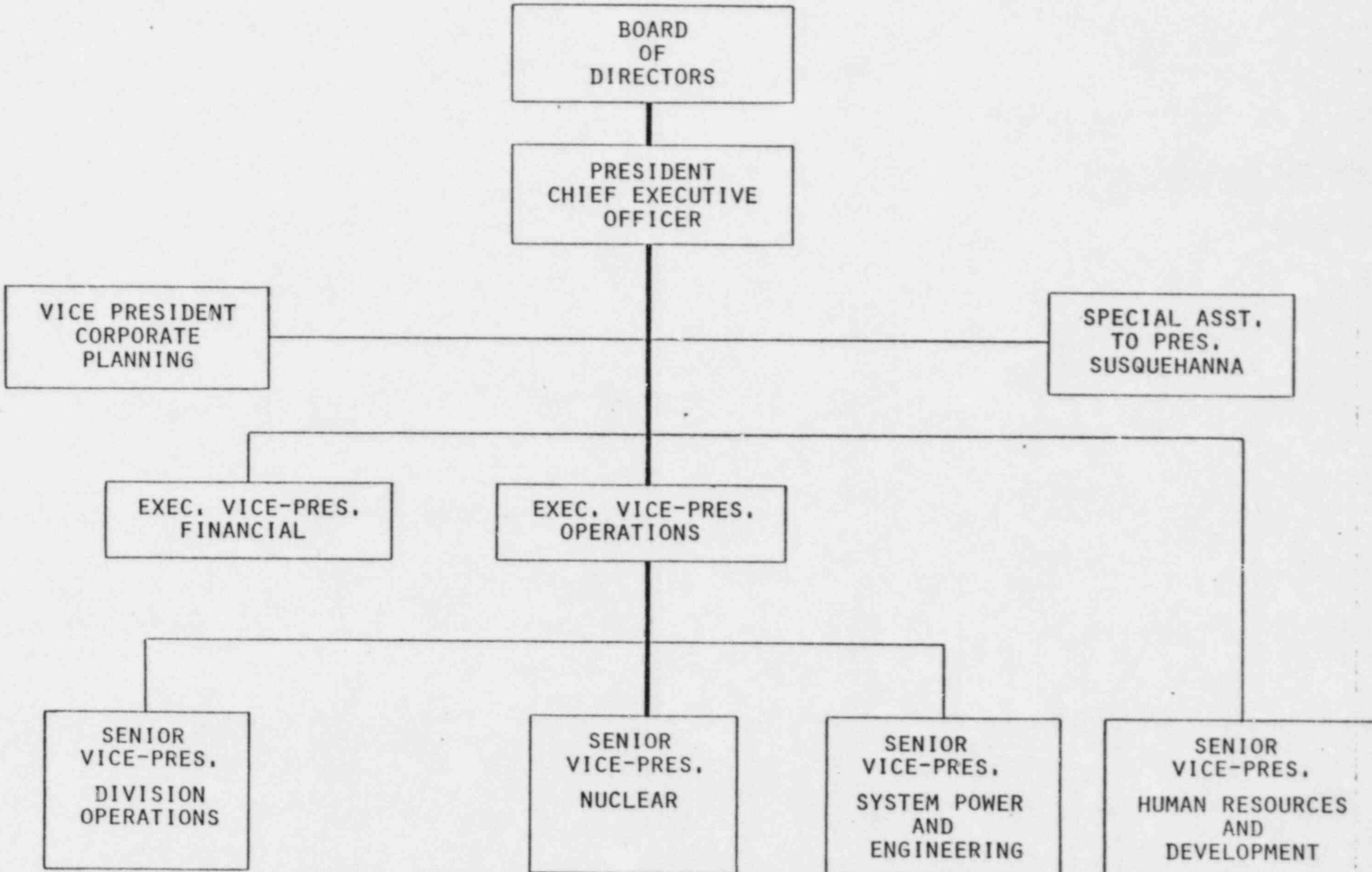
- BACKGROUND
- POST TMI ASSESSMENT
  - DESIGN
  - ORGANIZATION AND STAFFING
  - RADIATION MONITORING
  - EMERGENCY PLANS
  - COMMUNICATIONS
- RESULT

CORPORATE MANAGEMENT ORGANIZATION



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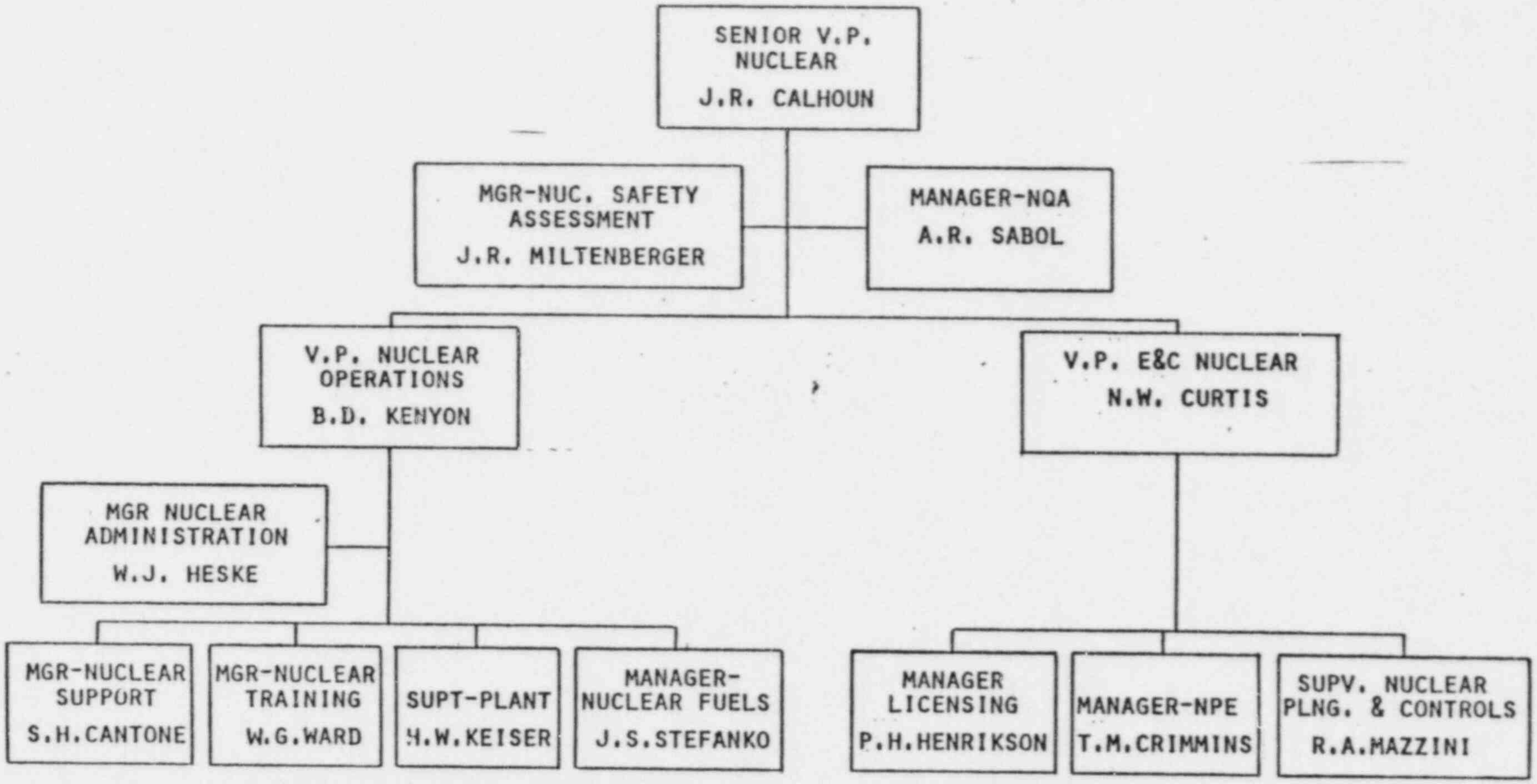
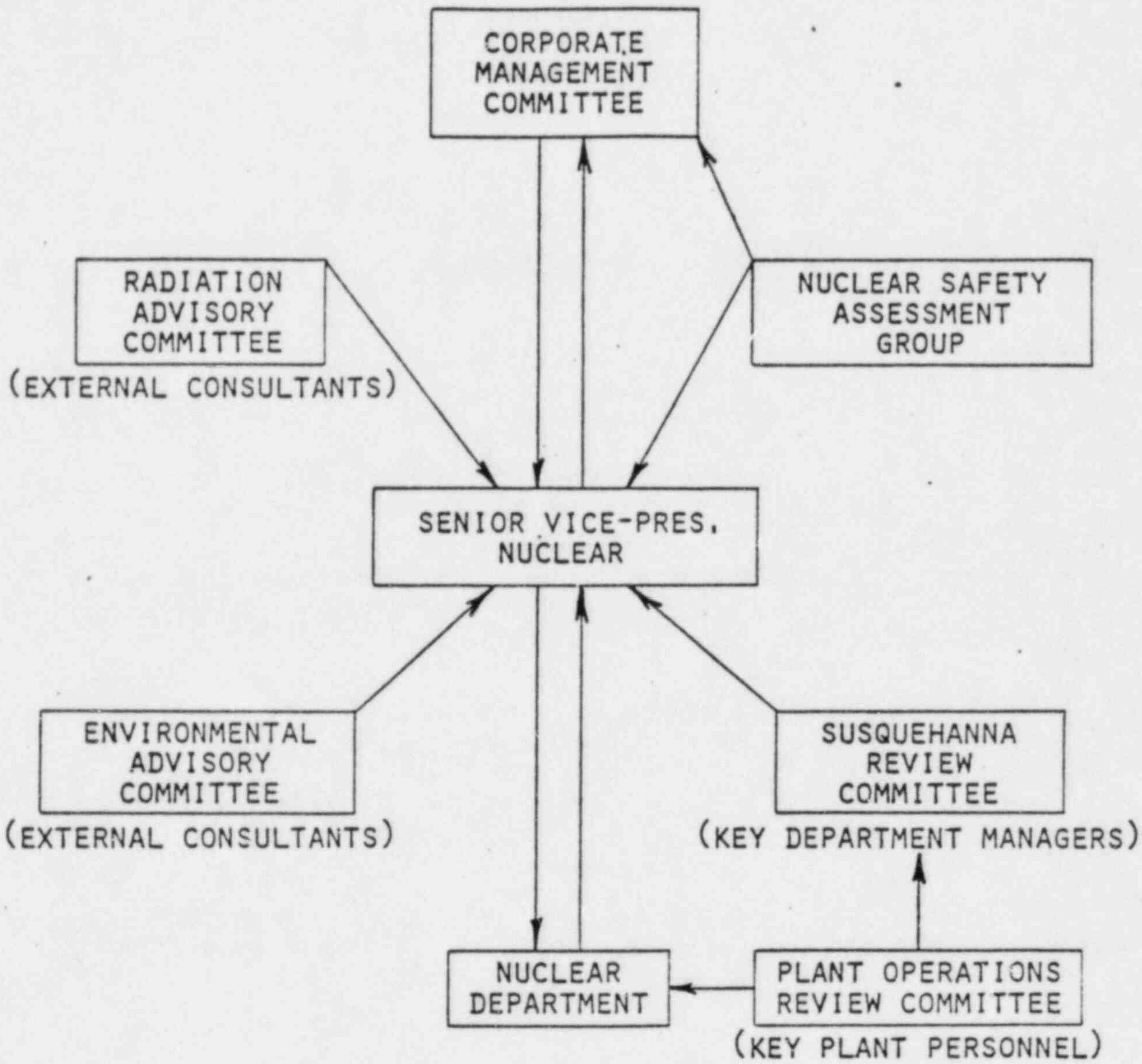
NUCLEAR DEPARTMENT ORGANIZATION



A-357

TRAINING

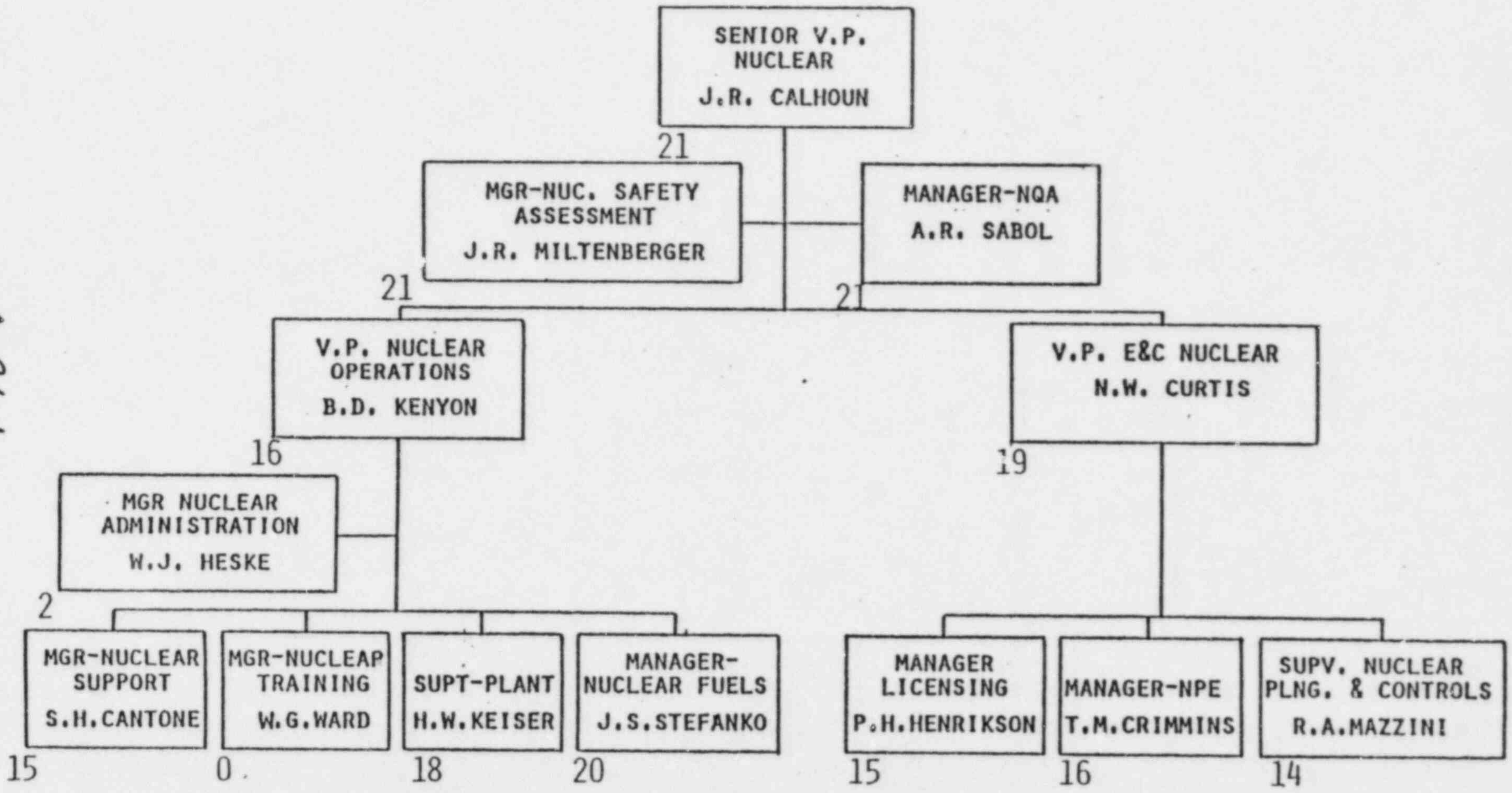
- CONSOLIDATION
- REPORTING RELATIONSHIP
- MANAGER OF NUCLEAR TRAINING
- COMPREHENSIVE
- SIMULATOR


CORPORATE MANAGEMENT COMMITTEE
RADIATION ADVISORY COMMITTEE
(EXTERNAL CONSULTANTS)
NUCLEAR SAFETY ASSESSMENT GROUP
SENIOR VICE-PRES. NUCLEAR
ENVIRONMENTAL ADVISORY COMMITTEE
(EXTERNAL CONSULTANTS)
SUSQUEHANNA REVIEW COMMITTEE
(KEY DEPARTMENT MANAGERS)
NUCLEAR DEPARTMENT
PLANT OPERATIONS REVIEW COMMITTEE
(KEY PLANT PERSONNEL)

NUCLEAR ORGANIZATION

- SINGULAR PURPOSE
- CLEAR RESPONSIBILITIES AND AUTHORITY
- EFFECTIVE PROCEDURE PROGRAM
- GOOD COMMUNICATIONS
  - VERTICAL
  - HORIZONTAL
  - EXTERNAL

NUCLEAR RELATED EXPERIENCE---YEARS

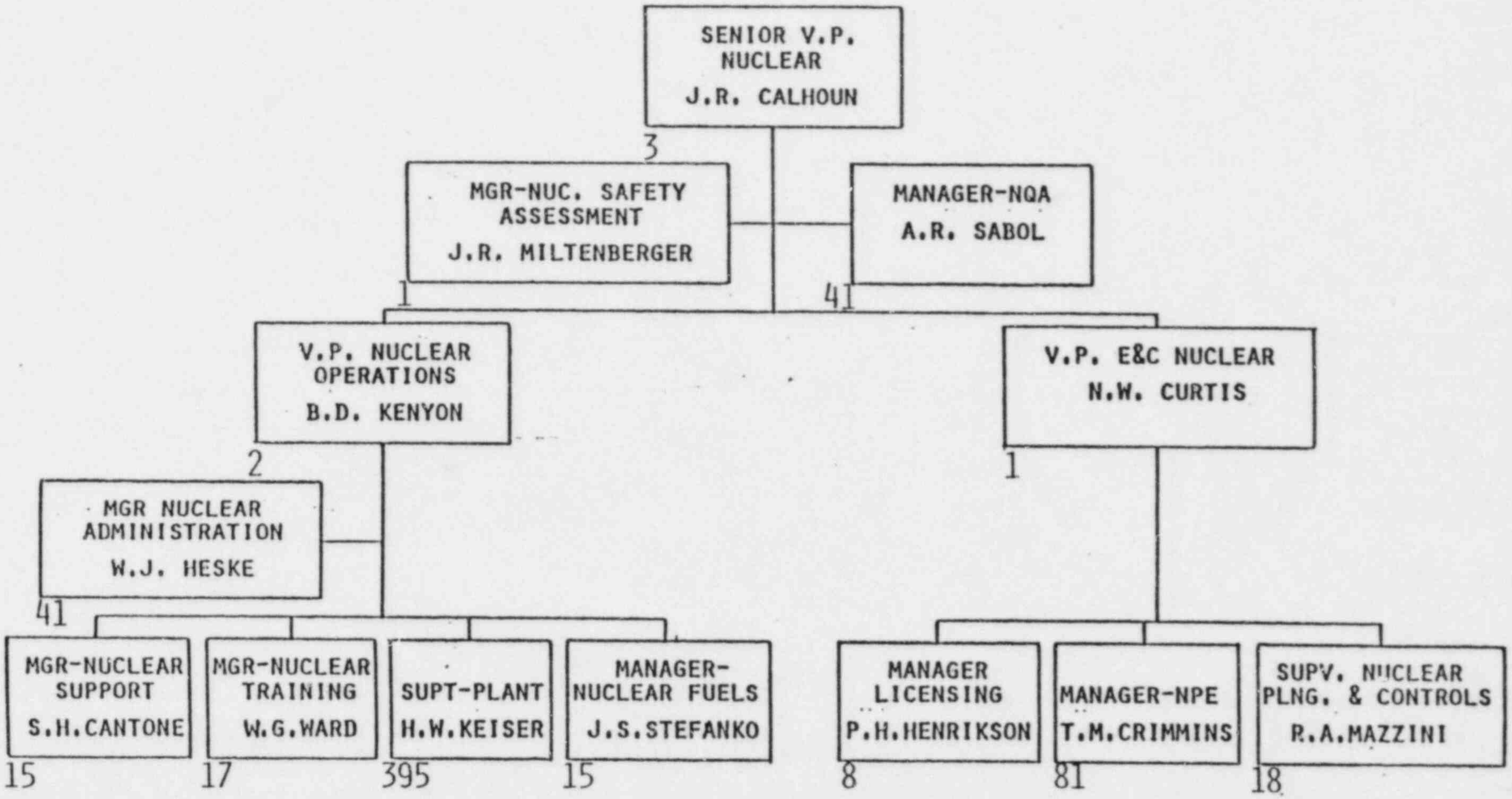


TOTAL -- 198 MAN YEARS

A-361

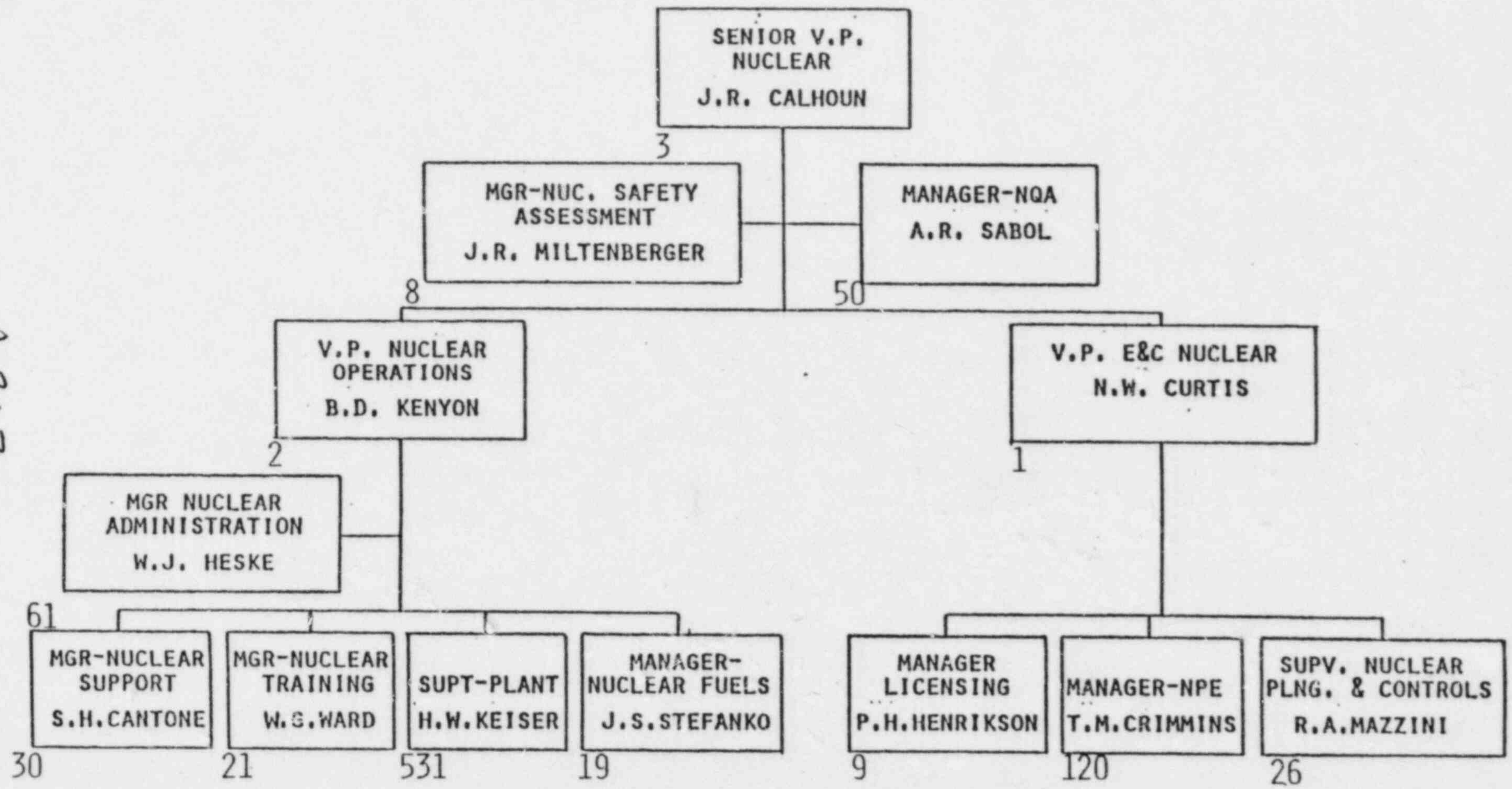
NUCLEAR DEPARTMENT MANPOWER AS OF 5/81

A-362



TOTAL PERSONNEL AS OF MAY 24, 1981 ... 732

NUCLEAR DEPARTMENT MANPOWER, BUDGETED 12/82



TOTAL PERSONNEL BUDGETED, END OF 1982 ... 881

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NUCLEAR DEPARTMENT EXPERIENCE

A-364

ON-SITE

TOTAL ---------------------- 1,388 MAN-YEARS

NUCLEAR ------------------ 1,038 MAN-YEARS

OFF-SITE

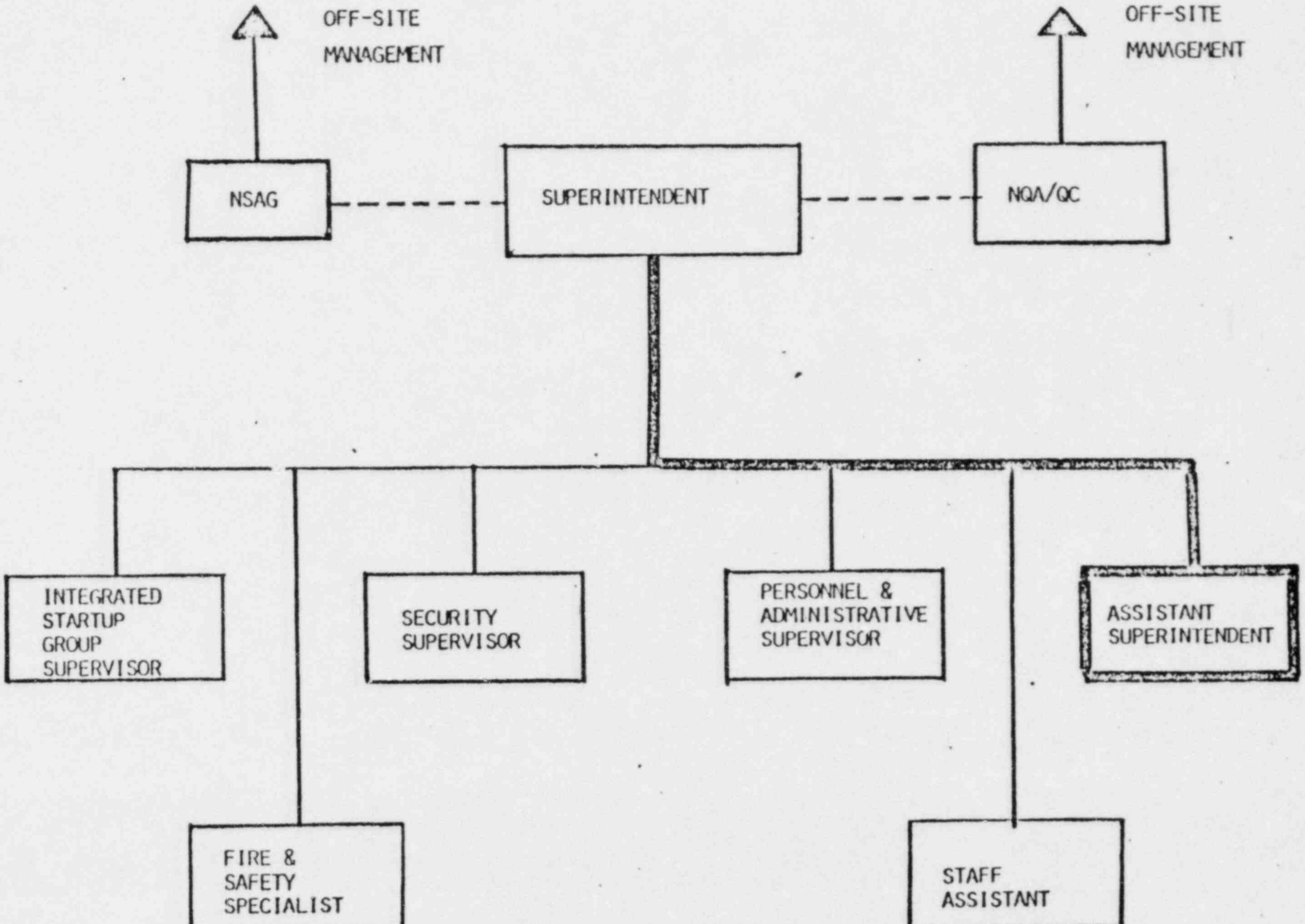
TOTAL ---------------------- 1,616 MAN-YEARS

NUCLEAR ------------------ 1,081 MAN-YEARS

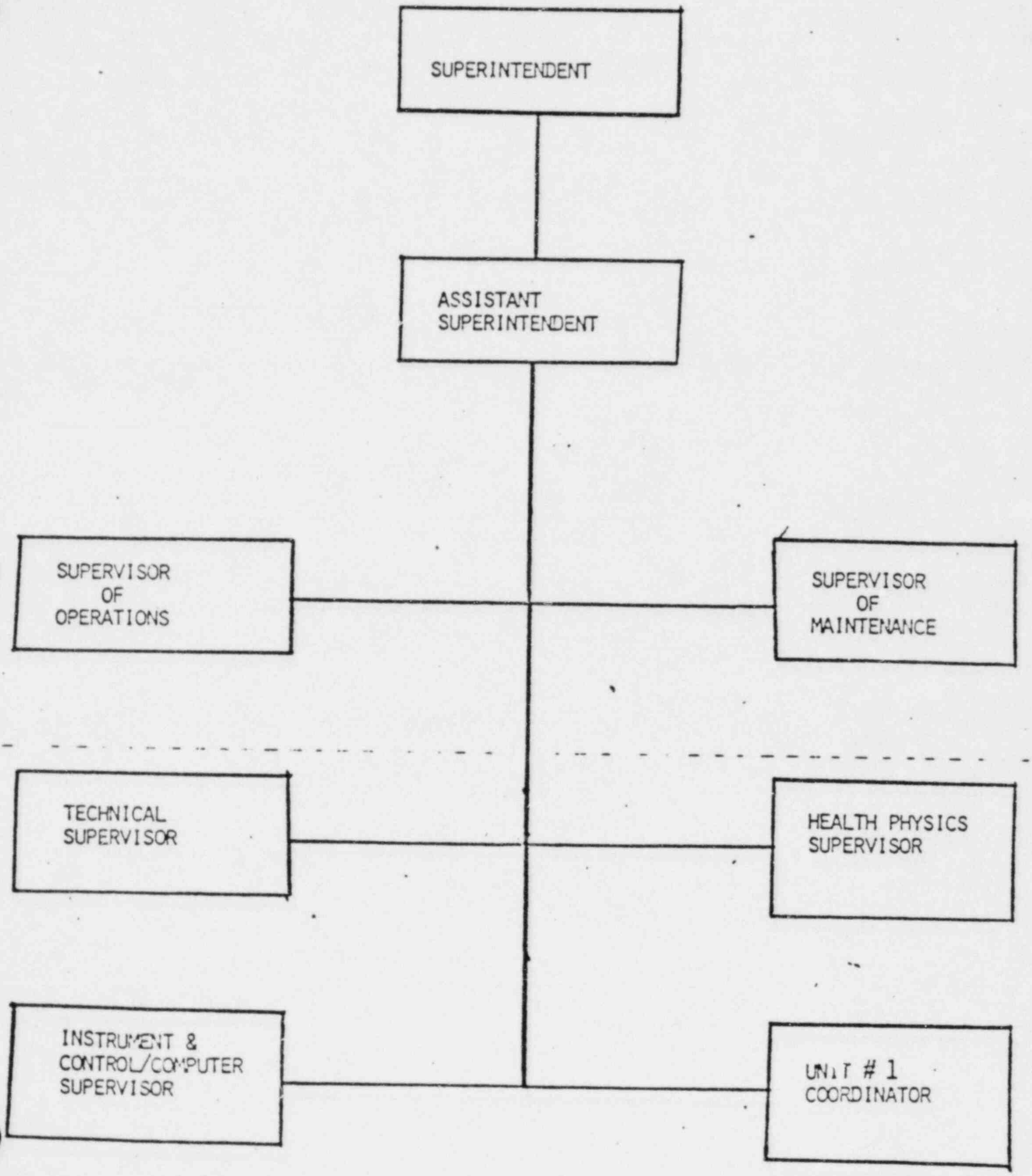
SUMMARY

- ORGANIZATION
- TOP MANAGEMENT INVOLVEMENT
- REVIEW & ASSESS
- STAFFING AND EXPERIENCE LEVELS
- INNOVATION
- GOAL

A-36 1/2


OFF-SITE MANAGEMENT
OFF-SITE MANAGEMENT
NSAG
SUPERINTENDENT
NQA/QC
INTEGRATED STARTUP GROUP SUPERVISOR
SECURITY SUPERVISOR
PERSONNEL & ADMINISTRATIVE SUPERVISOR
ASSISTANT SUPERINTENDENT
FIRE & SAFETY SPECIALIST
STAFF ASSISTANT

A-365


SUPERINTENDENT
ASSISTANT SUPERINTENDENT
SUPERVISOR OF OPERATIONS
SUPERVISOR OF MAINTENANCE
TECHNICAL SUPERVISOR
HEALTH PHYSICS SUPERVISOR
INSTRUMENT & CONTROL/COMPUTER SUPERVISOR
UNIT # 1 COORDINATOR

SUSQUEHANNA PLANT STAFF
ORGANIZATION/ RESPONSIBILITIES/ STAFFING

	YEAR END 1981	6-30-81
SUPERINTENDENT		
ADMINISTRATION	39	37
PERSONNEL ADMINISTRATION		
PROCUREMENT		
WAREHOUSING		
DOCUMENT CONTROL		
CLERICAL SUPPORT		
SECURITY	107	82
SECURITY PROGRAM IMPLEMENTATION		
TEMPORARY SECURITY PERSONNEL		
INTEGRATED STARTUP GROUP	34	27
EQUIPMENT AND SYSTEM TESTING		
ASSISTANT SUPERINTENDENT		
OPERATIONS	89	73
PLANT/ SYSTEM/ EQUIPMENT OPERATION		
MAINTENANCE	106	91
PREVENTIVE MAINTENANCE		
CORRECTIVE MAINTENANCE		
TECHNICAL	45	36
RESULTS ENGINEERING		
CORE MONITORING		
CHEMISTRY		
SHIFT TECHNICAL ADVISORS		

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ASST. SUPERINTENDENT CON'T	YEAR END 1981	6-30-81
INSTRUMENTATION & CONTROLS	39	34
PREVENTIVE MAINTENANCE		
CORRECTIVE MAINTENANCE		
COMPUTERS		
HEALTH PHYSICS	20	15
RADIATION PROTECTION		
TOTAL	479	395

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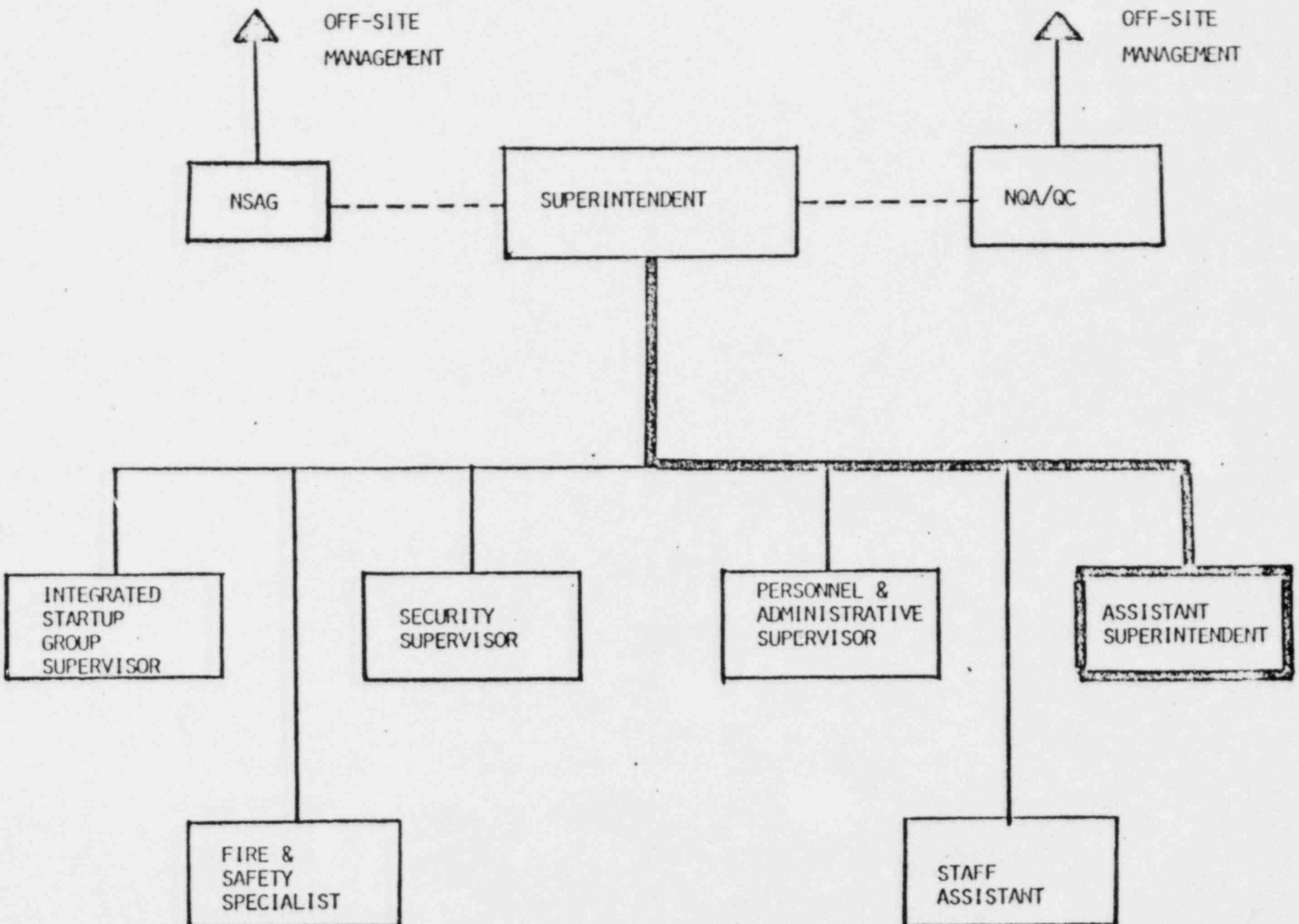
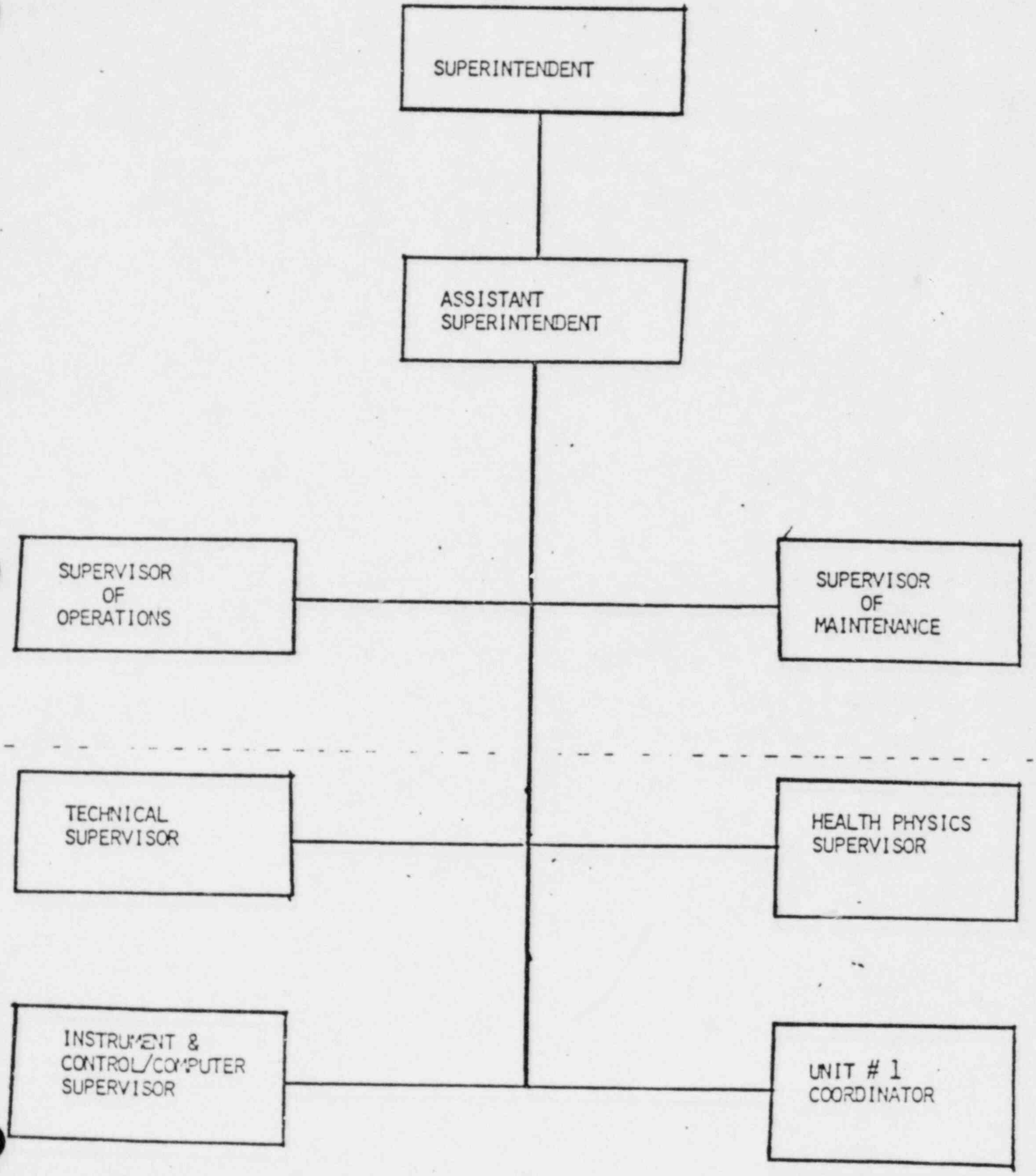
17.369



