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

REVISED
November 10, 1981

SCHEDULE AND OUTLINE FOR DISCUSSION
259TH ACRS MEETING
NOVEMBER 12-14, 1981
WASHINGTON, DC

Thursday, November 12, 1981, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. — Opening Session (Open)
   1.1) Report of ACRS Chairman re. activities of interest to ACRS (CM/RFF)

2) 8:45 A.M. - 12:45 P.M. — St. Lucie Plant Unit 2 (Open)
   Tab 2 ------------ 2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee regarding the request for an OL for this unit (WK/SKB)
   2.2) 9:15 A.M.-12:45 P.M.: Reports by and discussions with representatives of the NRC Staff and the Applicant
   (Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.)

12:45 P.M. - 1:45 P.M. — LUNCH

3) 1:45 P.M. - 5:45 P.M. — Callaway Plant Unit 1 (Open)
   Tab 3 ----------- 3.1) Report of ACRS Subcommittee regarding the request for an OL for this unit (MWC/RKM)
   3.2) Reports by and discussions with representatives of NRC Staff and the Applicant
   (Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.)

4) 5:45 P.M. - 6:30 P.M. — Preparation of ACRS Reports (Open)
   4.1) Discuss proposed ACRS report to NRC regarding the St. Lucie Plant Unit 2

Friday, November 13, 1981, Room 1046, 1717 H Street, NW, Washington, DC

5) 8:30 A.M. - 12:30 P.M. — Comanche Peak Steam Electric Station Units 1 and 2 (Open)

TAb 5 ----------- 5.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee regarding the request for an OL for this station (MB/HA)

5.2) 9:00 A.M.-12:30 P.M.: Reports by and discussions with representatives of the NRC Staff and the Applicant

(Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.)

12:30 P.M. - 1:30 P.M. — LUNCH

6) 1:30 P.M. - 3:00 P.M. — Systematic Evaluation Program (Open)

6.1) Report of ACRS Subcommittee regarding the scope and schedule for the SEP (WWM/RKM)

TAb 6 ----------

6.2) Reports by and discussion with representatives of the NRC Staff and representatives of the nuclear industry as appropriate

7) 3:00 P.M. - 3:30 P.M. — Future ACRS Activities (Open)

See Tab 8.1 --------- 7.1) Anticipated ACRS Subcommittee Activity

See Tab 8.2 ---------- 7.2) Anticipated ACRS Activity

7.3) ACRS Annual Report to the U.S. Congress on the NRC Safety Research Program - scope and organization of report regarding FY 1983 budget (CPS/SD)

8) 3:30 P.M. - 4:30 P.M. — Preparation of ACRS Reports (Closed)

8.1) Discuss proposed ACRS report to NRC on the Callaway Plant Unit 1

9) 4:30 P.M. - 6:30 P.M. ACRS Subcommittee Reports (Open)

Tab ------------------ 9.1) 4:30 P.M.-4:45 P.M.: Report of ACRS Subcommittee on Proposed NRC Rule (10 CFR Part 50) on Application of TMI-2 Lessons Learned to OL's (WMM/RKM)

See Handout ----------- 9.2) 4:45 P.M.-5:15 P.M.: Report of ACRS Procedures Subcommittee regarding proposed changes in scope of ACRS activities and procedures for assigning priorities for conduct of ACRS activities (CM/RFF)

See Handout ----------- 9.3) 5:15 P.M.-5:30 P.M.: Regulatory Activities regarding proposed NRC Regulatory Guides (CPS/SD)

See Handout ----------- 9.4) 5:30 P.M.-6:00 P.M.: Reliability of Electrical Power Supplies (JJR/RS)

See Tab 11.1-3) ------------ 9.5) 6:00 P.M.-6:30 P.M.: Human Factors considerations in the design/operation of nuclear powerplants including the qualifications and organization of management, operators and supporting infrastructure (DAW/RKM)

Saturday, November 14, 1981, Room 1046, 1717 H Street, NW, Washington, DC

10) 8:30 A.M. - 12:00 Noon — Preparation of ACRS Reports (Open/Closed)

10.1) 8:30 A.M.-12:00 Noon: Discuss proposed ACRS reports to NRC on:

- Callaway Plant Unit 1 (Closed)
- Comanche Peak Station Units 1 and 2 (Closed)
- St. Lucie Plant Unit 2 (Open)

(Portions of this session will be closed as necessary to discuss information which will be involved in adjudicatory proceedings.)

12:00 Noon - 1:00 P.M. — LUNCH

11) 1:00 P.M. - 1:30 P.M. — ACRS Subcommittee Reports (Open/Closed)

See Handout --- 11.1) 1:00 P.M.-1:30 P.M.: Reports of ACRS Subcommittees regarding:

11.1-1) Report of facility visits/meeting with Japanese representatives regarding Japanese regulatory policy, criteria and safety research activities (PGS/DO/MWC/HL/RFF) (Closed)

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

12) 1:30 P.M. - 2:00 P.M. — Miscellaneous Items (Open) - discuss miscellaneous items related to activities of members - MWCarbon attend International LMFBR Conference, Lyon, france

any prohibited transaction provisions to which the exemption does not apply; nor does the fact the transaction is the subject of an exemption affect the requirement of section 401(a) of the Code that a plan must operate for the exclusive benefit of the employees of the employer maintaining the plan and their beneficiaries.

(2) This exemption does not extend to transactions prohibited under section 4975(c)(1)(F) of the Code.

(3) This exemption is supplemental to, and not in derogation of, any other provisions of the Code, including statutory or administrative exemptions and transitional rules. Furthermore, the fact that a transaction is subject to an administrative or statutory exemption or transitional rule is not dispositive of whether the transaction is, in fact, a prohibited transaction.

Exemption

In accordance with section 4975(c)(2) of the Code and the procedures set forth in Rev. Proc. 75-26, 1975-1 C.B. 722, and based upon the entire record, the Department makes the following determinations:

(a) The exemption is administratively feasible;

(b) It is in the interests of the Plan and of its participant and beneficiaries; and

(c) It is protective of the rights of the participants and beneficiaries of the Plan.

Accordingly the sanctions resulting from the application of section 4975 of the Code, by reason of section 4975(c)(1)(A) through (E) of the Code, shall not apply to the sale of the stamp collection by Satloff to the Plan for $33,600: *Provided,* That this amount is not higher than the market value of the stamp collection as of the date of sale.

The availability of this exemption is subject to the express condition that the material facts and representations contained in the application are true and complete, and that the application accurately described all material terms of the transaction to be consummated pursuant to this exemption.

Signed at Washington, D.C., this 20th day of October 1981.

Ian D. Lanoff,

Administrator, Pension and Welfare Benefit Programs, Labor-Management Services Administration, Department of Labor.

[FR Doc. 81-31170 Filed 10-29-81; 8:45 am]

BILLING CODE 4510-29-M

MERIT SYSTEMS PROTECTION BOARD

Relocation of Headquarters Offices; Amendment

AGENCY: Merit Systems Protection Board.

ACTION: Notice; Amendment of Notice of Relocation of Headquarters Offices.

SUMMARY: This notice amends the notice of October 16, 1981 (46 FR 51094); the telephone number should read: 653-8897

EFFECTIVE DATE: October 30, 1981.

FOR FURTHER INFORMATION CONTACT: Frederick L. Foley, Acting Director, Personnel Management Division, Merit Systems Protection Board, Room 906, 1120 Vermont Avenue NW., Washington, D.C. 20419, 202-653-5916.

Merit Systems Protection Board.

Dated: October 26, 1981.

Ersa H. Poston,

Vice Chair.

[FR Doc. 81-31531 Filed 10-29-81; 8:45 am]

BILLING CODE 7400-01-M

NATIONAL ADVISORY COMMITTEE ON OCEANS AND ATMOSPHERE

Meeting Addendum

October 28, 1981.

An addition has been made to the Agenda for the November 2-4, 1981 meeting of the National Advisory Committee on Oceans and Atmosphere (NACOA) published in the **Federal Register** of October 22, 1981 (Page 51824). From 10:00 a.m.—11:00 a.m. on the morning of Monday, November 2, 1981, the speaker will be Mr. John Marcum of the Office of Science and Technology Policy.

Additional information concerning this portion of the meeting may be obtained through the committee's Executive Director, Steven N. Anastasion, whose mailing address is: National Advisory Committee on Oceans and Atmosphere, 3300 Whitehaven Street, NW., (Page Building #1, room 438), Washington, DC 20235. The telephone number is 202/653-7818.

Dated: October 28, 1981.

Steven N. Anastasion,

Executive Director.

[FR Doc. 81-31616 Filed 10-29-81; 8:45 am]

BILLING CODE 3510-12-M

NATIONAL SCIENCE FOUNDATION

Advisory Council, Task Group #19; Meeting

In accordance with the Federal Advisory Committee Act, Pub. L. 92-463, the National Science Foundation announces the following meeting:

Name: Task Group #19 of the NSF Advisory Council.

Place: Room 225, Baxter Hall, California Institute of Technology Pasadena, California 91125.

Date: Friday, November 13, 1981.

Time: 9:00 a.m. to 5:00 p.m.

Type of Meeting: Open.

Contact Person: Ms. Jeanne Hudson, Executive Secretary of the NSF Advisory Council, National Science Foundation, Room 518, 1800 G Street, N.W., Washington, D.C. 20550. Telephone: 202/357-9419.

Purpose of Task Group: The purpose of the Task Group, composed of members of the NSF Advisory Council, is to provide the full Advisory Council with a mechanism to consider numerous issues of interest to the Council that have been assigned by the National Science Foundation.

Summary Minutes: May be obtained from the contact person at above stated address.

Agenda: The Task Group is asked to consider the needs of organizations requiring policy research and analysis and to survey those NSF programs providing it. This will involve questions of coordination, possible overlap, and policy-making procedures as well as those of substance. In addition, the Council is asked to provide suggestions for the future development of NSF's science and technology policy resources.

Reason for Late Notice: Members could not reach agreement on meeting date and location.

Dated: October 27, 1981.

M. Rebecca Winkler,

Committee Management Coordinator.

[FR Doc. 81-31553 Filed 10-29-81; 8:45 am]

BILLING CODE 7555-01-M

NUCLEAR REGULATORY COMMISSION

[redacted]

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 November 12-14, 1981, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of th[illegible] meeting was published in the Federal Register on September 23, 1981.

The agenda for the subject meeting will be as follows:

Thursday, November 12, 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.

8:45 a.m.–12:45 p.m.: St. Lucie Plant Unit 2 (Open)—The Committee will hear and discuss the reports of its Subcommittee and consultants who may be present regarding the request for a full power operating license for this facility. Representatives of the Applicant and the NRC Staff will also make presentations and respond to questions regarding proposed operation of this unit.

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

1:45 p.m.–5:45 p.m.: Callaway Plant Units 1 and 2 (Open)—The Committee will hear and discuss the reports of its Subcommittee and consultants who may be present regarding the request for a full power operating license for this facility. Representatives of the Applicant and the NRC Staff will also make presentations and respond to questions regarding proposed operation of this plant.

Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter and information which will be involved in an adjudicatory proceeding.

5:45 p.m.–6:30 p.m.: Reports of ACRS Subcommittees (Open)—The Committee will hear and discuss reports of its Subcommittee and consultants who may be present regarding NRC regulatory activities related to proposed Regulatory Guides and preparation of the ACRS annual report to the U.S. Congress on the proposed NRC Safety Research Program.

Friday, November 13, 1981

8:30 a.m.–12:30 p.m.: Comanche Peak Steam Electric Station, Units 1 and 2 (Open)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present regarding the request for a full power operating license for this facility. Representatives of the NRC Staff and the Applicant will also make presentations and respond to questions regarding operation of this station.

Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter and information which will be involved in an adjudicatory proceeding.

1:30 p.m.–3:00 p.m.: Systematic Evaluation Program (Open)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present regarding the proposed NRC program for systematic evaluation of operating nuclear power plants. Representatives of the NRC Staff and the nuclear industry as appropriate will make presentations and participate in related discussion.

3:00 p.m.–5:00 p.m.: Alternate Decay Heat Removal Systems (Open)—The ACRS will hear the report of its Subcommittee and consultants who may be present regarding the proposed NRC action plan (Task Action Plan A-45) for Evaluation of Alternate Decay Heat Removal Systems. Representatives of the NRC Staff and the nuclear industry as appropriate will make presentations and participate in related discussion.