OFF-SITE MANAGEMENT
OFF-SITE MANAGEMENT
NSAG
SUPERINTENDENT
NQA/QC
INTEGRATED STARTUP GROUP SUPERVISOR
SECURITY SUPERVISOR
PERSONNEL & ADMINISTRATIVE SUPERVISOR
ASSISTANT SUPERINTENDENT
FIRE & SAFETY SPECIALIST
STAFF ASSISTANT


SUPERINTENDENT
ASSISTANT SUPERINTENDENT
SUPERVISOR OF OPERATIONS
SUPERVISOR OF MAINTENANCE
TECHNICAL SUPERVISOR
HEALTH PHYSICS SUPERVISOR
INSTRUMENT & CONTROL/COMPUTER SUPERVISOR
UNIT # 1 COORDINATOR

SUSQUEHANNA PLANT STAFF
ORGANIZATION/ RESPONSIBILITIES/ STAFFING

	YEAR END 1981	6-30-81
SUPERINTENDENT		
ADMINISTRATION	39	37
PERSONNEL ADMINISTRATION		
PROCUREMENT		
WAREHOUSING		
DOCUMENT CONTROL		
CLERICAL SUPPORT		
SECURITY	107	82
SECURITY PROGRAM IMPLEMENTATION		
TEMPORARY SECURITY PERSONNEL		
INTEGRATED STARTUP GROUP	34	27
EQUIPMENT AND SYSTEM TESTING		
ASSISTANT SUPERINTENDENT		
OPERATIONS	89	73
PLANT/ SYSTEM/ EQUIPMENT OPERATION		
MAINTENANCE	106	91
PREVENTIVE MAINTENANCE		
CORRECTIVE MAINTENANCE		
TECHNICAL	45	36
RESULTS ENGINEERING		
CORE MONITORING		
CHEMISTRY		
SHIFT TECHNICAL ADVISORS		

ASST. SUPERINTENDENT CON'T	YEAR END 1981	6-30-81
INSTRUMENTATION & CONTROLS	39	34
PREVENTIVE MAINTENANCE		
CORRECTIVE MAINTENANCE		
COMPUTERS		
HEALTH PHYSICS	20	15
RADIATION PROTECTION		
TOTAL	479	395

TRAINING PROGRAM

SUSQUEHANNA STEAM ELECTRIC STATION
SIMULATOR FACTS

CONTRACT AWARDED	OCT-76
TRAINING START	OCT-79

CAPABILITIES:

A.374

INITIAL START CONDITIONS	27
SIMULATION SPEEDS	3
BACKTRACK	10 MINS
MALFUNCTIONS	225
CRYWOLF	1583

SIMULATOR USAGE

PROCEDURE CHECKOUT

UNCOVER PLANT DESIGN PROBLEMS

HUMAN FACTOR ENGINEERING

TRAINING

NRC

A-375

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OPERATIONS TRAINING PROGRAMS

1. LICENSED OPERATOR CANDIDATES
2. NUCLEAR PLANT OPERATORS
3. AUXILIARY SYSTEM OPERATORS

LICENSED OPERATOR CANDIDATE TRAINING

- FUNDAMENTALS
- RESEARCH REACTOR
- BWR SYSTEMS
- BWR SIMULATOR
- PLANT-SPECIFIC SYSTEMS
- OPERATING BWR ASSIGNMENT

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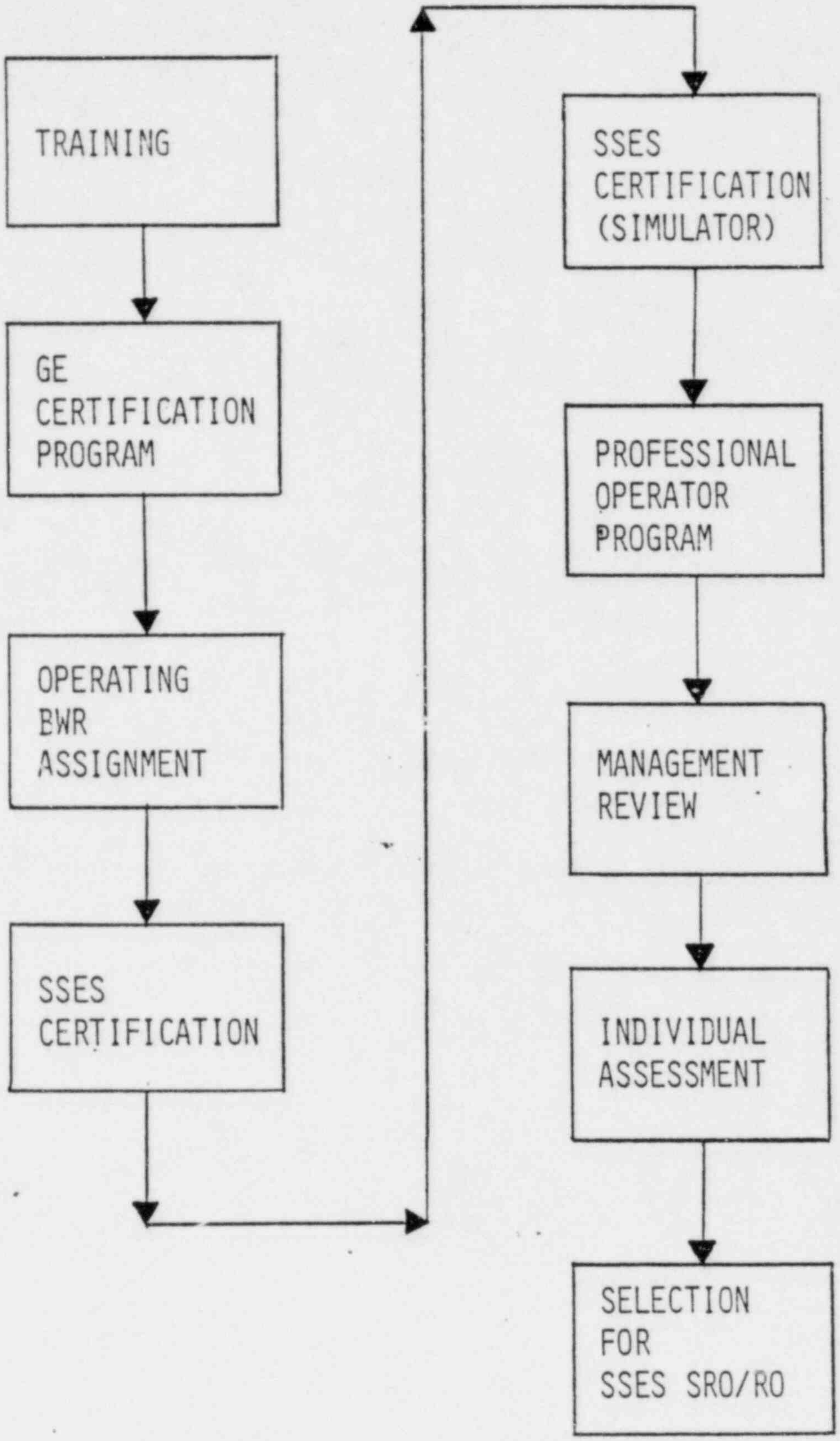

TRAINING
GE CERTIFICATION PROGRAM
OPERATING BWR ASSIGNMENT
SSES CERTIFICATION
SSES CERTIFICATION (SIMULATOR)
PROFESSIONAL OPERATOR PROGRAM
MANAGEMENT REVIEW
INDIVIDUAL ASSESSMENT
SELECTION FOR SSES SRO/RO

CURRENT ASO / NPO TRAINING PROGRAM

AUXILIARY SYSTEM OPERATOR TRAINING

ORIENTATION

FUNDAMENTALS

PLANT SYSTEMS

AUXILIARY SYSTEMS

NUCLEAR PLANT OPERATOR TRAINING

ORIENTATION

FUNDAMENTALS

NUCLEAR PLANT SYSTEMS

SHIFT TECHNICAL ADVISOR TRAINING

SUBJECT MATERIAL

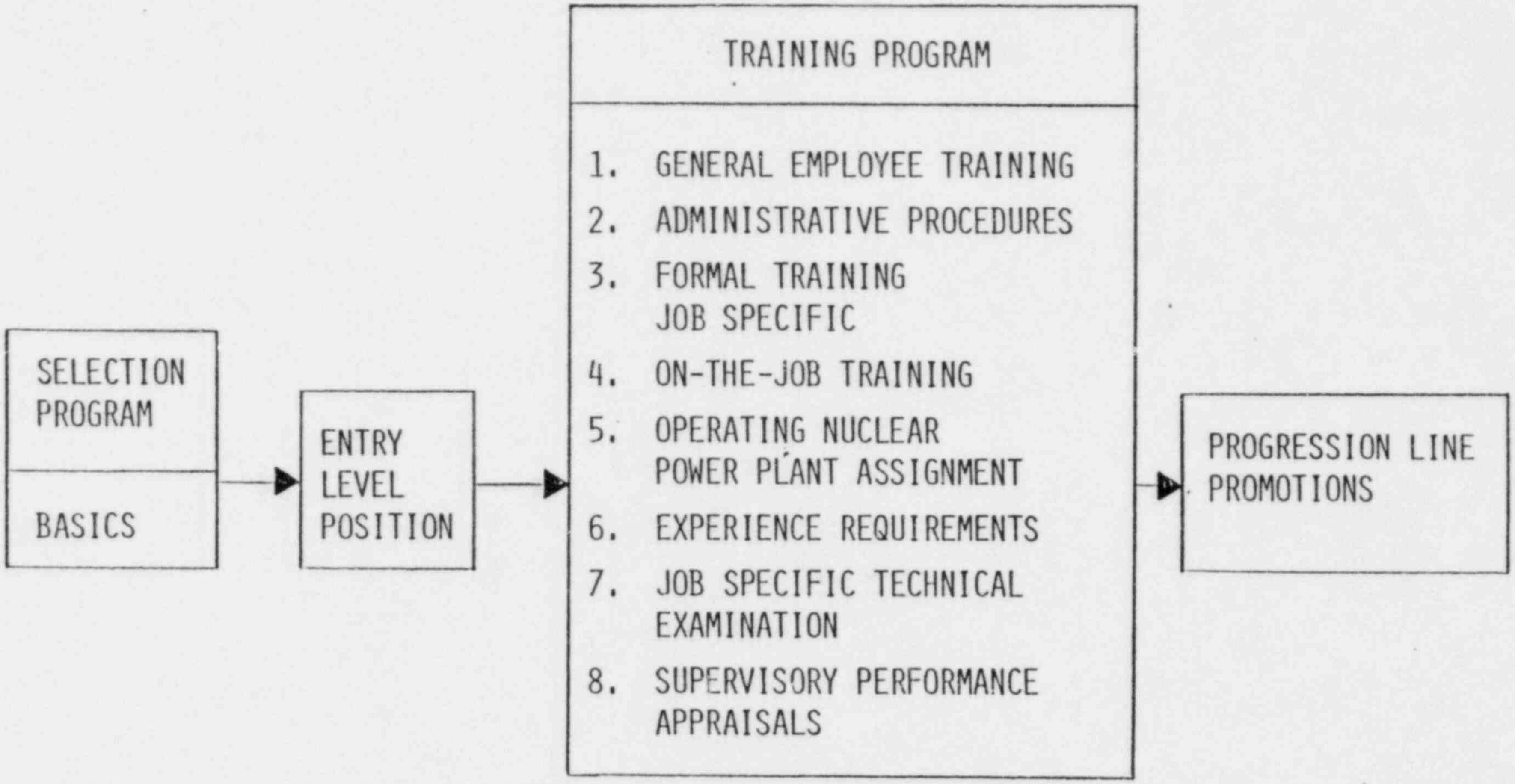
- PLANT SYSTEMS
- ADVANCED NUCLEAR THEORY
- THERMOHYDRAULICS
- TRANSIENT ANALYSIS
- CHEMISTRY
- HEALTH PHYSICS
- STARTUP TESTING
- INSTRUMENTATION & CONTROLS
- ELECTRICAL THEORY

A-380

SHIFT TECHNICAL ADVISOR TRAINING

MAJOR ELEMENTS

- FORMAL CLASSROOM . . . 762 HOURS
- SIMULATOR . . . . 182 HOURS
- 8-HOUR WRITTEN EXAMINATION
- SIMULATOR DEMONSTRATION
- ORAL EXAMINATION


SELECTION PROGRAM
BASICS
ENTRY LEVEL POSITION
TRAINING PROGRAM
1. GENERAL EMPLOYEE TRAINING
2. ADMINISTRATIVE PROCEDURES
3. FORMAL TRAINING JOB SPECIFIC
4. ON-THE-JOB TRAINING
5. OPERATING NUCLEAR POWER PLANT ASSIGNMENT
6. EXPERIENCE REQUIREMENTS
7. JOB SPECIFIC TECHNICAL EXAMINATION
8. SUPERVISORY PERFORMANCE APPRAISALS
PROGRESSION LINE PROMOTIONS

A-382

PURPOSE OF PRESENTATION

PHILOSOPHY OF TRAINING

FUNCTIONS AND STRUCTURE

GENERAL OVERVIEW OF PROGRAMS

PHILOSOPHY OF TRAINING

ORGANIZATION

A-384

- REPORTS TO VICE-PRESIDENT
- EXIST FOR ONE BWR PLANT
- DEMONSTRATED COMPANY COMMITMENT

PHILOSOPHY OF TRAINING

EDUCATIONAL

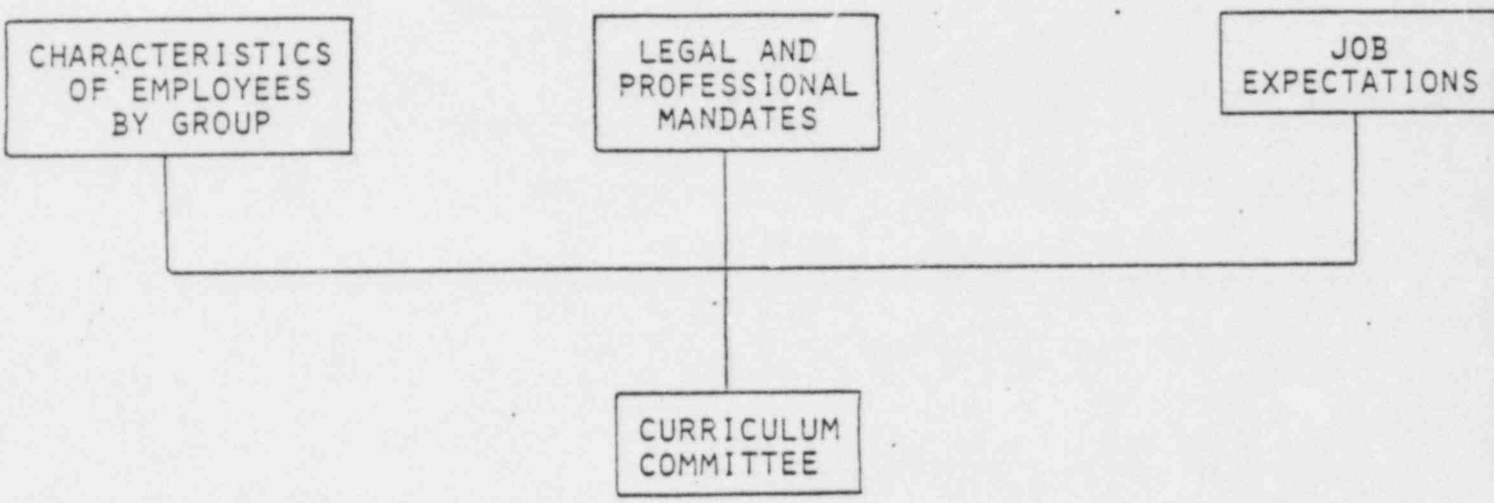
- THEORY/PRACTICE
- REALISTIC ENVIRONMENT
- CURRICULUM DEVELOPMENT SYSTEM
- INSTRUCTIONAL MATERIAL SYSTEM
- FORMATIVE AND SUMMATIVE EVALUATION

A-385

CURRICULUM COMMITTEE
CONCEPT

ASSUMPTIONS

1. NO ONE KNOWS EVERYTHING ABOUT JOB COMPETENCIES FOR A POSITION
2. THE MOST IMPORTANT JOB COMPETENCIES MUST BE KNOWN



TRAINING SUPERVISOR	RESPONSIBLE INSTRUCTOR	STENOGRAPHER
OCCUPATIONAL LINE SUPER.	OCCUPATIONAL EXPERT WORKER	ADVISOR

A-386

FUNCTIONS

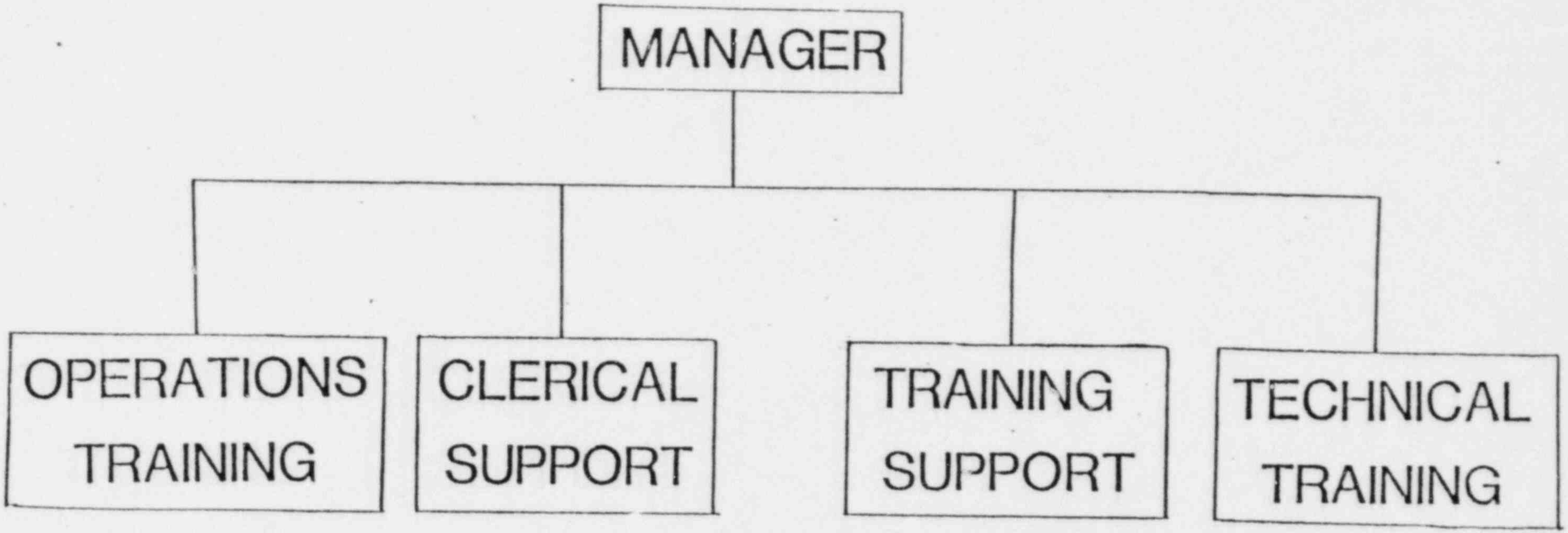
- TEACHING

- TESTING

- RECORDS

A-387

STRUCTURE



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CONTROL ROOM

ADVANCED CONTROL ROOM

1971 – CONTROL ROOM OPTIMIZATION STUDY

1974 – OPERABILITY ANALYSIS

1980 – PP&L HUMAN FACTORS ENGINEERING ASSESSMENT

1981 – NRC HUMAN FACTORS ENGINEERING ASSESSMENT

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ADVANCED CONTROL ROOM

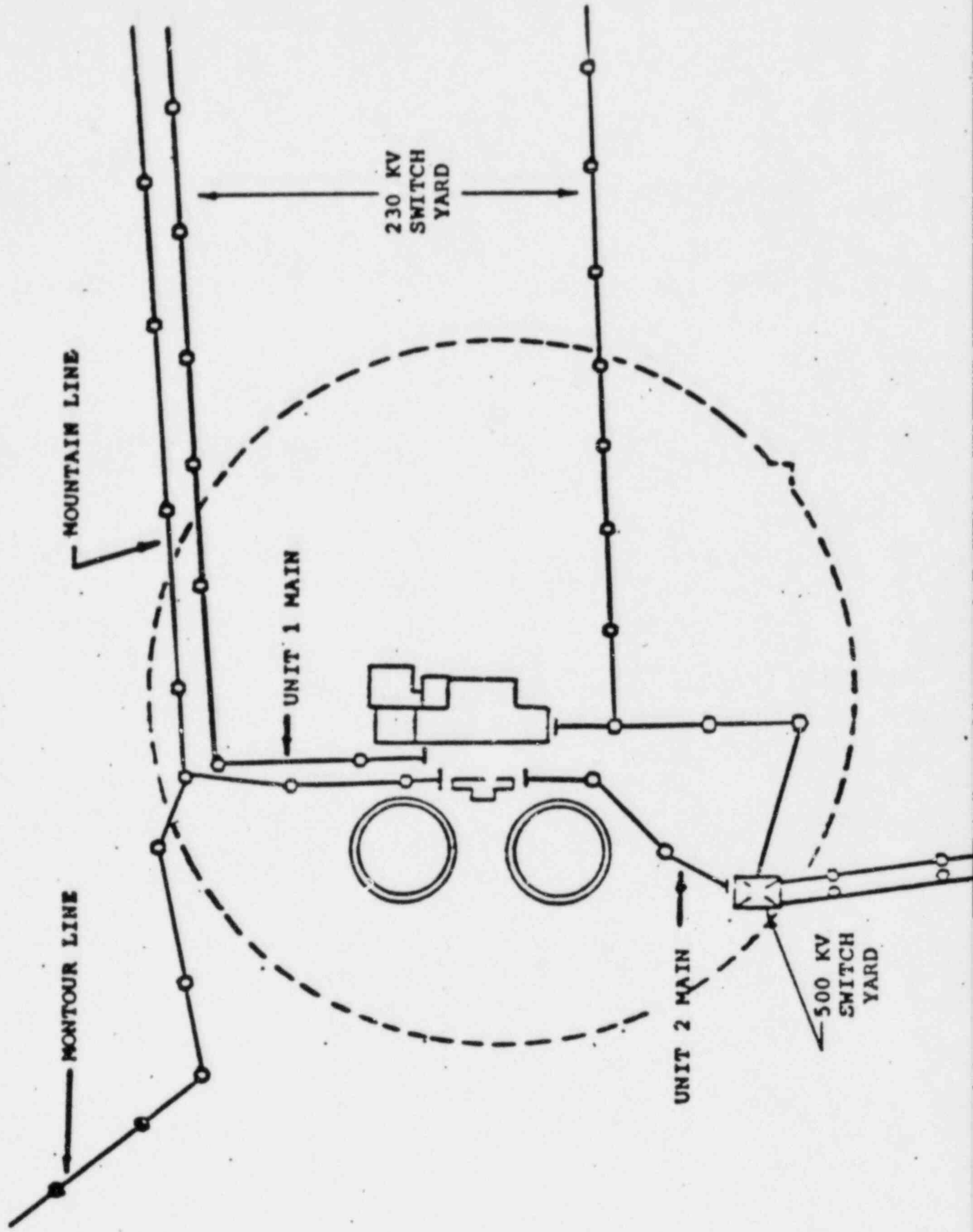
- DESIGNED WITH THE OPERATOR IN MIND
- EXTENSIVE USE OF ADVANCED GRAPHICS AND ALPHA-NUMERIC DISPLAYS
- FOLLOWED HUMAN ENGINEERING PRINCIPLES
- LIVING FEATURE OF THE PLANT

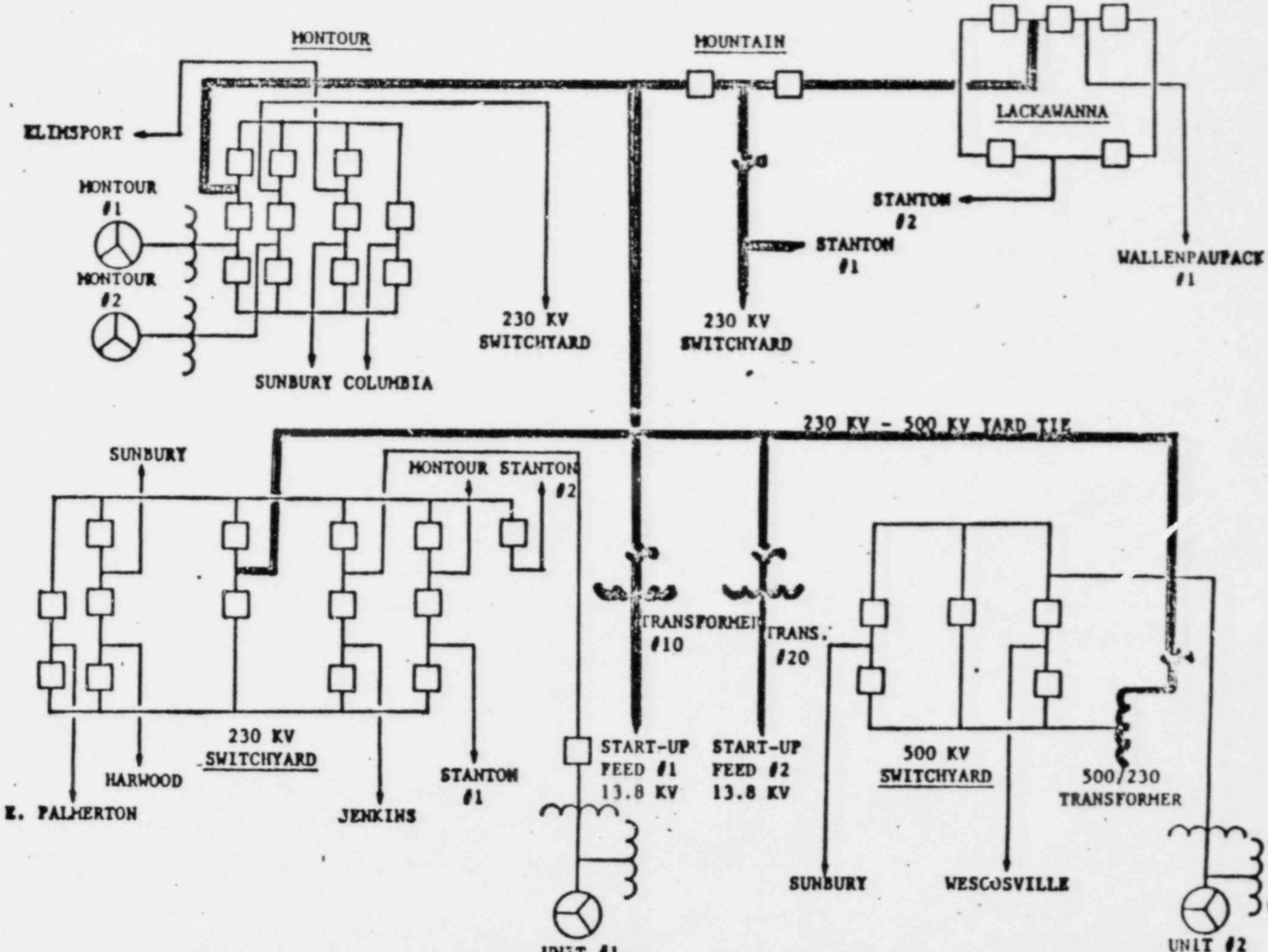
A-391

STATION BLACKOUT

STATION BLACKOUT

- SITE ELECTRICAL DISTRIBUTION
- NRC GENERIC LETTER 81-04
- SUMMARY OF BLACKOUT EVENT
- SIMULATED BLACKOUT TEST


MOUNTAIN LINE
230 KV SWITCH YARD
UNIT 1 MAIN
MONTOUR LINE
UNIT 2 MAIN
500 KV SWITCH YARD


MONTOUR
MOUNTAIN
LACKAWANNA
ELIMSPORT
MONTOUR #1
MONTOUR #2
STANTON #2
STANTON #1
WALLENPAUPACK #1
230 KV SWITCHYARD
230 KV SWITCHYARD
SUNBURY COLUMBIA
230 KV - 500 KV YARD TIE
SUNBURY
MONTOUR STANTON #2
TRANSFORMER #10
TRANS. #20
230 KV SWITCHYARD
HARWOOD
E. PALMERTON
JENKINS
STANTON #1
START-UP FEED #1 13.8 KV
START-UP FEED #2 13.8 KV
500 KV SWITCHYARD
500/230 TRANSFORMER
SUNBURY
WESCOSVILLE
UNIT #1
UNIT #2

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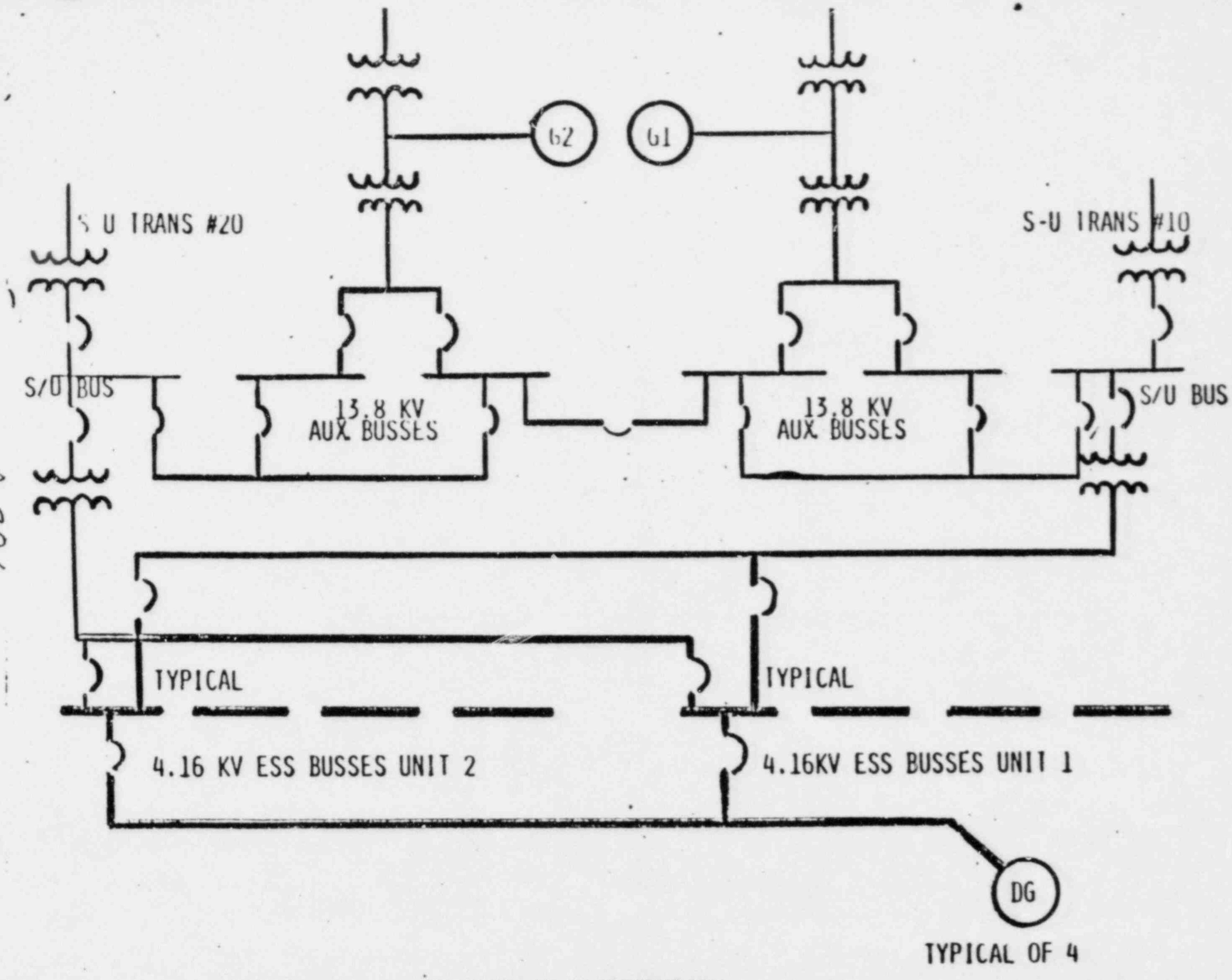
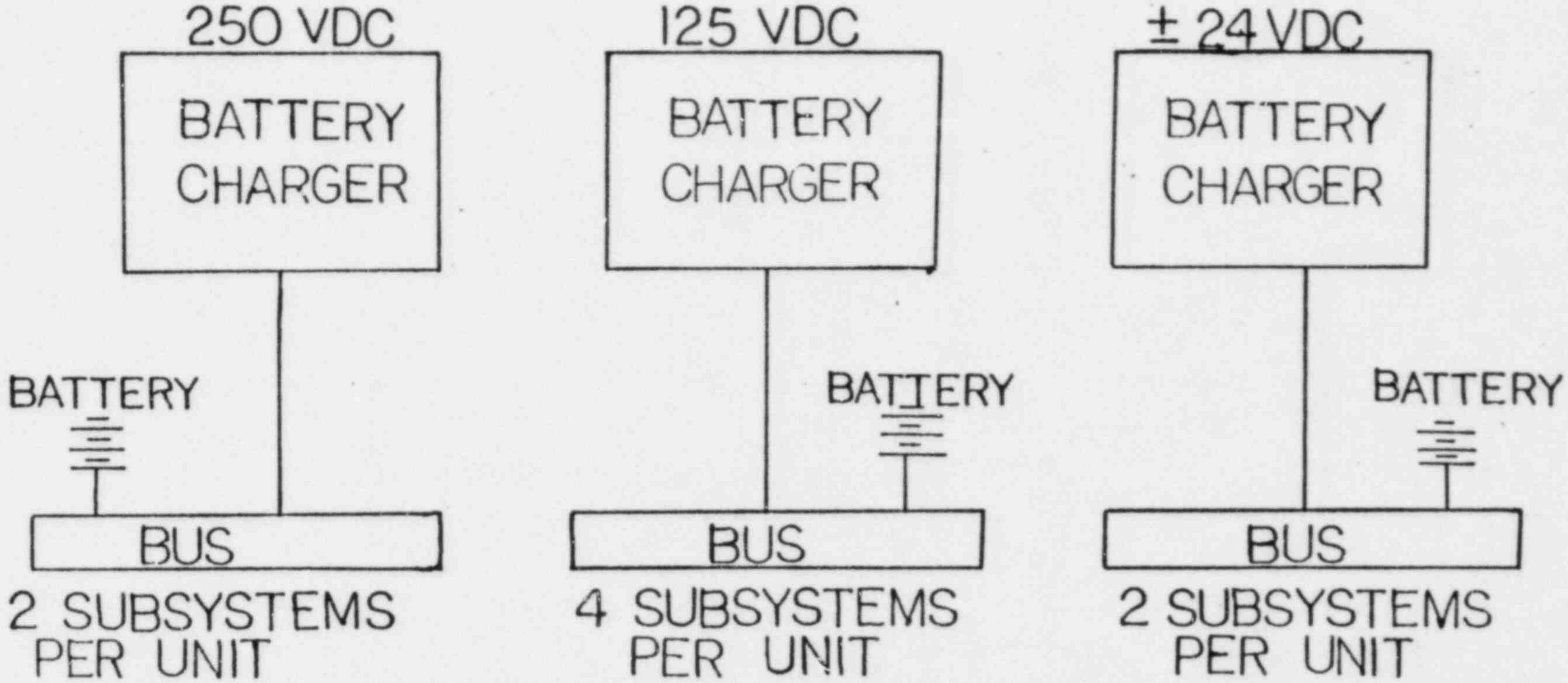