5:00 p.m.–5:30 p.m.: Future ACRS Activities (Open)—The members of the Committee will be briefed by members of the NRC Staff regarding the proposed NRC Staff organization and plan of action for accelerated review of the Clinch River Breeder Reactor. Anticipated activities for future Committee and Subcommittee meetings will also be discussed.

5:30 p.m.–6:15 p.m.: ACRS Subcommittee Reports (Open)—The Committee will hear and discuss reports of designated subcommittees regarding safety related issues including proposed changes in ECCS Evaluation Models (10 CFR Part 50, Appendix K); proposed NRC Rule (10 CFR Part 50) on application of TMI-2 Lessons Learned to Operating Licenses; and Japanese regulatory policy, criteria, and safety research activities.

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

Saturday, November 14, 1981

8:30 A.M.–12:30 P.M.: ACRS Reports to the NRC (Open/Closed)—The Committee members will discuss proposed ACRS reports to the NRC regarding the matters considered during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to the matters being discussed and to discuss information which will be involved in adjudicatory proceedings.

1:30 P.M.–2:30 P.M.: ACRS Subcommittee Reports (Open)—The Committee members will hear and discuss the reports of designated ACRS Subcommittees on a safety related matters including the reliability of electrical power supplies at nuclear power plants; human factors in the design and operation of nuclear plants, including the qualifications of management, operators and supporting infrastructures; and on ACRS procedures.

2:30 P.M.–3:30 P.M.: Concluding Session (Open)—The Committee will complete discussion of the items noted above and will discuss other miscellaneous matters related to nuclear safety and regulation.

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

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 relating to the matter being considered (5 U.S.C. 552b(c)(4)), information which will be involved in an adjudicatory proceeding (5 U.S.C. 552b(c)(10)), and information considered privileged and provided in confidence by a foreign source (5 U.S.C. 552b(c)(4)).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the ACRS Executive Director, Mr.

Raymond F. Fraley (telephone 202/634-3265), between 8:15 A.M. and 5:00 P.M. EST.

Dated: October 26, 1981.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 81-31582 Filed 10-29-81; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Reactor Fuel; Meeting

The ACRS Subcommittee on Reactor Fuel will hold a meeting on November 18, 1981 in Room 1046 at 1717 H Street, N.W., Washington, D.C. The Subcommittee will discuss with the NRC Staff the fuels research program. Discussion will focus on the budget levels for 1983 in preparation for the annual CRS Report to Congress on the NRC Safety Research Program.

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

The entire meeting will be open to public attendance except for those sessions of this meeting that may be closed to discuss the NRC Safety Research Program and Budget for 1983 as required (Sunshine Act Exemptions (2), (6), and (9)b). To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, November 18, 1981

8:30 a.m. until the conclusion of business.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, will exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentation by and hold discussions with representatives of the NRC staff, their consultants, and other interested persons regarding this review.

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 repesent oral statements and the time allotted therfore can be obtained by a prepaid telephone call to the cognizant Designated Federal Empolyee, Mr. Paul Boehnert (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m. est.

I have determined, in accordance with section 10(d) Pub. L. 92-463 that it may be necessary to close sessions of the meeting as noted above to discuss matters which relate solely to the internal personnel rules and practices of the agency (Exemption (2)), to discuss information of a personal nature, the disclosure of which would constitute a clearly unwarranted invasion of personal privacy (Exemption (6)), and to discuss preliminary information the release of which would be likely to significantly frustrate the Committee in the performance of its statutory function (Exemption (9)b). The authorities for such closure are Exemptions (2), (6) and (9)b to the Sunshine Act, 5 U.S.C. 552b(c)(2)(6)(9)b.

Dated: October 26, 1981.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 81-31583 Filed 10-29-81; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-99]

Babcock & Wilcox Co.; Proposed Issuance of Orders Authorizing Dismantling of Facility, Disposition of Component Parts and Termination of Facility License

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of orders authorizing the Babcock and Wilcox Company (the licensee) to dismantle the Lynchburg Pool Reactor (the facility), a pool-type nuclear reactor located in Lynchburg, Virginia, to dispose of the component parts in accordance with the plan set out in the licensee's application dated July 23, as supplemented September 23, 1981, and to terminate the facility license. The reactor is covered by Facility Operating License No. R-47.

Prior to issuance of any orders, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations.

By November 30, 1981, the licensee may file a request for a hearing with respect to issuance of the subject orders and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Requests for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the action under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Issue Date:

June 14, 1982

MINUTES OF THE
259TH ACRS MEETING
NOVEMBER 12-14, 1981
WASHINGTON, DC

CERTIFIED

The 259th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Chairman C. Mark at 8:30 a.m., Thursday, November 12, 1981.

[Note: For a list of attendees, see Appendix I. D. A. Ward was not present on Thursday. W. Kerr, H. W. Lewis, D. Okrent and M. S. Plesset were not in attendance 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 written or oral statements to the Committee. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC.

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

I. Chairman's Report (Open to Public)

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

A. Limited Appearance Statement by Joette Lorion

The Chairman informed the Committee to note a written statement submitted by Joette Lorion of the Center for Nuclear Responsibility at the October 30-31, 1981, St. Lucie Plant Unit 2 Subcommittee Meeting. Ms. Lorion requested that the Committee defer review of the St. Lucie Plant because of difficulties that the Center had experienced in obtaining copies of the SAR for review. The Members concluded that this was not an adequate basis to defer the ACRS review.

B. Retirement of Member of the ACRS Staff

The Chairman informed the Committee of the planned retirement of James M. Jacobs, ACRS Technical Secretary, at the end of the year. J. Jacobs was awarded a service pin for thirty years of government service.

C. Meritorious Service Award

Morton W. Libarkin was notified of a meritorious service award for exemplary management of the Project Reviews Staff. The award, a silver medallion, will be presented at the annual awards ceremony in January, 1982.

II. Meeting on St. Lucie Plant Unit 2 (Operating License) (Open to Public) [Note: Gary Quittschreiber was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Report

W. Kerr, Chairman of the St. Lucie 2 Subcommittee, briefly described the St. Lucie 2 plant as a second unit of a two unit plant, presently planned for operation at the same site. He pointed out that since the site was on a sparsely inhabited island, emergency planning for the site is somewhat unusual. He indicated that the emergency plans had not yet been completed to comform to Appendix E and therefore could not receive final approval by the NRC. He reminded the Committee that in connection with the Atomic Licensing Boards consideration of the Construction Permit for St. Lucie 2, the Licensing Board had concluded that the station blackout, which they define to be the loss of all a.c. power, should be a design basis accident.

Florida Power and Light had calculated, after further investigation of the probability of loss of offsite power and the ability to restore it, that the probability of loss of all a.c. power without restoration for a four hour period was 5×10^{-9} per year. A short discussion took place between P. Shewmon and W. Kerr concerning a blackout that occurred in Florida a few years ago. W. Kerr indicated that no formal intervention in the operating license procedure is expected at this time but there were presentations from four people at the subcommittee meeting. Their statements, as written, were part of the meeting notebook (see Appendixes IV and V).

W. Kerr indicated that a number of unresolved issues remain, including completion of emergency planning. None of these, however, appeared insoluble on some sort of a reasonable schedule. W. Kerr indicated that D. W. Moeller and C. P. Siess were also present at the subcommittee meeting, as well as H. Etherington during a portion of the meeting. D. W. Moeller commented that he had particular interest in the public health impact of a Class 9 or major accident in the plant. He said that after examining the environmental statement for St. Lucie 2, he was satisfied that the applicant had done an adequate job in assessing the environmental impact as far as they had gone. In terms of the aquatic and marine pathway of radionuclide intrusion into the environment, he was concerned that there was no discussion of plutonium or assessment of the impact of plutonium on the environment. D. W. Moeller found inconsistencies in the Florida Power and Light analysis of latent cancers and acute fatalities from major airborne releases. The acute fatalities for St. Lucie were 100 times what are expected for the Comanche Peak plant and yet the latent cancers are roughly the same. He hoped to get a clarification of that matter. Another item of concern was the population density and growth which D. W. Moeller indicated the Staff would discuss. D. Okrent asked D. W. Moeller about an analytical parameter used in the calculation of latent effects of radiation exposure, 25 rem in 30 years for large numbers of people. He questioned how groups like the BEIR Committee or other groups would view this dose limit. The discussion continued centered around a BEIR Committee report figure of 100 mrem per year as an acceptable dose rate with no observable health effects. This was a number quite a bit smaller than the 25 rem in 30 years. D. Okrent clarified for the Committee that he and D. W. Moeller were referring to two related but different items.

P. Shewmon brought up a different item which appeared on the first page of the status report for St. Lucie in the meeting notebook. It concerned the thermal power level for St. Lucie 2 which was reported as 2570 MWt, but the design thermal power was shown as 2700 MWt. The Committee expected to have the Applicant clarify this matter. D. Okrent brought up a question concerning emergency operating procedures for St. Lucie 2.

V. Nerses, NRC Project Manager for St. Lucie 2, indicated that the actual emergency operating procedures had not yet been distributed. D. Okrent requested that V. Nerses, as soon as convenient, acquire typical emergency operating procedures for each of the three PWR types and provide them to the Committee.

R. L. Tesdesco, NRC Staff, informed D. Okrent that the NRC Project Manager for Callaway indicated that the Applicant planned to discuss generic emergency operating procedures in the afternoon session. R. C. Axtmann asked a question about testimony at the subcommittee meeting concerning the rising level of the ocean. W. Kerr offered R. C. Axtmann the collection of information that was presented at the subcommittee meeting. His interpretation of the facts were that since the plant was designed for hurricanes that will produce water levels greater than an equivalent rise in the level of the ocean due to a so-called greenhouse effect, the plant was not actually at hazard.

B. <u>Site and Plant Description</u>

W. Derrickson, FPL, presented a brief discussion of the plant, the site, the project and the current status of the Operating License Application (see Appendix VII). P. Shewmon inquired as to what the ultimate heat sink was for the plant. W. Derrickson indicated that it was the Atlantic Ocean. Water is brought in from the Atlantic and discharged to the Atlantic. In answer to a question by J. J. Ray, W. Derrickson indicated that there were separate control rooms for Unit 1 and Unit 2.

W. Derrickson presented a brief report on the status of construction on Unit 2. He indicated that the plant had been under construction since June 1977 when FP&L received its Construction Permit. FP&L is currently about three weeks behind its original schedule developed in the spring of 1977. From a handout table entitled <u>Selected Quantity Status</u> (see Appendix VII), W. Derrickson selectively discussed the percentage of completions for certain installed pipes and conduits. In answer to a question by D. Okrent, W. Derrickson indicated that the reason FP&L was able to avoid the long delays that most of the utilities experienced was, because of experience with the NRC on three previous units, FP&L was able to anticipate some of the impacts.

E. W. Dotson, FP&L, then answered the previous question by P. Shewmon concerning the difference between 2560 and 2700 MWt power ratings. E. W. Dotson indicated that this is a fairly ordinary procedure for Florida Power and Light in that they buy the equipment installed in the plant for greater design and flexibility. As an example, he continued, Unit 1 is in the process now of applying for a stretch power rating to increase the power level. Therefore, actual plant analyses are based on 2560 MWt, with a margin so it is possible to increase the power level at some later date. He indicated that this two step procedure had been a company policy on fossil plants for quite some time because FP&L can ordinarily buy slightly larger pieces of hardware for very little increase in expense.

P. Shewmon was concerned that condenser leaks might put salt water into the secondary side of the plant. W. Derrickson indicated that both condensers had titanium tubes. He indicated that Florida Power and Light experience with titanium tubes had been very, very good. P. Shewmon was not entirely satisfied by the answer.

C. Discussion and Review of SER Open Items

1. NRC Staff Presentations

V. Nerses began his presentation by presenting a viewgraph of recent and projected licensing milestones for the St. Lucie project. He corrected the closing date for the comments on the Draft Environmental Statement, which was shown as December 1, 1981, to December 14, 1981. The final Environmental Statement is to be issued on January 15, 1982, another correction, not 1981. Eighteen open items were presented on two viewgraphs (see Appendix VI). V. Nerses gave a one sentence explanation of each item individually. P. Shewmon raised the question of whether the NRC had reviewed the ability of a St. Lucie 2 operator to distinguish between a steam line break and a small break LOCA. He explained that the problem comes up with regard to pressurized thermal shock and how the operators are trained to react to this situation or situations of a similar character. V. Nerses indicated that he did not know the answer to this question and suggested to the Committee that the question be deferred until the discussion of emergency operating procedures later in the session.

2. Florida Power and Light Response

E. W. Dotson concurred with the NRC's assessment of open items and had no additional comments.

D. Discussion of OL Review Issues by FP&L

1. Organization and Management

J. Williams, FP&L, presented a series of viewgraphs entitled Utility Technical Capability and Organization (see Appendix VII). A general organization chart of Florida Power and Light was first presented and discussed briefly. A second chart gave a more detailed breakdown of the Nuclear Energy Department of which J. Williams was Director. J. Williams briefly discussed the backgrounds of the managers underneath him in the organization chart. He reported that K. N. Harris, the Assistant Manager for Nuclear Energy, was formerly the plant manager at St. Lucie 1. H. E. Yaeger is currently the Turkey Point Site Manager. He described the Nuclear Services Group which provides technical services and plant support at the corporate level for operating nuclear plants of FP&L in five specific areas. The areas shown on the chart were technical support, codes and inspections, licensing, emergency planning, and health physics. He indicated that the physical operation of St. Lucie 2 when licensed will be incorporated into the existing Unit 1 organization. No major changes are contemplated to the St. Lucie 2 site organization other than manpower. A third viewgraph entitled Abbreviated Plant Organization presented the activities of the plant staff divided into its four major functional areas of maintenance, technical support, quality control and operations.

J. W. Williams then described three independent groups which routinely evaluate FP&L's nuclear plants. The first of these, the Company Nuclear Review Board (CNRB) oversees nuclear operations at the corporate level. He stated that the CNRB functions to provide independent review of designated operating nuclear plant activities. The next viewgraph described a St. Lucie facility review group (FRG) currently functioning on site for St. Lucie Unit 1. The activities of this group are to be extended to Unit 2 upon the issuance of the Unit 2 Operating License. The FRG functions on site to advise the plant manager in all matters related to nuclear safety. His last viewgraph described in brief an independent safety engineering group which was set up to perform independent reviews of plant operations in accordance with the guidelines of NUREG-0737. J. W. Williams indicated that this organization had evolved over the last 10 years during the time FP&L had successfully managed three nuclear plants.

J. J. Ray questioned the role of alternate members of the St. Lucie FRG. He also pointed out that there did not appear to be a regular member on this committee who represented training responsibilities. A short discussion concluded when J. W. Williams indicated that he was not aware of a specific person designated to represent the training supervisor (an alternate member of the FRG) when he was not present.

D. W. Moeller questioned the lack of radiation protection competence on the Company Nuclear Review Board. D. W. Williams indicated that K. N. Harris, although not a health physicist, is one of FP&L's radiological duty officers and will provide that input to the CNRB. R. J. Acosta also has quite a lot of plant experience in the area of radiological safety. He is also an alternate member. D. W. Moeller then asked the NRC Staff if this was an acceptable procedure. V. Nerses referred the question to H. C. Dance, Chief of Reactor Project Section 2 C of Region II. H. C. Dance indicated that a company nuclear review board is required to have expertise in 8 or 9 specific technical areas, one of which is health physics. H. C. Dance indicated that he would check on D. W. Moeller's question. He was confident, however, that the NRC already had looked into that matter. H. C. Dance also fielded a question from D. W. Moeller at the subcommittee meeting as to whether it was NRC policy to require outside members on the company nuclear review board. H. C. Dance indicated that the NRC would expect FP&L to provide the expertise from an in-house source or from outside. Either is acceptable so long as they provide the expertise. If they did not have the expertise inhouse, there is a specific requirement in a technical specification also consistent with the ANS Standard that the utility would go outside to a consultant or an outside organization to acquire that expertise. This is typical of all facility technical specifications, not only those of FP&L. D. W. Moeller asked if someone on the Committee could present the Committee's position on the matter of a totally inhouse review board. D. Okrent indicated that this had been a recurring question over the years and the Committee had never taken any formal position as to whether the Committee was really happy with the position just enunciated by the Staff, or whether there should always be someone from the outside on the review board.

2. Operator Selection and Training

P. L. Fincher, Training Supervisor for St. Lucie Plant Units 1 and 2, presented Florida Power and Light's approach for training the operations personnel for Unit 2. He stated that the process used for selection of operating candidates involved a screening examination administered by Memphis State University, which was aimed at determining the candidate's aptitude and capabilities for entering and completing an operator training program. He indicated that it also includes a psychological review. P. L. Fincher indicated that qualified candidates are subjected to an inhouse management review by personal interview conducted by himself and the operations supervisor at the plant. The final culmination of the selection process involves a review of the three items or three parts of the screening process by plant management before final selections are made.

The next major topic covered by P. L. Fincher was the Licensed Operator Training Program (the entire presentation is described in Appendix IX). Other major topics covered in the presentation included licensed operator requalification implementing 10 CFR 55 Appendix A, the St. Lucie Unit 1 licensing program, the difference between Units 1 and 2 training, and the License

Candidate Simulator Program and the License Requalification Simulator Program. P. L. Fincher, in answer to a question posed by P. Shewmon concerning the staffing for Unit 2, indicated that current staffing plans call for a minimum of 43 licensed personnel to operate St. Lucie Units 1 and 2 which includes shift supervisor, a senior reactor operator on each unit, and two reactor operators on each unit. He indicated that 43 people will provide enough personnel to run six shifts. He indicated that FP&L currently had in training and already qualified as licensed operators on Unit 1, 62 people with 1 still to fill a slot. P. Shewmon posed two questions of P. L. Fincher concerning training of instrument people and procedures on secondary water chemistry control and operation with leaks. P.L. Fincher indicated that the instrumentation control technicians at St. Lucie and at Turkey Point, before being assigned as instrument technicians, have to complete a certain level of training on instrumentation and controls through the apprentice program or through prior education. He said that the instrumentation specialists are certified or qualified to operate on the various types of instrumentation on a generic basis. The ensuing discussion concerned identification or certification of an instrument technician's qualifications during an operating incident. P. L. Fincher deferred the question to K. N. Harris from the Nuclear Energy Dept. of FP&L. He indicated that FP&L had developed a procedure whereby system engineers would direct the work of an FP&L instrument and control (INC) specialist.

K. N. Harris indicated that FP&L has built a very strong qualified staff of INC people from its experience at Turkey Point and St. Lucie 1. He indicated that plant systems are assigned by an engineer in the INC Department such that each system does have an assigned engineer who is responsible for that system and will be expected to respond to any problems that come up in the system. The supervisor on shift would expect the assigned engineer to respond to a problem. Selection of a qualified instrumentation technician is not random or based upon union contracts. K. N. Harris clarified that it is the supervisor for that system (specific people who are expert on that system) who designates or establishes who is qualified. P. L. Fincher added that the supervisor would respond, as well as the technician, to do the work to ensure proper handling of the problem.

W. M. Mathis questioned whether an operator licensed for Unit 1 will also be licensed to operate Unit 2. P. L. Fincher indicated affirmatively and qualified his statement indicating that in the opinion of FP&L the differences between Unit 1 and Unit 2 are not of sufficient magnitude to preclude dual licenses. J. J. Ray established that P. L. Fincher was the alternate member of the Facilities Review Group. P. L. Fincher explained that the plant manager and the operations superintendent to whom Fincher reported would represent his viewpoint on the Facilities Review Group when he could not attend. F. Miraglia, NRC Staff, interjected a comment related to dual licensability. He indicated the Staff position that at this time an operator should be sufficiently trained on either Unit 1 or Unit 2 and that while it is desirable perhaps in the future to have operators licensed on both units, the Staff feels until sufficient operating experience is gained by Unit 2 operators and the differences between Unit 1 and 2 are clearly understood, dual licensability would not be acceptable to the NRC Staff.

W. Kerr questioned if the Staff would refuse to license an operator who was licensed on Unit 1 for operation on Unit 2 unless he relinquished his license on Unit 1. F. Miraglia explained that the operator would have to be designated to operate at just one of the units. He could not be transferred between units. Further discussion elicited the fact that the Staff would not refuse to license someone who was qualified on Unit 1 for operation on Unit 2, but, while he is operating Unit 2, the Staff will refuse to recognize that he is licensed to operate Unit 1. W. Kerr as well as K. N. Harris expressed being confused by the NRC Staff position. W. Kerr suggested that FP&L discuss the matter with F. Miraglia.

D. W. Moeller brought up a question of feedback on experiences with the steam generator tubes on Unit 1 for the operators of Unit 2. He indicated that he was looking at feedback material in general such as that which would be found in the LER file. The question was deferred to the presentation by R. R. Jennings.

3. Feedback to Operators of Nuclear Plant Operating Experience

R. R. Jennings indicated that shift technical advisors who report to him do a great deal of the operating experience assessment and feedback function. He indicated that various sources of information are first filtered by the Program Administrator at the Corporate Office and reviewed for applicability with the assistance of engineering and design groups at the Corporate Office. R. R. Jennings indicated that there are quite a few sources used by FP&L including NRC input in various forms such as notices, circulars and bulletins, INPO Safety

Evaluation Reports, and significant operating experience reports. P. G. Shewmon questioned the process for determining what distinguishes between what is significant to a licensed operator and what is significant to an instrument and control technician. R. R. Jennings indicated that procedure guidelines used by shift technicians and the technical supervisor make sure internal reports or material from external sources are sent to the proper individuals. In answer to one of P. Shewmon's questions, R. R. Jennings indicated that FP&L spent $8 million retubing its condensers with titanium tubes to prevent seawater leaks. He said they also learned from industry experience that copper in the feedtrain was bad and stainless steel tubes were used as replacements in the feedwater heaters. P. G. Shewmon thought that FP&L might be able to avoid some of the trouble they are having with the steam generators if they paid more careful attention to air and seawater leaks. R. R. Jennings indicated that FP&L does pay careful attention to seawater leaks. He indicated that FP&L has procedures for immediate reductions in power, isolating the water box, and draining. He also indicated that FP&L has a procedure for isolating a quarter of a condenser so that the operator can reduce power and not have to shutdown and stop the seawater incursion by draining the water box. K. N. Harris indicated that FP&L has a procedure whereby the unit load is changed based upon chloride ingress. "Immediate action is to be taken to isolate the condenser inleakage and loads to be reduced, the maximum blowdown established until such time as the water chemistry is brought back into specifications." P. Shewmon continued by stating that procedures do not always work. K. N. Harris stated that the secondary water chemistry experience on St. Lucie for the last two cycles testifies to the fact that procedures do work.

4. Emergency Operating Procedures Concerning ATWS

J. H. Barrow presented a brief discussion of the objectives, hierarchy of priorities, format and content of emergency operating procedures (see Appendix X).

D. Okrent cited a letter dated September 15, 1981 to Kenneth Baskin, Chairman of the CE Owners Group, copy to Donald James of FP&L, in which the NRC Staff raised some questions concerning the CE emergency procedure guidelines. He asked J. H. Barrow to explain some of the concerns in the letter and how FP&L is addressing them. J. H. Barrow attempted to defer the question to the NRC Staff. D. Okrent insisted upon an answer from FP&L.

W. Windecker, the licensed operator on St. Lucie Unit 1 currently working on the startup of Unit 2 stated that he had some input into the writing of these emergency procedures. He offered to answer the question. Mr. Windecker's answer was general in nature. He stressed that FP&L had several discussions with the NRC and had taken their guidance into account in designing these procedures. As the discussion continued, W. Windecker indicated that after the accident at Three Mile Island, inadequacies were pointed out in these procedures. Since that time and under a time constraint, FP&L has been trying to prepare the most workable set of emergency operating procedures. M. Bender questioned what dependence FP&L had placed on the material provided by Combustion Engineering in the development of its procedures. W. Windecker indicated that FP&L had taken the CE information as being "solid and firm and what we could use." M. Bender was able to elicit from W. Windecker the fact that the CE procedures were symptomatic probably containing definite shortcomings. W. Windecker indicated that FP&L and CE are not in agreement on which procedures are right and which are not right at the present time. D. Okrent pressed W. Windecker about identification of some of the gaps in the CE procedures.