INHOUSE AC DISTRIBUTION

A-396

STATION BATTERIES



A-397

INITIAL ENGINEERING EVALUATION

PREPARED PROCEDURES

DEVELOP TRAINING

PHASE 1

SIMULATED BLACKOUT TEST

ENGINEERING EVALUATION

PHASE 2

INCORPORATE RESULTS INTO

- PROCEDURES

- TRAINING

PHASE 3

SITE AC POWER

- 10 SOURCES TO TRANSFORMER 10

- 9 SOURCES TO TRANSFORMER 20

- 4 DIESEL GENERATORS

A-400

STATION BLACKOUT EVENT

INITIATING CONDITIONS

- LOSS OF OFFSITE POWER

- EMERGENCY DIESEL GENERATORS FAIL TO START

AUTOMATIC ACTIONS

- MAIN GENERATOR AND TURBINE TRIP

- REACTOR SCRAM

- REACTOR AND CONTAINMENT ISOLATIONS

- SAFETY RELIEF VALVES ACTUATE

- HPCI AND RCIC AUTOMATIC INITIATIONS

- LOAD SHEDDING

- AC/AIR OPERATED EQUIPMENT FAILED CONDITION

OPERATING CREW RESPONSE

CONTROL REACTOR LEVEL

CONTROLLED REACTOR PRESSURE REDUCTION

A-403

SECURE DC LOADS

ALTERNATE MONITORING PARAMETERS

PREPARE FOR CONTINGENCY ACTIONS

RESTORATION OF EMERGENCY DIESELS

PROJECTED OFFSITE POWER AVAILABILITY

PREPARE FOR POWER RESTORATION

A.404

INITIATE NECESSARY CONTINGENCY ACTIONS

POWER AVAILABLE - RESTORE INHOUSE LOADS

TEST PURPOSE

SIMULATE LOSS OF AC POWER TO SELECTED SYSTEMS

- REACTOR
- PRIMARY CONTAINMENT
- HPCI, RCIC

MONITOR RESULTANT SYSTEM PERFORMANCE

SIMULATED BLACKOUT TEST GUIDELINES

ALL BUSSES REMAIN ENERGIZED

ALL INSTRUMENTATION AVAILABLE

DIESEL GENERATORS OPERABLE

LOW PRESSURE ECCS OPERABLE

PROTECT EQUIPMENT

COMPLY WITH TECHNICAL SPECIFICATIONS

CONTINUOUS PARAMETER MONITORING

PREDEFINED CUTOFF POINTS

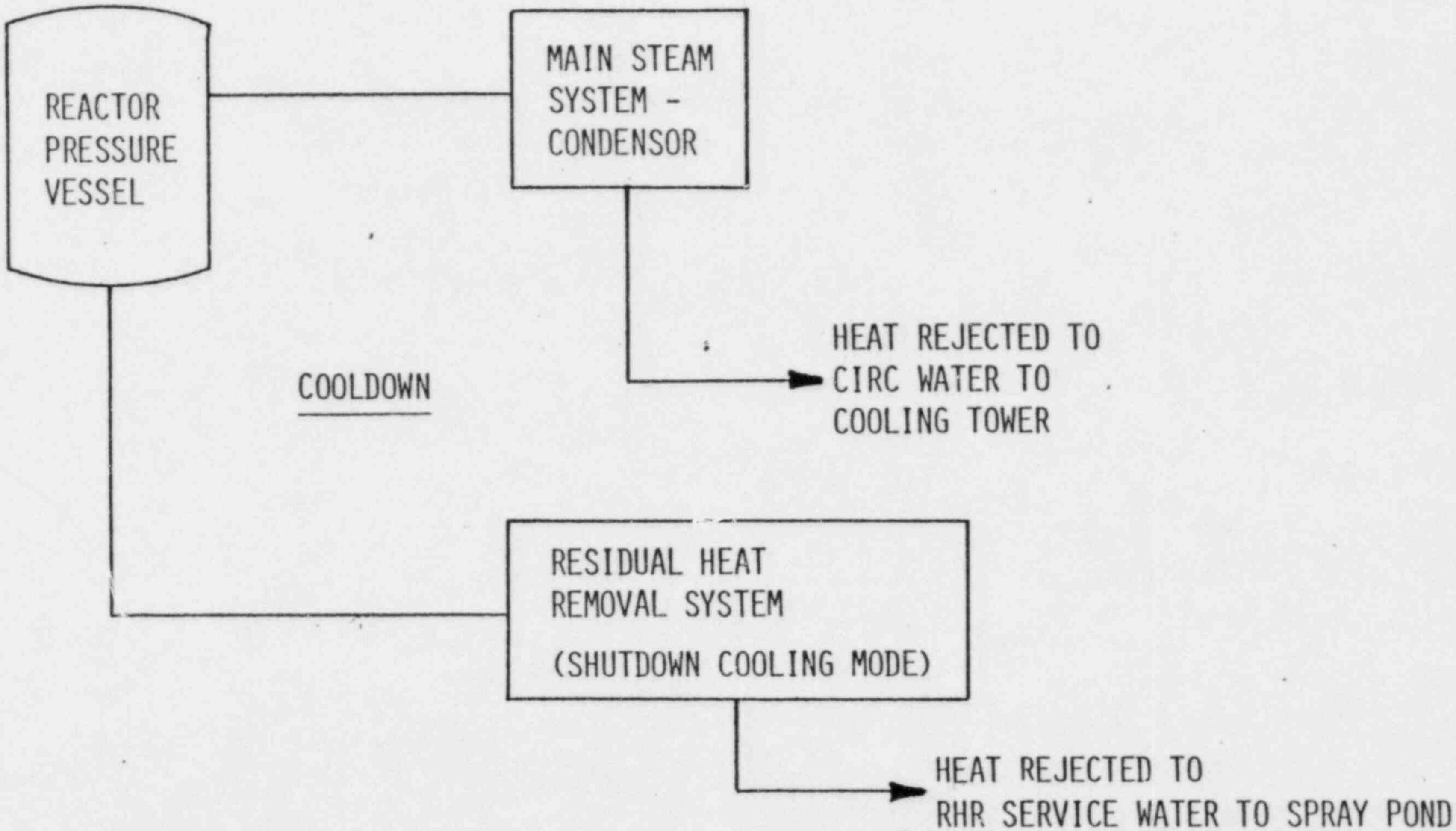
CONTINGENCY ACTIONS

TEST TERMINATES WHEN

- SUFFICIENT DATA
- CONTINGENCY ACTION

DECAY HEAT REMOVAL

DECAY HEAT REMOVAL
NORMAL



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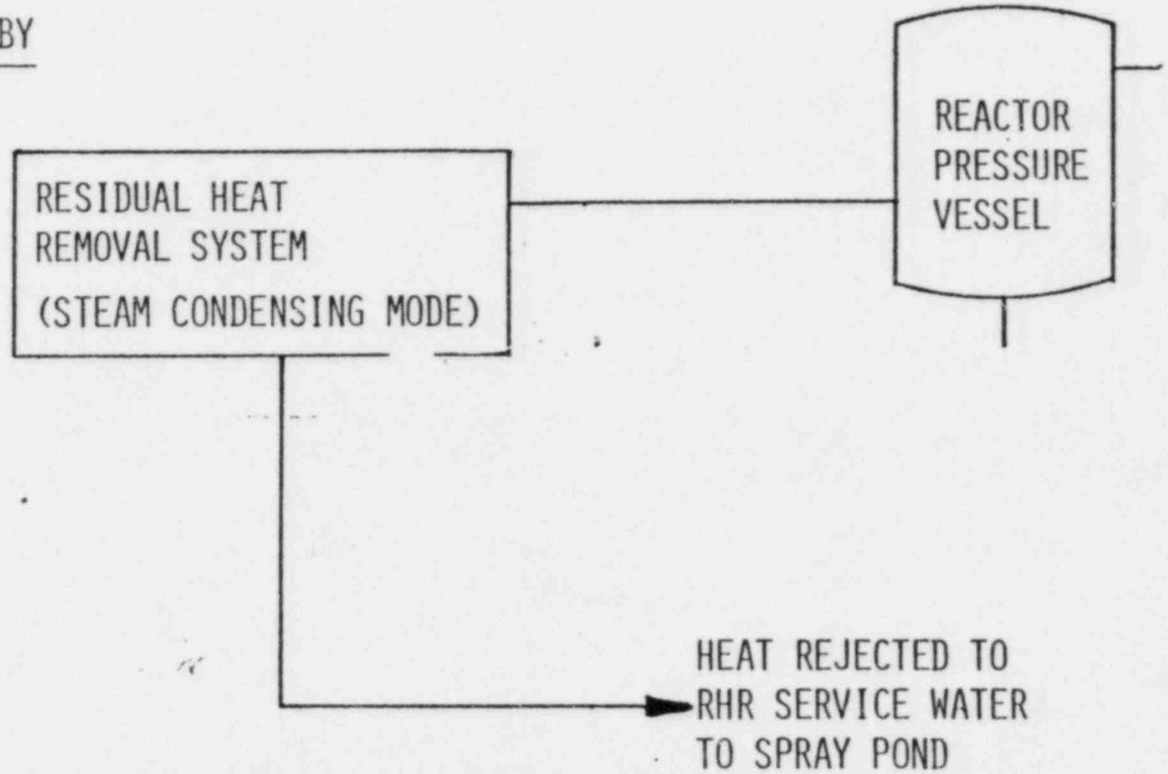
DECAY HEAT REMOVAL
NORMAL

HOT STANDBY



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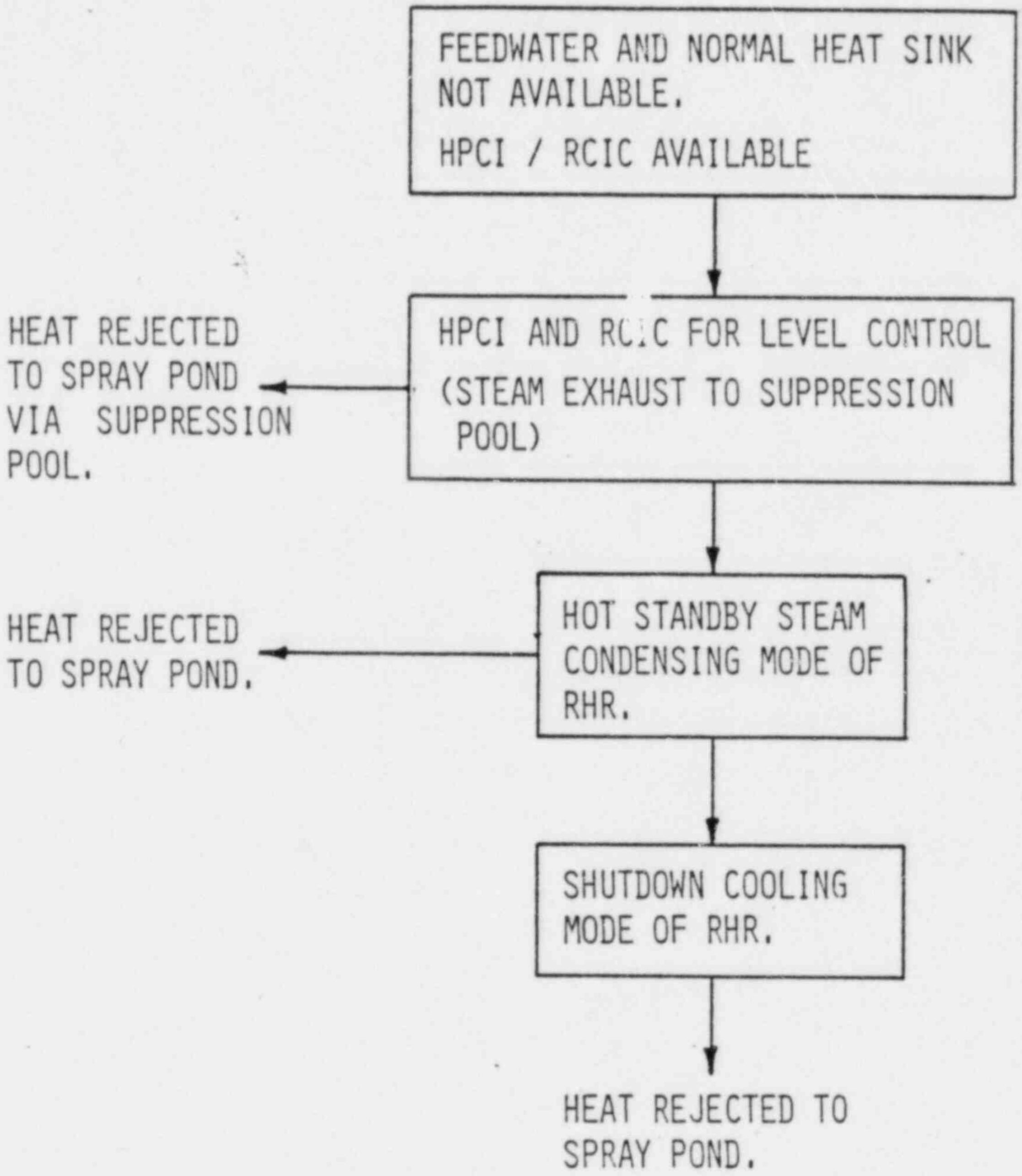
DEGRADED MODE OF DECAY HEAT REMOVAL



DEGRADED MODE OF DECAY HEAT REMOVAL

HPCI AND RCIC NOT AVAILABLE
REACTOR ISOLATED FROM NORMAL
HEAT SINK.

↓

INITIATE CS / LPCI;
DEPRESSURIZES (SRV OR ADS)

↓

WHEN CONDITIONS STABILIZE,
SHUTDOWN COOLING MODE OF
RHR TO REJECT HEAT TO SPRAY
POND.

4 ADDITIONAL JUSTIFICATION FOR T-QUENCHER LOADS

5 REVIEW OF SUBMERGED DRAGLOADS

6 IE BULLETIN 79-27 & 80-06

7 FIRE REVIEW OF ALTERNATE SAFE SHUTDOWN SYSTEM

8 MODIFICATION OF ADS LOGIC

9 COMMON REFERENCE LEVEL FOR REACTOR VESSEL LEVEL INSTRUMENTATION

10 EMERGENCY PREPAREDNESS

11 EMERGENCY SUPPORT FACILITIES

12 LONG-TERM EMERGENCY PREPAREDNESS

13 HEAVY LOADS GENERIC LETTER

14 SCRAM DISCHARGE VOLUME GENERIC LETTER

COMPARISON OF PRINCIPAL DESIGN FEATURES OF SUSQUEHANNA AND SIMILAR FACILITIES

DESIGN FEATURE	SUSQUEHANNA	LASALLE
RATED THERMAL POWER, MEGAWATTS	3,293	3,293
TOTAL REACTOR CORE FLOW RATE, POUNDS PER HOUR	100,000,000	106,500,000
NUMBER OF FUEL ASSEMBLIES	764	764
NUMBER OF CONTROL RODS	185	185
NUMBER OF RECIRCULATION LOOPS	2	2
RECIRCULATION PUMP FLOW RATE, GALLONS PER MINUTE	45,200	47,250

A-415

NUMBER OF JET PUMPS	20	20
NUMBER OF CORE SPRAY LOOPS	2	1
NUMBER OF LOW PRESSURE COOLANT INJECTION PUMPS	4	3
MAXIMUM POWER PER FUEL ROD LENGTH, KILOWATTS PER FOOT	13.4	13.4
MAXIMUM CENTERLINE FUEL TEMPERATURE, DEGREES FAHRENHEIT	3,435	3,325
MINIMUM CRITICAL POWER RATIO	1.23	1.24
TOTAL PEAKING FACTOR	2.51	2.25

A-446

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SER OUTSTANDING ISSUES

A-418

SER OUTSTANDING ISSUES

1 TURBINE MISSILES

2 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

3 STEAM BYPASS OF THE SUPPRESSION POOL

BWR SCRAM SYSTEM

A-419

NUREG - 0785

AEOD REPORT

SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS

IN BWR SCRAM SYSTEM

A-420

NRC RECOMMENDATIONS

- UPGRADE CRD-HYDRAULIC CONTROL UNIT EXHAUST LINES AND SCRAM DISCHARGE VOLUME PIPING TO "HIGHEST STANDARDS FOR DESIGN, FABRICATION, INSTALLATION, TESTING, ISI QA AVAILABLE (ASME III CLASS 1).

- PROVIDE REDUNDANT RELIABLE BREAK DETECTION INSTRUMENTS IN SDV AREA.

A-421

- DEVELOP EMERGENCY OPERATING PROCEDURES AND TRAINING PROGRAMS FOR SDV PIPING BREAK MITIGATION.

- CONSIDER IMPROVING SCRAM EXHAUST VALVE CLOSURE RELIABLIITY.

- IMPROVE MAINTENANCE PRACTICES ASSOCIATED WITH SDV PIPING AND CRD HCU MANUAL VALVES.

SUSQUEHANNA COMPARISON TO AEOD RECOMMENDATIONS

- CRD PIPING (INCLUDING SDV & SDIV) DESIGNED AND BUILT TO ASME III CLASS 2 REQUIREMENTS.

  REACTOR COOLANT PRESSURE BOUNDARY IS AT CRD DOUBLE SEALS WHICH PROVIDE PASSIVE FLOW RESTRICTION.

- MULTIPLE BREAK DETECTION DEVICES OR METHODS AVAILABLE TO OPERATOR

  - AREA RADIATION MONITORS
  - REACTOR BUILDING SUMP LEVEL ALARMS
  - CRD HIGH TEMPERATURE ALARMS
  - REACTOR BUILDING VENTILATION HIGH RADIATION ALARM
  - REACTOR BUILDING VENTILATION ISOLATION
  - OPERATOR OBSERVATION

- EMERGENCY OPERATING PROCEDURES UNDER DEVELOPMENT. THEY WILL ADDRESS APPROPRIATE OPERATOR ACTIONS.

- SCRAM EXHAUST VALVE HAS A FAIL OPEN DESIGN. THIS IS ESSENTIAL TO FAIL SAFE SCRAM. NO MODIFICATION ANTICIPATED.

- MAINTENANCE PRACTICES WILL BE IN ACCORDANCE WITH NUREG-0785.

SUSQUEHANNA

- EVALUATION ON NUREG - 0785 ONGOING

- STATUS TO DATE

  - SSES IS A MARK II CONTAINMENT WITH INHERENT DESIGN IMPROVEMENTS OVER THE MARK I CONCEPT WHICH FORMED THE BASIS FOR NUREG - 0785

    - WATER TIGHT ECCS PUMP ROOMS
    - IMPROVED SEPARATION BETWEEN SDV's AND ECCS PUMP ROOMS
    - CRD MAKE-UP PUMPS IN TURBINE BUILDING
    - 2 -250 GPM REACTOR BUILDING SUMP PUMPS

DRAFT NUREG 0803

PROPOSED CONCLUSIONS

NRC I. PIPING INTEGRITY

A. PERFORM QA AUDIT OF SDV PIPING

B. REVIEW HCU/SDV MAINTENANCE, SURVEILLANCE, MODIFICATION PROCEDURES

C. PROPOSE PROGRAM OF INSERVICE INSPECTION

PP&L I. ANTICIPATE COMPLIANCE

NRC II. MITIGATION CAPABILITY -

UPGRADE EMERGENCY OPERATING PROCEDURES AND OPERATOR TRAINING

PP&L II. ANTICIPATE COMPLIANCE

NRC III. ENVIRONMENTAL QUALIFICATIONS -

CONFIRM OR UPGRADE ENVIRONMENTAL QUALIFICATION OF EQUIPMENT REQUIRED TO INDICATE AND/OR MITIGATE THE CONSEQUENCES OF AN SDV BREAK

PP&L III. UNDER EVALUATION

ATWS

A-425

THE ATWS EVALUATION CONCERNS ARE

- RADIOLOGICAL CONSEQUENCES (10 CFR 100)
- PRIMARY SYSTEM PRESSURE ($<$LEVEL C)
- PRIMARY CONTAINMENT (PRESS/TEMP)
- FUEL INTEGRITY (CORE COOLABLE GEOMETRY) (NEUTRON OSCILLATIONS)
- LONG TERM SHUTDOWN
- AVAILABILITY
- SCHEDULE

A-426

SUSQUEHANNA'S ANALYSIS WILL ACCOUNT FOR ALL OF THESE CONCERNS AND WE ARE COMMITTED TO HAVE ASSURANCE THAT ALL ELEMENTS OF THE ATWS ISSUE WE EMPLOY WILL ACHIEVE THESE GOALS.

ATWS RESOLUTION

PROPOSED NRC RULE-vs-PP&L COMMITMENT

	SECY 80-409	SSES
• PLANT SPECIFIC ANALYSIS	YES	YES
• RECIRC PUMP TRIP (RPT)	YES	YES
• OPERATOR TRAINING (OT)	YES	YES
• SCRAM DISCHARGE VOLUME (SDV)	YES	YES
• CONTAINMENT ISOLATION	YES	YES
• HPCI IMPROVEMENT	YES	YES
• ALTERNATE ROD INSERTION (ARI)	YES	*
• LOGIC (CHANGE)	YES	*
• AUTO STANDBY LIQUID CONTROL	YES	*

* A PLANT SPECIFIC ANALYSIS IS BEING CONDUCTED FOR SSES. WHEN THIS ANALYSIS HAS BEEN COMPLETED WE WILL TAKE PRUDENT STEPS TO ASSURE OVERALL SAFETY OF THE GENERAL PUBLIC AND THE PLANT.

A-428

TENTATIVE SCHEDULE FOR
ATWS IMPLEMENTATION

TOPIC	UNIT 1	UNIT 2
• PLANT SPECIFIC ANALYSIS	1/1/82	1/1/82
• RECIRCULATION PUMP TRIP	FUEL LOAD	FUEL LOAD
• OPERATOR TRAINING	FUEL LOAD AND ON-GOING	FUEL LOAD AND ON-GOING
• SCRAM DISCHARGE VOLUME MODIFICATIONS	FUEL LOAD (PARTIAL) 12/31/82 (COMPLETED)	FUEL LOAD

Issues Closed Since The Sub committee Meeting

- Turbine missile
- Additional justification of T-quencherloads
- IE Bulletin 80-06
- TMI item II.K.3 item 27
- Heavy loads
- Upgraded emergency support facilities

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2

OPEN ISSUES AS OF AUGUST 7, 1981

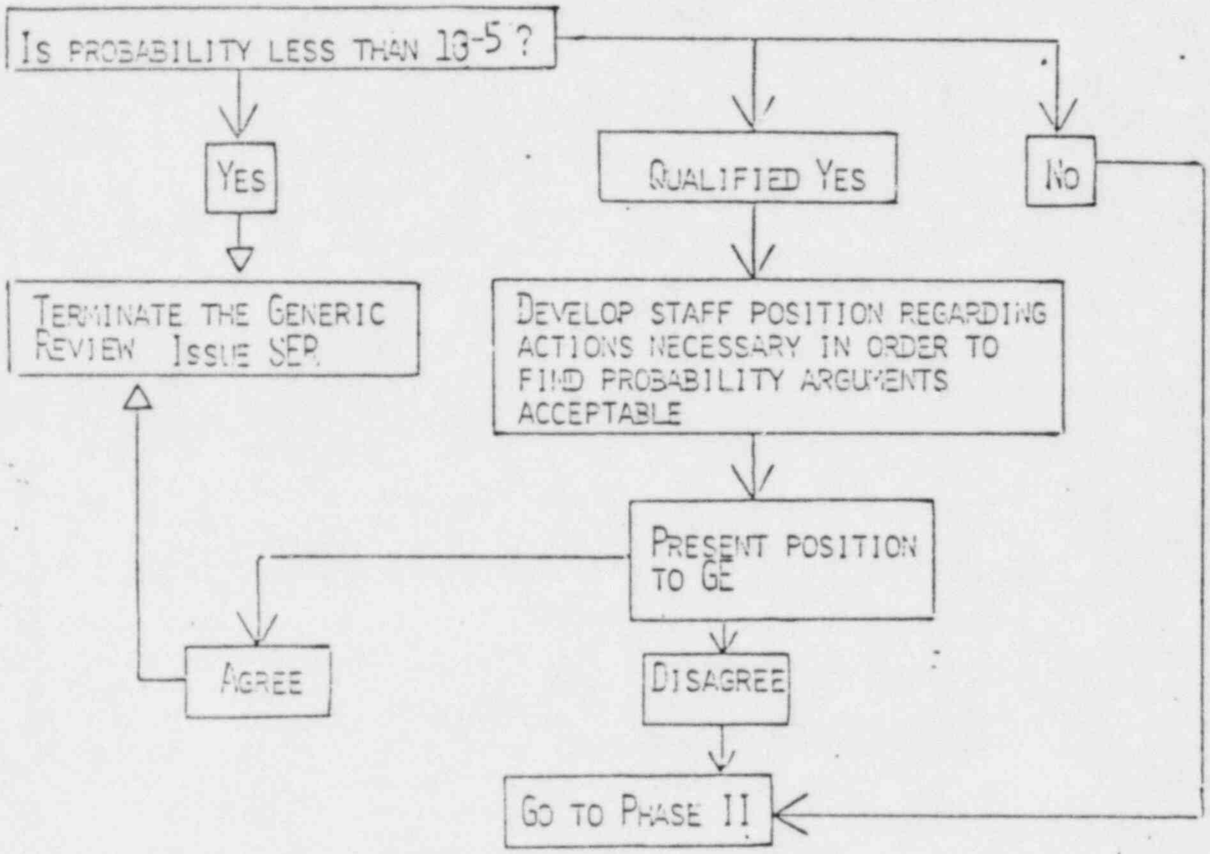
ITEM	NEXT ACTION	DATE OF NEXT ACTION
EQUIPMENT QUALIFICATION OF ELECTRICAL EQUIPMENT	APPLICANT	11-1-81
STEAM BYPASS OF THE SUPPRESSION POOL	APPLICANT	8-10-81
REVIEW OF SUBMERGED DRAGLOADS	APPLICANT	9-1-81
IE BULLETIN 79-27	APPLICANT	8-14-81
REVIEW OF ALTERNATE SHUTDOWN SYSTEM	APPLICANT	8-14-81
TMI ITEM II.K.3 ITEM 18 ADS LOGIC	APPLICANT	10-1-81
III.A.1.1 EMERGENCY PREPARDENESS	APPLICANT	8-12-81
LONG TERM EMERGENCY PRE-PARADNESS	APPLICANT	9-1-81
SCRAM DISCHARGE PIPE BREAKS	STAFF	EARLY AUGUST

REVIEW CHRONOLOGY

- DRAFT NUREG 0785 ISSUED BY AEOD ON APRIL 3, 1981
- NRC 10 CFR 50.54(F) LETTER ISSUED APRIL 10, 1981 REQUESTING:
  - (1) 45 DAY GENERIC EVALUATION
  - (2) 120 DAY PLANT SPECIFIC RESPONSE
- GE SUBMITTED NEDO-24342 BY LETTER DATED APRIL 30, 1981
- NRC GENERIC REVIEW COMMENCED MAY 1, 1981
- NRC LETTER DATED JULY 7, 1981 EXTENDED PLANT SPECIFIC RESPONSE TIME PENDING RECEIPT OF NUREG 0803
- NUREG 0803 SCHEDULED FOR ISSUANCE AUGUST 1981

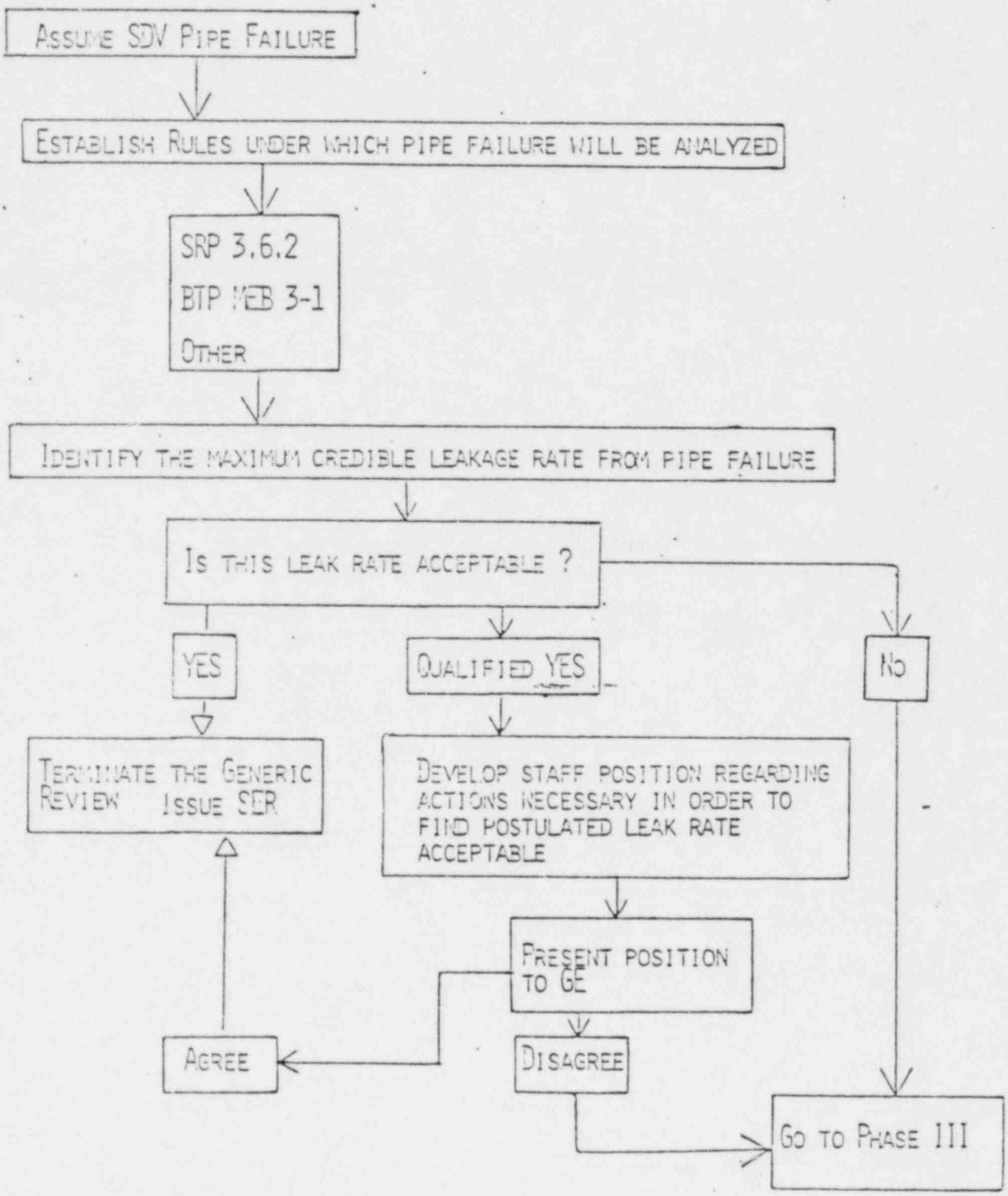
PHASE I REVIEW

PROBABILITY ASSESSMENT



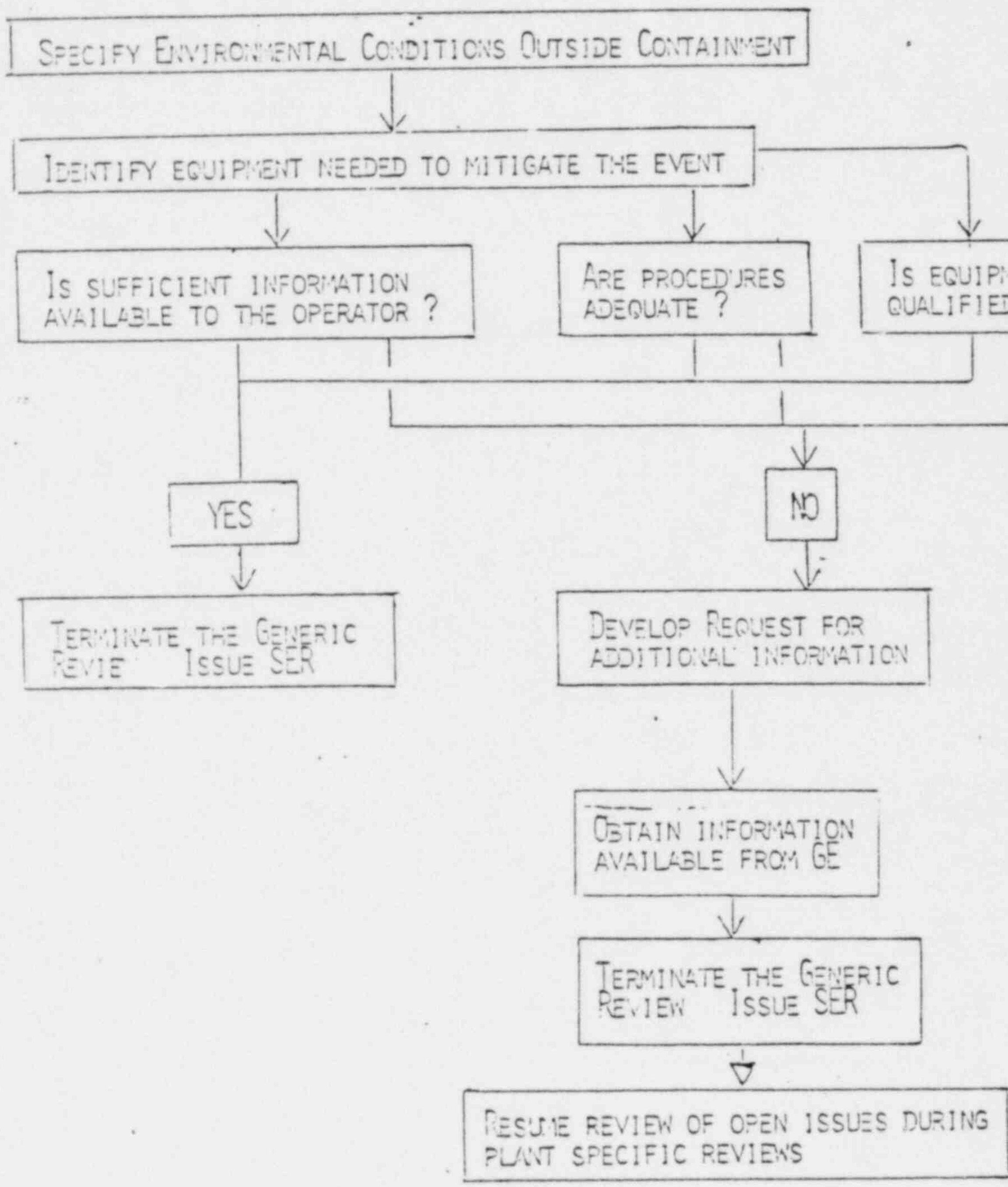
PHASE II REVIEW

PIPE FAILURE



PHASE III REVIEW

MITIGATION



INTEGRATED RISK ASSESSMENT

MAJOR ASSUMPTIONS:

- PIPING INTEGRITY -- SDV PIPE FAILURE FREQUENCY OF 10^{-4} PER PLANT YEAR
- MITIGATION CAPABILITY -- PROCEDURES FOR DIAGNOSIS AND MITIGATION ARE ADEQUATE
- ENVIRONMENTAL QUALIFICATION -- ADVERSE ENVIRONMENT DOES NOT AFFECT EQUIPMENT RELIABILITY

CONCLUSION:

- FREQUENCY OF CORE MELT IS ESTIMATED TO BE LESS THAN 10^{-6} PER PLANT YEAR
- ANALYZED SEQUENCE OF EVENTS IS NOT A DOMINANT CONTRIBUTOR TO CORE MELT

SUMMARY OF GUIDANCE ADOPTED IN NUREG 0803

o PIPING INTEGRITY	AS-BUILT INSPECTION ISI/SURVEILLANCE SEISMIC DESIGN HCU-SDV MAINTENANCE
o MITIGATION CAPABILITY	EMERGENCY PROCEDURES
o ENVIRONMENTAL QUALIFICATION	DETECTION FUNCTION DEPRESSURIZATION FUNCTION MITIGATION FUNCTION

APPENDIX XXXVI
WATERFORD 3: BACKGROUND INFORMATION

Pages A437-A439 has been deleted as Predecisional Information.

MINUTES
ACRS SUBCOMMITTEE ON WATERFORD-3
NEW ORLEANS, LA

DATE ISSUED:
JULY 2, 1981

PURPOSE: To discuss various matters as part of the ACRS review of Waterford-3 for an operating license.

ATTENDEES:

D. Ward, Subcommittee Chairman
J. Ray, Subcommittee Member
Z. Zudans, ACRS Consultant
I. Catton, ACRS Consultant
R. Pearson, ACRS Consultant

S. Blade, NRC
E. Pederson, NRC
L. Maurin, LP&L (Louisiana Power & Light)
T. Garrett, LP&L
F. Drummond, LP&L
J. Edwards, LP&L
W. Alphonso, LP&L
R. Hartman, Los Alamos Technical Research

There were no written or oral statements made by members of the public. Attached is the meeting agenda and a list of documents considered by the Subcommittee.

Plant Status and Schedule

Suzanne Black (NRR) summarized the plant schedule and the remaining open items. A CP was issued in November 1974. Construction is 89% complete. Fuel loading is scheduled for October 1982. The SER is scheduled to be issued in early July. At that time, there may still be about 27 open issues. There are expected to be 17 open issues remaining by the August full committee meeting. A list of open issues is attached.

Dr. Catton inquired about the toxic gases hazard analysis, which is one of the open issues. The details were not known. This will be discussed further at the next Subcommittee meeting. Dr. Catton also requested information on the open issue of analysis of steam voiding in the reactor vessel resulting from a secondary coolant system transient. Mr. Ward and Mr. Ray asked what the schedule impact would be if ACRS review were delayed. Mr. Blake, an attorney for LP&L, replied. There is no direct tie between the ACRS schedule on the ASLB hearings, however, there are indirect effects, particularly for contention items.

A-440

Staff Evaluation of Plant Management

E. Pederson (NRR) summarized NRR's review of the plant's management. A site visit has not been performed yet. The Staff, as per NUREG-0731, required that the title of the individual in charge of nuclear operations be changed from director to vice-president. There was no change in the scope of organizations that report to this individual. No changes in authorities and responsibilities of this individual were provided to the Subcommiteee. There are two levels of vice-presidents between the vice-president of nuclear operations and the president of LP&L. Dr. Zudans expressed concern that the individual in charge of nuclear operations does not have sufficient status within the organization. Dr. Catton noted the same was true for the head of training.

Currently, there are 160 LP&L and 206 contractor employees at the site. Mr. Ray noted that this is not a desirable ratio of LP&L to contractor people. Currently, procedures are being written by contractor personnel.

Due to a lack of prior nuclear experience, advisors will be assigned to the vice president and plant manager. Mr. Ray suggested that personnel be temporarily assigned to operating plants to compensate for the lack of experience among the operators.

There are three safety review groups: (1) the Safety Review Committee (SRC), (2) the Plant Operating Review Committee (PORC); and (3) the Onsite Safety Review Committee (OSRC). There will be a total of 15 shift technical advisors.

Design Comparison with Other CE Plants

L. Maurin (LP&L) compared the design of Waterford-3 to other CE plants. The NSSS is the same as San Onofre 2&3, and is similar to ANO-2. The containment is similar to St. Lucie-1, since both are from Ebasco.

Plant Staffing

Mr. Maurin (LP&L) summarized the staffing and organization. LP&L, as a company, does not have an engineering organization. Waterford-3 will have an offsite engineering group totalling about 50 people. Minor backfits can be performed by LP&L while major backfits are performed by contractors. LP&L is planning to hire about 100 additional people for Waterford-3. Recruiting consultants are being utilized.

The position of assistant plant manager was created to provide more nuclear experience to the organization. The plant manager has no proor experience in a nuclear power plant.

Mr. Ray recommended that the training program give sufficient emphasis to maintenance personnel.

Quality Assurance

Mr. T. Garrett (LP&L) described the quality assurance programs. LP&L is developing a fossil quality assurance program. Mr. Ray commended LP&L on this. The QA group develops and audits the QA program.

Safety Review Groups: Feedback of Operating Experience: Use of INPO

F. Drummond (LP&L) discussed these topics:

- o The PORC is an onsite review group composed of site management and is responsible for reviewing matters within operatng such as technical specifications. It reports to the plant manager.