D. Okrent asked V. Nerses of the Staff if there were any significant matters with respect to the emergency operating procedures that needed to be addressed. W. G. Kennedy indicated that the Staff has older documents with NRC interim approval for use as a technical basis for procedures for upcoming plants. Therefore, he indicated that the Staff has a technical basis for St. Lucie Unit 2 and in the long-term expects to have acceptable guidelines from the CE Owners Group. D. Okrent questioned why the Staff is still looking at older documents two years later. W. G. Kennedy indicated that the Owners Group's initial submittal was received by the NRC in June or July and the NRC Staff has sent comments to CE on problems in this second submittal.

D. Okrent noted that the Committee was especially concerned about a third paragraph in the CE Owners Group guidelines that required the operator to diagnose a specific event before entering the procedures. The Committee was especially concerned with the approach taken by an operator who was not able to diagnose a specific event. In this case the operator was directed to the inadequate core cooling guideline, the plant status, and appended tables that address critical safety functions. W. S. Windecker indicated that FP&L did not follow the CE guidelines in that particular case. The discussion revealed that it was still not possible to exactly determine what the operator would do if he could not diagnose the specific event that was taking place.

W. Kerr redirected the discussion to the P. Shewmon question of how an operator tells the difference between a small break LOCA and a break in the main steam line. W. S. Windecker attempted to explain the characteristics of the difference between these two incidents. He stated that all licensed operators have attended a simulator training course in Windsor, CT, where the various effects of different LOCAs is demonstrated. J. Ebersole entered in a discussion describing a scenario where certain service systems fail giving indications of a problem but leaving the degree of severity of the consequences of these failures unknown. He posed the case of one train of the component coolant failing or a d.c. power train failing. W. S. Windecker answered this question by indicating that FP&L had for a long time, various off normal procedures which were not considered emergency procedures. Component cooling water malfunction was one of them. A generator tube rupture was another one.

The discussion then turned on FP&L dependence upon having natural circulation with steam generators. P. L. Fincher of FP&L indicated that there were conditions that could occur in a plant such as a major LOCA, a steam line break, or situations where voiding occurs in the reactor coolant system where the possibility exists that natural circulation would not be possible. Under those circumstances in the case of a LOCA, the high pressure safety injection system would inject coolant into the reactor coolant system and pump it out through the break.

J. Ebersole pointed out that the St. Lucie design differs from the standard CE design in that St. Lucie has PORVs.

P. L. Fincher indicated that FP&L specifically requested that PORVs be made a part of the design as they are a part of the Westinghouse design at Turkey Point.

D. Okrent then asked the Staff if the Staff had a kind of schedule for development of approved operating procedures. W. G. Kennedy indicated that NUREG-0737 required the submittal of the analysis by January 1, 1982. The NRC was expected to approve that and cause it to be implemented at the first refueling outage after January 1, 1982. For plants licensed after that, it would be required before the plant was licensed.

W. G. Kennedy indicated that because of the difficulty in approving those procedures, the Staff is considering changing the schedules. D. Okrent asked where the difficulty occurred.

W. G. Kennedy indicated that basically the submittals have not all been strongly enough symptom-based to handle multiple casualties. They tended to be event oriented and, therefore, specific events with complications were not adequately handled. J. H. Barrow, FP&L, indicated his concern about the state of flux in the industry with respect to emergency operating procedures, but indicated that FP&L intended to make sure that they had symptomatic procedures that addressed all possible events. D. Okrent stated his own feeling that there should be a very real interest in emergency operating procedures in the industry.

J. Ebersole noted that this applicant, through its own efforts, caused an alteration in CE design by putting PORVs in the plant. He asked the Staff what their position was on whether they considered this the proper thing to do. R. L. Tedesco in answer to this question about whether or not other CE plants should have PORVs, indicated that it was the subject of a long term resolution dealing with rulemaking on inadequate core cooling in degraded cores. He did indicate that there were benefits.

5. A.C. Power System Reliability Including Station Blackout

J. Franklin presented a discussion of the Florida Power and Light Power Supply System Transmission Facility supplying the St. Lucie plant, the onsite a.c. and d.c. power systems and the station blackout event (see Appendix VII). J. Franklin indicated that the FP&L transmission system forms a portion of the Florida State transmission network with several ties to other utilities within the state and ties with the Georgia State transmission system to the north. The St. Lucie site is tied to the FP&L system at a midway substation by three physically independent, 240 kv transmission lines. He indicated that the transmission lines terminate at the 240 kv switchyard at the St. Lucie site in a four way breaker and a half arrangement. The 240 kv switchgear was described in detail in accord with a viewgraph which is a part of Appendix VII. J. Ebersole raised a question about the intertie between the safety related d.c. buses considered by many to be a degrading influence on their reliability. J. Franklin of FP&L discussed the double breaker intertie between the two systems in detail. He indicated that there was a mechanical interlock through a key switch which also provided an electrical interlock.

J. Franklin indicated that if St. Lucie experienced a single d.c. bus failure, it would have one bus remaining and some access to additional supplies of power.

J. Franklin then described the St. Lucie station blackout event (see Appendix VII viewgraph entitled Total Loss of AC Power). He indicated that conclusions of the analysis of the event were that subcooled natural circulation is maintained for a minimum of four hours. Decay heat removal capability is maintained in excess of four hours, and FP&L's battery capacity is sufficient to operate required equipment in excess of four hours with selective load reduction. In answer to a question by J. J. Ray, J. Franklin indicated that the mean time to restoration of transmission to the station had been determined by FP&L to be approximately 27 minutes with a high confidence level. He also indicated that St. Lucie has blackstart capability within the system. P. Armond, FP&L, indicated that FP&L had blackstart capability at The Turkey Point plant and the Ft. Everglades plant which is about 120 miles south of the St. Lucie site. M. P. Armond continued to discuss diesels and gas turbines in response to several prompting questions by J. J. Ray.

J. Ebersole brought up another topic concerned with salt spray in a hurricane condition deactivating insulators in the switch yard. J. Franklin indicated that FP&L experience was that the plant was not likely to shutdown because rain would keep the insulators clear. J. Ebersole then asked if there were any cables at St. Lucie which are of an emergency category which are normally in a dry environment but are subjected in rare circumstances to submergence. J. Ebersole was concerned how the Applicant could validate after a period of years that under a submerged environment a cable presummably qualified for submerging might in fact be degraded and incapable of operating under submerged conditions. J. Franklin was not exactly sure how to answer the question. P. O. Chopra, NRC Staff, was not able to point to NRC interest or activity in this subject. J. J. Ray and J. Ebersole agreed that this was potentially a generic issue in that it is a recurring industry-wide condition under various circumstances.

6. Shutdown Capability Outside Control Room

C. L. Fisher, FP&L, indicated that his talk would concern the capability to shutdown from outside the control room in the unlikely event that the control room had to be evacuated. He indicated that transfer of the controls from the control room to a hot shutdown panel would be made by actuating manual transfer switches which are located outside the control room (see viewgraph presentation in Appendix XI). C. L. Fisher's presentation consisted primarily of describing the functions to accomplish a cooldown cold shutdown. They are: reactor coolant circulation, decay heat removal, boration and makeup, and depressurization. After discussing these four processes, he concluded that the plant could be maintained in hot standby condition entirely from the hot shutdown panel. The plant could be cooled down to cold shutdown outside the control room using the hot shutdown panel and equipment designed for handling a LOCA.

In response to an inquiry from J. Ebersole, C. L. Fisher indicated that the St. Lucie Plant is capable of handling multichannel failures in the control room by using transfer switches. G. Harrison, NRC, indicated that the Staff position on smoke and fumes in the control room assumes that fire does cause damage to electrical cables and equipment. Reliance is placed on the alternate shutdown panel.

D. W. Moeller questioned how control room operators were going to breath if the control room had an internal source of fumes or toxic gases. The essence of his question was why the Staff or the Applicant did not examine the control room ventilation system to determine whether the capability for use of outdoor makeup air existed. With outdoor makeup capability, the air within the control room could be maintained at an acceptable level for breathing and avoidance of evacuation in the event of an internal source of fumes. G. Harrison indicated that the Fire Protection Branch of NRC assumes evacuation. J. Ebersole questioned what degree of evacuation would take place in a two unit plant or multi-unit station if the site were to experience another TMI-2 level of contamination of the environment and a degree of leakage larger than that experienced during the TMI-2 accident. G. Harrison indicated that to his knowledge in fire protection, the Staff had not defined any points for evacuation including fire or smoke. It was assumed that the plant operator would be the one to determine that. The discussion was effectively tabled when V. Nerses indicated that the proper answer to the evacuation question would have to come from the Accident Evaluation Branch of NRC.

7. <u>Instrumentation to Follow the Course of a Serious Accident</u>

B. Pagnozzi, FP&L, gave a brief history of St. Lucie 2 instrumentation criteria. He explained FP&L's commitment to Regulatory Guide 1.97, Rev. 2 (see Appendix VII for slide presentation). A question and answer discussion that followed covered the areas of performance of core exit thermocouples, and the question previously asked of the ability of reactor operators to distinguish between a small break LOCA (pressure drop) and an overcooling incident. M. S. Plesset who asked the question initially was not entirely satisfied with either the Applicant's answer or the response of the NRC Staff. He said that he still considered this an open question and thought that, if there were an unambiguous way for an operator to know quickly whether he has a small break LOCA or an overcooling transient, that would be very significant and useful. He did not feel confident that this was available at present.

J. Ebersole remarked that he noticed a common problem in looking at instrumentation following the course of an accident which includes PORVs, blocked valves, level indicators, and pressurized heaters. He felt that the NRC was not providing environmental qualifications and control for these items even though they must all face a hostile environment.

B. Pagnozzi indicated that FP&L is buying the best available equipment on the market following programs set up by the key instrumentation suppliers, and actively procuring equipment to meet those seismic and LOCA envelopes within the containment design and the outlying areas for harsh environments. R. L. Tedesco indicated that the Staff has a qualification program for safety equipment and also a program on valves and their operability. M. Bender qualified J. Ebersole's question as not involving the capacity of valves to operate under certain conditions, but the environment that surrounds the valves. R. L. Tedesco stated that he did not think that the NRC program considered the external environment. M. Bender continued the discussion by asking FP&L if they were confident that the equipment being provided could survive the environment in which it needed to work. E. W. Dotson, FP&L, indicated that FP&L is in the process of submitting all instrumentation qualifications that it is performing to the NRC as supporting evidence for one of the open items by the end of November 1981. NRC is to select on an audit basis from those qualifications and field check the instrument qualifications.

8. Control Room Design Changes Resulting from TMI Experience

B. Pagnozzi displayed a viewgraph (see Appendix XI, last page) which showed additional instrumentation and controls implemented in accord with the requirements of NUREG-0696, -0737 and Regulatory Guide 1.97, Rev. 2. He explained, in detail, each of the several items added since the occurrence of the Three Mile Island accident.

9. Emergency Planning

H. D. Johnson, FP&L, briefly discussed changes in emergency planning resulting from a population increase, the wind direction and its relation to evacuation. H. D. Johnson indicated that as population grows, certain emergency parameters need to be reevaluated. These include the following:

- Estimates of evacuation time
- Account of improvements and additions to the transportation network
- Augmenting of the public warning system
- Plan for additional facilities for hosting and care of evacuees.

There were no questions from the Committee.

H. D. Johnson indicated that FP&L is committed to conform and comply with the emergency planning rule in 10 CFR 50 part 47, section B as well as NUREG-0654, Rev. 1, which has recently been made into a Reg. Guide. In response to a question by D. W. Moeller, H. D. Johnson indicated that two 12 volt heavy duty batteries were used as the system for backup power for a public address alarm system. A partial wind rose is attached as Appendix XII which H. D. Johnson used to explain the predominance of trade winds in the northwest, north northwest, and west northwest sectors. In answer to D. W. Moeller's question, FP&L was able to show that for a high percentage of the time the direction of the wind will have no bearing on the selection of the evacuation route. At no time will wind direction be a significant deterent or hazard to people evacuating the plant.

J. Sheetz, FP&L, attemped briefly to describe FP&L's policy in restricting population growth at the St. Lucie plant site. He indicated that FP&L had no control over the private actions of the people of Florida to choose home sites. He added that FP&L does not attempt to influence elected officials in St. Lucie and Martins Counties, Florida or the counties further away to modify any zoning regulations established within their jurisdiction for the purpose of population density control. He did indicate that FP&L has attempted to develop an evacuation plan which would safely allow the timely evacuation of the island.