- o The SRC is an offsite corporate level group responsible for auditing plant activities. It also oversees the PORC and SORC.
- o The OSRC is responsible for independent review of plant staff and activities. It reports to the offsite support manager. The details of the duties of this group have not been formulated yet.

Mr. Ward questioned the rigid separation between the OSRC and the offsite Engineering Staff. However, NUREG-0737 requires this rigid separation. The Subcommittee agreed that this is undesirable. Dr. Catton noted that none of the safety review groups are comprised only of engineers and executives. He urged the inclusion of someone expert in human factors. The Subcommittee agreed.

Personnel Training and Qualificaton

D. Lester (LP&L) gave an overview of training. A position of Supervisor of Training was recently created. This individual will oversee all training activities for Waterford-3. The training budget for this year is $2.0M. TERA Corporation is being used to assist in recruiting.

LP&L is currently using the CE simulation which simulates Calvert Cliffs. A simulator will be purchased and is expceted to be ready by 1985. Dr. Pearson cautioned that training on a simulator that does not adequately represent the plant can be counterproductive.

R. Armstrong summarized operator training. The training includes: four weeks of academic refresher; power plant fundamentals; 3 weeks research reactor training; 10 weeks at ANO-2; and 8 weks simulator. Dr. Catton inquired whether the training emphasized rote button pushing or fundamental understanding. The reply was that there is some combination of the two. Dr. Pearson asked whether operator candidates are given ability testing. The reply was

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they are not. EEOC does not allow it.

Maintenance and I&C technicians are trained to service equipment through vendor courses. Health physics technicians are being trained by Houston Lighting and Power.

J. Edwards (LP&L) provided information on use of simulators. Palo Verde has recently completed their simulator. Waterford-3 may utilize that simulator in addition to the CE simulator until 1985. Mr. Edwards said simulators are an extremely effective tool. Dr. Pearson indicated that this is an overstatement. Mr. Ward and Dr. Catton cautioned that simulators have limitations.

Human Factors Engineering

W.Alphonso (LP&L) and R. Hartman (Los Alamos Technical Associates) described the human factors engineering work. The decision was made in 1975 to purchase an advance computer system. A consultant is being employed to assist in utilizing the computer's capabilities. Consultants are also evaluating the control board and other plant controls. There are 21 CRTs in the control room. Currently, it is not planned to put emergency procedures on the computer as an operator aid.

J. Edwards described the process by which the operator responds to annunciators. Alarms will flash on the dedicated CRT terminals as they occur. Procedures are still in the development stage. CE is performing the work.

Dr. Pearson inquired whether plots of pressue and temperature of the core were available to the operator. The reply was that although the capability exists, the function is not currently available.

Future Meetings

The Waterford-3 Subcommittee will meet in Washington, D.C. on August 4, 1981 to continue its review of the plant.

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LIST OF DOCUMENTS

1. NRC presentation on plant status and open issues - 6 slides
2. NRC presentation of Evaluation of Organization and Staffing - 14 slides
3. LP&L presentation on comparison with other plants, design changes since CP, and organization and staffing - 29 slides
4. LP&L presentation on Training and Qualification of Personnel, Human Engineering - 52 slides

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OPEN ISSUE STATUS (6/15/81)

. RESOLVABLE BY SER ISSUANCE

- NON-TMI ISSUES

1. FLOOD PROTECTION - COOLING TOWER AREA

2. DIESEL ENGINE PIPING

3. TURBINE MISSILES

4. EMERGENCY FEEDWATER CONTROL

5. I&E BULLETIN 80-06

6. SINGLE FAILURE OF CONTROL SYSTEM

7. R.V. SUPPORT LOADS

8. ORGANIC MATERIALS

9. SOLID WASTE STORAGE

- TMI ISSUES

1. II.D.3 VALVE POSITION INDICATION

TOTAL ISSUES - 10

- RESOLVABLE BY AUGUST ACRS MEETING
  - NON-TMI ISSUES
    1. RCP SHAFT BREAK
    2. PRESSURE TRANSIENT ANALYSIS - SHIELD BLDG. ANNULUS
    3. DNBR
    4. CPC
    5. SITE HAZARDS (TOXIC GAS AND EXPLOSIVE MATERIALS)
    6. SHUTDOWN WITHOUT LEAVING CONTROL ROOM
    7. BORON DILUTION EVENT PROCEDURES AND ALARMS
    8. Q-LIST
  - TMI ISSUES
    1. II.B.3 POST-ACCIDENT SAMPLING

TOTAL ISSUES - 10

- RESOLVABLE POST-ACRS
  - NON-TMI ISSUES
    1. FIRE PROTECTION (INCLUDING ALTERNATE SHUTDOWN - 9/81)
    2. LICENSEE QUALIFICATIONS - 11/81
    3. PSI-ISI - 11/81
    4. APPENDICES G & H - 10/81
    5. EMERGENCY PREPAREDNESS
       - ON-SITE - 9/81
       - OFF-SITE - EARLY 82
    6. ENVIRONMENTAL QUALIFICATIONS - 11/81
    7. SEISMIC QUALIFICATIONS - 10/81
    8. STEAM VOIDING IN RV ANALYSIS - 8/81
    9. FW LINE BREAK ANALYSIS - 9/81
    10. LOSS OFFSITE - RCP TRIP DURING MSLB TRANSIENT ANALYSES - 8/81
    11. CLARIFICATION OF TRANSIENT ANALYSES WITH POTENTIAL FOR FUEL DAMAGE - 8/81
  - TMI ISSUES
    1. OPERATIONAL SAFETY (I.A.1 TASKS) - 11/81
    2. ORGANIZATION & MANAGEMENT (I.B.1.2) - 11/81
    3. OPERATING PROCEDURES (I.C TASKS) - 8/81-11/81
    4. CONTROL ROOM REVIEW (I.D.1) - 11/81
    5. CONTAINMENT SYSTEM DESIGN (II.E.4.1 AND 2) - 11/81
    6. ICC INSTRUMENTATION (II.F.2) - 11/81

SUMMARY OPEN ISSUE STATUS

AS OF 6/15 - 37

RESOLVABLE BY SER - 10

RESOLVABLE BY ACRS MEETING - 10

RESOLVABLE IN POST ACRS SUPPLEMENTS - 17

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ACRS WATERFORD-3 SUBCOMMITTEE MEETING SUMMARY
WASHINGTON, D.C.
AUGUST 5, 1981

The purpose of this meeting was to coninue the ACRS review of Waterford-3 for an operating license. The meeting agenda is attached.

HIGHLIGHTS

1. Open Items - Attached is a list of open items for Waterford-3. The Subcommittee discussed items 1,2,3.4,7,8,9, and 10 during the meeting. LP&L indicated that the open items did not represent areas of controversy with the Staff, rather, they were areas where reviews were simply not complete. The Subcommittee did not see any problems with the open items discussions under agenda Item 2. The NRC Staff commented that these items are expected to be resolved without difficulty.

2. Hazards Considerations - Hazards considerations stemming from the weather and from toxic releases from nearby industries were discussed. Safety related plant systems are all located within the "nuclear island," which is flood protected to 30 feet (above mean sea level). The maximum credible flooding, including levee failure and hurricane surge, is less than this. The plant has been analyzed for 360 mph tornado winds.

There are 16 nearby industrial facilities, and Mississippi river shipping traffic. The principal hazardous materials were indicated to be chlorine and ammonia, and also CS_2 and HCl. The control room automatically isolates from chlorine and ammonia. Mr. Ebersole inquired whether the effects on plant electrical equipment has been considered. The answer was no. Dr. Catton questioned the LP&L analysis of toxic material dispersions during periods of atmospheric stagnation or near stagnation. Dr. Siess asked the Staff why after 10 years of licensing activity on Waterford-3 site hazards is still an open item.

3. Station Blackout - The plant has been analyzed as capable of tolerating loss of AC power for two hours.

4. Ultimate Heat Sink - The ultimate heat sink is contained within the nuclear island. It consists of two trains of dry and wet cooling towers. A wet-plus-dry tower is required for removing heat for seven days following shutdown. After seven days, decay heat is reduced to the point where a dry tower alone is sufficient. Cooling tower operation requires 600 kW.

5. Organization and Staffing Discussions - LP&L plans on having 15 shift technical advisors (STA). Each STA would be on STA duty for two 24-hour days per month. The duties of the STAs when they are not on shift duty have not yet been clearly established.

Middle South Utilities (MSU) is the parent holding company of LP&L. MSU also includes Arkansas Power & Light, and Mississippi Power & Light. The services and support that MSU provides to Waterford-3 is mainly in the areas of core follow, licensing, reload analyses, and fuel purchasing. MSU, however, also intends to perform plant audits of Waterford-3, ANO, and Grand Gulf.

Dr. Catton asked the NRC Staff what their assessment was of Waterford-3 staffing. Dr. Hanauer replied that Waterford-3 was in the worst shape of any plant in recent years.

6. Use of PRA - E. O'Donnell indicated areas where reliability analyses were being used for Waterford-3. These include the auxiliary feedwater systems, AC/DC power reliability, the use of fault tree analyses in the development of operating procedures, and to assess the information available to the operator from various accident scenarios.

7. Containment Capabilities - The Containment would begin to loose its integrity at about 95 psig. The design pressure is 44 psig. Definitive information was not provided on hydrogen mixing.

8. Control of Base Mat Settlement - The base mat is monitored to control level. The base mat can be controlled by wells to control the hydraulic uplift on the plant.

WATERFORD 3

OPEN ITEMS

AS OF 8/5/81

ITEM		REVIEW TO BE COMPLETED BY
(1)	FIRE PROTECTION	10/81
(2)	LICENSEE QUALIFICATIONS	11/81
(3)	PSI/ISI	11/81
(4)	EMERGENCY PLANNING	
	ON-SITE	09/81
	OFF-SITE	EARLY 82
(5)	ENVIRONMENTAL QUALIFICATION	11/81
(6)	SEISMIC QUALIFICATIONS	10/81
(7)	STEAM VOIDING IN REACTOR VESSEL ANALYSIS	08/81
(8)	FEEDWATER LINE BREAK ANALYSIS	09/81
(9)	LOSS OF OFFSITE POWER OR TRIPPING OF THE REACTOR COOLANT PUMPS DURING A MAIN STEAM LINE BREAK	08/81
(10)	CLARIFICATION OF TRANSIENT ANALYSES WITH POTENTIAL FOR FUEL DAMAGE	08/81
(11)	REACTOR COOLANT PUMP SHAFT BREAK ANALYSIS	09/81
(12)	THERMAL-HYDRAULIC DESIGN	09/81-03/82
(13)	Q-LIST	08/81
(14)	TURBINE MISSILES	08/81
(15)	PLANT PROCEDURES	11/81
(16)	INDEMNITY REQUIREMENTS	
(17)	TMI ISSUES	
	OPERATIONAL SAFETY (I.A.1 TASKS)	11/81
	ORGANIZATION AND MANAGEMENT (I.B.1.2)	11/81
	OPERATING PROCEDURES (I.C TASKS)	08/81-11/81
	CONTROL ROOM REVIEW (I.D.1)	11/81
	CONTAINMENT SYSTEM DESIGN (II.E.4.2)	09/81
	ICC INSTRUMENTATION (II.F.2)	11/81

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ACRS SUBCOMMITTEE MEETING ON WATERFORD-3

AUGUST 5, 1981

WASHINGTON, D.C.

1. Opening Remarks by Subcommittee Chairman — 5 min — 8:30 am

2. Open Items — 40 min — 8:35 am
   - o Fire Protection M Horrell (Ebasco)
   - o Analysis of Steam Voiding in RV M. Jacob (CE)
   - o Feedwater Line Break Analysis J. Young (CE)
   - o Loss of Offsite Power - RCP Trip during MSLB Analysis M. Green (CE)
   - o Transient Analyses with Potential for Fuel Damage M. Green (CE)

3. Staff comments on Open Items — 10 min — 9:35 am

4. Hazards Considerations — 30 min — 9:50 am
   - o Control Room Habitability
   - o Balance of Plant Habitability, Diesel Operation

   J Mauro (Ebasco)

   - o Floods
   - o Tornadoes & Hurricanes

   D Hunter (Ebasco)

BREAK — 10 min — 10:35 am

5. AC/DC Power Reliability — 30 min — 10:45 am
   - o Station Blackout - E Senac (LP&L)
   - o Stability Analysis - J Saacks (LP&L)

5. AC/DC Power Reliability - (cont'd)

   - Consideration of Invertor Failures
   - Consequences of Inadvertent Disconnection of DC Power with Plant Trip
   - Capability of Diesels to Handle Loads

   E Senac (LP&L)

6. Emergency Planning - R Azzarello (LP&L) 15 min 11:30 am

   - Consideration of Flood or Hurricane
   - Emergency Support Facilities
   - Role of NRC, FEMA, State, Local, and Other Agencies

7. Ultimate Heat Sink - PV Guildys (Ebasco) 20 min 11:55 am

LUNCH 60 min 12:25 pm

8. Organization and Staffing 60 min 1:25 pm

   - Clarification of Change in Responsibilities and Authority of Head of Nuclear Operations - LV Maurin (LP&L)
   - STA Qualification, Duty Schedule - D Lester (LP&L)
   - Qualifications of Plant Chemist - D Lester (LP&L)
   - Use of Middle South Services - DC Gibbs (MSS)
   - Inclusion of Human Factors and Training and Outside Technical Expertise in Safety Review Group FJ Drummond (LP&L)
   - Training Policy and Procedures for Habit Interference D. B. Lester (LP&L)
   - Comparison of staffing and qualifications with ANO-2 at a similar stage D. B. Lester (LP&L)

9. Use of PRA - EP O'Donnell (Ebasco) 20 min 2:55 pm

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10.	Seismology and Geology Information Since CP - J Ehasz (Ebasco)	15 min	3:25 pm
BREAK		10 min	3:50 pm
11.	Systems Interactions - M Horrell (Ebasco) o Seismic o Air/Water o Internal Flooding	20 min	4:00 pm
12.	Hydrogen Capabilities - W Krotiuk (Ebasco) o Mixing o Utilization of EPRI Work o Pressure Capability of Containment, Shield Wall	20 min	4:30 pm
13.	Control of Base Mat Settlement - M Pavone (Ebasco) o Monitoring	10 min	5:00 pm
14.	Summary Remarks - L V Mauvin (LP&L)	5 min	5:15 pm
15.	Staff Comments	5 min	5:20 pm
16.	Executive Session	5 min	5:25 pm
ADJOURN			5:30 pm

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Pages A457-A468 has been deleted as Predecisional Information.

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

June 25, 1981

APPENDIX XXXVIII
WATERFORD 3: CONSULTANTS' REPORTS

MEMORANDUM TO: David Bessette

FROM: Ivan Catton

Ivan Catton

SUBJECT: WATERFORD NUMBER 3 SUBCOMMITTEE MEETING, JUNE 18-19, 1981, NEW ORLEANS

The presentation by the utility showed that they were very serious about their responsibilities as a nuclear utility. Their efforts in the human factors arena should be encouraged. They have put together a plan for integrating the control room and the operator that is better than most. The Waterford control room was conceived in about 1975, yet has more CRTs than many of todays designs. The software that will back up the CRTs is well thought out and may be one of the industries better systems.

The utilities recruiting problems appear to be more serious than those of others. We were told that they have increased the pay grade of all nuclear grades. How they plan to alleviate their recruitment problems should be followed in more detail. This is an industry wide problem and the Waterford site does not appear to be a desirable place to work.

Waterford has a unique ultimate heat sink. The containment building and the heat sink become a nuclear island - needing only electricity to be self sufficient. The heat sink is a forced air cooling system that operates by forcing air through vertical finned tubes. The air is exhausted into a well that is open at the top. The fans can operate off the auxiliary diesels, if necessary. Further, we were told that the cooling tower could operate in a natural circulation mode if the fans were lost. We lacked sufficient description of the thermal hydraulics of the system to decide if it could operate in the natural circulation mode. How well it operates in this mode is of sufficient interest that more details should be requested.

The containment air circulation system is different than most other plants. The air return is at the containment floor elevation and the flow into the containment building is at the top of the cylindrical wall of the building. The inflow is in the form of jets directed towards the top of the containment dome. The inflow is through several large openings in a ring header. According to the Ebasco engineers, the A&E, the result will be a well mixed containment under all circumstances. This approach is different and a number of questions came to mind. We should request further discussion of the concept as it is somewhat unique.

The TMI issues were discussed by Susan Black. Unfortunately they were only enumerated and their disposition was not clearly stated. This is another area that further discussion is needed.

The Waterford Number 3 site is in the midst of a number of chemical producers (Hooker Chemical among others). The potential for toxic gas transport to the plant site needs more attention. The analysis done at this time uses the Pasquill method (or Turner). We were told that up to 12% of the time were calms - later corrected to 5%. At the time, nobody knew whether these calms were many of short duration or few of long duration. If the calms are of long duration, a day or so, then methods other than the Pasquill method must be used. Further consideration must be given to the toxic chemical question at Waterford because NRC guides do not include calms and as is well known all serious air pollution episodes have occurred during calms.

On the surface the health physics program seems to be adequate. It should be noted, however, that none of the health physicists are certified. Further, it does not appear as if they will have encouragement from the management levels to do so. It has been my experience that certification and enthusiasin for the job go together. The utility should be encouraged (or told) to offer incentives for their health physcists to become certified.

Safety review committees have been set up at several levels. The utilitie's good intentions seem to be lost here. The makeup of the several safety review committees does not include the operator training interests nor does it include anyone else with human factors or physcological inclinations. Safety review committees serve a very important function and if they are to serve it they must have proper balance. The utility should be encouraged (or told) to reconsider the makeup of their safety committees to properly reflect the TMI-2 incident.

The reactor operator training program has a philosophy that does not seem to be responsive to TMI-2 lessons. In response to a question as to whether the training approach was one of understanding the basic concepts of energy and mass balances or a more militaristic approach, we were informed that it was the latter. The topic of "thermal-hydraulics" was givenonly secondary consideration. A very brief look at the Waterford lesson plans seem to bear out this initial impression. It it my view that understanding the basic physical processes will lead to better decision making when under stress. Apparently this position is not unanimous. In that some of use attempt to promolgate it at every opportunity, maybe the views of others (NASA, Air Force and Navy) should be solicited.

During the two days a number of topics that might be appropriate for the 4 August meeting came up. To summarize, they are as follows:

1. Recruitment,

2. Thermal hydraulics of the ultimate heat sink including its capability for heat removal without fans,

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3. Containment air circulation and how one is assured that it accomplishes complete mixing,

4. TMI-2 issues and how they have been satisfied,

5. Toxic chemicals and local meteorology,

6. Approach to health physics,

7. Contention items before ASLB and their disposition by the staff and the applicant.

Further topics that were a result of the site visit are:

8. Spent fuel pool thermal hydraulics,

9. Utility position on vessel coolant level sensing, and

10. How one sets priorities for use of available auxiliary power and how these priorities are incorporated into emergency procedures in a meaningful way.

cc: D. Ward

COPY

MEMORANDUM TO: D. Ward, Chairman
Waterford-3 Subcommittee OL Review

FROM: I. Catton
ACRS Consultant

SUBJECT: WATERFORD -3

1. Staffing

Dr. Hanauer's comment - "Waterford-3 falls far short of any plant seen in a number years"

I think Hanauer is correct and I don't think Waterford-3 understands or believes him. This is a problem.

2. Atmospheric Dispension

I don't believe the meteorology is handled correctly. Whenever a wind blows you have a situation where reasonable estimates can be made and mixing usually alleviates the problem. Under circumstances of low wind speeds, stron inversions, etc. we not only don't know how to make the calculation but have the most danagerous atmospheric conditions. For example crud from 4-corners flows down the Rio Grande and Albuquerque has smog. This really a generic question and is related to 10 CFR 100.

3. At our last Subcommittee meeting, we were given a presentation on the "systems Integration Program." It showed how they plan to pull training, procedures, and the control room together. It should be part of the full committee presentation.

4. Hydrogen

Sprays will mix the containment air. Stratification will only be a problem if there are no sprays.

5. Systems Interactions

Systems interacting with one another are a continuing problem. The nature of people makes me uneasy when "awareness" is to be depended on. Some or group needs to be given responsibility or you can be sure it will not be done well.

A-472

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June 26, 1981

MEMORANDUM

TO: D. E. Bessette, ACRS Staff

FROM: Dr. Richard G. Pearson, ACRS Consultant RGPearson

SUBJECT: Waterford 3 Review, June 18-19, 1981--Comments

1. In general within my area of expertise, human factors engineering, I was impressed with the qualifications and presentations made by the Los Alamos Technical Associates group, especially insofar as they have the Lockheed group behind them on control room design.

2. A principal concern I shared with Ivan Catton involves the composition of the three "Review and Audit" groups:

   a. SRC--Safety Review Committee;
   b. PORC--Plant Operating Review Committee;
   c. OSRS--Onsite Safety Review Subgroup.

   At present the composition of these groups is restricted to technical personnel (engineers; physicists) and executive/management level personnel. I would like to see a broader mix of people including some who can address issues in the areas of human performance, selection, training, medical and human factors, and performance appraisal. Ultimately operational safety will be dependent upon effective human performance involving not only operators but also maintenance personnel. There should be some "expertise" involved in auditing such performance determinants as vision of operators, noise, illumination, work-rest schedules, stress, boredom, seating, job performance aids, display factors, and control design.

   Consequently I would recommend:

   a. Personnel with human factors, medical, and/or industrial/organizational psychology expertise on the SRC. Possibly LP&L could include the head of the Personnel Department (I assume they have such a department). Also, perhaps a physician. Do they have a corporate medical staff?
   b. That the training manager (superintendent) serve on the PORC.
   c. Someone in the human factors or industrial/organizational psychology area on the OSRS. Th.s person should have concerns (and relevant expertise) relative to the control room environment, operator performance, and maintenance personnel effectiveness.

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If such personnel are not or will not be available with LP&L, then the company should consider the employment of consultants from reputable firms or universities.

3. In the areas of training I have some concerns. Some operators will have had experience at other plants; others will have been trained as fresh recruits on simulators offsite. The major problem here is what is termed "habit interference". That is, habits (skills) acquired in one setting with particular control-display relationships, whether operational or simulated, may conflict with performance in the new setting. In short, old habits must be "unlearned" and new ones must be well practiced, otherwise the operator is likely to resort to his old habits in emergency situations (i.e. under stress) and make errors. This is a well known phenomenon in human performance, and is a causative factor in many accidents. Ultimate installation of an onsite simulator with good fidelity in terms of the actural Waterford 3 control room operation will be an important milestone. Until then, the habit interference problem must be addressed both in terms of training and operational practice as well as by the philosophy of the training superintendent. With regard to the latter point the figure presented by Joe Edwards (presentation A/I 3.6) relating to simulator retraining and operator effectiveness is hypothetical. While the theory here is relevant to the effects of practice on the simulator, it is only valid under conditions of high simulation fidelity.

4. In conclusion you may wish to consider as <u>Agenda Items</u> for the proposed August 4th meeting the following:

   a. Composition of the "Review and Audit" groups;
   b. Qualifications of proposed training superintendent, and clarification of training policy and procedures to deal with the "habit interference" issue.

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COPY

August 5, 1981

MEMORANDUM TO: Dave Ward

FROM: Dick Pearson

SUBJECT: WATERFORD III CONCERNS

1. LP&L appears to have taken our comments to heart regarding safety committee composition. I do hope that their Industial/organizational Psychologist is a "competent" person (there are some incompetents in this field, too!). The availability of the Lockheed group is an asset. I would hope that, for the long haul, they acquire services of a competent human factors engineer on a periodic, consulting basis (or hire one full time as some utilities have done); there aren't many available in this region of the country, i.e. Mid-South. Such a person might have to be imported from Texas, Oklahoma, or Florida if Lockheed personnel are no longer available.

2. I do not feel strongly about their need for a corporate physician. There are some medical concerns which they recognize and for which they assert they will use consulting physicians (hopefully, board-certified in Occupational Medicine). Some of the "other" medical problems (e.g. glare; vision; postural complaints) can be handled by some human factors engineers if she/he has competence in anatomy, physiology, and biomechanics.

3. Articulation of the solution to the "habit interference" issue was poor. Lockheed personnel should have made the presentation here. May point of concern is for the transfer from the "old" job and/or simulator training to the Waterford III operation. It would be desirable to have R/O's go thru a series of high stress, simulated problems on the Waterford III control room design, e.g. LOCA. Again, the Lockheed consultants know the problems here and should be able to provide effective guidance. Parenthetically, I'm sure this is an industry-wide problem. so NRC Staff should be the watchdog on this issue. If Dr. Goodman (NRC -Human Factors) goes on the NRC Staff Review to Waterford-III, he should follow-up on this.

   Again, the qualifications of the to-be-hired Training Director will be of concern with regard to the above issues.

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Franklin Research Center
A Division of The Franklin Institute

June 23, 1981

Mr. David A. Ward, Chairman
E.I. Du Pont de Nemours & Company
Savannah River Laboratory
Building 773-A, A-219
Aiken, SC 29808

re: ACRS Subcommittee Meeting on Waterford 3,
June 18-19, 1981, New Orleans, LA

Dear Mr. Ward:

During site visit we observed the ultimate sink in a form of 5 cell dry-cooling tower. Forced air circulation is used to remove the heat, however, it was stated that natural circulation may accomplish the task as well. A more detailed description of the path of heat removal to this and other heat sinks, under various operating conditions, with corresponding heat balances at various points of the path should be made available to this Committee for information.

Waterford 3 appears to be well layed out, no large add-on pipe whip supports dominate the scene.

Control Room, in particular the added clusters of four (4) Color CRT screens indicate early commitment (pre TMI-2 accident) to computerized plant operational aids. Subsequent presentations on the same subject reinforced my initial impression about the comprehensiveness and the quality of LP&L treatment of human engineering aspects. Waterford 3 computer facility has a broad range of potential support capabilities not as yet undefined. I believe LP&L systems approach to human factors analysis and utilization of aerospace technology will finally produce a system I have always advocated for Nuclear Power Plant operation.

With respect to the LP&L organization, it is noted that VP for Power Production has the responsibility for Nuclear and Non-nuclear Operations and Quality Assurance. I do not feel comportable with such an arrangement, however, LP&L and NRC Staff appear to be happy with it.

Although it was brought out that Waterford 3 Operating License Application has been contested, no detail of issues was available at this meeting. Hopefully, we will have details at the August 4, 1981 meeting.

Cotton - OC loads

A-476

Site hazards due to the proximity of a chemical plant, readings on base mat differential settlement (reactor building and turbine building), composition of safety review groups (i.e., inclusion of training and human factors personnel in these groups), need to be addressed in greater detail during the upcoming meeting.

Training program appears to be well designed and under a good management.

Unless something comes up during the presentations at August 4, 1981 meeting, I see no major issues impacting the review of Waterford 3 Operating License.

Very truly yours,

Zenons Zudans
Senior Vice President

ces

cc: David Bessette, ACRS

A-177

LOUISIANA POWER & LIGHT COMPANY
WATERFORD SES - UNIT NO 3

CURRENT STATUS (6/28/81) OF BULK CONSTRUCTION COMMODITIES

	PERCENT COMPLETE
ELECTRICAL RACEWAY	86
ELECTRICAL CABLE	76
ELECTRICAL TERMINATIONS	64
PIPING 2½ INCHES AND LARGER	97
PIPING 2 INCHES AND SMALLER	85
LARGE PIPE HANGERS	88
INSTRUMENTATION TUBING	66
HVAC DUCTWORK	95
HVAC SUPPORTS	97
OVERALL CONSTRUCTION STATUS	89

WATERFORD 3 SES

LP&L STARTUP STATUS 6/30/81

ACTIVITY	% TOTAL EFFORT	ACTIVITY % COMPLETE	% TOTAL EFFORT COMPLETE
1. Prerequisite Testing	40	6	3
2. Prerequisite Procedures	17	85	14
3. Preoperational Testing	16	0	0
4. Preoperational Procedures	7	70	5
5. Operational Testing	17	0	0
6. Operational Procedures	3	80	2
	100	NA	24

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LOUISIANA POWER & LIGHT COMPANY
WATERFORD SES - UNIT NO 3

PROJECT MILESTONE SCHEDULE

Preliminary Safety Analysis Report Filed	-	December 1970
Limited Work Authorization Issued	-	April 1972
Construction Permit Issued	-	November 1974
First Nuclear Concrete Placed	-	December 1975
NSSS Vessels Set	-	January 1979
480V Energized	-	September 1980
Safety Evaluation Report Issued	-	July 1981
Hot Functional Testing	-	June 1982
Fuel Loading	-	October 1982*
Commercial Operation	-	April 1983

*Applicant projection, also verified as realistic and achievable by NRC's Case Load Forecast Panel in January 1981.

ACRS

LOUISIANA POWER & LIGHT COMPANY
WATERFORD SES UNIT NO. 3
NSSS DESIGN COMPARISON

Item	Waterford 3	SONGS 2 & 3	ANO-2
1. RATED CORE THERMAL OUTPUT, MWT	3,390	3,390	2,815
2. NOMINAL PRIMARY SYSTEM PRESSURE, PSIA	2,250	2,250	2,250
3. NOMINAL INLET TEMP., F	553	553	553
4. NOMINAL OUTLET TEMP., F	611	611	612
5. AVE THERMAL OUTPUT, KW/FT	5.34	5.34	5.34
6. MAX THERMAL OUTPUT, KW/FT	12.5	12.5	12.5
7. MAX CLAD SURFACE TEMP., F (AT NOMINAL PRESSURE)	657	657	657
8. MAX FUEL TEMP., F (AT 100% POWER)	3,420	3,420	3,420
9. NUMBER OF FUEL ASSEMBLIES	217	217	177
10. NUMBER OF SPACER GRIDS PER ASSEMBLY	11	11	12
11. CORE EQUIVALENT DIAMETER, IN	136	136	123
12. CORE ACTIVE HEIGHT, IN	150	150	150
13. FUEL DISCHARGE BURNUP, MWD/MTU (AVE FIRST CYCLE)	12,730	12,730	12,500
14. NO. OF CONTROL ASSEMBLIES (FULL/PART-LENGTH)	83/8	83/8	73/8
15. BORON CONC. FOR CRITICALITY, PPM (COLD/HOT)	899/832	899/832	1004/987
16. NUMBER OF STEAM GENERATORS	2	2	2
17. NUMBER OF REACTOR COOLANT PUMPS	4	4	4

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ACRS

LOUISIANA POWER & LIGHT COMPANY
WATERFORD SES UNIT NO. 3
BOP DESIGN COMPARISON

	ITEM	WATERFORD 3	ST. LUCIE
1.	CONTAINMENT	STEEL VESSEL SURROUNDED BY CONCRETE SHIELD BUILDING	STEEL VESSEL SURROUNDED BY CONCRETE SHIELD BUILDING
2.	CONTAINMENT VESSEL ID, FT.	140	140
3.	CONTAINMENT VESSEL HT., FT.	240.5	232
4.	CONTAINMENT FREE VOL., FT^3	2,677,000	2,500,000
5.	CONTAINMENT DESIGN INTERNAL PRESSURE, PSIG	44	44
6.	CONTAINMENT VESSEL THICKNESS, IN. (VERTICAL/HEAD)	1.9/.95	1.9/.95
7.	SHIELD BUILDING, ID, FT	148	148
8.	SHIELD BUILDING, HT, FT	249.5	241
9.	CONCRETE THICKNESS, FT (VERTICAL/DOME)	3/2.5	3/2.5
10.	NO. OF CONTAINMENT SPRAY PUMPS	2	2
11.	NO. OF CONTAINMENT FAN COOLERS	4	4

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ACRS FULL COMMITTEE MEETING
LOUISIANA POWER & LIGHT CO.
WATERFORD SES UNIT NO 3
AUGUST 6, 1981

UNIQUE DESIGN FEATURES

ULTIMATE HEAT SINK
- DRY/WET COOLING TOWER COMBINATION

CONTAINMENT AND SHIELD BUILDING DESIGN
- DUAL CONTAINMENT CONCEPT
- ANNULUS MAINTAINED AT NEGATIVE PRESSURE AT ALL TIMES

CONDENSATE AND REFUELING WATER STORAGE POOLS
- INTEGRAL PART OF NPIS

PLANT COMPUTER
- EXTENSIVE CAPABILITIES

ACRS FULL COMMITTEE MEETING
WATERFORD SES UNIT NO. 3
WASHINGTON, D.C.
AUGUST 6,1981

SUMMARY OF TMI (NUREG 0737)
SER OPEN ITEMS

JOHN HART
EBASCO

A-483

SER SEC. 1.7 (OUTSTANDING ISSUES):

"(25)TMI ISSUES (SECTION 22)

OPERATIONAL SATETY (I.A.1 TASKS)

ORGANIZATION AND MANAGEMENT (I.B.1.2)

OPERATING PROCEDURES (I.C. TASKS)

CONTROL ROOM REVIEW (I.D.1)

CONTAINMENT SYSTEM DESIGN (II.E.4.2)

ICC INSTRUMENTATION (II.F.2)

VALUE POSITION INDICATOR (II.D.3)"

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I.A.1

I.A.1.1 STA

- PROPOSED STA PROGRAM SUBMITTED 4/20/81
- ACCEPTED BY NRC IN SER

I.A.1.2 SHIFT SUPVR ADMIN DUTIES

- LPL COMMITTED TO REQUIREMENTS IN FSAR AMENDMENT 16 (3/81)

I.A.1.3. SHIFT MANNING

- NRS ACCEPTED LPL PROPOSED MINIMUM SHIFT CREW COMPOSITION IN SER
- FSAR AMENDMENT 16 LPL COMMITTED TO PROCEDURES GOVERNING OVERTIME

NRC STAFF WILL REPORT FINDINGS AFTER AUDIT - 11/81

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

- LPL HAS COMMITTED TO PROVIDE AN ISEG PER REQUIREMENTS OF I.B.1.2 BY FUEL LOAD. WILL BE CALLED "ONSITE SAFETY REVIEW SUBGROUP" (OSRS)
- OSRS WILL PROVIDE INDEPENDENT REVIEW OF PLANT STAFF ACTIVITIES BEGINNING 4TH QTR 1981

NRC STAFF WILL REPORT FINDINGS AFTER AUDIT - 11/81

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I.C. TASKS

I.C.1. GUIDANCE FOR EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

I.C.8. PILOT MONITORING OF SELECTED EP's FOR NTOL APPLICANTS

- SELECTED EP's SUBMITTED 5/19/81
- WALK THROUGH AT PALO VERDE SIMULATOR AND W-3 CONTROL ROOM CONDUCTED 7/27/81
- NRC TO COMPLETE REVIEW BY 8/15/81

I.C.2. SHIFT AND RELIEF TURNOVER PROCEDURES

I.C.3. SHIFT SUPERVISOR RESPONSIBILITIES

I.C.4. CONTROL ROOM ACCESS

I.C.5. PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

I.C.6. PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

- LP&L COMMITTED TO REQUIREMENTS IN FSAR AMENDMENT 16 (3/81)
- NRC WILL REPORT FINDINGS AFTER AUDIT - 11/81

I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

- CE WILL REVIEW S/U TESTS AND EOP'S

I. D. 1. CONTROL ROOM REVIEW

- PCRA 4/15/81
- NUREG-700 SUBMITTAL - 6/82

II. E.4.2. CONTAINMENT SYSTEM DESIGN

- ALL PARTS CLOSED EXCEPT: MINIMUM CONTAINMENT SETPOINT STUDY
- SUBMITTAL SCHEDULED - 12/81

II. F. 2. ICC INSTRUMENTATION

- SUBCOOLED MARGIN MONITOR INCORPORATED IN DESIGN
- RVLMS - UNDER EVALUATION
- CORE EXIT T/C MONITORING SYSTEM

II. D. 3. VALUE POSITION INDICATION

- DESIGN SUBMITTED 7/81
- APPROVED BY STAFF. WILL BE CLOSED IN SSER 1

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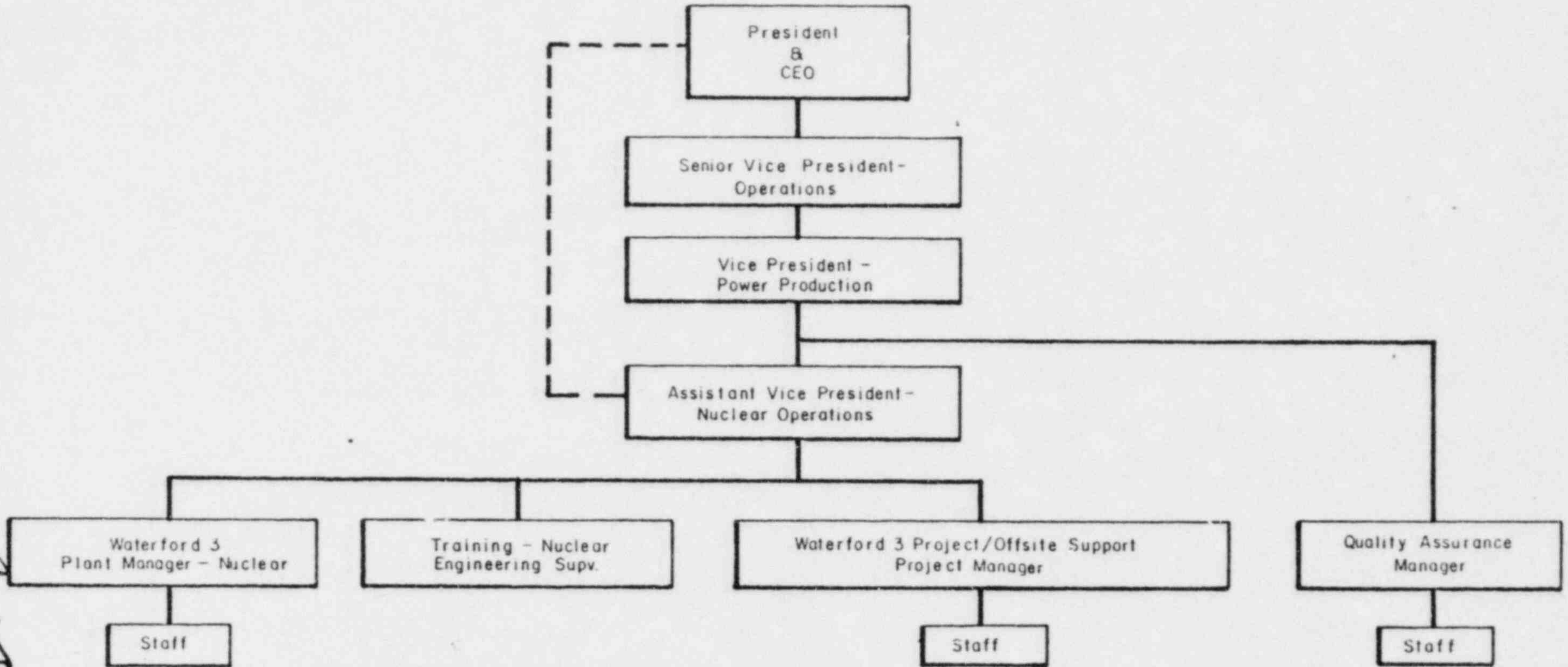
8/6/81

LP&L WATERFORD 3 CORPORATE ORGANIZATION



- Operation & Maintenance of the Facility

- Training Coordination both onsite and offsite

- Offsite Technical Support
- Offsite Engineering Support
- Offsite Environmental Health & Safety Support
- Offsite Administrative Services

Quality Assurance

— — — Parallel Reporting Path During Emergency or Critical Situations

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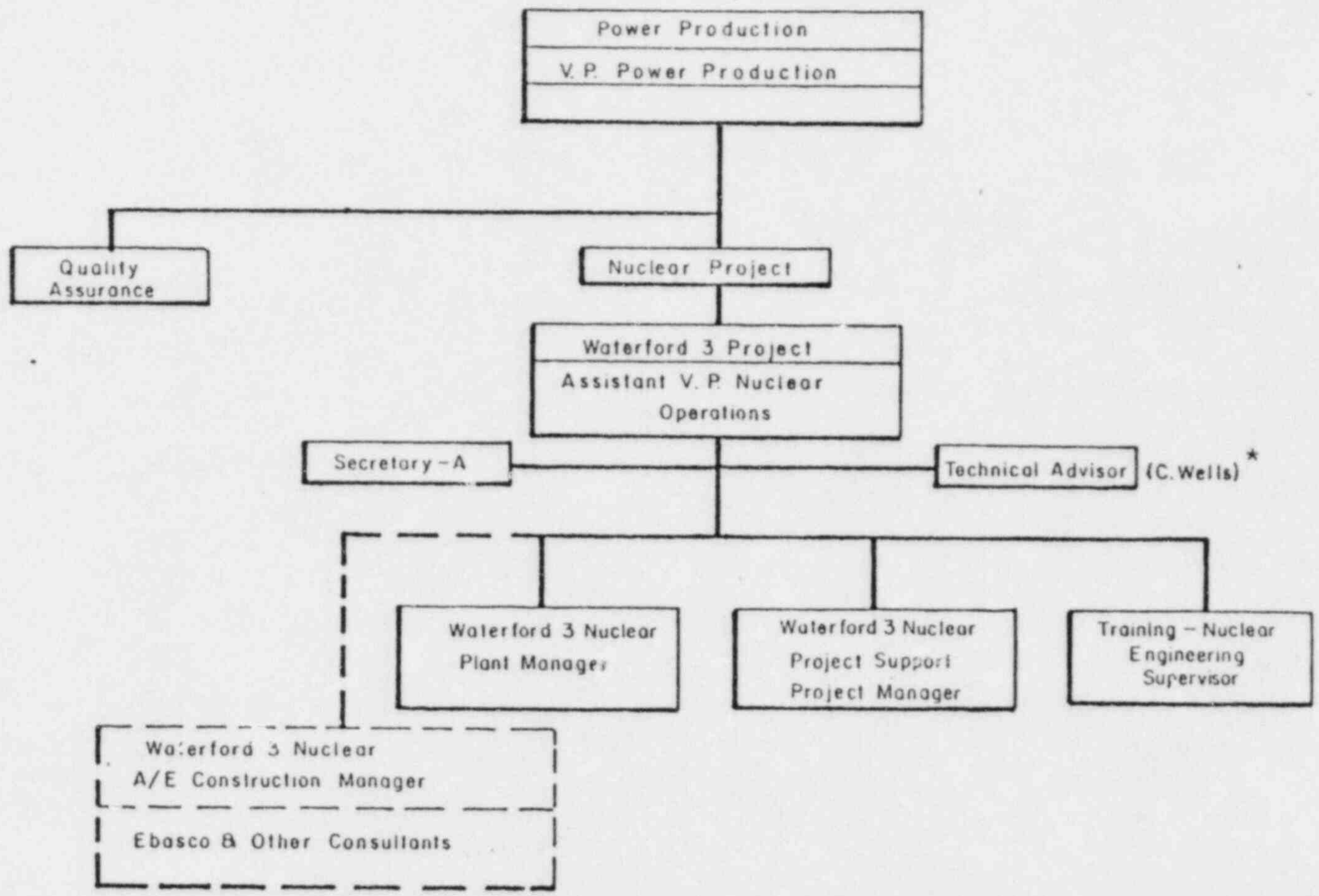
8/6/81



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PROJECT OFFSITE SUPPORT GROUP

WATERFORD 3 PROJECT/OFFSITE SUPPORT
Project Manager
E.J. DRUMMOND

Project Cont. Contract Adm.
- Engineering Office Supvsr. — X
- Engineering Technician

Onsite Safety Review
- Engineering Supervisor — ✓ (Accepted)
- Engineer
- Utility Engineer

Technical Services
- Engineering Supvsr. — X

Licensing
- Engineer
- Utility Engineer

Nuclear Eng. & Fuel Mng
- Engineer — X

Chem/Radiochem.
- Engineer — ✓

Rad. Control/Health Phy
- Engineer

Support
- Utility Engineer — X

Engineering
- Engineering Supvsr. — X

Maintenance
- Engineer

Refueling/Operations
- Engineer — X

Site Support
- Engineer — X

Support
- Utility Engineer — ✓
- Associate Engineer II/I — X, X

Mechanical Engineering
- Engineer

Civil Engineering
- Engineer — X

I & C Engineering
- Engineer — X, X

Electrical Engineering
- Engineer — X

X - POSITION FILLED
✓ - JOB OFFER MADE

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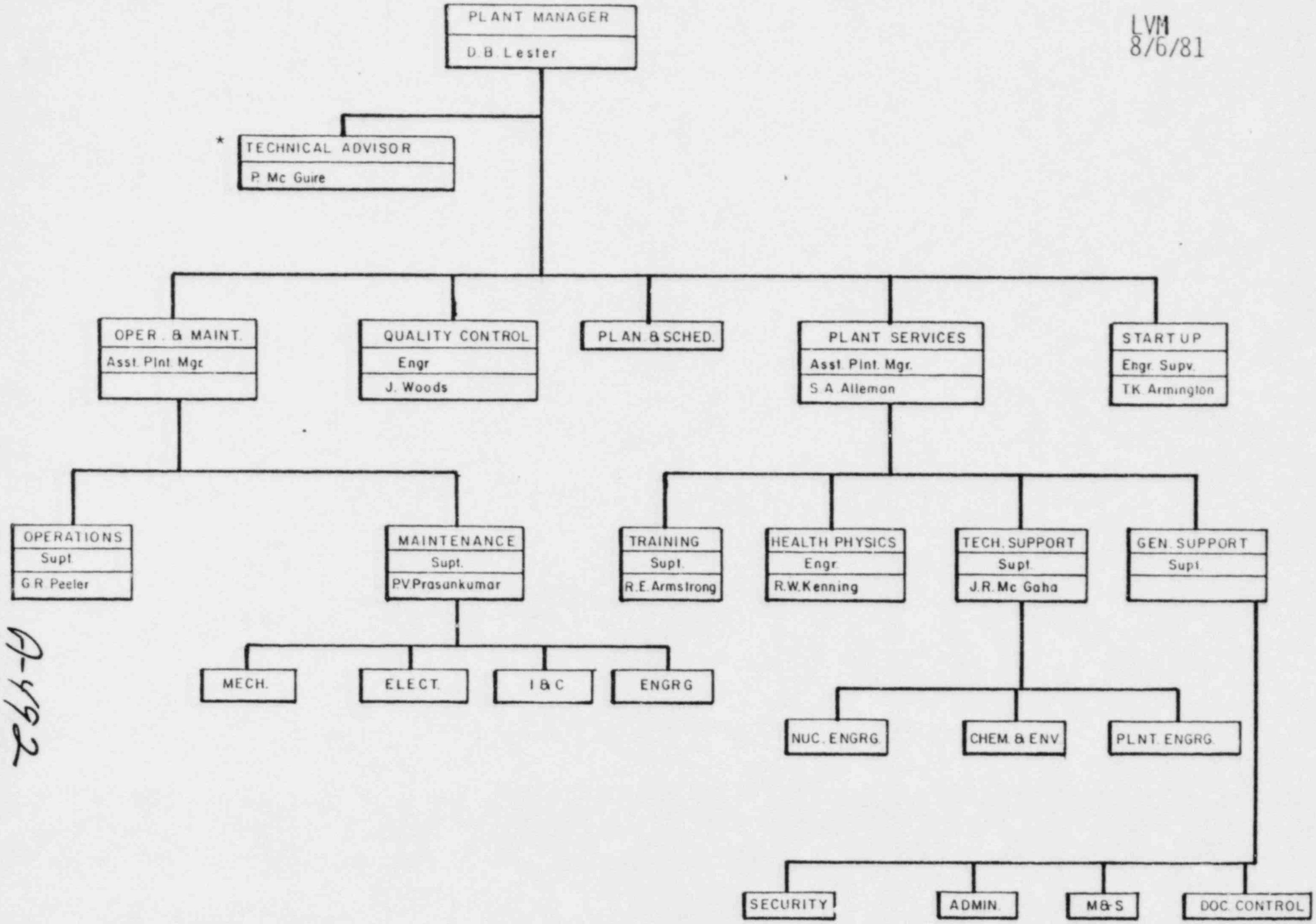
8/6/81



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8/6/81

SITE STAFFING

	AUTHORIZED	ON BOARD
PLANT STAFF	267	174
PLANT CONTRACT STAFF	-	207
STARTUP STAFF	14	11
STARTUP CONTRACT STAFF	-	131

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LVM
8-6-81

OPERATIONS

CLASSIFICATION	TOTAL AUTHORIZED	ON BOARD	SHIFT COMPLEMENT
OPS. SUPT.	1	1	
NOS	12	6	2
NPO	12	6	2
NAO	18	23	3
NAO-TRAINING	12	3	-
NAO-TRAINING (FOR START-UP)	12	11	-
	66	50	7

OTHERS

HP TECH.	1
RAD/CHEM. TECH.	1
STA	1

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8-6-81

OPERATOR QUALIFICATIONS

NAME	CLASS	LP&L SERVICE		NUCLEAR NAVY	COMM. EXP.	TRAINING COMPLETED					
		YR.	MO.			1	2	3	4	5	6
Peeler, G.R.	Ops.Supt.			x	x	x	x		x	x	x
Booher, R.	NOS		2	x	x	x	x	x	x	x	x
Bourgeois, M.	NOS	3	7	x		x	x	x	x	x	x
Edwards, J.	NOS	3	6	x		x	x	x	x	x	x
Ellard, J.	NOS	4	6	x		x	x	x	x	x	x
Smith, W.	NOS	3	7	x		x	x	x	x	x	x
Olsen, D.	NOS			x	x	x	x	x	x	x	x
Comeaux, J.	NPO	3	9	x		x	x	x	x	x	x
Pendergrass	NPO	3	9	x		x	x	x	x	x	x
Bowers, A.	NPO	3	0	x		x	x	x	x	x	x
Ortego, D.	NPO	2	11	x		x	x	x	x	x	x
Hoffpauir,J.	NPO	2	7	x		x	x	x			
Jones, M.	NPO	1	6	x		x	x	x			x
Beechem, C.	NAO	2	0	x		x	x	x	P		
Brinkley, R.	NAO		11	x		x	x	x	P		
Brown, K.	NAO		3	x		x	x	P			
Brown, T.	NAO		4	x		X	x	x	P		
Bumgardner,P.	NAO		4	x		x	x	x	P		
Burns, K.	NAO		6	x		x	x	x	P		
Collins, V.	NAO		4	x			x	x	x	P	
Fugate, C.	NAO		7	x		x	x	P			
Henderson, L.	NAO	2	1	x		x	x	x			x
Hampton, J.	NAO		4	x		x	x	P			
Kiech, G.	NAO		1	x				P			
Ledford, J.	NAO		4	x		x	x	P			
Macias, R.	NAO		7	x		x	x	x	P		
Meyers, B.	NAO	3	7			x	x	x	x	x	x
Miller, H.	NAO		4	x		x	x	P			
Miller, R.	NAO		7	x		x	x	x	P		
Mitchell, D.	NAO		11	x		x	x	x	P		
Tillman, M.	NAO	3	6	x		x	x	x	x	x	x
Timmons, R.	NAO	2	1	x		x	x	x	P		
Wemett, G.	NAO		7	x		x	x	x	P		
Vest, A.	NAO		6	x		x	x	x	P		
York, R.	NAO	1	4	x		x	x	x	P		x
Lietzke, B.	NAO			x							
Greer, H.	NAO-T		3								
Matherne, R.	NAO-T										
Legendre, K.	NAO-T										
Alexander, V.	NAO-T										
Champagne, G.	NAO-T										
Dennis, R.	NAO-T										
Oubre, R.	NAO-T										
Boudreaux, M.	NAO-T										
West, J.	NAO-T										
Matherne, B.	NAO-T										
Martin, R.J.	NAO-T										
LeBlanc, L.C.	NAO-T										
Delvin, M.J.	NAO-T										
Darnall, J.J.	NAO-T										

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x = Yes P = In Process

1) Power Plant Fundamentals
2) Advanced Academic Training
3) Research Reactor Training
4) Observation Training
5) Simulator
6) Plant Specific Lectures Series

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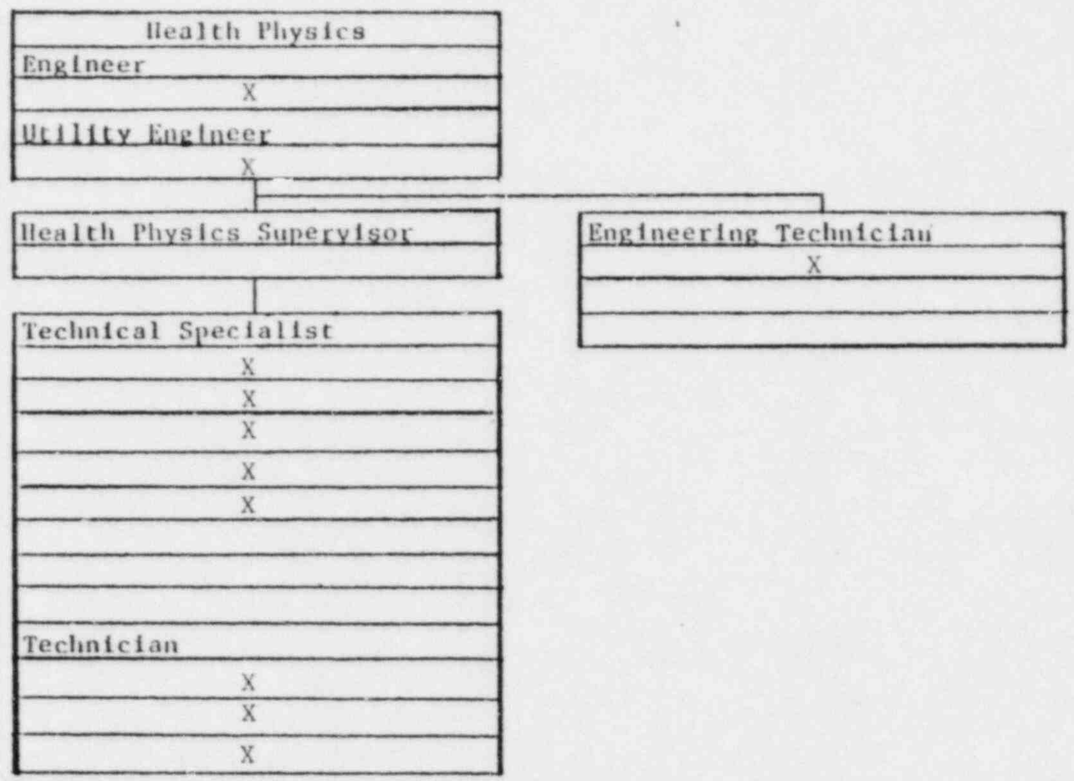
8/6/81

HEALTH PHYSICS ORGANIZATION



X - POSITIONS FILLED

There is another Associate Engineer I with an M.S. in Health Physics who is assigned to this Organization but is on another organization chart.

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8/6/81

HEALTH PHYSICS PERSONNEL

NAME	CLASSIFICATION	WF3 SERVICE Yr./Mo.	NUCLEAR NAVY YRS.	ELT	COMM. NUC. OPR. EXP. Yr./Mo.	OTHER HP EXP. Yr./Mo.
R.W. Kenning M.S. HP	Engineer	2-10	-	-	2-0	-
D.H. Espenan M.S. HP	Utility Engineer	1-0	-	-	0-1	1-1
L.R. Simon B.S. Ind. Tech.	Engineering Technician	3-3	-	-	0-3	6-4
R.T. Meschter A.S. HP	Technician	0-6	-	-	4-3	0-9
N.I. Huber	Technician	1-8	U.S. Army Nuclear 17 yrs.	x (Army)	0-5	-
D.L. Hoel	Technician	3-3	6	x	0-4	-
D.A. Landeche	Technician	1-9	6	x	0-2	-
D.B. Stevens	Technician	1-9	-	-	1-2	2-2
M.W. VanDerHorst	Technician	1-6	-	-	1-6	-
D.M. Hall M.S. HP	Ass. Eng. I	0-1	-	-	3-4	-
J.L. Bickham	Technician	0-1	-	-	4-8	2-2
J.H. Herring B.S. Biology	Technician	0-1	-	-	2-6	-

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CHEMICAL & ENVIRONMENTAL ORGANIZATION

- Chemical & Environmental Engineer
  - Utility Engineer
    - X
    - X
  - Radiochemistry Chem.& Env.Supvr.
    - Tech. Spec.
      - X
      - X
      - X
      - X
    - Technician
      - X
      - X
      - X
  - Chem.& Environ. Chem.& Env.Supvr.
    - X
    - Tech. Spec.
      - X
      - X
    - Technician
      - X

X - POSITIONS FILLED

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8-6-81

MAINTENANCE

CLASSIFICATION	TOTAL AUTHORIZED	ON BOARD
MECHANICS	23	11
ELECTRICIANS	10	6
INST. TECHS.	21	17
ENGR. TECHS.	10	4
SUPERVISORS	9	5

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8-6-81

MAINTENANCE DEPARTMENT QUALIFICATIONS

- THERE WILL BE TRAINED MAINTENANCE PERSONNEL IN SUFFICIENT QUANTITIES TO SAFELY AND EFFICIENTLY MAINTAIN WATERFORD 3.

- TRAINING WILL MEET THE REQUIREMENTS OF:
  1) ANSI/ANS-3.1
  2) REGULATORY GUIDE 1.8 - REV. 1R (REISSUED MAY 1977)

- PLANT PROCEDURES REQUIRE QUALIFICATION ON SYSTEMS AND EQUIPMENT. GOAL IS FOR TWO PEOPLE TO BE QUALIFIED ON EVERY SYSTEM OR MAJOR EQUIPMENT.

- TRAINING COMMITMENT

  1) 40,000 WORK HOURS EXPENDED ON TRAINING TO DATE (HALF IN I&C).

  2) 16,000 WORK HOURS EXPENDED ON TRAINING THUS FAR THIS YEAR.

  3) 20,000 WORK HOURS TO BE EXPENDED ON TRAINING IN THE BALANCE OF 1981.

A-500

LVM
8/6

SER INTERIM CONCLUSION (1)

"The applicant's recent proposal to assign experienced advisors to the Assistant Vice President and the Plant Manager is a step in the right direction for augmenting the staff's experiential knowledge."

S T A T U S

Technical Advisor to Assistant Vice President - Nuclear Operations	On Board 8/10/81
Technical Advisor to Plant Manager	On Board 5/28/81

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SER INTERIM CONCLUSION (1) (Continued)

"The staff considers in similar light the applicant's proposal to cover each operating shift with experienced personnel and to require PWR operating experience for the Assistant Plant Manager, Operations and Maintenance, and the Operations Superintendent."

STATUS

		Inplant Commercial Operating Nuclear Experience
Assistant Plant Manager - O&M	Offer Made	5 years
Operations Superintendant	On Board 7/27/81	7 years
NOS 1	On Board	4 years
NOS 2	On Board	10 years
NOS 3	Accepted	2 years
NOS 4, 5, 6	In Process	
On Site Safety Review Engineering Supervisor	Accepted	8 years
Technical Services Support Engineer	On Board	6 years

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8/6/81

TRAINING IN OPERATING NUCLEAR PLANTS
(Existing LP&L Employees)

All Operating License Candidates	10 wks. (minimum)
Offsite Support/Proj. Mgr.	10 wks.
Asst. Plant Mgr. - Plant Services	6 mos.
Start-up Manager	12 mos.
Maint. Supt.	6 wks.
H. P. Engr.	24 mos.
Chem. & Env. Engr.	8 mos.
Nuclear Engr.	30 mos.

SER INTERIM CONCLUSION (2)

"The applicant should accelerate its plan to acquire the needed corporate and plant staff personnel so that the organizational units will be functioning smoothly during the preoperational startup and test program."

STATUS

1. Personal involvement of management/executive personnel

2. Improvements in Recruiting/Retention Process/Techniques

   A. Pay Grade Increase for Technical Nuclear Positions.

   B. Improvements in Recruiting Techniques

      1) 3 Full-Time Recruiters On-Site.

      2) Local/National Advertising.

      3) Emphasis on "Selling" LP&L

      4) Improved Moving Policies for New Employees.

A-5-04

WATERFORD 3 RECRUITING

November 1, 1980 through July 31, 1981

	First Quarter*	Second Quarter	July 1981	TOTAL
Interviews	198	116	50	364
Offers	83	63	15	161
Acceptances	42	35	7	84
In Process				27
Offers Outstanding				7
LP&L Employees Interviewed	88	5	8	101
LP&L Transfers	6	1	0	7

* First Quarter represents November 1, 1980 through March 31, 1981.

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LVM
8/6

SER INTERIM CONCLUSION (3)

"After the applicant has staffed the offsite support groups and plant operating and support groups with supervisory personnel, and these groups have been functioning for several months, the staff will complete its review, including an audit team visit, and will report the results of that review in a supplement to this report."

STATUS

Monthly report to Dr. Hanauer on recruiting and staffing.

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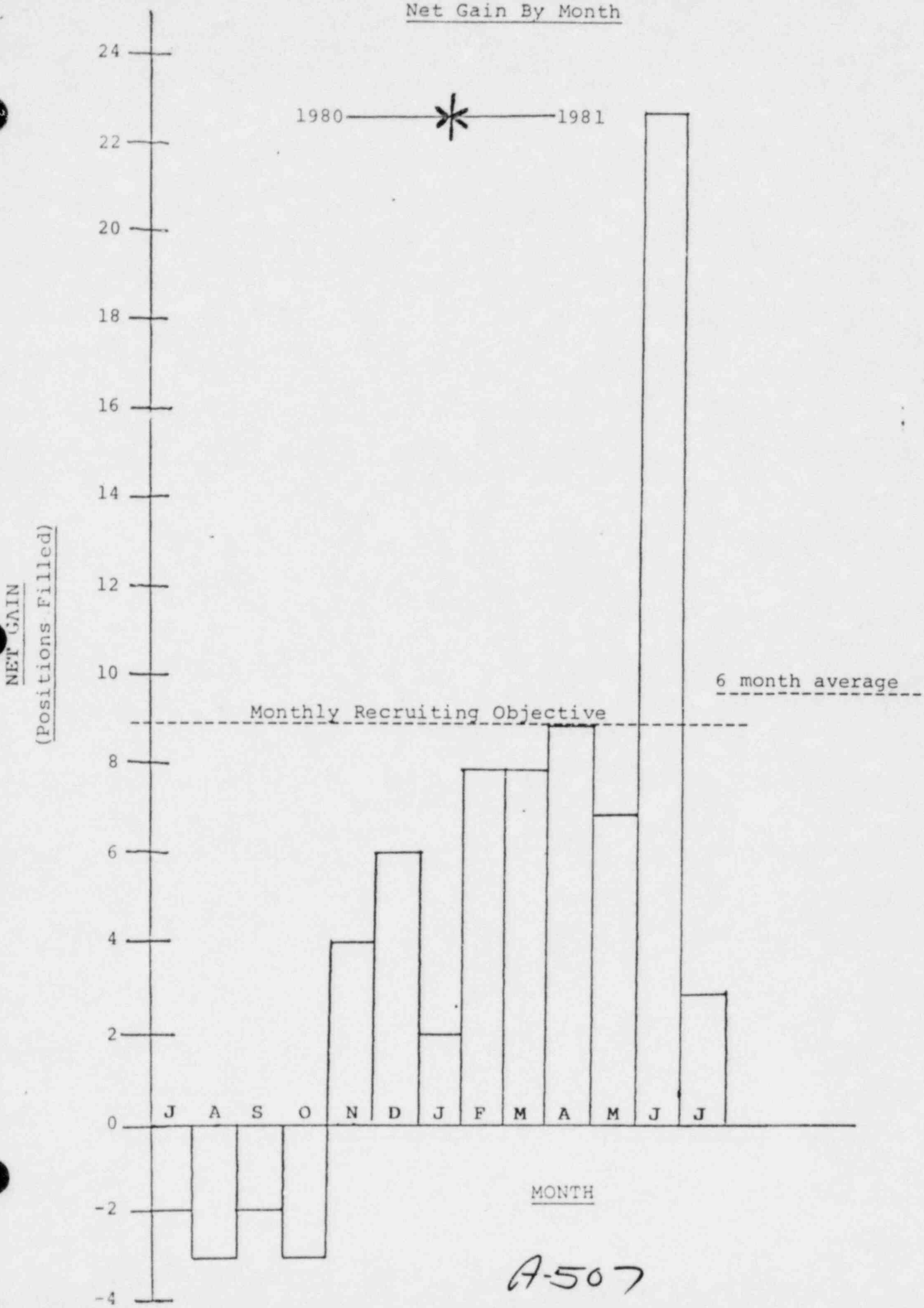
WATERFORD 3 PLANT STAFFING

Net Gain By Month



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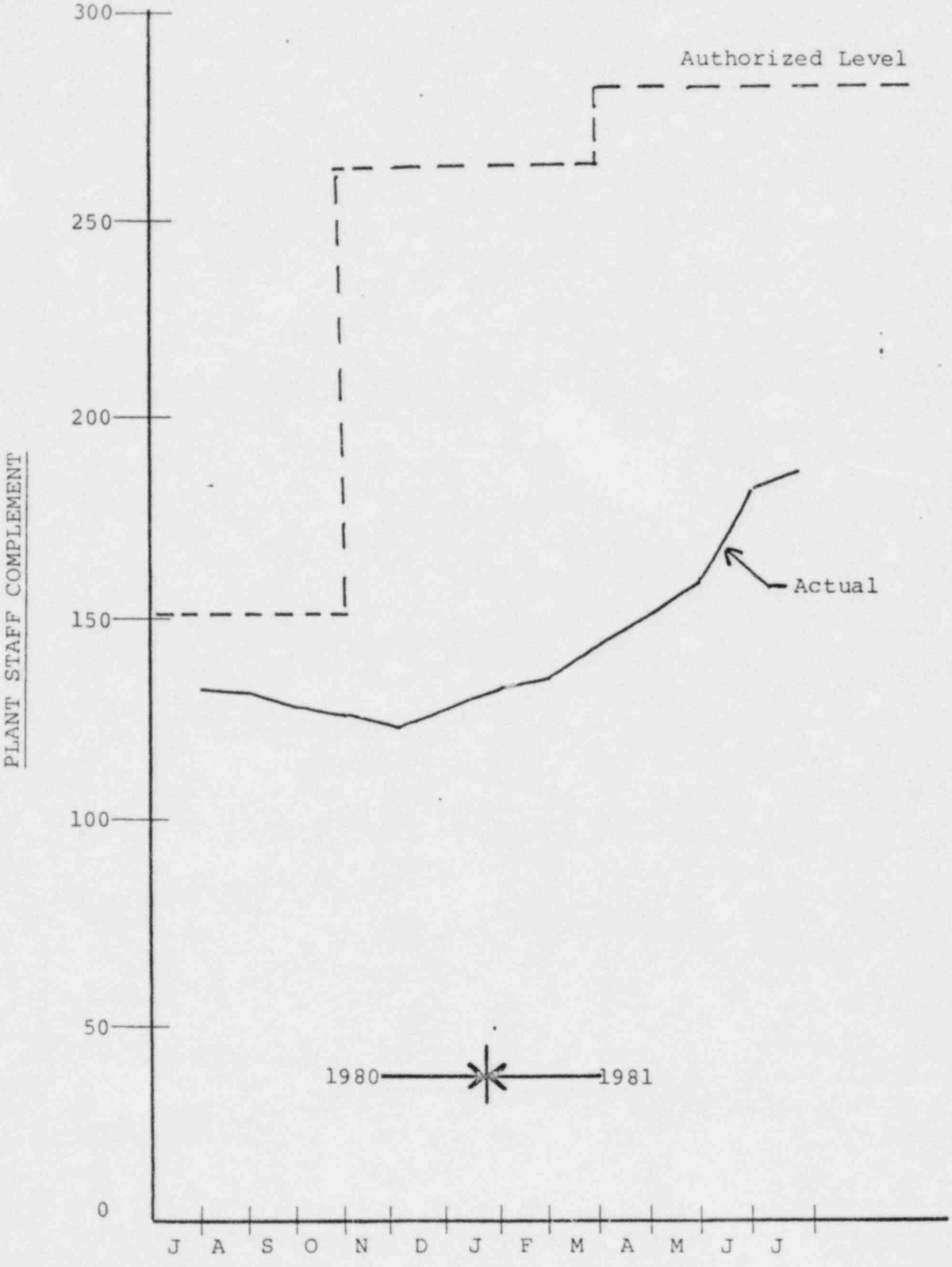
<u>CUMULATIVE PLANT STAFFING</u>



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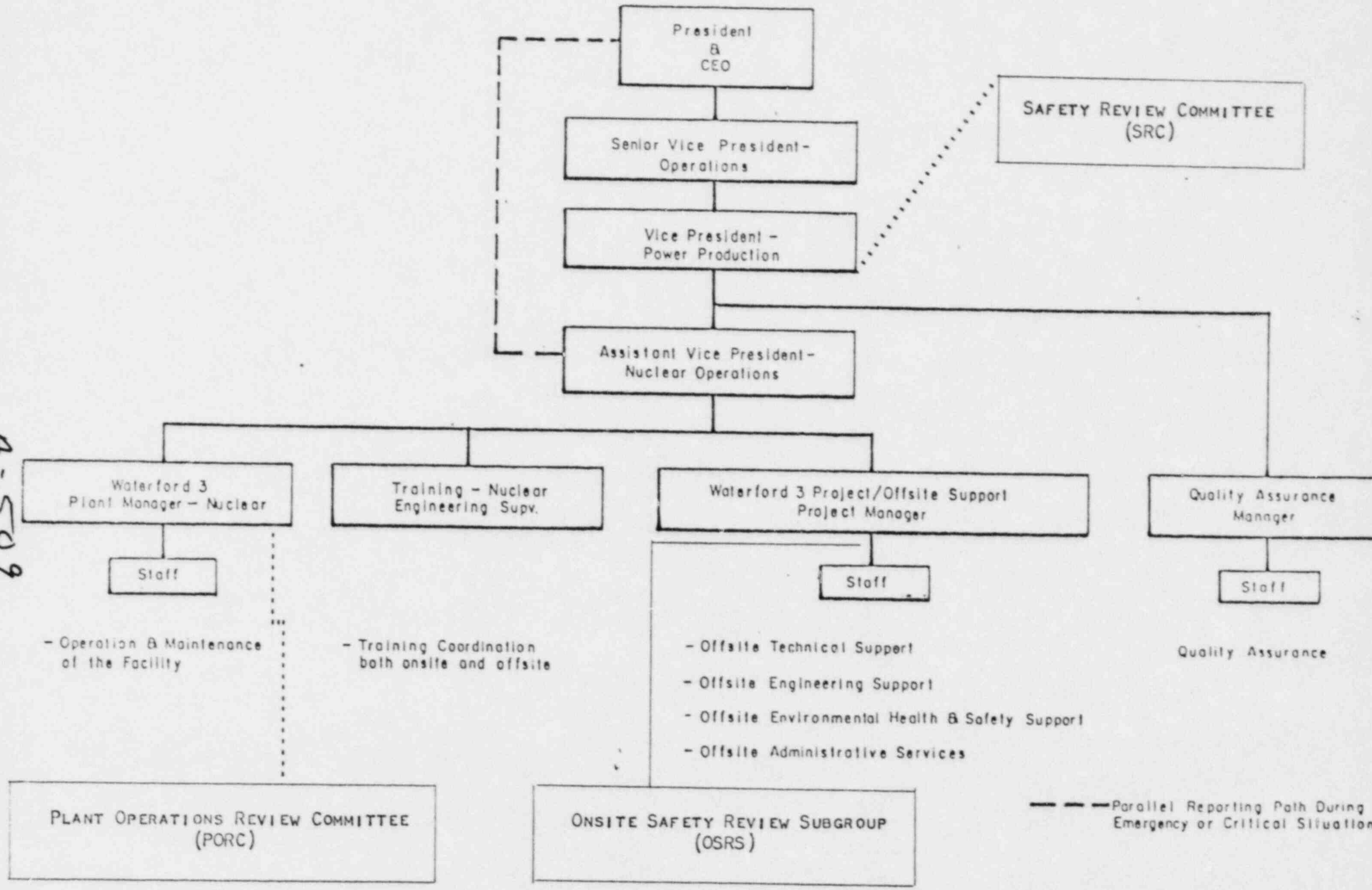
ACRS

8/6/81

LP&L WATERFORD 3 CORPORATE ORGANIZATION



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8/6/81

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PLANT OPERATIONS REVIEW COMMITTEE

- DESIGNATED ASSISTANT PLANT MANAGER
- TECHNICAL SUPPORT SUPERINTENDENT
- MAINTENANCE SUPERINTENDENT
- OPERATIONS SUPERINTENDENT
- NUCLEAR ENGINEER
- HEALTH PHYSICS ENGINEER
- INSTRUMENTATION & CONTROL ASSISTANT SUPERINTENDENT
- QUALITY CONTROL ENGINEER

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8/6/81
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SAFETY REVIEW COMMITTEE

- ASSISTANT VICE PRESIDENT - NUCLEAR OPERATIONS
- WATERFORD 3 PROJECT/OFFSITE SUPPORT - PROJECT MANAGER
- QUALITY ASSURANCE MANAGER
- MANAGER SYSTEM NUCLEAR OPERATIONS (MIDDLE SOUTH SERVICES)
- WATERFORD 3 PLANT MANAGER
- ONSITE SAFETY REVIEW SUBGROUP - ENGINEERING SUPERVISOR
- WATERFORD 3 PROJECT/OFFSITE SUPPORT GROUP TECHNICAL SERVICES - ENGINEERING SUPERVISOR
- TRAINING - NUCLEAR ENGINEERING SUPERVISOR

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8/6/81

ACRS

ONSITE SAFETY REVIEW SUBGROUP

- FIVE DEDICATED, FULL-TIME, SITE-BASED ENGINEERS
- TECHNICAL DISCIPLINES AS FOLLOWS:
  - ELECTRICAL
  - INSTRUMENTATION & CONTROLS
  - MECHANICAL
  - RADIATION PROTECTION
  - NUCLEAR
- QUALIFICATIONS AS FOLLOWS:
  - SUPERVISOR
    - BS IN ENGINEERING OR PHYSICAL SCIENCE
    - 8 YEARS RESPONSIBLE EXPERIENCE
    - 3 YEARS NUCLEAR EXPERIENCE
  - OTHER MEMBERS
    - BS IN ENGINEERING OR PHYSICAL SCIENCE
    - 2 YEARS NUCLEAR EXPERIENCE IN DISCIPLINE

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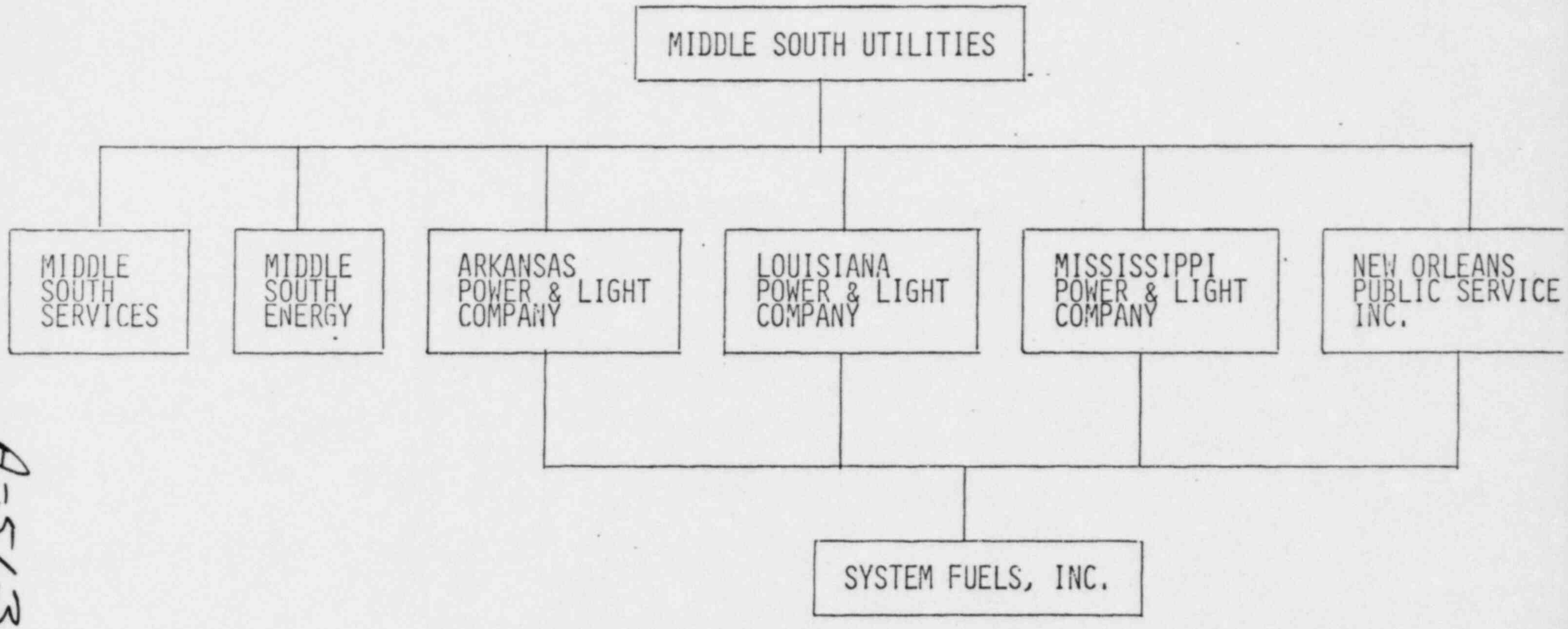
MIDDLE SOUTH UTILITIES ORGANIZATION



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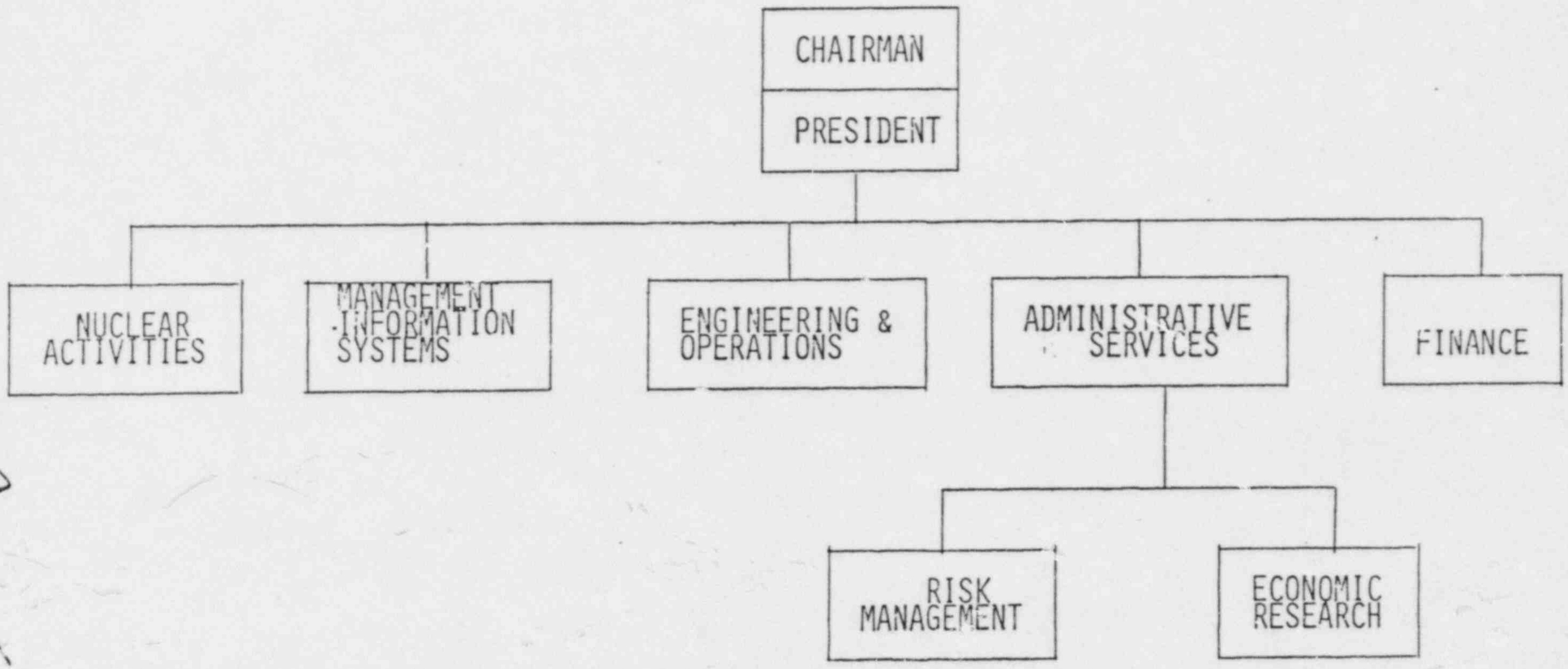
MIDDLE SOUTH SERVICES, INC. ORGANIZATION

(SIMPLIFIED)



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THE RESPONSIBILITY OF MIDDLE SOUTH SERVICES, INC.