D. W. Moeller questioned J. Sheetz about a draft proposed siting rule that the NRC has been developing. This rule requires the licensee once a year to survey potentially adverse offsite developments such as the potential consequences of adverse population growth. C. P. Siess pointed out that the survey did not imply that the applicant was to do something about the adverse condition. A short discussion of projected population took place. K. P. Twine of Ebasco Services participated.

L. Soffer, NRC, presented NRC's policy on population growth around nuclear power plants (see Appendix XIII). With respect to St. Lucie, the Staff concluded that the site was typical of other nuclear plants. The end of life population figures exceeded slightly Regulatory Guide 4.7 trip levels (projected above average) but were not beyond the range of other plants. D. W. Moeller read a sentence from an ACRS letter entitled Report on Proposed Rule on Reactor Site Criteria, which stated the Committee concern that, if the rule were to place limits on only the average population density as a function of distance from the reactor, with no limitations on density within an angular sector, the rule would permit a large densely populated city to be located near a plant. He indicated his personal misgivings and the Committee concern about this matter in the past. D. W. Moeller cited his concern about the tremendous population growth in St. Lucie County. He felt that perhaps the Planning Board did not clearly understand the potential long term evacuation difficulties high population density would cause.

R. Cordell, NRC, described the Staff analysis for the assessment of groundwater releases from the St. Lucie site. D. W. Moeller questioned why the analysis does not discuss plutonium. R. Cordell indicated that the dose factors for plutonium are very low compared to other elements such as cesium and strontium. He stated that plutonium was neglected from the assessment of doses because the contribution was negligible. D. W. Moeller cited 1900 curies of plutonium 239 (half life of 25,000 years) in the core at the time the hypothetical accident occurred. R. Cordell indicated that he would look into the matter but felt confident that the Staff had considered all core fission and activation products in the assessment of liquid pathway dose.

D. J. Perrotti, NRC Staff, explained that a weakness identified in the emergency plan for St. Lucie gave instructions to site worker evacuees to go from the site north to a site assembly station and adjacent public park area, but did not permit the emergency coordinator on the site to give alternate routes to these evacuees. In the SER, D. J. Perrotti indicated that the applicant had agreed to add this provision to the plan as of a September 4 commitment. He added that these instructions pertain to site evacuation of nonessential, onsite workers. D. J. Perrotti indicated that there is no standard upon which to gauge the acceptability or nonacceptability of evacuation times. Chairman Mark asked a question about buildings that offer shelter on Hutchinson Island. D. J. Perrotti indicated that most of the buildings in that part of the country are made of cement block, stucco, or cinder block-stucco (CBS). He offered to pass on the shielding factor to the Committee at a later time. He added that the Staff did not mean to imply that the evacuation time which is a tool used by the local emergency preparedness decision makers is the only tool which they would use to determine what protective measures to take in the event of an accident.

10. Miscellaneous Carryover Items from the Subcommittee Meeting

J. Sheetz, FP&L, briefly informed the Committee of new data which had been obtained which showed that a postulated Hutchinson Island fault does not exist below the Island (see Appendix XIV). He indicated that a marine seismic reflection survey was conducted this past summer to investigate the hypothesized fault. The first recorded discussion of the alleged fault occurred in an unpublished master's thesis. No faults of any kind were found in the sediment sequence. However, several areas of localized and possibly connected warping were found.

Chairman Mark asked the Committee whether it was necessary to hold a short, closed session on the matter of industrial security. After a few short questions were answered by J. Sheetz, the Committee proceeded to a discussion of their position on writing a letter to the Applicant. The Committee agreed that it could write a letter which may have conditions of qualification, but which would be generally supportive of granting of the operating license.

III. NRC Briefing of Analysis Errors Found at the Diablo Canyon

F. Miraglia, NRC Staff, presented the background summary of errors detected to date, the reverification program that the Utility had proposed to NRC in the first week of November, tentative Staff conclusions, and NRC's own proposal for reverification (see Appendix XV). Errors detected to date include the inappropriate application of the containment annulus diagram. As the Utility was engaged in the reanalysis effort, another error was discovered - the incorrect distribution of seismic response spectra. As a result of the technical order and Staff inspection, the same package of information that transmitted the inappropriate diagram also transmitted weight and weight distribution for equipment in the containment annulus. In addition, during the reanalysis effort a number of additional errors were detected by the Utility that were unrelated to the initial errors. C. P. Siess noted that the fan coolers that were not affected by the first reappraisal, were now affected by the incorrect weights and weight distributions. J. Ebersole disclosed a potential generic problem where seismic competence is determined by analysis. Analysts do not go far enough to evaluate the performance of necessary equipment. This came up in the case of cracked battery cases that should have been seismically qualified upon installation. F. Miraglia indicated that the Staff had reached some tentative conclusions. It appears that there was a lack of rigor and formality in the design control used by the utility, in that the QA system provided by PG&E did not establish a formal interface with QA controls between them and URS-Bloom. During the discussion that ensued, C. P. Siess suggested that errors like those at Diablo Canyon might be generic. He questioned whether the Staff's audit program was adequate.

IV. Callaway Plant Unit 1 (Open to Public)

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

A. Report of the ACRS Subcommittee

M. W. Carbon indicated that the Callaway Subcommittee visited the site last week to hold a meeting (see Appendix XVI). He indicated that the plant was similar to Comanche Peak and that the operating license ASLB hearing was contested. Unit 2 has been canceled. He felt that the SNUPPS organization added technical strength to the organization of Union Electric, and he urged the Committee to listen for that kind of information during the presentation. M. W. Carbon thought that the Committee should pay particular attention to the applicant's presentation of emergency operating procedures and instrumentation to follow inadequate or degraded core cooling. He felt that the Subcommittee identified a lack of commercial nuclear experience in the utility's organization as an area of concern. J. J. Ray, another member of the Subcommittee, added that Callaway is in an ideal position from the viewpoint of reliability and power supply since it is peripherally surrounded by other power pools. Should there be an area-wide blackout affecting those ties, they have black start capability. The Committee held a short, general discussion during which a generic document from Westinghouse entitled Summary of Westinghouse Owner Groups Emergency Response Guideline Program was mentioned. D. Okrent expressed interest in these emergency measures. A. Passnater, Union Electric, explained the design of four simulators available for training of Callaway reactor operators.

B. Union Electric's Discussion of Its Organization and Management

A. Passnater, acting as coordinator for the presentation, introduced D. F. Schnell, Vice-President of the Nuclear Department of Union Electric. D. F. Schnell explained his qualifications, a functional organization chart, and then the qualifications of other major managers at Union Electric (see Appendix XVII). He indicated that perhaps 12 to 15 additional people were to be added to the home office organization between now and fuel load. D. F. Schnell mentioned an independent safety engineering organization which now has seven people budgeted and will have an additional five people and a supervisor named to fill that responsibility. In response to a question by J. Ebersole, D. F. Schnell indicated that the ISEG would report functional abnormalities at the plant site. Committee Members questioned D. F. Schnell extensively about the relationship of the SNUPPS organization to Union Electric and the lack of commercial nuclear power plant operational experience as contrasted with total experience shown and Callaway experience. D. Okrent questioned the reliance of Union Electric on SNUPPS experience and the experience of the architect/engineer, Bechtel Corp. D. F. Schnell attempted to explain where the expertise in Union Electric itself was to be found. Some discussion of the safety oriented groups - Nuclear Safety Review Board, Onsite Review Committee and the Independent Safety Engineering Group - took place. M. W. Carbon was concerned that the training program for shift technical advisors was less than adequate. D. F. Schnell introduced J. F. McLaughlin who discussed technology transfer and the startup organization.

C. Operation Staffing and Training

J. F. McLaughlin explained that Union Electric has retained a startup staff of highly experienced engineers and technicians. He added that it is a commonly accepted that startup experience is the best experience for developing operations expertise. He explained how the technology transfer would take place by having reactor operators and equipment operators operate the equipment in the systems under the direction of the startup engineers. J. N. Kaelin, Superintendent of Startup at Callaway, explained the Callaway test program for startup as shown in a viewgraph which was taken from the SNUPPS C FSAR Section 14.2. The second transparency presented the total organization and indicated where Union Electric personnel were assigned in the organization.

J. N. Kaelin indicated that with the exception of himself, almost all of the commercial experience was Callaway startup experience, whereas most of the total experience was coal plant experience. It was pointed out by M. W. Carbon, that of the 44 startup engineers and supervisors that have direct responsibility for the systems, only 12 of those are Union Electric permanent people.

M. A. Stiller, Superintendent of the Callaway Nuclear Power Plant, presented in a chart the functional organization that exists onsite today (see Appendix XVII). He indicated that the present staff is approximately 80% of that projected at fuel load. Manpower levels as a fraction of the number of individuals assigned and those available at the plant site as of November, are shown on the above mentioned chart.

Operations manning will be divided into six complete shifts to provide for adequate relief and retraining without extended overtime. M. A. Stiller indicated that Union Electric determined that their best interest would be served by developing their own staff through the selection of key experienced personnel from the existing organization who had some nuclear background or capabilities. They would then complement them with experienced nuclear personnel hired from the outside. M. A. Stiller indicated that other nuclear experience shown in the charts is predominately Navy. Slides were presented of the selection criteria for operating personnel, technicians and the sources of operating experiences followed by the organization. In answer to a question by D. W. Moeller, M. A. Stiller indicated that Dr. Hughes, Supervisor of the Independent Safety Engineering Group, has assigned responsibility for following LERs and assuring that the various people who are to implement the lessons learned really do it.

P. T. Appleby, Superintendent of Training for Union Electric, defined the Training Department in a block diagram and explained the qualifications of the training supervisors (see Appendix XVII). He defined the different phases of the training program, concentrating in part specifically on the licensed operator training program. A floor plan of the Callaway training center was shown as well as discussion of site technical advisor training. P. T. Appleby indicated that Union Electric presently has a staff of 20 instructors and expects to increase the staff to a level of 22 instructors in the future. D. W. Moeller questioned Union Electric about how they incorporate the lessons learned and LERs into the training program. P. T. Appleby indicated that these reports are reviewed by individuals almost constantly and also go out to other superintendents within the department. P. Shewmon asked about the criteria and training certification of instrumentation control technicians.

D. Open Items from SER

1. Presentation of Licensee Conditions and Confirmatory Items

G. E. Edison, NRC Licensing Project Manager on the Callaway plant, presented a summary of open items, noting that there were eight items plus 5 TMI related items (see Appendix XVIII). There was considerable discussion about item 10, the Fire Protection Program.

J. Ebersole indicated that Union Electric had gone to the lowest possible interpretation of Reg. Guide 1.75, even permitting redundant cables in common enclosures, not to mention a common room like a control room. D. F. Schnell, F. Schwoerer of SNUPPS, disputed that statement, indicating that SNUPPS had gone to great lengths to separate electrical cables. F. Schwoerer went into considerable detail explaining the cable spreading rooms in the SNUPPS plants and the auxiliary shutdown panel. The disagreement in the discussion was not resolved to the satisfaction of J. Ebersole. G. E. Edison responded to D. W. Moeller's concern involving TMI item 2.B.2, Plant Shielding for Access. He indicated that the Staff could not review the structures until the design was nearly complete and the shielding in place.

2. Applicant Response to Open Items from SER

R. L. Stright, SNUPPS, responded for the Applicant to the SER open items. He indicated that the first item on ice loads was not considered to be an unresolvable technical or licensing problem. He indicated that Union Electric is 75% complete on its pipe break analysis and viewed it as more of a confirmatory issue. He expected to submit information the following day to cover the cable tray seismic analysis and resolve that issue. R. L. Stright indicated that he did not understand the Staff problem about the pump and valve operability issue. He indicated that the Applicant had waited to complete the final shielding analysis in order that they would know the exact installation of the post accident sampling system and some of the implications of the operating procedures. As a final comment, he indicated that he did not understand the reason for a requirement for a surveillance program on control rods to be a license condition of concern. He indicated that SNUPPS and Union Electric worked with the Staff to propose an alternate program to resolve the issue in a different way. J. J. Ray commended the Staff Project Manager on his handling of the review of the Callaway plant.

D. Okrent asked Union Electric what their plans for the future were on a PRA for this plant. J. O. Cermak, SNUPPS Staff, indicated that Union Electric did not plan to do a probabilistic risk assessment for the Callaway plant at this time.