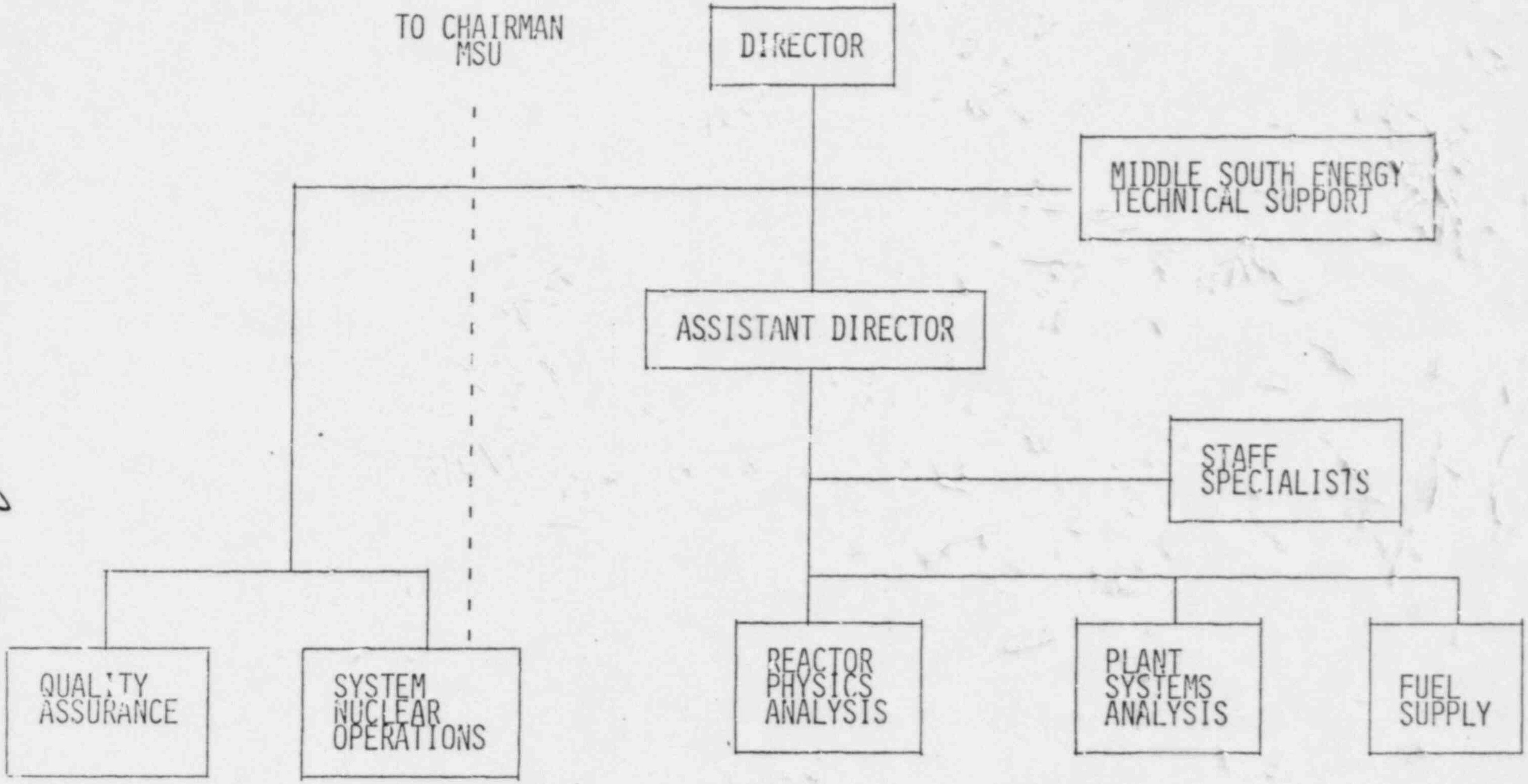
To provide quality and timely support to the Operating Companies and management of the Middle South Utilities System.

For Waterford 3, these services include:

- Risk Management
- Management Information Services
- Engineering
- Nuclear Activities
  - Physics Analysis
  - Plant Systems Analysis
  - Technical and licensing review and evaluation
  - Operations Monitoring and Support
  - Nuclear Fuel Supply
  - Quality Assurance
  - Nuclear Project Support

A-515

MIDDLE SOUTH SERVICES, INC. NUCLEAR ACTIVITIES DEPARTMENT



A-5/6

MIDDLE SOUTH SERVICES, INC.

NUCLEAR ACTIVITIES DEPARTMENT QUALIFICATIONS

- 32 PROFESSIONALS
- 26 DEGREED ENGINEERS
- 18 ADVANCED DEGREES INCLUDING 4 PH.D'S.
- CERTIFIED LEVEL III NDE EXAMINER
- CERTIFIED ASME SECTION III BOILER & PRESSURE VESSEL INSPECTOR
- CERTIFIED WELDING INSPECTOR TO SECTION 6.1 OF AWS STANDARD
- OVER 250 MAN-YEARS OF NUCLEAR POWER EXPERIENCE
- ORGANIZATION HAS DOUBLED IN LAST TWO YEARS
- 22 ADDITIONAL PROFESSIONALS AUTHORIZED THROUGH 1982
- PLAN TO HAVE 12 NUCLEAR ANALYSTS COMMITTED TO WATERFORD-3 BY SPRING 1984

A-517

NUCLEAR ANALYSIS SUPPORT SCOPE

RELOAD CYCLE DESIGN, SAFETY EVALUATION AND ENGINEERING SUPPORT

OPERATIONS ANALYSES

LICENSING ANALYSES

REACTOR SYSTEM DESIGN CHANGE EVALUATIONS

ENGINEER/OPERATOR TRAINING

TECHNOLOGICAL EXPERTISE

NEUTRON PHYSICS (ARMP)

REACTOR THERMAL-HYDRAULICS (COBRA)

PLANT SYSTEM TRANSIENT ANALYSIS (RETRAN)

FUEL ROD THERMAL-MECHANICAL PERFORMANCE (COMETHE)

SHIELDING ANALYSIS (MORSE/CG)

COMPUTER CODES DEVELOPED BY EPRI

NRC QUALIFICATION REPORT PROGRAM UNDER WAY

REA

8/6/81

THERMAL SCIENCES INTEGRATED TRAINING

A. ACADEMIC TRAINING - DR. F.J. BROWN AND ASSOCIATES

1)	OPERATOR TRAINING	20 HOURS
2)	STA TRAINING (AVAILABLE TO OPER)	90 HOURS
3)	ADVANCED ACADEMIC (FOR OPER)	90 HOURS

B. POWER PLANT FUNDAMENTALS

1)	NUS CORP. NET SERIES	40 HOURS
2)	CE SPECIAL TRAINING	40 HOURS

C. NSSS LECTURE SERIES

1)	CE STANDARD PRESENTATION	16 HOURS

* TOTAL CONTACT HOURS 206

* DOES NOT INCLUDE UNIVERSITY OF FLORIDA RESEARCH REACTOR TRAINING COURSE.

NOTE: DR. BROWN & ASSOCIATES AVAILABLE TO DO REMEDIAL TRAINING OR ADVANCED TRAINING AS REQUESTED BY LP&L.

A-519

8/6/81

ACRS

OVERVIEW OF SIMULATOR USES

- INITIAL LICENSE TRAINING
  - COLD LICENSE OPERATOR CANDIATES
  - HOT LICENSE OPERATOR CANDIATES
- ANNUAL REQUALIFICATION TRAINING
  - LICENSED OPERATORS
  - SHIFT TECHNICAL ADVISORS
- I&C TECHNICIAN TRAINING
  - REACTOR PROTECTION SYSTEM
- TRAINING INSTRUCTOR CERTIFICATION
- PROCEDURE DEVELOPMENT & VERIFICATION

A-520 .

AI 6.3-1

8/6/81

ACRS

TYPICAL CAPABILITIES OF A FULL SCOPE SIMULATOR

NORMAL OPERATION

- PLANT STARTUP - COLD TO HOT STANDBY
- TURBINE STARTUP AND GENERATOR SYNCHRONIZATION
- POWER ESCALATION TO 100% POWER
- POWER SYSTEM LOAD CHANGES (MANUAL AND AUTOMATIC CONTROL)
- REACTOR TRIP FOLLOWED BY RECOVERY TO 100% POWER
- OPERATIONS AT HOT STANDBY
- PLANT SHUTDOWN AND COOLDOWN TO COLD (REFUELING) CONDITIONS
- SURVEILLANCE TEST ON SAFETY-RELATED EQUIPMENT OR SYSTEMS (FROM CONTROL ROOM)
- PROCESS COMPUTER RESPONSE DURING NORMAL AND TRANSIENT OPERATION

ABNORMAL OR EMERGENCY OPERATION

- EMERGENCY OPERATING CONDITIONS (INCLUDING SATURATED CONDITIONS)
- ANTICIPATED TRANSIENTS WITHOUT SCRAM
- VARIABLE AND CONDITIONAL MALFUNCTIONS
- SIMULTANEOUS MALFUNCTIONS
- THREE-DIMENSIONAL CORE MODELS ACCURATELY CALCULATE FLUX TILT
- POWER OPERATION WITH REACTOR COOLANT PUMP(S) OUT OF SERVICE
- NATURAL CIRCULATION FOR DECAY HEAT REMOVAL
- POST LOSS OF COOLANT CONFIGURATIONS
- PRIMARY PLANT HYDROSTATIC TEST (FROM CONTROL ROOM)
- CORE PHYSICS TESTING AFTER FUEL LOAD OR RELOAD
- OPERATION WITH PLUGGED STEAM GENERATOR TUBES
- OPERATION WITH MISSING TURBINE BLADES

-2- A-521

AI 6.3-2

8/6/81

ACRS

LP&L SIMULATOR COMMITMENTS

- PLANT SPECIFIC SIMULATOR BY JANUARY, 1985
- SIMULATOR WILL MEET OR EXCEED THE REQUIREMENTS OF ANSI/ANS - 3.5 - 1979
  - FULL SCOPE SIMULATOR
  - UPDATED DATA BASE
- TRAINING COMMITMENTS PRIOR TO W-3 SIMULATOR AVAILABILITY
  - 1982 (22 weeks)
  - 1983 (16 weeks)
  - 1984 (16 weeks)

A-522

AI 6.3-3

8/6/81

ACRS

LP&L SIMULATOR COMMITMENTS

- PLANT SPECIFIC SIMULATOR BY JANUARY, 1985

- SIMULATOR WILL MEET OR EXCEED THE REQUIREMENTS OF ANSI/ANS - 3.5 - 1979

    - FULL SCOPE SIMULATOR

    - UPDATED DATA BASE

- TRAINING COMMITMENTS PRIOR TO W-3 SIMULATOR AVAILABILITY

    - 1982 (22 weeks)
    - 1983 (16 weeks)
    - 1984 (16 weeks)

A-523

WMA

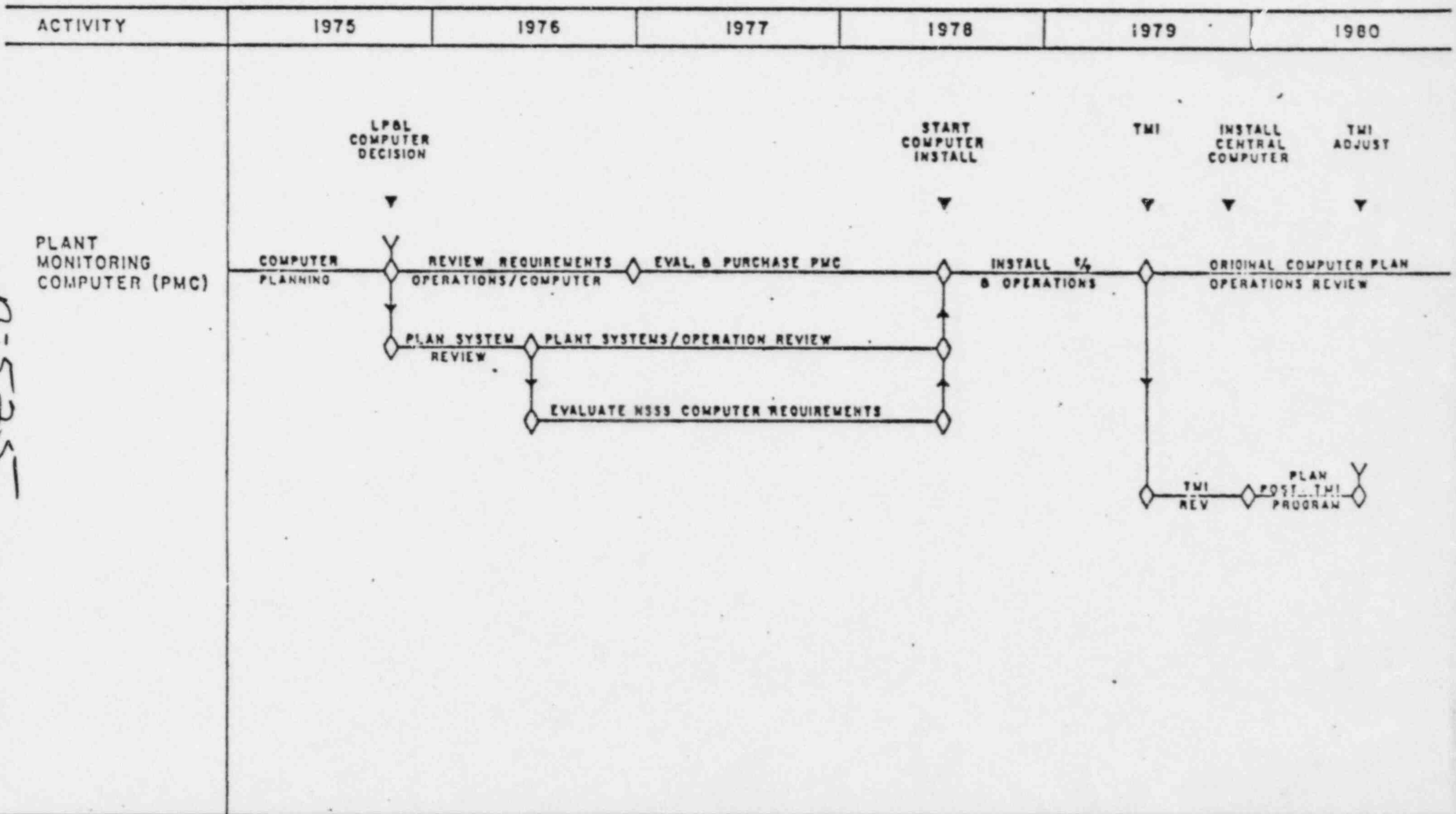
ACRS

AGENDA

- CONTROL ROOM REVIEW
- PLANT COMPUTER
- SAFETY PARAMETER DISPLAY SYSTEM
- OPERATOR PROCEDURES

A524

WATERFORD 3 ORIGINAL
PLANT OPERABILITY REVIEW
PROGRAM PLAN



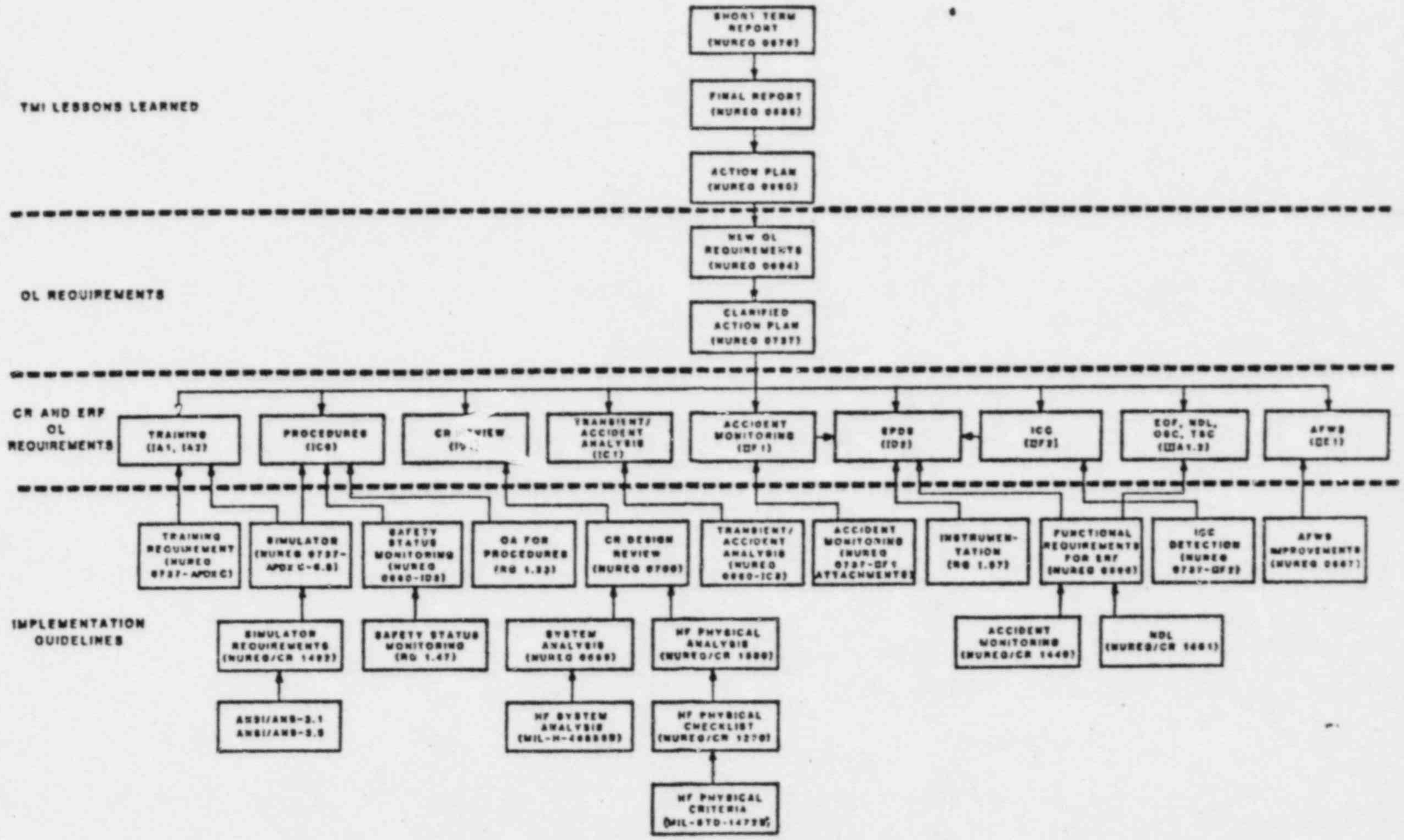
A-525.

INTEGRATED AND DIRECTED PROGRAM FOR NUREG 0700 OPERABILITY ASSURANCE REVIEW

- PROGRAM ELEMENTS
  - REQUIREMENTS
  - DEFINITION
  - INTERFACES
  - ORGANIZATION
  - MANAGEMENT
  - APPROACH AND RESULTS
- OPERABILITY IMPLEMENTATION PLAN

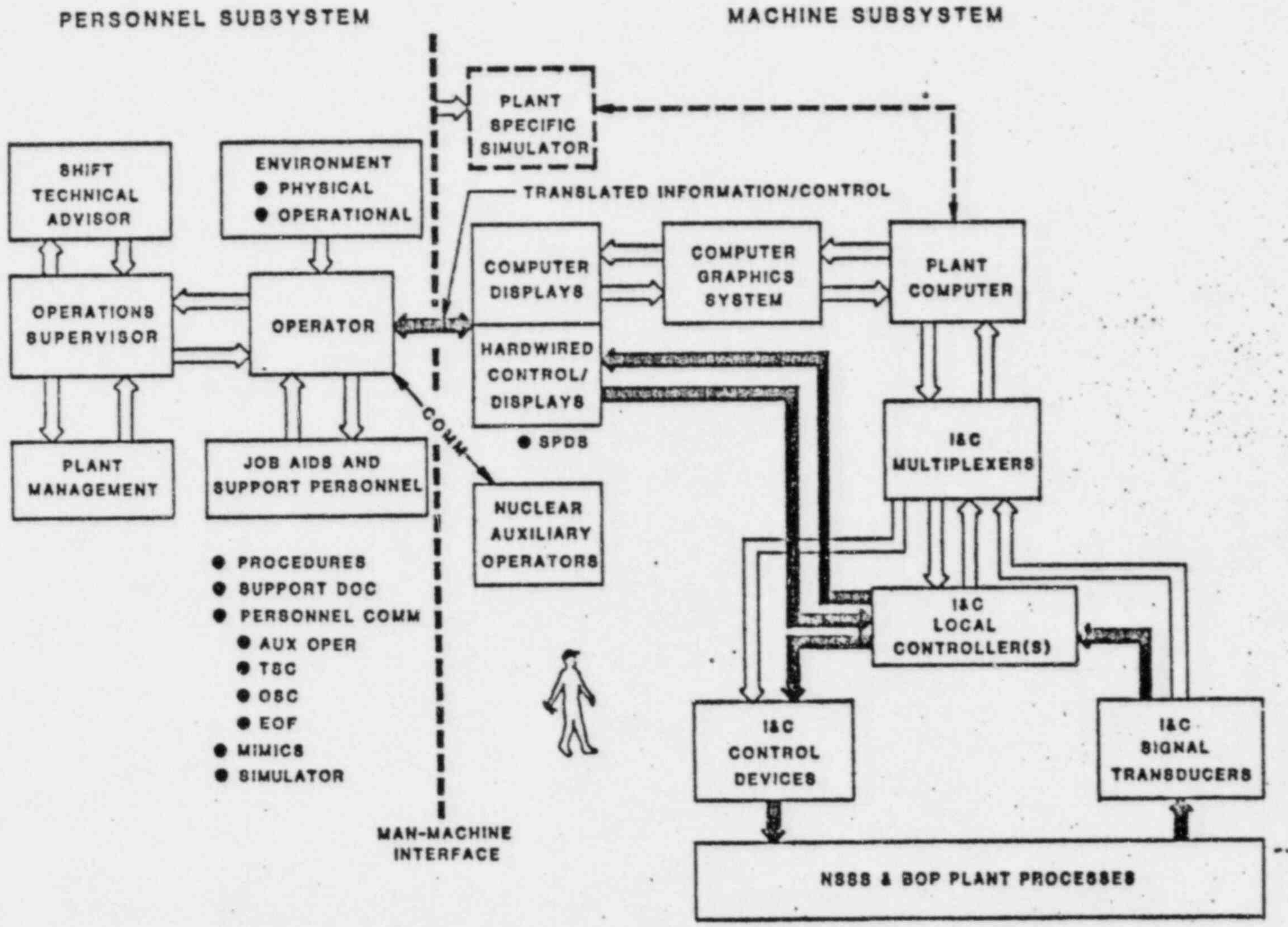
A-52C

NPP POST-TMI OL REQUIREMENTS
AND
IMPLEMENTATION GUIDELINES



A-527

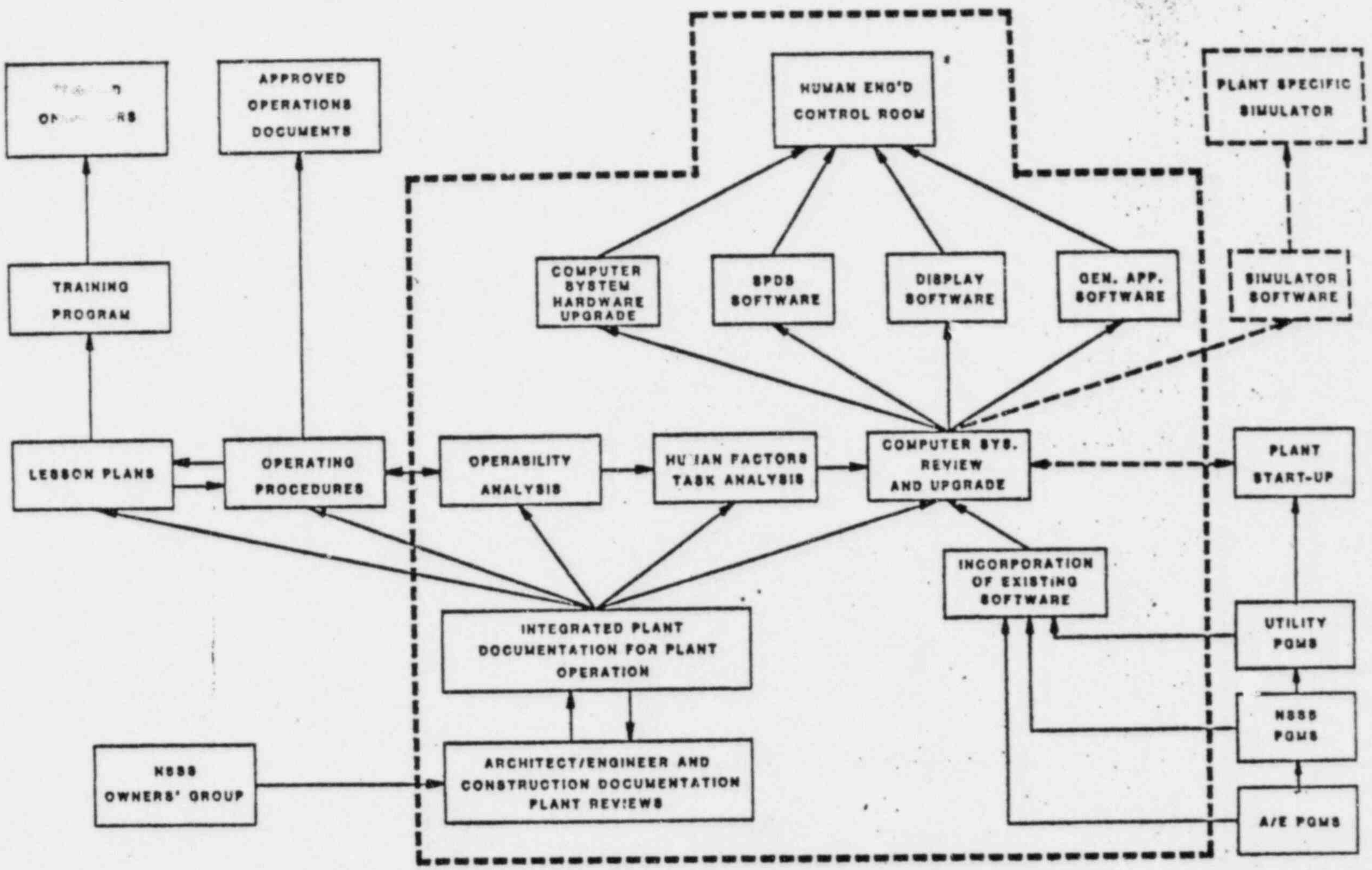
THE NPP MAN-MACHINE SYSTEM MODEL DEFINITION



A-528

PROGRAM INTERFACES

-SYSTEMS ENGINEERING INTEGRATION OF PLANT OPERABILITY REVIEW ACTIVIES-



POST-TMI EMPHASIS FOR OPERABILITY ANALYSES

PROGRAM ORGANIZATION

LP&L PROGRAM MANAGEMENT

↓

SYSTEMS ENGINEERING MANAGEMENT
LOS ALAMOS TECHNICAL ASSOCIATES

↓

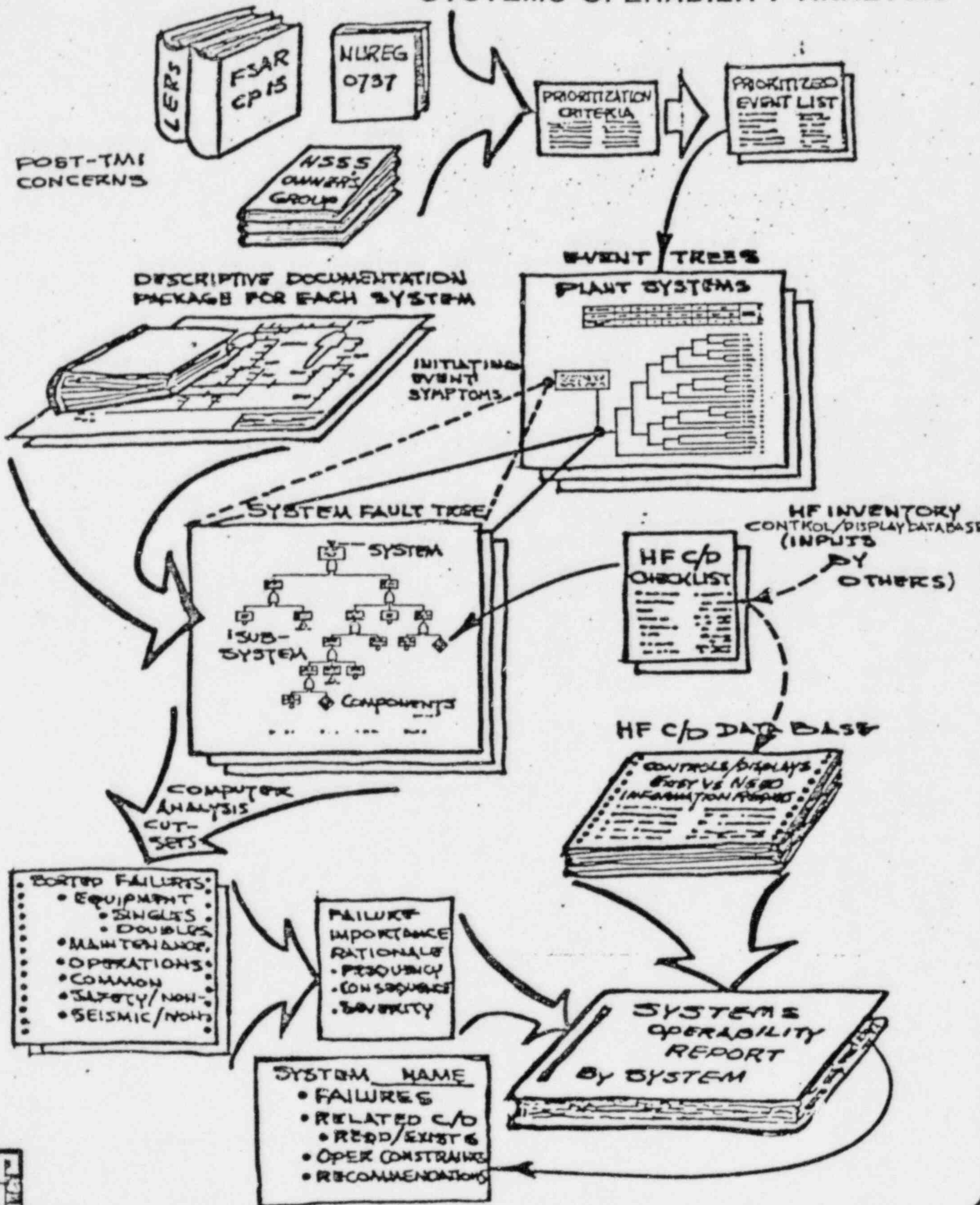
PROGRAM FUNCTIONS	CEOG	SAI	LP&L	EBASCO	LMSC	CSDL	PROGRAM ACTIVITIES SUPPORTED
POST TMI NSS	P						TRANSIENT RESPONSE (IC1) TRAINING, PROCEDURES (IC2)
OPERABILITY ANALYSIS		P		S	S		FAULT TREES, OPERABILITY REVIEW, INFORMATION/CONTROLS DATA BASE
SENIOR OPERATOR EXPERIENCE REVIEWS			P	S	S		REVIEW, EXPERIENCE GUIDANCE
EVENTS/CONCERNS ANALYSIS		S	S	P	S		EVENT SELECTION-RATIONALE, PRIORITIZATION, EVENT TREES
HUMAN FACTORS ENGINEERING	S				P		INFO RQMTS, CONTROL/DISPLAYS, TASK ANALYSIS, ENVIRONMENTS, PROCEDURES
OPERATIONS TRAINING	S		P		S		LESSON PLANS PROCEDURES
HARDWARE IMPLEMENTATION			P	P			RECOMMENDATION REVIEWS, CONFIGURATION CONTROL BOARD
PLANT COMPUTER DEVELOPMENT			S	S	S	P	REQUIREMENTS, RELIABILITY/AVAIL. SPDS PROCESSING-DISPLAY, OPERATOR INFO. SUPPORT, APP. PROGRAMS
PLANT COMPUTER START-UP			P			S	PLANT COMPUTER START-UP SYSTEM, CHECKOUT-TEST
PLANT REFERENCE DOCUMENTATION		S	S	P			COLLECT, REVIEW, ORGANIZE PLANT DOCUMENTATION DATA BASE

P PRIMARY RESPONSIBILITY **S** SUPPORTING RESPONSIBILITY

A-530

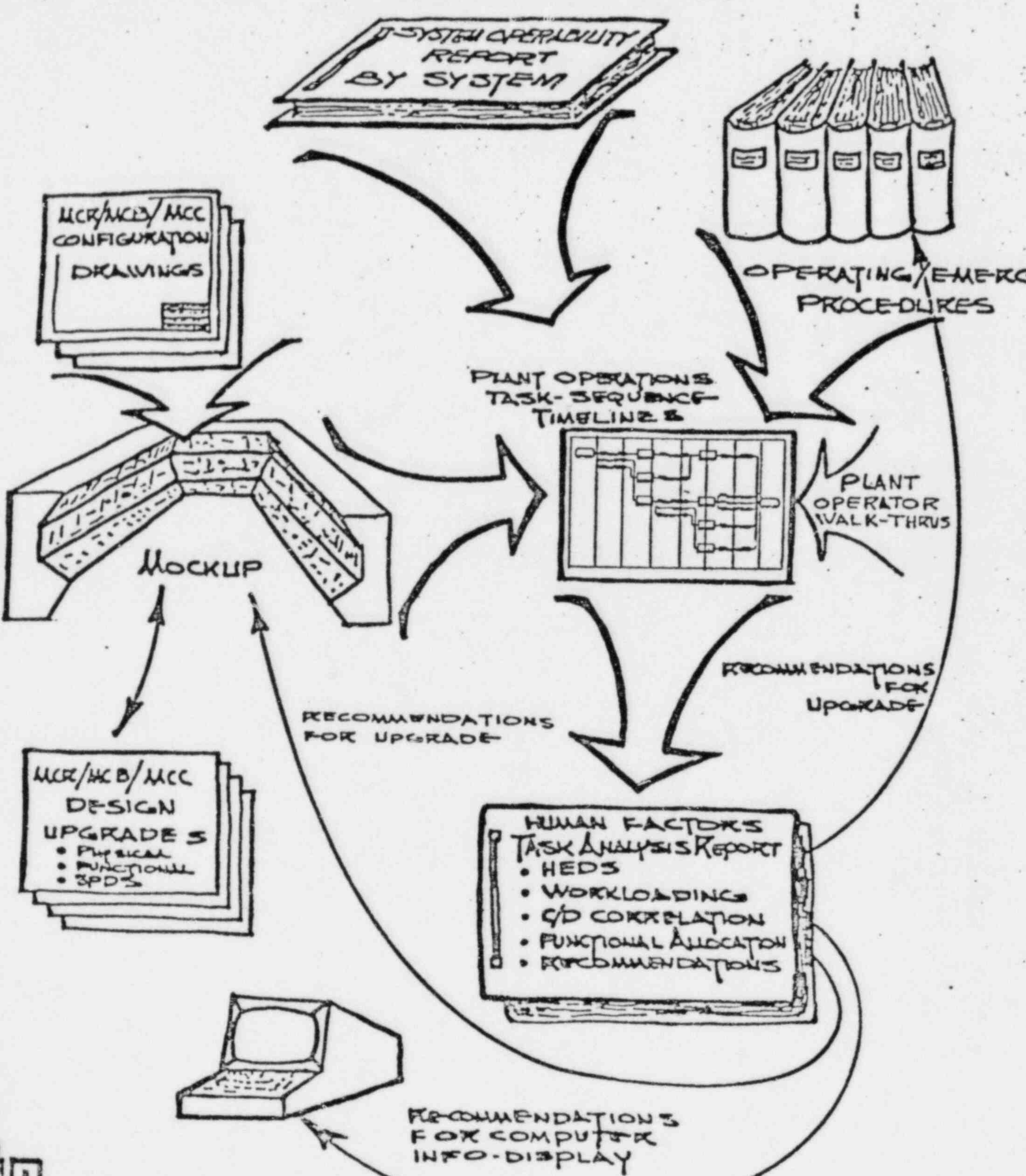
APPROACH AND RESULTS

-SYSTEMS OPERABILITY ANALYSIS-



APPROACH AND RESULTS

-HUMAN FACTORS ENGINEERING ANALYSIS-



PLANT COMPUTER SYSTEM REQUIREMENTS

THREE MAJOR AREAS:

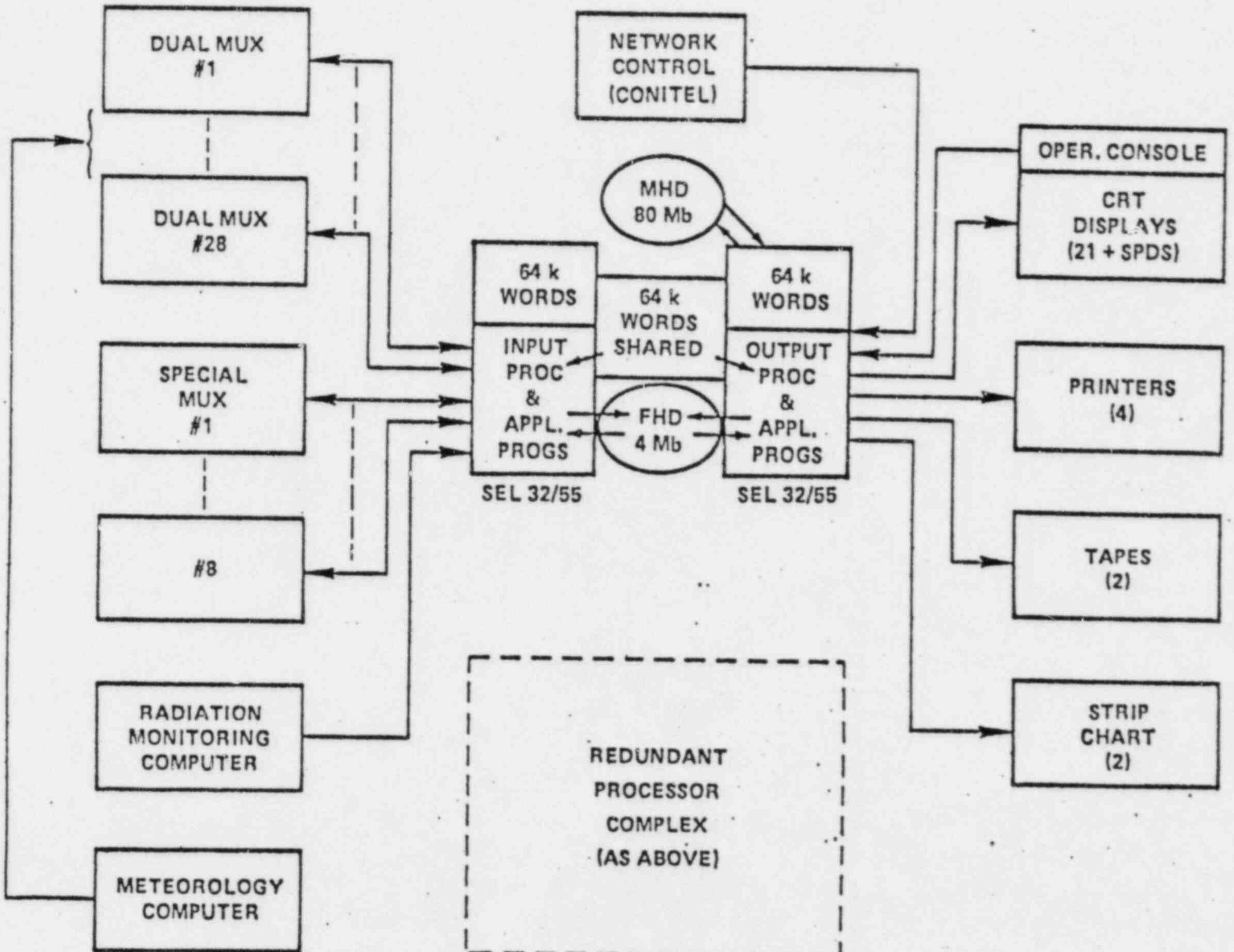
1) Control and sequencing

2) Monitoring and alarming

3) Evaluation and analysis

A-5-33

WATERFORD III PLANT COMPUTER SYSTEM



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A-534

UNIQUE COMPUTER SUPPORT CAPABILITIES
AT
WATERFORD 3

- DESIGN INITIATED 3 YEARS PRIOR TO TMI-2

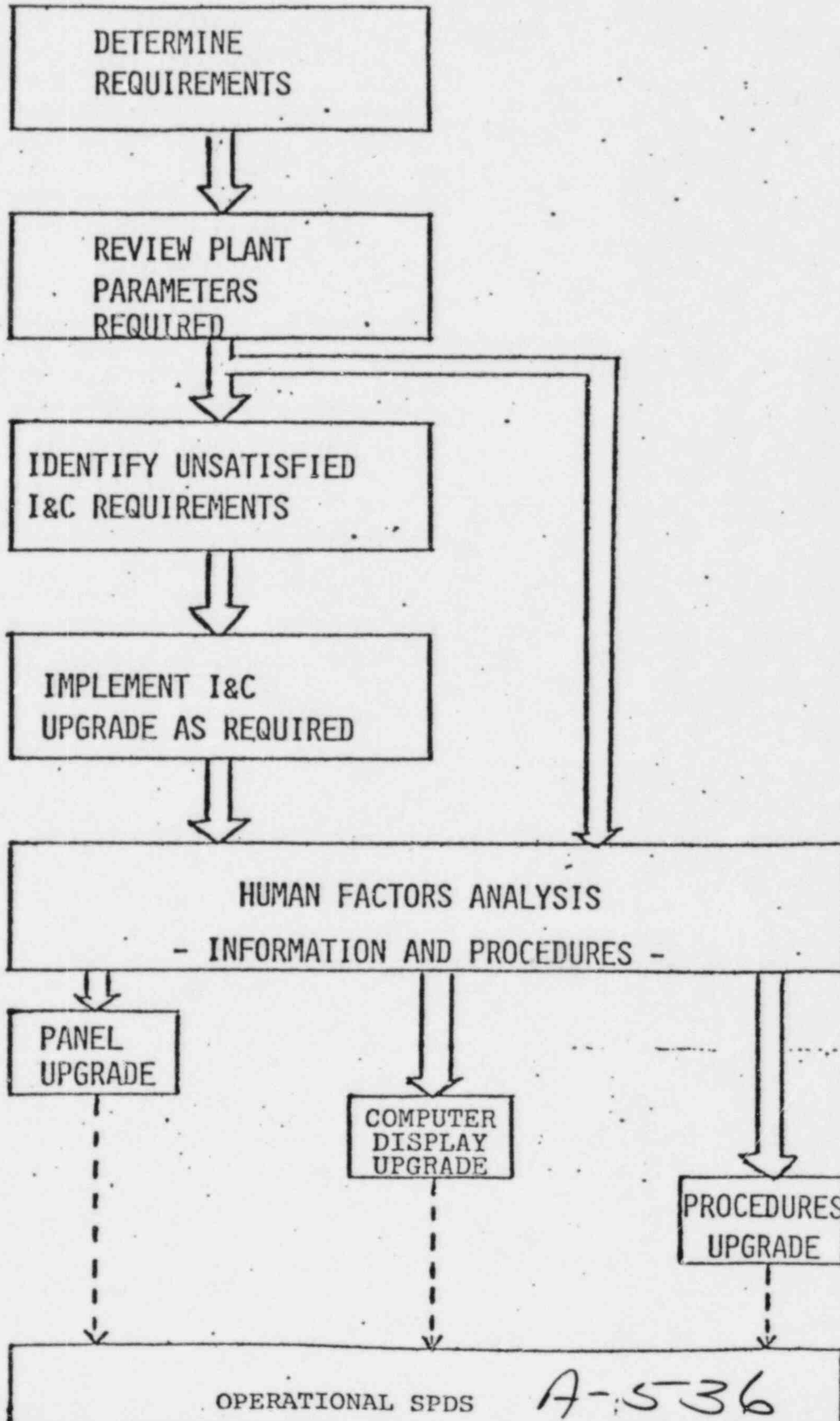
• I/O POINTS	6500	-- EXTENSIVE DATA ACQUISITION AND CONTROL FOR ON-LINE OPERATOR SUPPORT
• AVAILABILITY	>0.99	-- REDUNDANT COMPUTER HARDWARE
• OPERATOR SUPPORT	21 COLOR CRTs 7 KEY BOARDS	-- EXTENSIVE CAPABILITY FOR OPERATOR INFORMATION DISPLAY AND INTERACTION

THIS PROVIDES SIGNIFICANT ADVANTAGE FOR
HUMAN FACTORS IMPLEMENTATION/DEVELOPMENT

- ENHANCED CONTROL/DISPLAY CORRELATION
- ANNUNCIATOR PRIORITIZATION AND SEQUENCING
- ADAPTIVE MIMICS, TAILORED TO SITUATION
- INFORMATION PROCESSING AND DISPLAY FOR SPDS

A-535

SPDS DEVELOPMENT



A-536

INADEQUATE CORE COOLING INSTRUMENTATION

LP&L IS COMMITTED TO MEET II.F.2 FUNCTIONS

A. DESIGN HAS INCORPORATED A SATURATION METER

B. DESIGN WILL INCORPORATE CORE EXIT THERMOCOUPLE DISPLAY AND TRENDING

C. EVALUATING THE MOST EFFECTIVE METHOD OF INCORPORATING RCS SYSTEM INVENTORY MEASUREMENT WITH WATERFORD-3 INSTRUMENTATION DISPLAYS

- OPERATOR PROCEDURES

THE WATERFORD 3 EMERGENCY OPERATING PROCEDURES ARE BEING DEVELOPED IN ACCORDANCE WITH THE NUREG-0737 REQUIREMENTS.

THE C-E OWNERS GROUP HAS DEVELOPED CEN-152 AND CEN-156 WHICH BUILDS UPON EVENT ORIENTED PROCEDURES BY PROVIDING ADDITIONAL OPERATOR GUIDANCE. THIS FUNCTIONAL GUIDANCE WILL ALLOW THE OPERATOR TO ADDRESS CRITICAL SAFETY FUNCTIONS WHEN HE CAN NOT FOLLOW THE OPTIMAL RECOVERY PATH (EVENT RELATED).

THE PLANT STAFF OPERATIONS GROUP HAS WORKED CLOSELY WITH THE PROCEDURES & TEST REVIEW BRANCH IN THE DEVELOPMENT AND REVIEW OF EMERGENCY OPERATING PROCEDURES. THESE INCLUDE

- SUBMITTAL OF PROCEDURES FOR REVIEW AND COMMENT
- INCORPORATION OF COMMENTS INTO REVISIONS
- WALKSTHROUGHS AT PALO VERDE SIMULATOR AND WATERFORD 3 CONTROL ROOM

WMA

ACRS

CONCLUSION

THE SYSTEM APPROACH TO HUMAN FACTORS ANALYSIS AT WATERFORD 3 WILL ASSURE.....

- OBJECTIVE, COMPREHENSIVE REQUIREMENTS DETERMINATION

- PERFORMANCE CRITERIA TRACEABLE TO NRC CONCERNS AND PRIORITIZED EVENTS

- IMPLEMENTATION TAILORED TO WATERFORD 3 PLANT

- SUITABLE ADVANTAGES TAKEN OF EXTENSIVE PLANT COMPUTER SYSTEM

  - AVOIDANCE vs. MITIGATION
  - PHYSICAL ENHANCEMENT vs. DYNAMIC INFORMATION
  - MAN-PANEL vs. MAN-PROCESS INTERFACE
  - FLEXIBILITY FOR FUTURE IMPROVEMENTS

TOXIC CHEMICAL EVALUATION SUMMARY OF ANALYSIS AND RESULTS

- SITE CHARACTERISTICS
  - PERFORMED SURVEY OF STATIONARY, MOBILE PIPELINE SOURCES
  - IDENTIFIED OVER 100 SOURCES REQUIRING ANALYSIS
- CONTROL ROOM HABITABILITY ANALYSIS
  - PERFORMED DETAILED MODELING OF SOURCES
  - RESULTS ARE:
    - CHLORINE AND AMMONIA REPRESENT POTENTIALLY SEVERE HAZARD
    - CS_2 AND HCL REPRESENT LESSER HAZARDS
    - THE QUANTITY AND TYPES OF CHEMICALS COULD CHANGE IN THE FUTURE
- PROTECTIVE FEATURES – DEFENSE IN DEPTH
  - CONSERVATIVE ANALYSIS
  - REDUNDANT INSTRUMENTATION
    - CHLORINE DETECTORS
    - AMMONIA DETECTORS
    - BROAD RANGE DETECTORS
  - INDUSTRY HOTLINE
  - ODOR DETECTION TRAINING PROGRAM
  - PERIODIC SURVEY UPDATE

A-540

PROTECTIVE FEATURES

- REDUNDANT CL_2 + NH_3 DETECTORS

CHEMICAL	SET POINT (PPM)	EFFECTIVE ISOLATION TIME (SEC)	IDLH (PPM)	CONCENTRATION (PPM) 2 MINUTE	PEAK
CL_2	1	5	25	32	36
NH_3	5	0	500	120	380

A-541

- RESPIRATORY PROTECTION
- BROAD RANGE DETECTORS FOR ORGANIC COMPOUNDS
- INDUSTRY HOTLINE
  - INDUSTRY ⟶ EOC ⟶ WATERFORD
  - WEEKLY COMMUNICATIONS TESTS
- ODOR DETECTION TRAINING PROGRAM
- PERIODIC SURVEY UPDATE

PROTECTION OF NON–ESSENTIAL PERSONNEL

EMERGENCY SCENARIO

- NOTIFICATION OF CHEMICAL RELEASES
  - HOTLINE
  - DETECTORS
  - ODOR
- ASSESS CONDITIONS
  - PARISH EOC RECOMMENDATIONS (HOTLINE)
  - METEOROLOGICAL DATA
  - TOXIC CHEMICAL MONITORS
- INITIATE PROTECTIVE ACTIONS

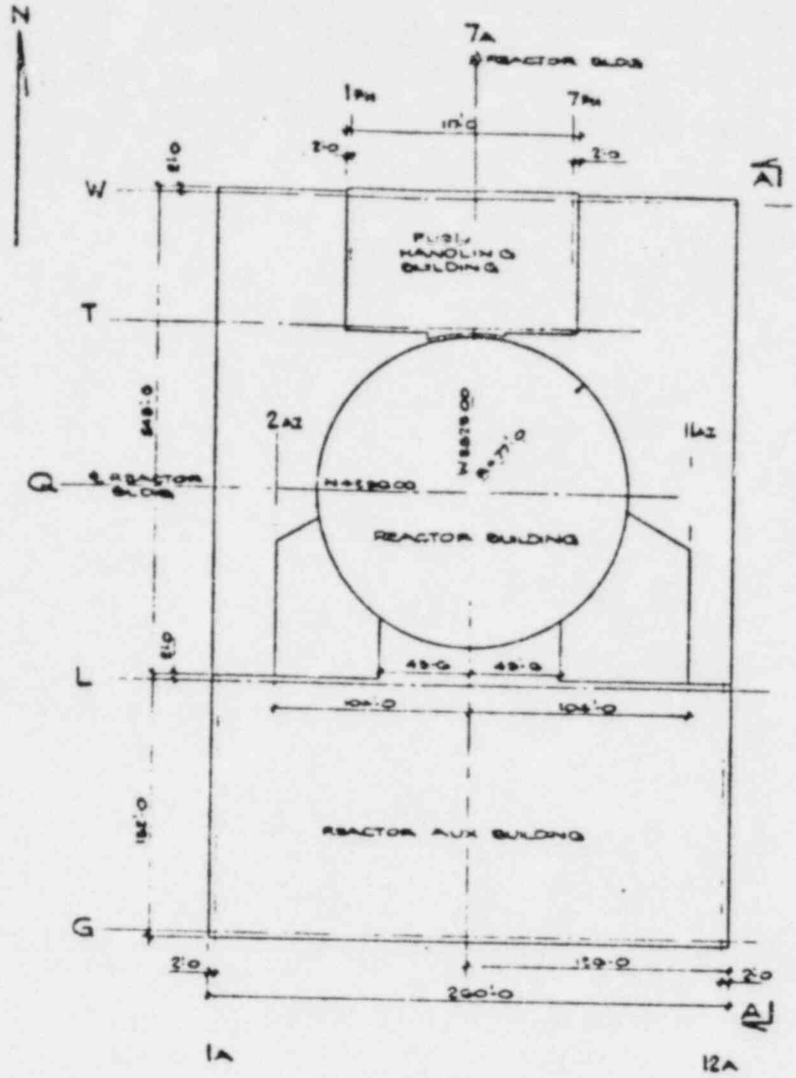
PROTECTIVE ACTION OPTIONS:

- EVACUATE SITE
  - EOF (8 MILES SE)
  - ST JOHN THE BAPTIST CHURCH (6 MILES NE)
  - OTHER PER EOC RECOMMENDATIONS
- REMAIN ONSITE
  - TAKE SHELTER

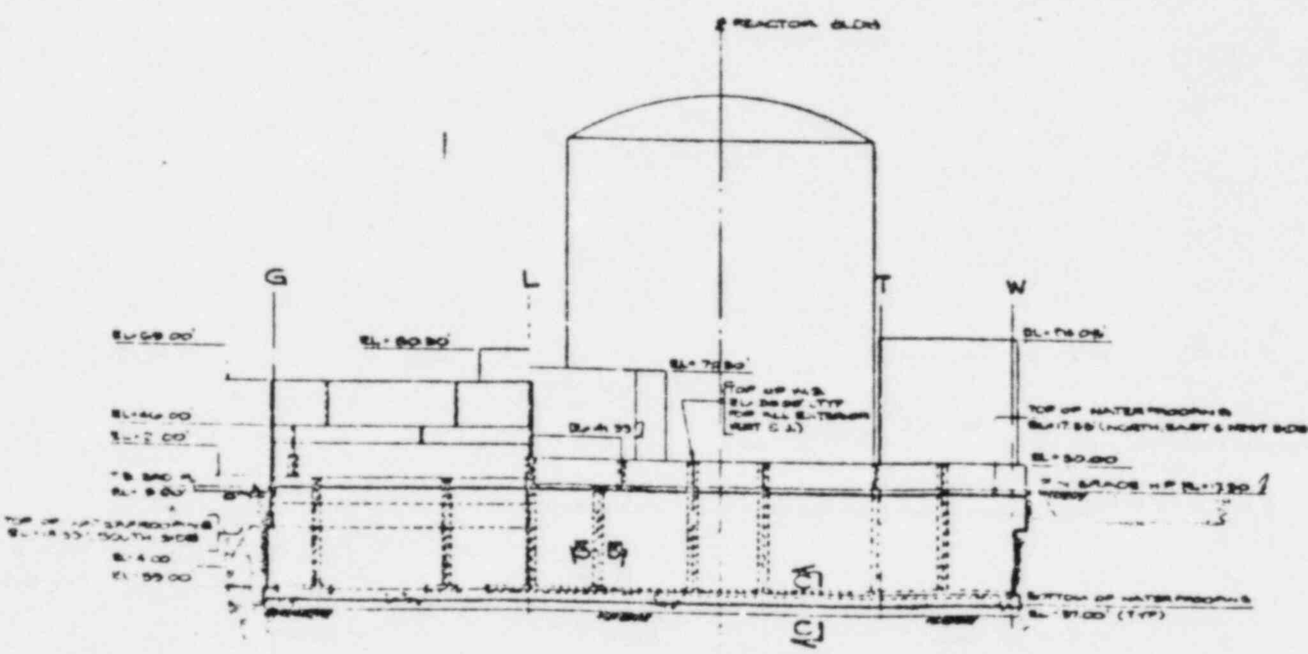
DIESEL GENERATOR OPERATION

- LOSS OF OFF-SITE POWER
- TOXIC CHEMICALS AIRBORNE AT DG AIR INTAKE AT CONCENTRATIONS GREATER THAN 20 PERCENT
  - NOT CONSIDERED A CREDIBLE EVENT

A-543

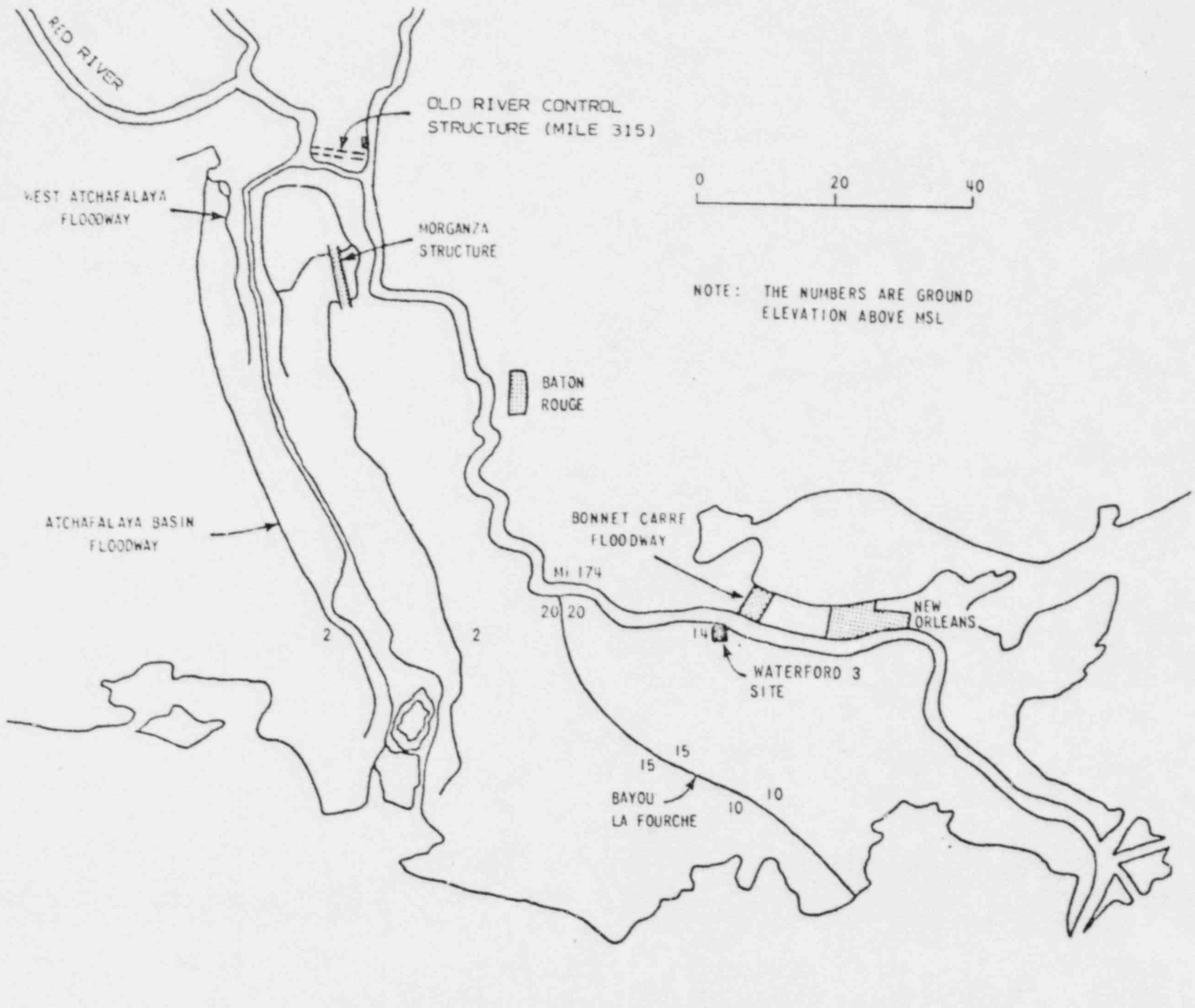


PLAN

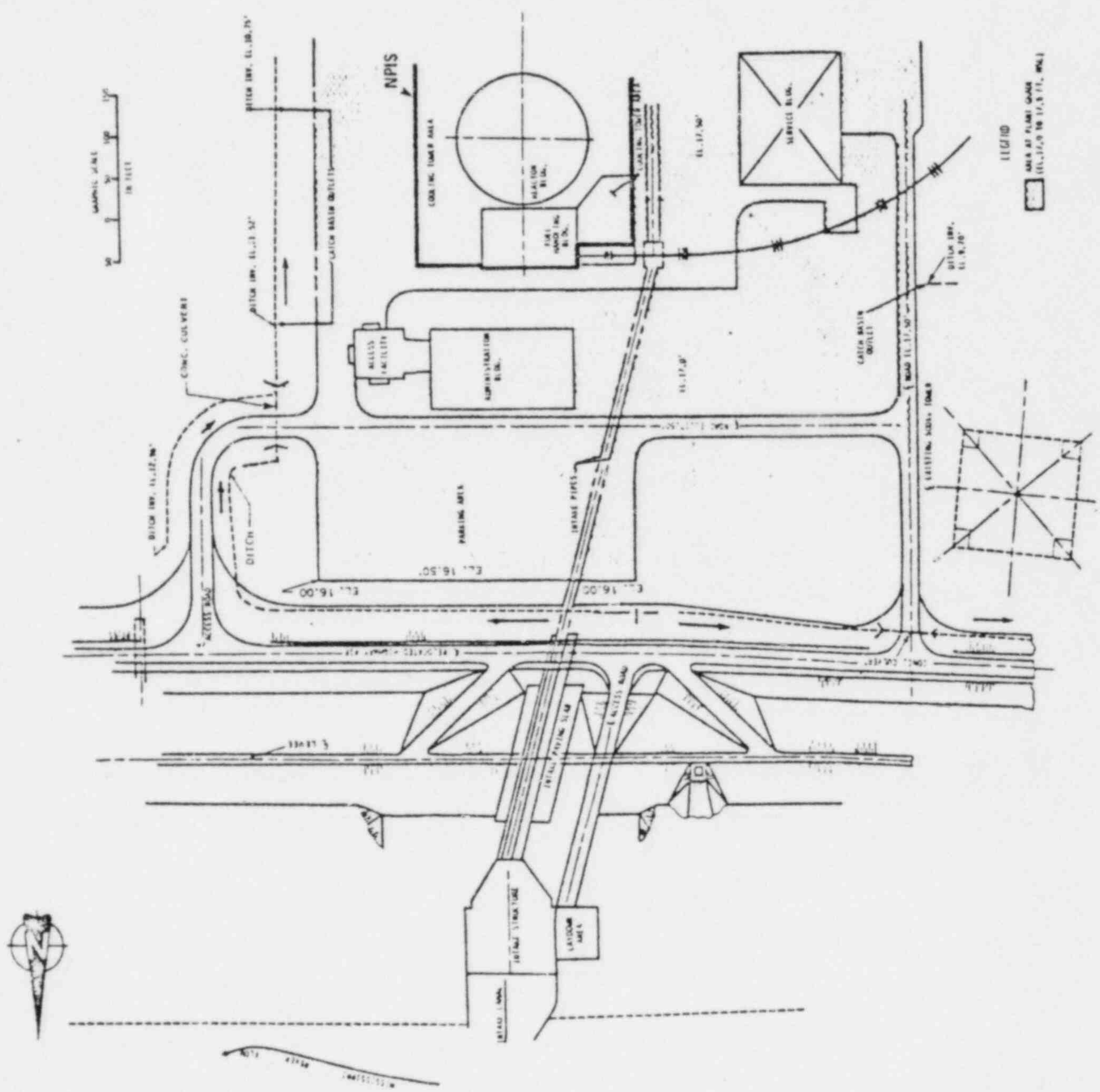


ELEVATION AA

A-544


RED RIVER
OLD RIVER CONTROL STRUCTURE (MILE 315)
WEST ATCHAFALAYA FLOODWAY
MORGANZA STRUCTURE
0
20
40
NOTE: THE NUMBERS ARE GROUND ELEVATION ABOVE MSL
BATON ROUGE
ATCHAFALAYA BASIN FLOODWAY
BONNET CARRE FLOODWAY
Mi 174
20
20
2
2
14
NEW ORLEANS
WATERFORD 3 SITE
15
15
10
10
BAYOU LA FOURCHE

A-545


NPIS
REACTOR BLDG.
SERVICE BLDG.
ADMINISTRATION BLDG.
ACCESS FACILITY
PARKING AREA
INTAKE PIPES
INTAKE STRUCTURE
DITCH
CONC. CULVERT
CATCH BASIN OUTLETS
EL. 16.00
EL. 16.50
LEGEND

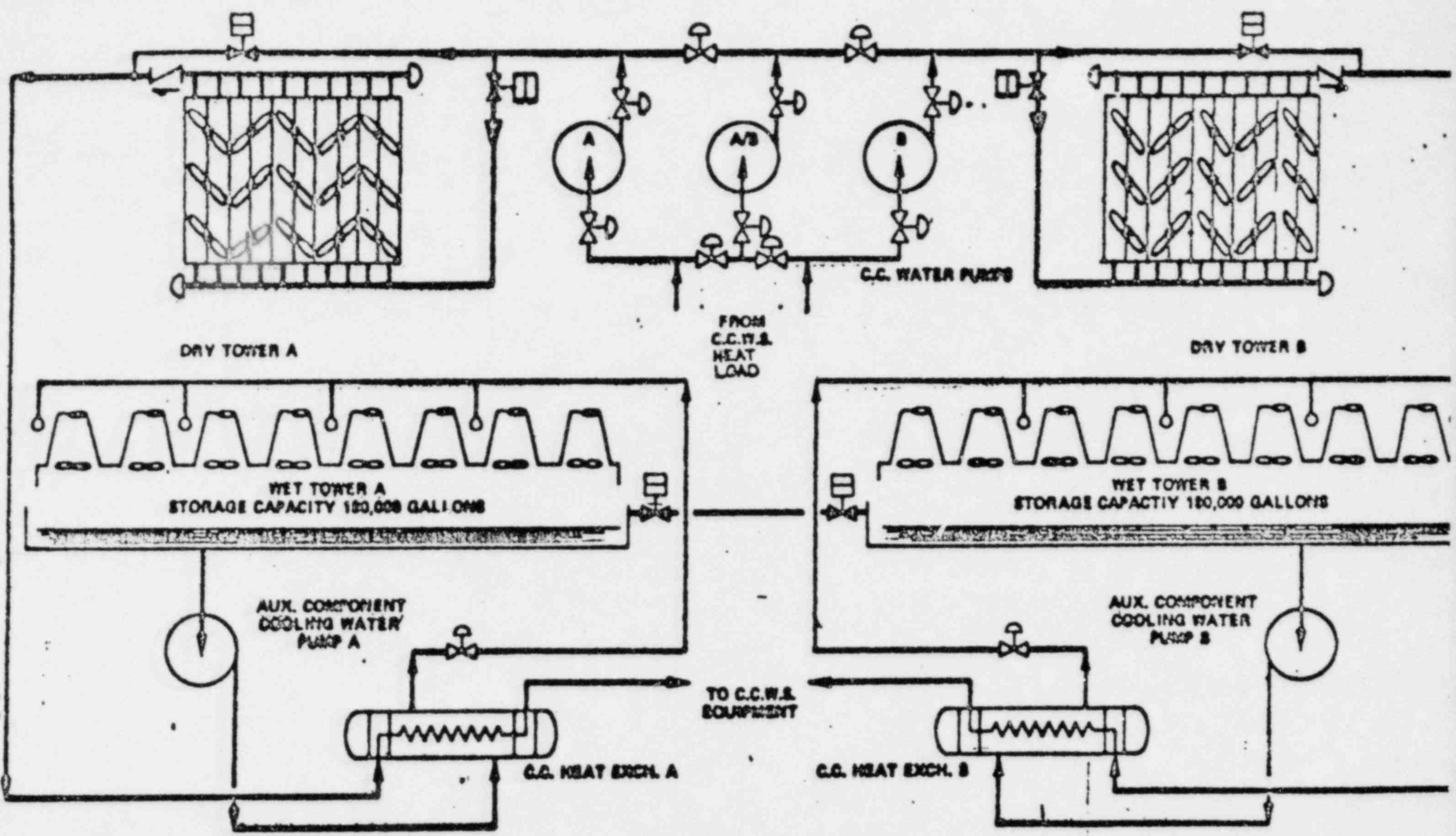
A-546

WATERFORD SES NO. 3
ULTIMATE HEAT SINK (UHS)

PRESENTATION INCLUDES:

1. SYSTEM DESCRIPTION
2. PHYSICAL ARRANGEMENT
3. LIMITING DESIGN CONDITIONS

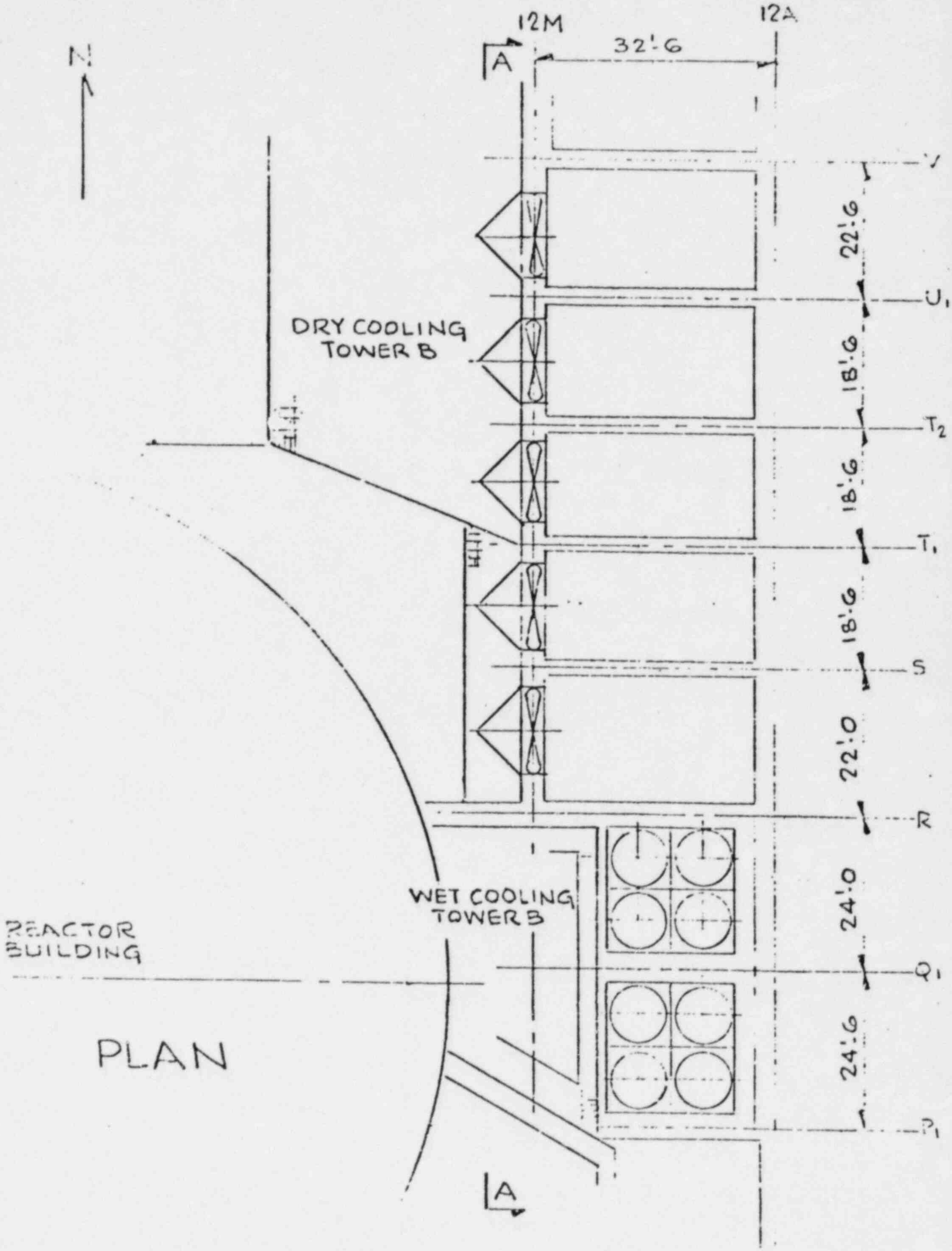
A-547



A-548

LOUISIANA POWER & LIGHT
Waterford Steam Electric Stat

SIMPLIFIED FLOW DIAGRAM (
WET & DRY COOLING TOWER


N
12M
12A
32'-6
A
22'-6
18'-6
18'-6
18'-6
22'-0
24'-0
24'-6
V
U1
T2
T1
S
R
Q1
P1
DRY COOLING TOWER B
WET COOLING TOWER B
REACTOR BUILDING
PLAN
A

A-549

WATERFORD UNIT 3
APPLICATION OF PRA TO POST-TMI CONCERNS

- EMERGENCY FEEDWATER SYSTEM RELIABILITY ANALYSIS (NUREG 0737 II.E.1)
- ELECTRIC POWER SYSTEM RELIABILITY ANALYSIS
- COMPUTER RELIABILITY ANALYSIS (NUREG 0737 I.D.2)
- CONTROL ROOM SYSTEMS REVIEW (NUREG 0737 I.D.1)
- EMERGENCY PROCEDURES DEVELOPMENT (NUREG 0737 I.C.1, I.C.7)

A-550

EMERGENCY FEEDWATER SYSTEM RELIABILITY ANALYSIS

SYSTEM UNAVAILABILITIES – ORIGINAL STUDY

• LOSS OF FEEDWATER	3.7 E-4
• LOSS OF OFFSITE POWER	5.1 E-4
• STATION BLACKOUT	2.2 E-2

SYSTEM UPGRADING

- QUALIFY MOTOR DRIVEN PUMPS AS 100% CAPACITY
- COMPUTER SURVEILLANCE OF DISCHARGE VALVE POSITION INDICATION TO REDUCE PROBABILITY OF MAINTENANCE CMF

SYSTEM UNAVAILABILITIES – UPGRADED STUDY

• LOSS OF FEEDWATER	1.4 E-5
• LOSS OF OFFSITE POWER	3.9 E-5
• STATION BLACKOUT	2.6 E-2

FURTHER EVALUATION

- RELIABILITY ANALYSIS OF EFW PUMP DISCHARGE THROTTLE VALVE CONTROLS TO ENSURE NO UNDUE DEGRADATION OF SYSTEM RELIABILITY FOR FREQUENT DEMANDS