E. Emergency Planning

N. G. Slaten, Supervising Environmental Engineer for Union Electric, presented a summary of the status of the Union Electric Emergency Plan (see Appendix XII). In answer to a Committee question by R. C. Axtmann, N. G. Slaten indicated that Missouri does have an organization that can handle or plan an evacuation in the event of tornados, chemical spills, or a problem dealing with civil defense matters. N. G. Slaten presented two slides, one of which showed a ten mile radius called a plume pathway exposure zone, the area which generally involves the most detailed planning. He indicated that most of the zone is within Callaway County, with smaller amounts in Osage and Montgomery and a minute area in Gascanade Counties. All of the counties have joined together with Callaway acting as the lead county. D. W. Moeller questioned whether Union Electric felt that its primary responsibility in terms of emergency planning was to notify the Missouri Disaster Planning and Operations office and the Callaway County Sheriff's office. N. G. Slaten indicated that as far as notification was concerned that was the primary responsibility. In response to another Moeller question, N. G. Slaten indicated that Union Electric plans to install a siren network to cover the EPZ which will have multizone sounds but will not be a public address system. Brochures would tell the population to tune to the emergency broadcast system when they heard this siren. Another slide showed the locations of the emergency response facilities, highlighted by a technical support center located within the protected area ajacent to the service building. The service building would be expected to contain the office of the people who would man the technical support center.

The question of control room design was deferred until a later time since the control room remains an open item until late 1982.

F. Emergency Operating Procedures and Instrumentation Related to Degraded Core Cooling

A. P. Neuhalfen, Superintendent of Operations for Callaway, offered to discuss Westinghouse Owners Group procedures development in a generic session with the ACRS. In addition, he offered a four volume procedures document for subcommittee review. The offer was accepted by D. Okrent. A. P. Neuhalfen presented Union Electric's current development and implementation of emergency procedures as well as format, philosophy, and coordinated use of the separate portions of these procedures (see Appendix XVII). He indicated that current development consists of following the Westinghouse Owners Group guidelines which were presented to utilities the week of September 28, 1981. A chart entitled, Coordinated Use of Emergency Response Guidelines, was presented. It was indicated that the guidelines provide a means of continuously monitoring the plants critical safety functions through the use of status trees. A. P. Neuhalfen went into considerable detail explaining the nature of these procedures through the use of example status trees. J. J. Ray elicited from A. P. Neuhalfen that some criticality guides have not been written by Union Electric but are currently under development by the Westinghouse Owners Group. Several fairly technical questions were addressed to A. P. Neuhalfen concerning various accident scenarios and actions which an operator could take after evaluating the indications on his instrumentation. Members of the Committee felt that even with these operator procedures, difficult decisions will have to be made that may point out that good operator training and good operating procedures may not be all that is needed.

P. Shewmon presented a question on Union Electric's operating procedure for their full flow demineralizer. J. Ebersole brought up a question concerning Westinghouse's design of their vessel level indication system. He pointed out that breaks in the tubing system of the Delta P Cells could invite the problem of confusing indications in the case of a small break accident. A. P. Neuhalfen indicated that this was not the sole means of level detection for the core. The DP Cells are used in conjunction with the core exit thermocouples and the core monitor. A discussion of a small-break accident scenario took place between J. Ebersole and J. O. Cermak of SNUPPS.

G. Decay Heat Removal

F. Schwoerer discussed the functional requirements of a cold shutdown as shown on the slide which is the last page of Appendix XVII. He indicated that the SNUPPS plants are designed to meet the guidelines of Regulatory Guide 1.139 which says that you should be able to go to cold shutdown after an assumed loss of offsite power coincident with a safe shutdown earthquake. A discussion took place involving the dryout of a steam generator. Westinghouse personnel contributed specific design information to this discussion.

H. NRC Staff Remarks on Commercial Experience of the Applicant

K. R. Baker from Region III indicated that the Licensee was found to be somewhat short on commercial experience. The Staff has imposed on the Licensee as part of the license condition that they have on shift for one year or until the reactor reaches 100% power one individual with a minimum of one-year on shift license experience in a similar commercial operating plant. M. W. Carbon asked K. R. Baker if the Staff thought the plant superintendent or superintendent of operations at Callaway had need for advisory people. K. R. Baker answered that the Staff was only imposing the requirement of an individual on shift. If Union Electric has this experience on shift, he does not have to provide anybody in an advisory capacity.

I. Closing Remarks to the Applicant

The Committee agreed that they were in a position to write a letter in favor of granting the operating license in certain conditions. D. Okrent noted that the Applicant should be encouraged to do a good job developing emergency procedures.

V. OL Review of Comanche Peak Steam Electric Station Units 1 and 2 (Open to Public)

[Note: H. Alderman was the Designated Federal Employee for this portion of the meeting.]

A. Report of the ACRS Subcommittee

M. Bender, Chairman of the Comanche Peak Subcommittee, briefly described Comanche Peak Steam Electric Generating Station as being situated on a very good, quite remote site. He pointed out certain special features of the plant which included hafnium control rods, N-16 power detectors instead of the old system which measured the incremental temperature for the purpose of scramming the reactor, and the fact that this is the first plant implementing IEEE-323 as one of its requirements. He also cited the seismic and environmental protection qualification of certain protected instrumentation. He indicated that the utility group that will operate this plant has established an agressive, young operating contingent. The plant appears to be well constructed and the number of open issues does not appear to be large, with most of a somewhat routine nature (see Appendix XIX).

B. NRC Staff Overview of Plant and Operational Schedule

S. B. Burwell, Licensing Project Manager, gave a brief overview of the OL review (see Appendix XX). His second slide listed the unique features of the Comanche Peak Station which were touched upon by M. Bender. Each of the five features was discussed separately in an individual slide. S. B. Burwell showed a chart of the open items since supplement 1 of the SER had been issued. He indicated that there was no difference between the Staff and the Applicant on these open items. It was just that the Staff review has simply not been completed. S. B. Burwell discussed each of the non TMI open items, individually. He discussed nine TMI open items shown on a separate viewgraph. The last slide of the presentation indicated two license conditions on which there was continuing discussion and disagreement. The first item concerns two manual valves in which misposition open or closed could prevent or degrade the ECCS function. The Staff has taken the position that these are sample points of vulnerability. The second item involves a required inspection of the low pressure turbine disc at the first refueling outage. This is because the industry has experienced difficulty with cracking of low pressure turbine discs. The Applicant disagrees with the Staff on inspecting the turbine that early in its life. The problem is that neither the Applicant nor the Staff have the information needed to make a sound decision on this matter at this time. After an informational question and answer session with S. B. Burwell, M. Bender remarked that the Staff had done an exceptionally good job at distilling down the matters of disagreement that needed resolution.

C. Project Overview - TUGCO

H. Schmidt of Texas Utilities Generating Co., listed the participants to this hearing, including the owner utilities. He described the plant, briefly reviewing construction milestones, including an estimated fuel loading date of June 1983. He indicated that construction on Unit 1 was 89% complete and that it was 52% complete on Unit 2. He concurred in S. B. Burwell's discussion of the open items. H. Schmidt then showed a few color slides of the construction process giving the Committee an overview of how the plant was laid out. P. Shewmon was concerned about drought and the drying up of Lake Grandbury. H. Schmidt described the connecting pipe lines to the Brazos River and to Lake Grandbury which is an onstream lake on the Brazos River. A short discussion on the restricted use of ground water and the condition of the water tank took place.

M. Bender brought up the question of inspection of the turbines to the Staff. W. S. Hazelton explained that it was Staff practice that ever since the turbine disc cracking problem occurred, to encourage utilities to inspect turbines that they felt to be subject to stress corrosion cracking significantly prior to a possible failure. P. Shewmon remarked that this Staff policy appears to have little to do with reactor safety. It is just a very conservative policy. R. L. Tedesco, NRC Staff, pointed out that the objective of the policy was to maintain integrity and preclude the probability of failure. M. Bender concluded that there is a risk assessment aspect to this problem that the Staff does not seem to be applying very rationally. He felt that this overemphasis might be out of proportion to the significance of the problem.

D. TUGCO Discussion of Organizational Capability

B. R. Clements, Vice-President for Nuclear at TUGCO, defined TUGCO as the operating department of Texas Utilities and Texas Utilities System, Inc. (TUSI) as the Engineering Construction Department in a viewgraph (see Appendix XXI). He presented the Comanche Peak operational organization, breaking down the corporate nuclear organization by personnel on board and authorized for 1984, and the nuclear experience in man-years. The chart of the Nuclear Operations Department was accompanied by tables showing the authorized manning levels and the plant nuclear experience. B. R. Clements then mentioned the hiring of EDS Nuclear for the startup group. M. Bender asked what kinds of EDS Nuclear skills TUGCO was using. B. R. Clements explained that EDS Nuclear is helping in procedure writing, procedure checkout, procedure and program development and other areas of nuclear expertise. B. R. Clements pointed out that TUGCO has agreed to the Staff requests to have a person with commercial nuclear experience assigned to each shift as an advisor to the shift supervisor during the early portion of Comanche Peak operation. He also indicated that TUGCO has available many consultants including Westinghouse, NUSAC, Quadrex and others available on a full-time basis.

D. Okrent questioned what was meant by operating experience in the early days of operation. B. R. Clements indicated that this meant while proceeding to 100% power. D. Okrent then turned to the Staff for an explanation of what was exceptional about 100% power requirement. L. P. Crocker, NRC Staff, indicated that this was a measurable figure that occurs on the order of about a year after the plant goes into operation. He pointed out that they also would have completed their startup test program by that time. D. Okrent seemed concerned about the technical depth in the TUGCO organization and questioned whether the organization had the capability to do systems analysis with computer codes. B. R. Clements indicated a discussion of their Operations Review Committee at the Subcommittee review on November 11, 1981. He indicated that advisors would come from various academic and industrial sources in Texas and nationwide.

This Committee would have voting members external to TUGCO, will meet once a month until the beginning of operation, and then, according to NRC regulations. The Committee discussed the membership of the Corporate Review Committee. D. Okrent asked if there was any individual or group in the TUGCO organizational setup whose only function and responsibility was public health and safety. B. R. Clements indicated that there was no special group with only that function. He indicated that the Independent Safety Engineering Group would have this as one of their functions but not their only function. D. Okrent pressed the Staff for a policy statement on the question of whether there should be one particular group or individual within utility organizations with only the function of protecting the public health and safety.

E. Training Programs

R. B. Seidel briefly identified the different types of training programs currently conducted (see Appendix XXI). He described the systems and fundamentals program in a viewgraph which covered the topics in that program. He described the maintenance training program, the program for mechanical maintenance personnel, the technician training programs, specialty training for the chemistry area, and a shift technical advisor training program. The STA program used the Westinghouse training center at Zion, Illinois. Additional slides outlined the operator training programs, initial licensed operator program, the replacement training program, the requalification training program and the simulator training program. R. B. Seidel, in answer to a question from J. Ebersole, indicated that there was a consistent theme in the overall training program to emphasize the performance and importance of safety systems. R. B. Seidel defined the types of procedures at Comanche Peak (see Appendix XXI), the steps in procedure operation, and the status of procedure writing at the Comanche Peak station. B. R. Seidel explained that the Station Operations Review Committee (SORC) reviews, votes and acts on all procedures that are written by the plant manager. In answer to a question by M. Bender, he indicated that the SORC Committee is responsible for reviewing any safety questions concerning the operating staff. SORC has representatives from operations, maintenance, engineering, chemistry, health physics and quality assurance. The plant manager is the chairman. SORC reviews all safety questions including procedures. M. Bender questioned the status of emergency operating procedures at Comanche Peak. R. B. Seidel indicated that TUGCO originally developed their own emergency procedures based upon the guidelines available and the best information from Regulatory Guides and other plants. He indicated that they are currently revising these procedures in accord with new Westinghouse Owners Group guidelines that were issued since the TMI-2 incident.

P. Shewmon asked a question about secondary water chemistry. D. W. Braswell of TUGCO indicated that they had looked at the secondary chemistry program of other utilities and decided that in order to ensure that the integrity of the secondary system for steam generators was maintained, they would add full flow condensate polishing demineralizers. P. Shewmon again asked if a procedure entitled Actions to be Taken for Off-Controlled Point Chemistry Conditions would be developed. D. W. Braswell said that it would be in place in the second quarter of next year. P. Shewmon asked if he might see a copy of the procedure when it is developed. M. Bender asked H. Schmidt if it would be possible to get a copy of the procedure. H. Schmidt agreed.

F. Safety Parameter Display System (SPDS)

R. Estes, Lead I&C Engineer for Texas Utilities, described the response facility computer system which also provides TUGCO SPDS top level displays. He showed actual photographs of typical displays. He also explained that there are redundant central processer units. SPDS display parameters were described as based upon critical safety trees. Shown were the parameters that the operator would need to make the determination on those trees. J. Ebersole questioned whether these enhanced systems would provide too much information which would lead the operator possibly down a trail to trouble. R. Estes explained that the system in no way affected the reactor trip or the Engineered Safety Features Actuation System. B. R. Seidel added that this is simply an aid to enhance the normal operating procedures that are in place. R. Estes went to considerable detail describing the function keyboard and the types of parameters the operator would see.

G. Loss of A.C. Power

R. D. Calder spoke about the reliability of the station electric power and a.c. power system at Comanche Peak and also the survival time for loss of all a.c. power. He explained the voltage transmission network (see Appendix XXI). A slide entitled Diesel Facts defined the 7000 kw diesels attached to each train. Members of the Committee asked several questions concerning the monthly tests of the diesels. R. D. Calder described the reactor protection system d.c. power supply (labeled slide 5 in the presentation on Reliability of Station Electric Power and D.C. Power System). R. D. Calder described the symptoms of the loss of all a.c. power. J. Ebersole expressed the belief that there might not be a clean break of power as it goes from full power to no power. He questioned the criteria for undervoltage tripout on losing a.c. power. R. D. Calder then described the operator goals for this type of event. He indicated that Westinghouse, through the Owners Group, had developed a generic procedure for this event which tied the top of core uncovery to decay heat removal on the order of 100 hours for the Comanche Peak Plant. A slide then explained d.c. decay heat removal.

Additional charts described d.c. power supply and emergency lighting as well as the communications capabilities for the station in the event of loss of a.c. power.

J. Ebersole asked whether TUGCO had examined the auxiliary feedwater supply for subtle dependencies on the a.c. system. J. J. Ray asked TUGCO of its experience with failure of diesels to start. This lead to a request by M. Bender to D. Jones of TUGCO to check with San Onofre to see if data is available on the failure of diesels to start. D. Woodlan of Texas Utilities indicated that they had looked into the case of a gradual degradation of voltage and had installed alarms to alert the operator to this condition so that he could respond and take action. S. B. Burwell of the Staff, in answer to a question from J. J. Ray, indicating that after the Millstone experience, a requirement was placed on utilities concerning a gradual loss of a.c. power.

H. Hydrogen Control and Engineering Changes to Inert Containment

F. W. Madden, Technical Support Group, Comanche Peak, explained the current design basis and design features of the hydrogen control system. He indicated that the hydrogen purge system was left up to operator discretion. In response to TMI Lessons Learned, TUGCO explained that they had added vent lines to the top of the reactor vessel with remotely operated valves that provide the capability of venting the reactor coolant system from these high points, and also installed a post-accident sampling system. F. W. Madden then discussed the preliminary analysis to take account of a new proposed rule for dry PWR containments (SECY-81-245A). E. J. Bond, Gibbs and Hill, in answer to a question by D. A. Ward concerning the yield pressure on equipment hatches and gaskets, indicated that there was a factor of 1.5 to 2 in yield pressure over allowable pressure.

In answer to a question by M. Bender which came up at the Wednesday subcommittee meeting, F. W. Madden indicated that TUGCO felt that preinerting the containment would not be very desirable at Comanche Peak. He concluded that post-accident inerting using CO_2 would be the optimum choice and indicated that TUGCO would design a storage system for approximately 500,000 gallons or 500,000 pounds of CO_2. In summary, he concluded that it is technically feasible, but would be a major undertaking from both an engineering and expense point of view. D. W. Moeller noted that a previous slide showing the hydro-purge system indicated purging through charcoal filters. D. W. Moeller wanted to know the capacity of the charcoal filters. S. Kumar, Gibson and Hill, indicated that he did not have the details on the filters but indicated that the hydrogen purge system was used purely as a backup system. He said that the plant had a redundant recombiner system which was designed to limit the hydrogen concentrations below 4%. M. Bender asked F. W. Madden to provide information about dependency of the system on the composition of radionuclides in the containment environment.

M. Bender asked D. W. Moeller to amplify the question about plutonium which occurred yesterday. D. W. Moeller indicated that he had discussed the matter with the NRC Staff, had received an answer to most of the question, and a promise of a written response to the rest of the question.

P. Shewmon brought up an item in Nucleonics Week which discussed insignificant seismic design changes that were very costly to Comanche Peak. H. Schmidt indicated that this concerned installation of seismic supports on piping systems and cable trays. The matter, he continued, concerned the extensive amount of analysis and reanalysis and redesign being done. H. Schmidt explained that the acceleration of the SSE did not change but analysis iterations regarding piping supports did not converge as fast as expected.

I. Application of Probabilistic Risk Assessment Analysis to Comanche Peak

R. D. Calder, Manager of Technical Support at Comanche Peak, indicated that TUGCO agrees with the industry position that PRA is a valuable tool to weigh alternatives and improve the safety of nuclear power plants. He indicated that it is not TUGCO's intent to do a full blown PRA for Comanche Peak. What they are doing is reliability analyses and a reliability study of the auxiliary feedwater system for cases of loss of feedwater and offsite power and loss of all a.c. power. TUGCO has shown that their system has a high reliability factor for the loss of feedwater and offsite power and a medium reliability for the loss of all a.c. power. In response to a question by W. Kerr, R. Werner of TUGCO indicated that a determination of the reliability of the auxiliary feedwater system was made using the techniques recommended in NUREG-0611. R. Werner indicated that his conclusion that the system was very reliable was made using the same procedure in NUREG-0611 used by the Commission in their study.

J. Ebersole asked R. D. Calder if TUGCO had looked into putting diversity in the Westinghouse scram system to improve reliability.

He lauded TUGCO for having crossed-tied certain systems such as the component cooling system between Units 1 and 2. J. Ebersole suggested that TUGCO go to cross ties to improve simple redundancy on critical service systems. He was also concerned about a valve operability assurance program that did not convey realization that TUGCO knew whether certain critical valves could in fact interrupt very large flows which they are subject to during pipe breaks. R. D. Calder deferred the answers to these questions.

J. Review of Systems Interaction

R. D. Calder indicated that systems interaction is an unresolved safety issue, A-17, Recommendation 9 of the Long Term Lessons Learned. R. D. Calder indicated that procedurized, interdisciplinary review, which could be called a systems interaction, was conducted by the architect/engineer, Gibbs and Hill. He indicated that TUGCO had a dedicated systems engineering group called the Damage Study Group which does hazard analysis. R. D. Calder answered a question from J. Ebersole indicating that a comprehensive study had been made to study the influences of nonseismic equipment on seismic equipment. A discussion took place of a criteria GDC-19 for designing a control room to prevent undue entrance of poison to the operator. F. W. Madden of TUGCO indicated that a control room habitability analysis of the shielding and the plant ventilation was to protect the control room to meet GDC-19 and was based on the design basis containment leak rate. R. D. Calder indicated that systems interaction was used on control systems failure analysis, heavy loads analysis and LER review.

D. W. Moeller thanked the Applicant for his written comments on the control rods and the N-16 monitoring system (see Appendix XXII). He questioned the reason why TUGCO looked at gross gamma count instead of simply at the higher energy N-16 gammas. F. Thompson of Westinghouse indicated that that was what they were originally doing, but it was too sensitive to environmental conditions. Since the system could not be qualified to the appropriate environment, it was decided to go along with the gross gamma count. P. Shewmon questioned the connection between N-16 gammas and overtemperature. F. Thompson indicated that the N-16 system is a power meter used in place of the typical delta T measurement to go into the overpower delta t protection system and delta p protection system. F. Thompson indicated that the N-16 system allows a direct power measurement instead of the old method of indication of allowable power levels as a function of pressure and temperature. In answer to a question by J. Ebersole, F. Thompson indicated that the system did not serve to detect failed fuel, but detected it because of additional gamma counts.

K. Committee Caucus

Chairman Mark found from a consensus among the Committee Members that the ACRS could provide a letter to Comanche Peak on the operating license. D. Okrent felt it important to add certain remarks since he would not be present on Saturday. His remarks concerned the state of emergency operating procedures, independent participation in the corporate safety review board, appropriate sophistication of the plant organization with respect to commercial nuclear operating experience, and utility knowledge of possible kinds of serious accidents that could occur and their consequence. M. Bender indicated that TUGCO was not ready to commit to a system of reactor vessel water level indicators. R. D. Calder indicated that TUGCO had, nevertheless, done extensive studies of the different systems that were available. M. S. Plesset commended TUGCO on its deliberate approach.

VI. Review of Probabilistic Risk Assessments for Nuclear Power Plants

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

A discussion took place of the way to handle the review of the PRAs that were being submitted to the NRC Staff on Limerick, Zion, Indian Point and Big Rock Point. The Committee recognized that a National Lab would be assigned to do a fairly extensive review of the Zion PRA by August 1982, while a short term review would be conducted by Brookhaven National Laboratory. D. Okrent agreed to organize a group of ACRS consultants (approximately 12) to review the PRA for Zion. An objective of this ACRS study would be to compare the consultants's findings with the quick look analysis assigned by the NRC to Brookhaven National Laboratory.

VII. Report of the ECCS Subcommittee Concerning Proposed Changes in 10 CFR 50 Appendix K

M. S. Plesset, Subcommittee Chairman, noted that the Committee has previously objected to the piecemeal review of 10 CFR 50 Appendix K. He indicated that last August, General Electric had applied for an exception to Appendix K, proposing a shift in the actual power distribution to better utilize fuel.

He indicated that the Staff should be able to evaluate these BWR, ECCS evaluation models by January or February of 1982. This item was therefore deferred as a future ACRS activity.

VIII. Systematic Evaluation Program

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

W. M. Mathis, Subcommittee Chairman of the Reactor Operations Subcommittee, discussed the October 29 Subcommittee meeting whose purpose was to have a briefing by the Staff of the current status of the Systematic Evaluation Program.

M. W. Mathis referred to the first introduction of the subject, a letter of June 14, 1966 to the then Chairman of the ACRS, D. Okrent (see Appendix XXVI). This letter suggested that a periodic 10 year review of operating power reactors be conducted. A similar letter in November 1970 (see Appendix XXIV) went to the Chairman of the Commission signed by Spence Bush that again recommended a comprehensive review. Another letter in October 1979 (see Appendix XXV) to the Commission signed by M. W. Carbon complained about the lack of progress on the SEP program. W. M. Mathis indicated that the program is moving along at a reasonable pace now and the first plant will come to the ACRS full Committee for review as early as March 1982. He indicated that the Reactor Operations Subcommittee recommends that each of the SEP plants be first reviewed by the individual plant subcommittees with the Operations Subcommittee providing assistance through some overlap of membership.

W. Russell of the NRC Staff began a presentation on the SEP by indicating that the Staff is reviewing the possibility of combining the deterministic SEP type approach with a probabilistic perspective. He added that the Staff is looking at some of the open issues on Palisades from a risk perspective. W. Russell explained a priority ranking system the Staff expects to use which will be based upon a point score safety significance to identify the basis for backfits to these plants. D. Okrent asked questions about the point system. W. Russell indicated that the intent was to give more credit to preventing accidents and improving operational safety than to mitigating accidents. W. Russell indicated that the issue of the use of the point system was being reviewed again. He continued that backfitting on a plant was to be justified on a written basis and not by an absolute point count. W. Russell indicated the purpose of the SEP program was to review and document comparisons of the old facilities with what is currently required for licensing on a new plant, and to provide the basis for integrated and balanced backfit decisions (see Appendix XXVI). (The Staff briefing by W. Russell closely followed the material in Appendix XXVI).

J. Ebersole mentioned the cracked battery cases that had occurred at Diablo Canyon and inquired as to the depth of the seismic analysis review that the Staff does on the SEP plants. The Palisades Plant was explained in some detail as an example of how the program worked. W. Russell indicated that there are a large number of different types of issues that the Staff is reviewing and trying to integrate to make balanced and integrated backfitting decisions and integrated plant safety assessments. A tabulated schedule was presented showing the prospective dates for full ACRS Committee reviews. The plant, Palisades, was shown scheduled for March 1982. W. Russell explained the process by which the review was completed as a decision making process taking six months after the completion of the safety evaluation on the plant.