A-5-51

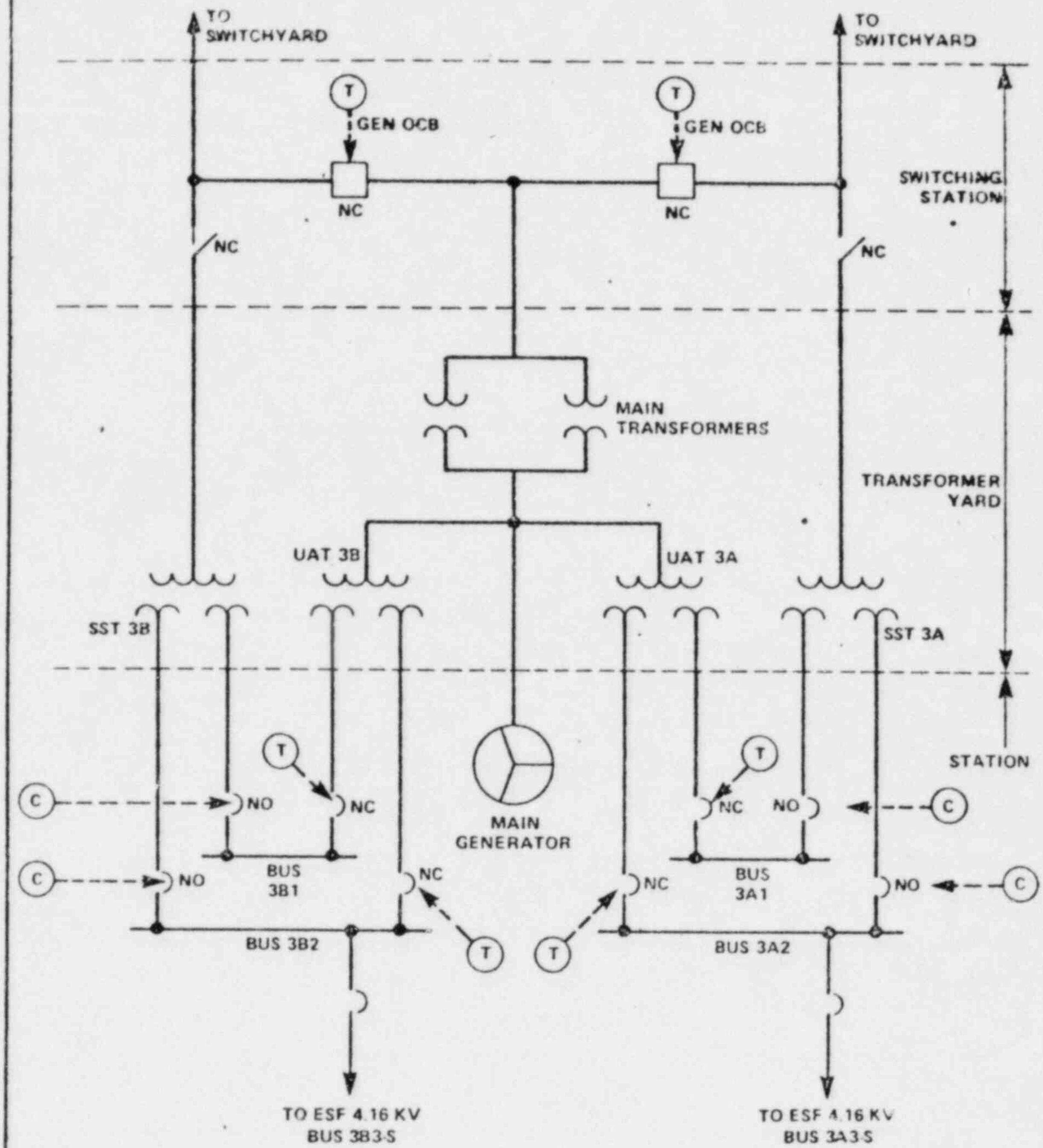
EJS

ACRS

AC/DC POWER RELIABILITY

- STATION BLACKOUT
- STABILITY ANALYSIS

A-552

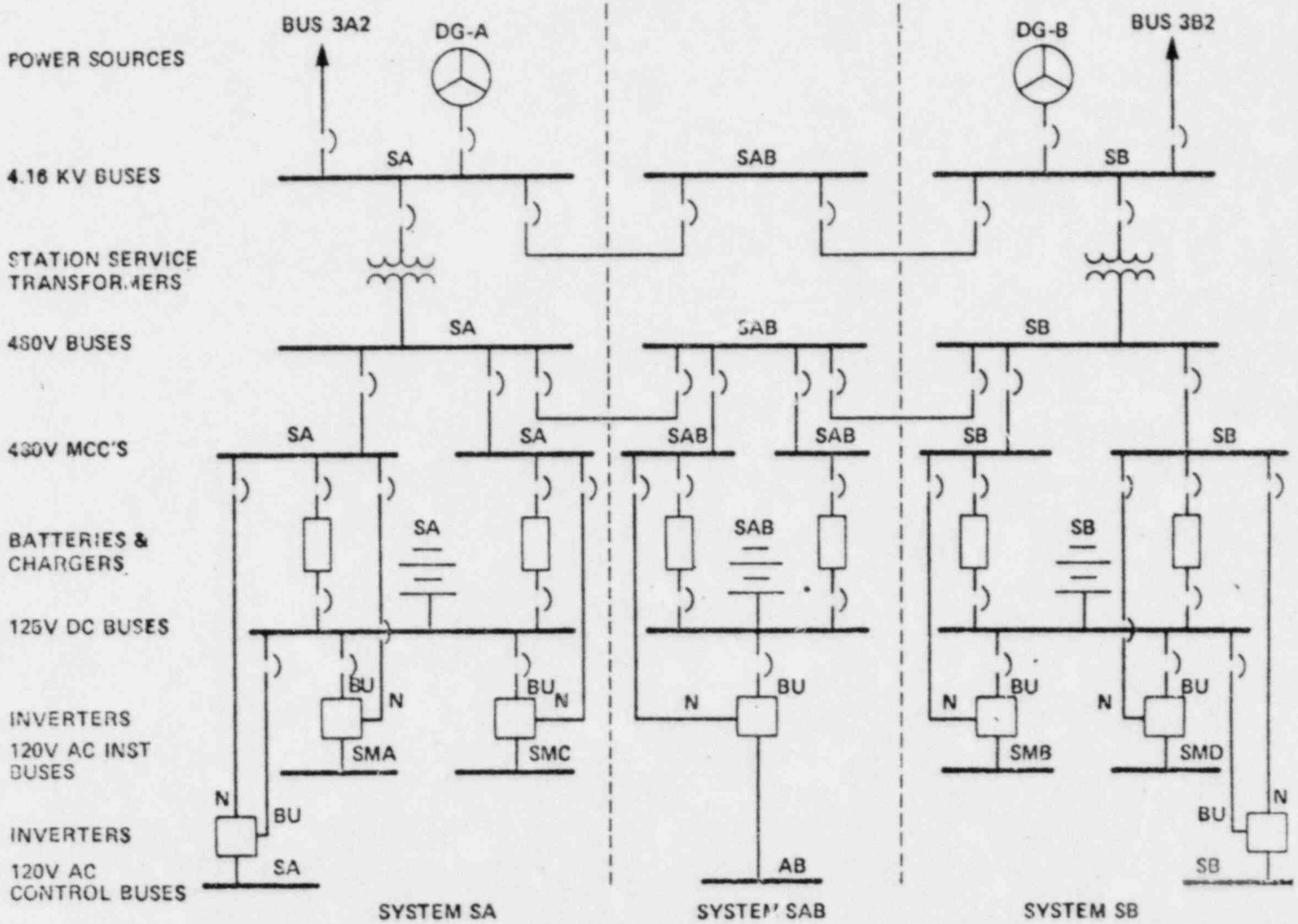


AMENDMENT NO. 11 (7/80)

LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	PREFERRED POWER SOURCE	Figure 8.2-3

A-55-3

WATERFORD UNIT 3
ONSITE AC-DC POWER SYSTEM

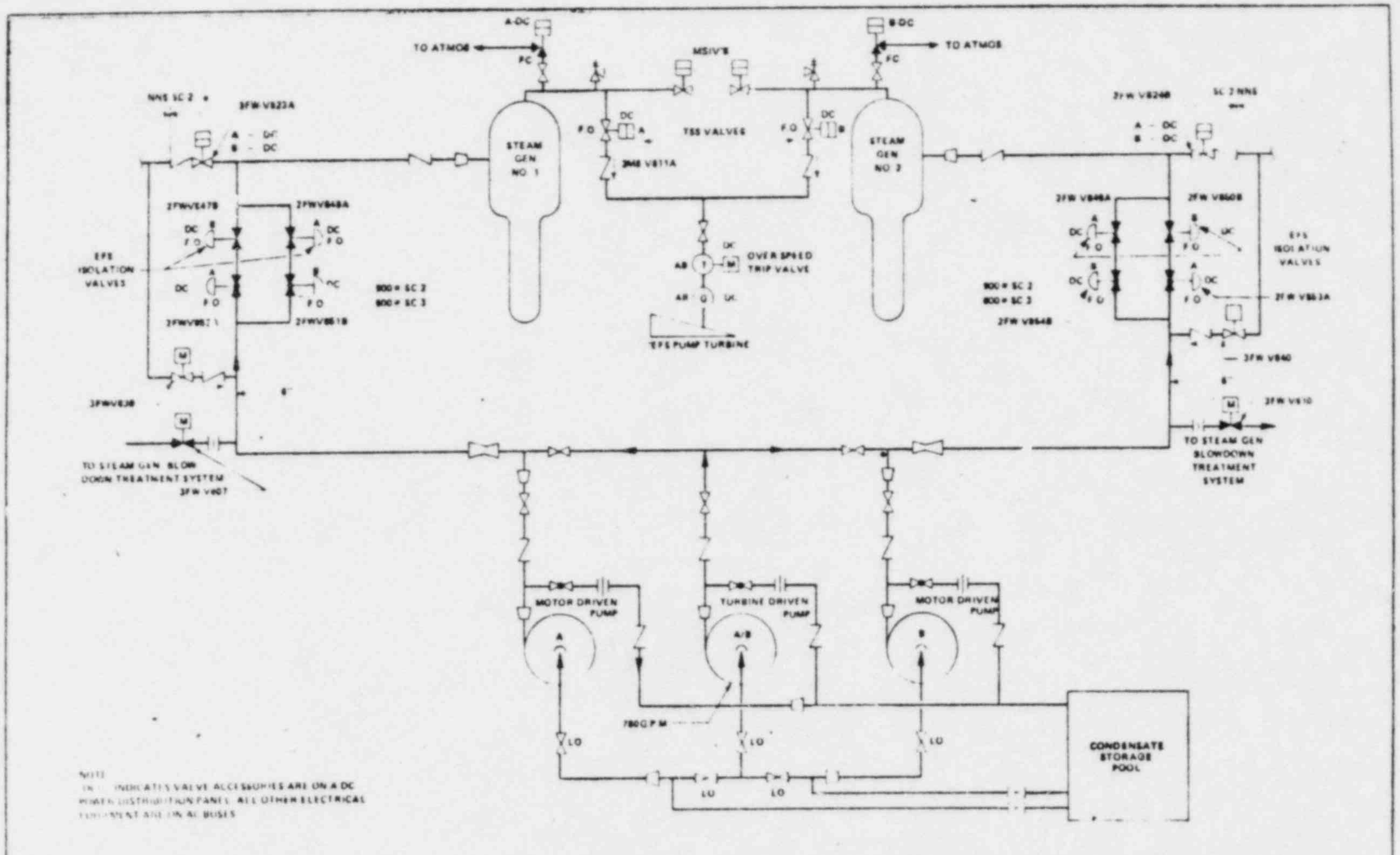


N - NORMAL POWER SOURCE
BU- BACKUP POWER SOURCE

RELIABILITY ENHANCEMENT FEATURES OF ONSITE EMERGENCY AC-DC POWER SYSTEM

- PHYSICAL AND ELECTRICAL INDEPENDENCE OF REDUNDANT SUBSYSTEMS
- CONSERVATIVE SIZING CRITERIA FOR COMPONENTS
- MULTIPLE POWER SOURCES
- AUTOMATIC LOADING AND SEQUENCING
- INTERLOCKS AND ALARMS ON AC "AB" BUS INTERTIES
- NO AUTO OR MANUAL DC BUS INTERTIES
- MALFUNCTION SURVEILLANCE ALARMS (UNDERVOLTAGE, GROUND DETECTION, INOPERABLE STATUS, ETC)
- DG SET RELIABILITY
  - 300 START RELIABILITY TESTS
  - REDUNDANT AIR STARTING SYSTEM WITH 10 START CAPACITY
  - PRE-OPERATIONAL AND PERIODIC TESTING AND MAINTENANCE
- SEPARATE BATTERY AND BUS FOR "AB" BUS CONTROLS AND MAJOR NON-SAFETY DC LOADS
- REDUNDANT CHARGERS FOR EACH BATTERY BUS
- TESTING AND MAINTENANCE PROCEDURES

A-5-5

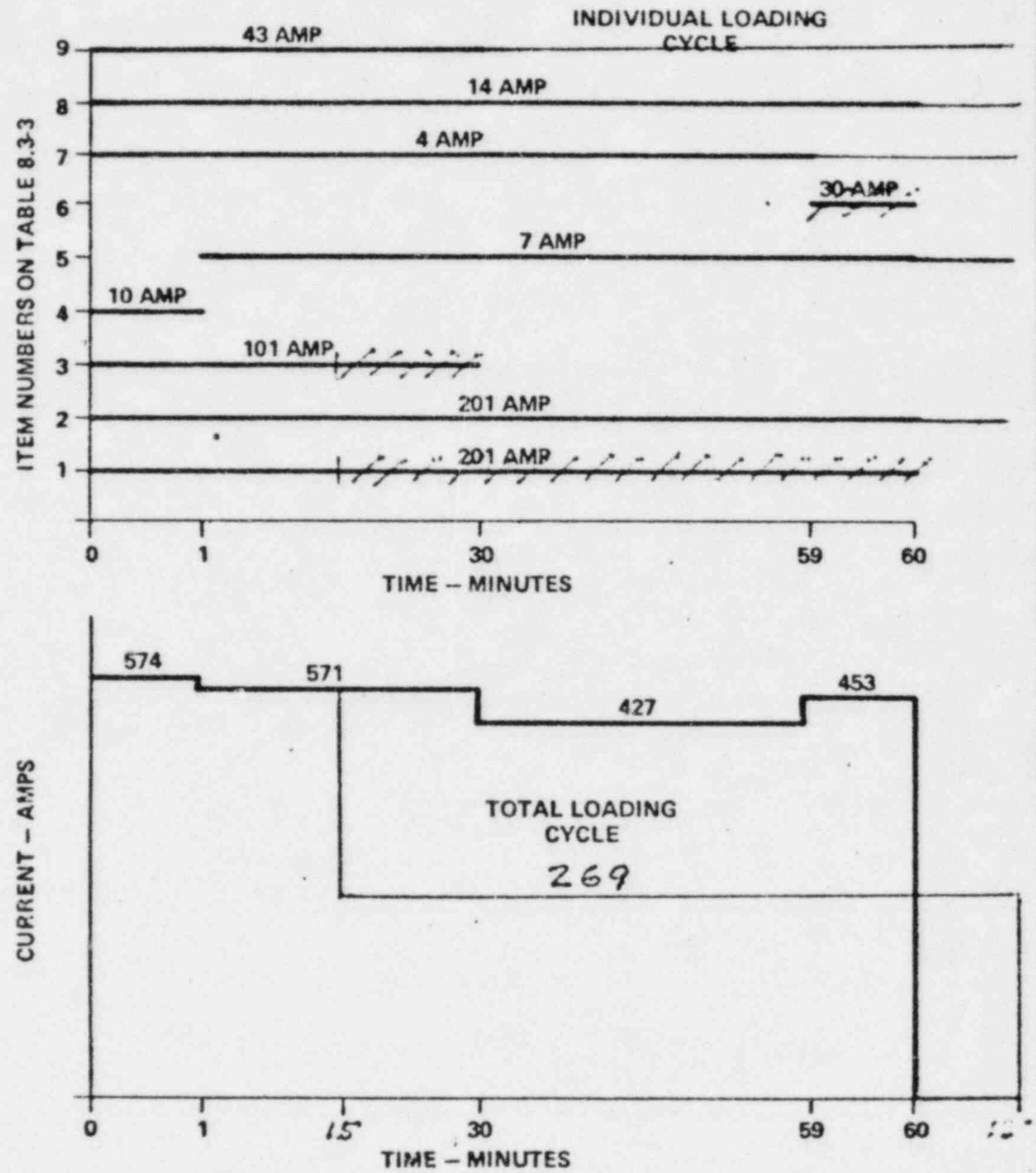


AMENDMENT NO. 18 (6/81)

LOUISIANA POWER & LIGHT CO.
Waterford Steam Electric Station

EMERGENCY FEEDWATER SYSTEM
FIGURE 10.4-6

A·556



AMENDMENT NO. 11 (7/80)

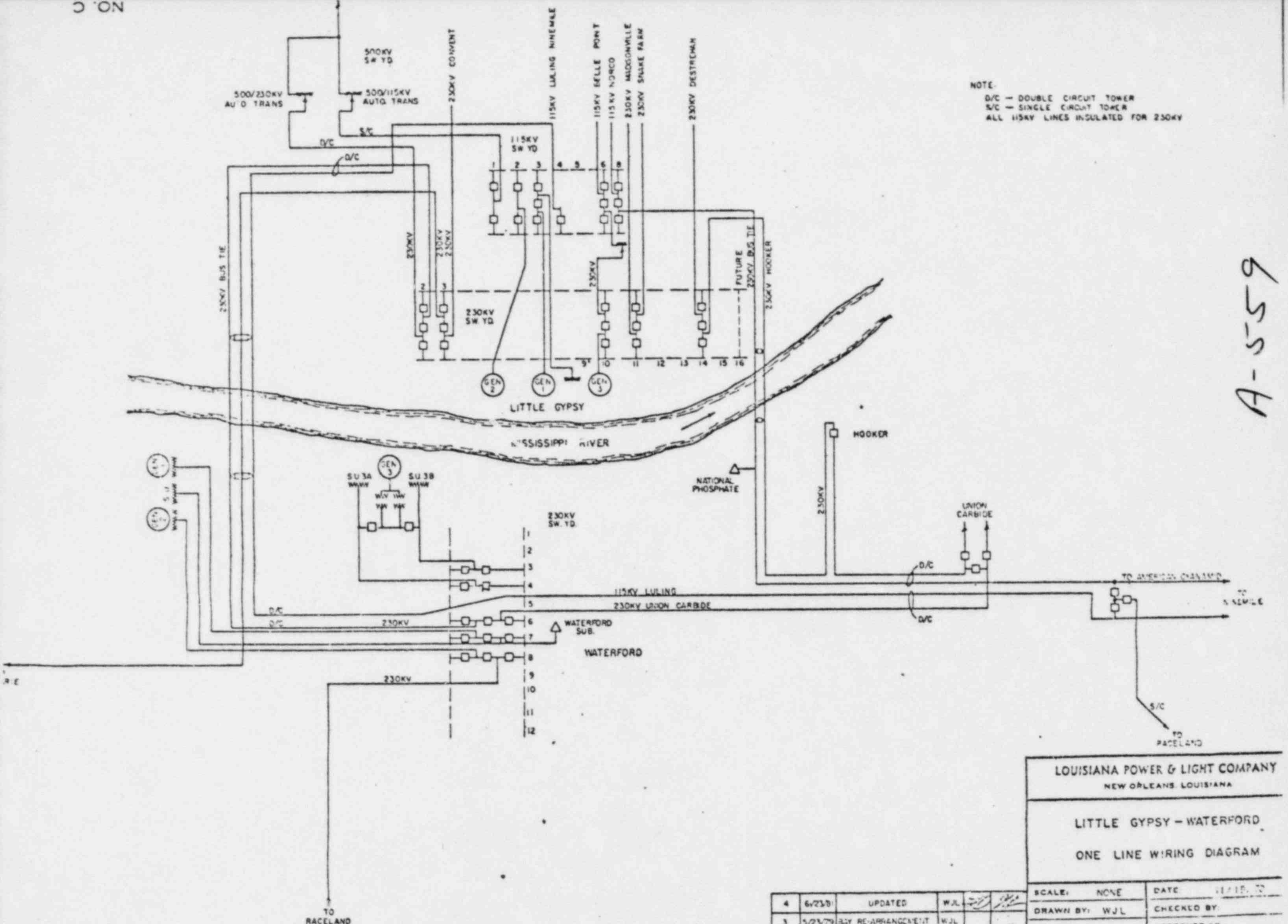
LOUISIANA POWER & LIGHT CO. Waterford Steam Electric Station	BATTERY 3A-S LOADING CYCLE	Figure 8.3-2

CAPABILITY TO WITHSTAND STATION AC BLACKOUT

• EFW CONDENSATE SUPPLY	>24 HRS
• DC POWER SUPPLY FOR EFW CONTROLS AND MONITORING (WITH SELECTIVE LOAD SHEDDING)	> 2 HRS

7-5-58

A-559



LOUISIANA POWER & LIGHT COMPANY
NEW ORLEANS, LOUISIANA

LITTLE GYPSY – WATERFORD

ONE LINE WIRING DIAGRAM

4	6/23/81	UPDATED	WJL		
3	5/23/79	BAY RE-ARRANGEMENT	WJL		
2	6/1/77	REDRAWN & UPDATED	WJL		
NO.	DATE	REVISION	BY	CH.	APP.

SCALE: NONE	DATE: 11/15/72
DRAWN BY: WJL	CHECKED BY:
TRACED BY:	APPROVED BY:

NO. C – 1 – 4

NO. C

Waterford 3
Stability Analysis

Initiating Event	Clearing Time	Line Out	Result
Loss of Ninemile Plant	---	---	Stable
3-Ø Fault on Switchyard Bus	6 cycles	---	Stable
3-Ø Fault on Switchyard Bus	6 cycles	Waterford-Gypsy 230	Stable
3-Ø Fault on Willow Glen	6 cycles	Gypsy-Willow Glen 500	Marginal

A-560

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

August 11, 1981

APPENDIX XL
ACRS REPORT ON ENRICO FERMI ATOMIC
POWER PLANT UNIT 2

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON ENRICO FERMI ATOMIC POWER PLANT UNIT NO. 2

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the ACRS completed its review of the application of the Detroit Edison Company (Applicant) for a license to operate the Enrico Fermi Atomic Power Plant Unit No. 2 (Fermi-2). A Subcommittee meeting was held in Washington, DC, on July 24, 1981 to consider this project. A tour of the facility was made on July 15, 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 reported on the construction permit application for this unit in its report dated March 9, 1971.

The Enrico Fermi plant is located in Frenchtown Township, Monroe County, Michigan. The nearest population center is the city of Monroe, Michigan about 5.5 miles west-southwest of the site.

Fermi-2 is equipped with a General Electric BWR-4 nuclear steam supply system with a rated power level of 3292 MWt and has a Mark I pressure suppression containment with a design pressure of 62 psig. The Applicant has performed a detailed evaluation of the containment's ability to withstand LOCA and relief valve hydrodynamic loads as required by the NRC for the Mark I Containment Program. As a result of this evaluation, extensive modifications were required and are underway. However, since the evaluation was performed prior to the issuance of the NRC report delineating the Staff's acceptance criteria (NUREG-0661 - Safety Evaluation Report, Mark I Containment Long-Term Program - Resolution of Generic Technical Activity A-7), the design has not yet been shown to be completely in conformance with this report. The Applicant has made a commitment to perform a plant unique analysis on the basis of the NUREG-0661 criteria and other requirements established by the Long-Term Program, including in-plant confirmatory tests to assess loads resulting from safety relief valve operation. The Applicant will submit this analysis to the Staff for audit review upon its completion. Subject to the results of this analysis, the NRC finds the Applicant's evaluation generally acceptable. This matter should be resolved in a manner satisfactory to the NRC Staff prior to full power operation. We wish to be kept informed.

We note that Detroit Edison has acted as its own architect-engineer for this project. The Applicant stated that this arrangement will result in a valuable carry-over of knowledge as people transfer from construction to plant operation activities. The NRC Staff has reviewed the Applicant's organization and management structure and has expressed some concern about the personnel transition. The Staff recommends that care be taken to assure that quality of construction and safety of operations are not compromised during the transition. We concur in this recommendation. To address a concern over a lack of commercial nuclear power plant operating experience, the NRC Staff is requiring that the control room staff be augmented with vendor personnel during startup. We recommend that the NRC assure that these personnel remain on site for a period of time which permits the necessary operating experience to be obtained by the Applicant's Staff.

The Applicant described the program and the philosophy for training of personnel. Training has a high priority and a training simulator has been ordered to aid in this effort. The simulator will be used for operator training and will also be used to train other plant personnel including managers and supervisors. It will also be used to test ATWS operating procedures. The NRC has reviewed the Applicant's ATWS procedures and finds them generally acceptable. The NRC should assure that the ATWS procedures and the associated simulator training are well coordinated.

The Applicant discussed provisions to address station blackout. In the event of a loss of all offsite AC power and loss of all onsite emergency diesel generators, the Applicant can call on a self-starting turbine-generator located onsite. While we recognize that this additional power source further lowers the probability of a station blackout, we recommend that the NRC Staff assure that procedures exist to address a station blackout event and that operating personnel are adequately trained in the use of these procedures. We wish to be kept informed.

Construction of this unit has taken a longer than usual time owing to financial difficulties and the impact of the TMI-2 accident. As a result, the Applicant has been required to perform a seismic reassessment of the structures, systems, and components required for safe shutdown based on currently accepted NRC design response spectra. This reassessment is still under way. Preliminary results indicate that there is sufficient margin in the original design to meet the NRC requirements and that only minor equipment changes will be required. This matter should be resolved to the satisfaction of the NRC Staff.

The NRC has begun review of the Applicant's emergency planning. Because of the plant's location, interaction with Canadian authorities is necessary. Responsibility for this interaction rests with the offices of the Federal Emergency Management Agency.

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The NRC Staff proposes to require the installation of core thermocouples in Fermi-2 as specified by Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The Applicant has not yet agreed to this requirement. The ACRS supported use of core thermocouples in BWRs in its letter of November 10, 1980 to the NRC Executive Director for Operations, but called attention to the need for further study to determine the appropriate vertical location of such thermocouples. Since most of the information of interest from thermocouples may be obtainable from a small number of thermocouples placed in a more accessible location, we recommend that this requirement be reevaluated.

The Applicant's security plan was discussed. We note with approval that security guards will be Detroit Edison employees.

As part of the NRC Staff review of plant fire protection provisions, the Applicant simulated a control room fire to demonstrate that a fire external to the control panels will not result in a loss of redundant shutdown functions. The NRC Staff has identified what it believes to be deficiencies in the test and the Applicant has responded in a recent submittal. We believe this item should be resolved in a manner satisfactory to the NRC Staff.

Other issues have been identified as Outstanding Issues in the NRC Staff's Safety Evaluation Report dated July 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 Enrico Fermi Atomic Power Plant Unit No. 2 can be operated at power levels up to 3292 MWt without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Detroit Edison Company, "Enrico Fermi Atomic Power Plant Unit 2 Final Safety Analysis Report," Volumes 1 - 11 and Amendments 1-37.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant Unit No. 2," USNRC Report, NUREG-0798, dated July 1981.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report, Mark I Containment Long-Term Program - Resolution of Generic Technical Activity A-7," USNRC Report, NUREG-0661, dated July 1980.

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

August 11, 1981

APPENDIX XLI
ACRS REPORT ON SUSQUEHANNA STEAM
ELIECTRIC STATION UNITS 1 AND 2

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the Advisory Committee on Reactor Safeguards completed its review of the application of the Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (Applicant) for a license to operate the Susquehanna Steam Electric Station Units 1 and 2. The units will be operated by the Pennsylvania Power and Light Company. A Subcommittee meeting was held in Washington, D.C. on July 23, 1981 to consider this project. A tour of the facility was made on July 2, 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 April 13, 1972.

The Susquehanna station is located in Luzerne County, Pennsylvania about 12 miles northwest of Hazleton and 15 miles southwest of Wilkes-Barre, the nearest cities having populations in excess of 25,000.

Each Susquehanna unit is equipped with a General Electric BWR-4 nuclear steam supply system with a rated power level of 3293 MWt and has a Mark II pressure suppression containment with a design pressure of 53 psig.

In connection with our review of the Susquehanna station, the NRC Staff discussed its generic resolution of the safety issues associated with the Mark II containment design and performance. This resolution is given in the Staff report NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria." This matter has received detailed review by the ACRS Subcommittee on Fluid Dynamics. We believe that the load definitions given in this report are conservative and acceptable. These load definitions are to be applied to BWR Mark II's on a case-by-case basis. We believe that the Susquehanna containment structures will meet these requirements.

The Applicant described the management organization and the technical personnel available for operation of the Susquehanna plant. Although this is the first nuclear power plant to be operated by this Applicant, both management and plant staff are made up of personnel with considerable background and expertise in commercial nuclear power plant operation. We commend the Applicant's efforts to obtain knowledgeable and experienced personnel.

The Applicant described the program and the philosophy for training of personnel. Training has a high priority as it had even prior to the TMI-2 accident. For example, a training simulator was ordered by the Applicant considerably before the accident at TMI-2 and is currently in use. The training program includes consideration of ATWS. The Applicant's training program appears sound and thorough.

The NRC Staff proposes to require the installation of core thermocouples in the Susquehanna station as specified by Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The Applicant has not yet agreed to this requirement. We supported use of core thermocouples in BWRs in our letter of November 10, 1980 to the NRC Executive Director for Operations but called attention to the need for further study to determine the appropriate vertical location of such thermocouples. Since most of the information of interest from thermocouples may be obtainable from a small number of thermocouples placed in a more accessible location, we recommend that this requirement be reevaluated.

The NRC Staff proposes to require a second meteorological tower at the Susquehanna site for the purpose of collecting additional data for use during an emergency. This issue is still being discussed with the NRC Staff. Additionally, there are several other issues concerning emergency planning which are identified by the NRC Staff in its Safety Evaluation Report and Supplement No. 1 as Outstanding Issues. We believe that these issues should be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

Another Outstanding Issue involves IE Bulletin 79-27, "Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation." The Applicant has stated that this IE Bulletin will be complied with prior to issuance of an operating license. We recommend that this issue be resolved in a manner satisfactory to the NRC Staff.

The Applicant is currently reviewing the issue of station blackout. Analytical work, development of operating procedures, and actual testing of equipment response to simulated blackout conditions are planned by the Applicant. We believe that the Applicant's proposed program is a satisfactory response to this issue.

The NRC Staff has identified other Outstanding Issues in its Safety Evaluation Report dated April 1981 and in Supplement No. 1 to that report dated June 1981 such as turbine missiles, review of the alternate shutdown system, and modification of depressurization logic. We believe the Outstanding Issues can be resolved, and recommend that this be done in a manner satisfactory to the NRC Staff before operation at full power.

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 Susquehanna Steam Electric Station Units 1 and 2 can be operated at power levels up to 32[illegible] MWt each without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:
1. Pennsylvania Power and Light Company, "Final Safety Analysis Report, Susquehanna Steam Electric Station, Units 1 and 2," with Amendments 1 through 35.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2, Docket Nos. 50-387 and 50-388," USNRC Report NUREG-0776, dated April 1981 and Supplement No. 1, dated June 1981.
3. U.S. Nuclear Regulatory Commission IE Bulletin No. 79-27, "Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation," dated November 30, 1979.

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

August 11, 1981

APPENDIX XLII
ACRS INTERIM REPORT ON WATERFORD
STEAM ELECTRIC STATION UNIT 3

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: INTERIM REPORT ON THE WATERFORD STEAM ELECTRIC STATION UNIT 3

Dear Dr. Palladino:

During its 256th meeting, August 6-8, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of Louisiana Power & Light Company (Applicant) for a license to operate the Waterford Steam Electric Station Unit 3 (Waterford-3). This project has been considered at Subcommittee meetings on June 18-19, 1981 in St. Charles Parish, Louisiana, and on August 5, 1981 in Washington, D.C. A tour of the facility was made by Subcommittee members on June 18, 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 unit in its report dated January 17, 1973.

Waterford-3 is located on the bank of the Mississippi River near Taft, Louisiana in St. Charles Parish. The city of New Orleans is approximately 25 miles east-southeast from the plant site and Baton Rouge is approximately 50 miles north-northwest. The largest town within 10 miles of the site is Reserve, Louisiana, which had a population of approximately 7000 in 1977.

Waterford-3 uses a Combustion Engineering nuclear steam supply system with a rated power level of 3410 MWt. The architect-engineer is Ebasco Services, Inc. The containment is a free standing steel pressure vessel enclosed within a reinforced concrete shield building. The containment building, auxiliary building, fuel handling building, and ultimate heat sink are located on a common base mat, forming a self-contained nuclear island.

Louisiana Power & Light (LP&L) is a part of Middle South Utilities (MSU). Although Waterford-3 is the first nuclear plant to be operated by the Applicant, the MSU system has two operating nuclear plants, Arkansas Nuclear One Units 1 and 2, which are being operated by Arkansas Power and Light Company. Two additional plants in the MSU system, Grand Gulf Nuclear Station Units 1 and 2, are under construction by Mississippi Power and Light. MSU provides some technical services to support the nuclear units in its system.

The Applicant described the management, the operating organization, and the status of staffing. The NRC Staff has not completed its review of these matters, but reported its conclusion that the management and staffing at Waterford-3 is less well established than at other nuclear plants at a similar time during their construction and startup schedule. The LP&L management has not yet been successful in putting together the team of experienced and qualified personnel which we believe will be necessary to successfully operate the plant. Of particular concern is the lack of nuclear experience throughout the organization and the apparent lack of appreciation by high-level management of the magnitude of the project it is undertaking. We believe that an extraordinary effort will be required to prepare the LP&L management and staff for operation of the Waterford-3 plant. We also believe that a more concerted effort is needed to build an integrated organization of LP&L and contractor personnel for startup and operation of Waterford-3. We recommend that the adequacy of management and staffing be established prior to fuel loading. We will continue to review this matter with the Applicant and the NRC Staff.

The Applicant described the three safety review committees which will be a permanent part of the Waterford-3 organization. We believe that better use could be made of experts from sources other than the Applicant's organization and its contractors to provide professional experience in areas such as training, human factors engineering, and reactor safety. We recommend that the Applicant make a greater effort to include recognized experts, especially on its Safety Review Committee.

Although a sincere effort has been made to establish a comprehensive training program at Waterford-3, it has suffered from a lack of professional direction. We believe the Applicant should move as soon as possible to employ a highly qualified professional for the key position of training director and provide him with the resources needed to build an effective program.

Waterford-3 is located in a highly industrialized area with an unusually large concentration of sources of hazardous substances from nearby industries and transportation routes. We believe the Applicant has done a commendable job in analyzing these hazards and providing for protection of the plant by both equipment design and administrative procedures. The NRC Staff has not completed its review of this matter, but we believe it can be resolved satisfactorily.

The Waterford-3 control room makes extensive use of a computer system for monitoring and control of the plant, and for evaluating plant performance. We commend the initiative the Applicant has shown in this area and the continuing effort to integrate the control room equipment with operating procedures and human factors considerations.

Waterford-3 has a unique ultimate heat sink design. It is contained within the nuclear island and is protected from extreme environmental effects. It consists of two trains of wet and dry cooling towers. Sufficient water is stored on the nuclear island to meet the needs for shutdown decay heat removal. We believe the design is acceptable.

The Applicant has performed an analysis of total loss of AC power. The DC power supply is capable of supplying essential loads for at least two hours and the condensate supply is sufficient for a longer period. We recommend that the Applicant expand this analysis to consider the effect of loss of space cooling on essential electrical equipment and to also consider the effect of coolant leakage from the primary system. Evaluation of these matters is a generic issue. Studies for this plant need not be completed prior to startup.

We note that a number of items have been identified as Outstanding Issues in the NRC Staff Safety Evaluation Report dated July 1981. These include some TMI-2 Action Plan requirements. We believe these issues can be resolved in a manner satisfactory to the NRC Staff, subject to the concerns on instrumentation for detection of inadequate core cooling expressed in the ACRS letter to the Executive Director for Operations dated June 9, 1981.

The Committee believes that, contingent on the Applicant's attainment of an adequate level of management and staffing, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that Waterford Steam Electric Station Unit 3 can be operated at power levels up to 3410 MWt without undue risk to the health and safety of the public.

We expect to report further on the adequacy of the staffing and management as progress is made toward improvement.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

References:

1. Louisiana Power & Light Company, "Waterford Steam Electric Station, Unit 3 Final Safety Analysis Report," with Amendments 1 through 20.

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit 3," Docket No. 50-382, USNRC Report NUREG-0787, July 1981.

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

August 12, 1981

APPENDIX XLIII
MEMORANDUM ON INTERACTIONS BETWEEN
SAFETY AND NON-SAFETY SYSTEMS

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

FROM: R. F. Fraley, Executive Director

RFFraley

SUBJECT: SEISMIC-INDUCED AND OTHER INTERACTIONS BETWEEN NONSAFETY AND SAFETY SYSTEMS

During its 256th ACRS Meeting, the Committee again raised with the NRC Staff the question of interactions between safety and nonsafety systems. The responses to these inquiries indicate that an organized and systematic approach to this matter has not yet been formulated for use in the licensing process. The Committee believes that this warrants attention, particularly in view of the large number of operating license applications which are currently under review.

For additional guidance as to the Committee's view of some suitable approaches to this matter, please refer to the following ACRS Reports:

- Systems Interaction Study for Indian Point Nuclear Generating Unit No. 3, October 12, 1979

- Report on TMI-2 Lessons Learned Task Force Final Report, December 13, 1979

- Diablo Canyon Nuclear Station, November 12, 1980

cc: R. L. Tedesco, AD/DL
T. Novak, AD/DL
G. Lainas, AD/SA
ACRS Members
J. McKinley, ACRS Staff
M. Libarkin, ACRS Staff
G. Quittschreiber, ACRS Staff
R. Major, ACRS Staff
J. Griesmeyer, ACRS Staff

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Discussion of Some Comments on NUREG-0739, J. Michael Griesmeyer and David Okrent, August, 1981

2. Memorandum, T. E. Murley to D. G. Eisenhut, R. H. Vollmer, R. J. Mattson and S. H. Hanauer, Generic Issues Tracking System (GITS), June 30, 1981

3. DRAFT, A "Recipe" for an ALARA Criterion for LWR Accidents, David Okrent and J. Michael Griesmeyer, July, 1981

4. Letter, R. B. Hubbard, MHB Technical Associates, to R. F. Fraley, Public Risk Associated with Low Power Testing Diablo Canyon Nuc ar Station, July 14, 1981

5. NUREG-0675 (Supple. 10), Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Station Units 1 and 2, Supplement 10, August 1980

6. Memorandum, R. F. Fraley to NRC Commissioners, Meeting of NRC Chairman and Commissioners with the ACRS on August 7, 1981, July 28, 1981

7. Memorandum, W. J. Dircks, NRC EDO, to S. J. Chilk, NRC Secretary, SECY-81-308, Request for Proposed (REP) No. RS-RES-81-173 Entitled, "Long Term Performance of Materials Used for High-Level Waste Packaging, June 19, 1981

8. Analysis with Respect to Periodic and Systematic Review of Regulations (TMI Action Plan Task IV.G.2): Section 50.49 pertaining to environmental and seismic qualification of electric equipment

9. Letter, S. H. Herwell, Atomic Industrial Forum, Inc., to H. R. Denton, NRR, regarding environmental qualification of safety-related electrical equipment and enclosures, July 2, 1981

10. Proposed Regulatory Guide 1.89 (Rev. 1), Environmental Qualification of Electric Equipment Important to Safety for Light-Water-Cooled Nuclear Power Plants, Draft 1, June 17, 1981

11. Letter, Rep. M. K. Udall, U.S. House of Representatives to Governor Richard Thornburgh, PA, regarding funding for cleanup of TMI-2, July 23, 1981

12. Letter, G. G. Sherwood, GE, to J. C. Mark, ACRS, Reliability of BWR High Pressure Water Systems Under ATWS Conditions, June 25, 1981

13. Memorandum, W. J. Dircks, NRC EDO to R. F. Fraley, ACRS, Reliability of BWR 5/6 High Pressure Core Spray System Under ATWS Conditions and Attach., Nov. 17, 1980

14. Memorandum, G. G. Sherwood, GE, to M. S. Plesset, ACRS, Anticipated transient With Scram - General Electric Comments on ACRS letters dated April 16, 1980, May 30, 1980

15. ACRS Consultant's report regarding meeting of the Electric Systems Subcommittee on Core Level Measuring Devices, May 28, 1981, Washington, DC, June 1, 1981

16. Memorandum, K. Kniel, NRC Staff to R. F. Fraley, Draft Action Plan for Task A-45, Shutdown Decay Heat Removal Requirements, May 22, 1981 and attach.

17. Memorandum S. H. Hanauer, NRC Staff to R. F. Fraley, NUREG-0799, Draft Criteria for Preparation of Emergency Operating Procedures, July 6, 1981

18. Letter, Nuclear Safety Oversight Committee to the President, July 23, 1981

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