The Committee Members and the Staff discussed the criteria and rationale used to determine what items were to be backfitted. W. Kerr asked whether one could quantify a certain percentage reduction in risk which would be considered substantial and could justify a recommendation for backfit. After the Staff presentation ended, P. Shewmon summarized for the Committee procedures to be employed in the particular project subcommittees in order to bring one of the plants to the full Committee for review. Chairman Mark asked a representative of the owners of SEP-reviewed plants present at the meeting if they recognized the benefits from the SEP study. R. A. Vincent of Consumers Power thought that of the 23 topics being considered for backfit on Palisades, very few in the Owners Group's judgment have any significant impact on risk. He supported the concept of SEP but felt that an inordinate number of man-hours and utility resources were required to complete many of the SEP topics. He hoped that in phase 3 of the program a very careful weeding out of topics considered insignificant would be done. J. Ebersole asked the Owners Group why no activity was voluntarily spent during the last 12 year period to possibly detect deficiencies in these plants that the industry knew about from more recent licensing cases. R. A. Vincent indicated that comparison of older plants to today's criteria would show that the differences are not that significant when considering the impact on risk. The conclusion of this short discussion ended the discussion of the SEP program.

IX. Report of the Subcommittee on Human Factors (Open to Public)

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

D. A. Ward, Subcommittee Chairman, reported on the status of proposed NRC NUREGs on Control Room Design Criteria (NUREG-0700), Evaluation of Control Room Design (NUREG-0801), Human Factors Acceptance Criteria for the Safety Parameter Display System (NUREG-0835), Criteria for Preparation of Emergency Operating Procedures (NUREG-0799), and Utility Management Guidelines and Technical Resources (NUREG-0731) and noted that a decision is needed regarding the degree to which the ACRS desires to be involved in consideration of these guides (see Appendix XXVII). No action was taken by the Committee regarding this matter.

Consultant reports were introduced from R. G. Pearson of North Carolina State University, W. M. Keyserling, and J. Buck. It was indicated in the Pearson report (see Appendix XXVIII) that procedures should receive highest priority in the program and that the proliferation of CRT units will lead to more visual human factors concerns. W. M. Keyserling noted that there were an insufficient number of human factor specialists (see Appendix XXIX); he favored more conventional displays over CRT units which might be more economical but not necessarily the best alternative. J. Buck was encouraged by the recognition of human factors concerns in balance of plant areas and maintenance (see Appendix XXX).

X. Report of the Regulatory Activities Subcommittee (Open to Public)

[Note: S. Duraiswamy was the Designated Federal Employee for this portion of the meeting.]

C. P. Siess, Chairman of the Regulatory Activities Subcommittee, reported on its review of proposed Rev. 1 to Regulatory Guide 1.23, Meteorological Programs for Nuclear Power Plants (see Appendix XXXI).

The Committee discussed the recommendation of the Regulatory Activities Subcommittee to concur with proposed implementation of Regulatory Guide 1.23. It was unable to endorse the Guide and deferred action until after the Guide has been reviewed by the Generic Requirements Review Committee (V. Stello, Chairman).

XI. Report of the Procedures Subcommittee

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

The need for additional advisory committees such as radiological protection and radioactive waste management was discussed. Action was deferred regarding this matter. Because of the many project reviews in the near future, the Subcommittee expressed concern that safety issues of more general importance might be unnecessarily curtailed unless action is taken to make ACRS project reviews more efficient (see Appendix XXXII). In order to make subcommittee reviews more responsive to the interests/concerns of individual members, the Subcommittee recommended that Members suggest topics of concern for the Project Subcommittee Chairman prior to the particular project review. The Subcommittee Chairman would explore these particular topics during the Subcommittee review in addition to those items considered important by the Subcommittee Chairman/members. The list of unresolved items identified by the NRC Staff and/or intervenors should not be the only topics used as the basis for subcommittee meetings except for those items considered of major safety significance. In view of scheduling difficulties associated with this scheme, it was proposed that the ACRS Staff should develop a list of items of concern/interest to individual Members for use by Project Engineers/Subcommittee Chairmen in planning their meetings. M. W. Libarkin has been assigned the responsibility for working up an initial list of such items with particular emphasis on the Watts Bar Units 1 and 2 review.

XII. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. Human Factors

During review of the Callaway plant, several members expressed concern regarding the qualifications of Shift Technical Advisors. Follow-up by the Human Factors Subcommittee was suggested.

2. CRBR

M. W. Carbon discussed the planned subcommittee meeting to be held on December 15-16, 1981, and expressed concern that the absence of certain subcommittee members because of scheduling problems would jeopardize the value of the meeting. The Committee agreed with M. W. Carbon that postponement of the meeting was the proper course of action.

3. AC/DC Power Systems Reliability

Time did not permit the report of the AC/DC Power Systems Reliability Subcommittee at the 259th full Committee meeting as scheduled. Subcommittee Chairman J. J. Ray, however, has committed to submit to the Committee for distribution by R. Savio a written report which will include as an attachment the report on AC/DC system reliability that was prepared by C. Ryder, ACRS Fellow.

4. Systematic Evaluation Program (SEP) Reviews

SEP reviews are scheduled for full Committee review starting in March 1982 with evaluation of the Palisades plant and subsequent reviews (total of 10) at the rate of one per month. Detailed review of these projects is to be done by the appropriate Project Subcommittee. The Subcommittee Chairman should plan to have at least one member from the Reactor Operations Subcommittee present at this review for continuity. The Chairman of the Reactor Operations Subcommittee should attend if no other Reactor Operations Subcommittee member is available.

5. Generic Items

The Procedures Subcommittee recommended and the full Committee endorsed assignment of an overview responsibility in this area to the Generic Items Subcommittee. The Subcommittee will specifically:

- Conduct a preliminary review of proposed generic issues (e.g. those proposed by individual Members, etc.) and report to the ACRS regarding disposition of the matter (e.g. refer to NRC Staff, refer to ACRS topical subcommittee, take no further action, etc.).
- Provide oversight of generic matters including the review, etc., of those items it is competent to deal with and assigning others to ACRS topical subcommittees as appropriate for review, etc., in the same manner as the Regulatory Activities Subcommittee provides oversight regarding proposed rules and regulations.

The Generic Items Subcommittee was asked to prepare a priority list for existing generic issues (including the NRC Category A, B and C Task Action Plan items) to be used as guidance regarding the activities of ACRS topical subcommittees who are/or will be working on them.

6. Three Mile Island 2 Action Plans

W. M. Mathis, Subcommittee Chairman, explained that the Proposed NRC Rule (10 CFR Part 50) on Application of TMI-2 Lessons Learned to OLs was being rewritten by the NRC Staff for issuance about February 1982. The public comment period has concluded and the Staff is currently incorporating relevant comments into the rule. The TMI-2 Action Plans Subcommittee shall initiate ACRS review of the proposed rule when it is available in February.

B. ACRS Reports, Letters, and Memoranda

1. Report on the St. Lucie Plant No. 2

The Committee prepared a report to the Commissioners recommending, subject to satisfactory completion of construction, staffing, and preoperational testing, the granting of a license to operate the plant at full power. H. W. Lewis and M. S. Plesset appended additional comments expressing concern about the post-TMI Unit 2 requirement that applicants for an Operating License demonstrate specific capability to detect the onset of inadequate core cooling by installation of hastily conceived instrumentation.

2. Report on the Callaway Plant Unit No. 1

The Committee prepared a report to the Commissioners of its review of the full power operating license for Callaway Plant Unit No. 1. The recommendation is for full power operation subject to certain issues requiring final resolution mentioned in the letter. M. W. Carbon appended comments concerning his belief that the NRC Staff's requirement for experienced, on-shift personnel during the initial operation of the plant is inadequate.

3. Report on Comanche Peak Steam Electric Station

The Committee prepared a report to the Commissioners of its review of the full power operating license for the Commanche Peak Steam Electric Station Units 1 and 2. If due consideration is given to the recommendations made in the letter, the Committee believes that there is reasonable assurance that the facility can be operated safely at full power.

C. Generic Safety Items

1. Westinghouse Owners Group Guidelines for Emergency Operating Procedures

As a result of questions raised during the review of the Callaway Nuclear Plant regarding emergency operating procedures for different types of PWR's, Westinghouse Electric Company agreed to provide the Committee with copies of its September 1981 document (4-volumes) for the preparation of emergency operating procedures. D. Okrent will receive a copy directly because of his specific interest in this matter and other members will be notified. Copies will be supplied by the ACRS Staff to Committee members on a demand basis.

D. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 260th ACRS Meeting, December 10-12, 1981 (see Appendix II).

2. Future Subcommittee Activities

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

E. M. Bender Requests of TUGCO

M. Bender requested that D. Jones of TUGCO contact the management of the San Onofre Unit 1 about the availability of data on the failure of diesel generators to start. M. Bender asked H. Schmidt to provide a copy of Secondary Chemistry operating procedures when they are available. F. Madden, TUGCO, was asked to investigate and report to the Committee the sensitivity of the hydrogen control purge to radionuclide inventory in the containment. J. Ebersole questioned whether occupancy of the control room could be maintained if there were containment leakage as a result of an accident. M. Bender requested that TUGCO investigate this matter by considering a TMI-2 fission product inventory with worse containment leakage than occurred during that accident.

F. Review of the Zion Station PRA

D. Okrent has agreed to organize a group of twelve consultants to review the probabilistic risk assessment recently submitted to the NRC Staff. One objective of this study will be to compare consultant findings with the "quick look" analysis assigned by the NRC Staff to Brookhaven National Laboratory.

G. Action to Examine the TMI Unit 2 Core

D. W. Moeller proposed a memorandum to the Commissioners questioning the delay in NRC inspection of the TMI Unit 2 core. The Executive Director agreed to provide the full Committee a status and schedule for the TMI-2 core inspection.

H. SECY-81-605 "Proposed Changes to the NRC-NRB/MOST (Korean Nuclear Regulatory Bureau/Ministry of Science and Technology Information Exchange Arrangement."

D. W. Moeller expressed an interest in this request by the Koreans for NRC assistance during nuclear emergencies. The ACRS Executive Director agreed to provide him with background information regarding this matter. (M. C. Gaske has been assigned followup.)

I. Fast Reactor Conference Entitled "International Topical Meeting on LMFBR and Safety Related Design and Operational Aspects" in Lyon, France

M. W. Carbon expressed an interest in attending this conference. Members endorsed his attendance. The Executive Director mentioned that expenses for the Canadian visitors expected in early December will consume the existing ACRS Fund for Foreign Visitors. The ACRS does not expect to receive additional funds from NRC for the purpose of entertaining other foreign guests.

J. Distribution of Documents to Members

Members agreed to a reduction in the processing and distribution of Category B reports to ACRS Members, particularly those generated after the ACRS CP review has been completed and before the OL review has started (e.g., reports of construction deficiencies, etc.).

K. Format/Scope of ACRS Meetings with NRC Commissioners

The Procedures Subcommittee recommended that although items of significant concern to Committee members could properly be discussed in the course of ACRS meetings with the Commissioners, such occasions were not well suited for collegial Committee reports on such issues. M. Bender and several other Members endorsed a policy where important ACRS reports might be used as the basis for a collegial briefing of the Commissioners since all of the Commissioners do not perceive ACRS reports in the same way.

The 259th ACRS Meeting was adjourned at 2:20 p.m., Saturday, November 14, 1981.

APPENDIXES
TO
MINUTES OF THE 259TH ACRS MEETING
NOVEMBER 12-14, 1081

ACRS-1924

DESIGNATED ORIGINAL

Certified By ____

ATTENDEES
259TH ACRS MEETING
NOVEMBER 12-14, 1981

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Carson Mark, Chairman
Paul G. Shewmon, Vice-Chairman
Robert C. Axtmann
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
M. Norman Schwartz, Technical Secretary
Herman Alderman
William M. Baldewicz
Stuart K. Beal
William M. Bock
Paul A. Boehnert
Joseph Donoghue
Sam Duraiswamy
David C. Fischer
J. Michael Griesmeyer
Elpidio G. Igne
Morton W. Libarkin
John A. MacEvoy
Richard K. Major
Thomas G. McCreless
John C, McKinley
Thomas McKone
Austin Newsome
Gary R. Quittschreiber
Christopher Ryder
Richard P. Savio

NRC ATTENDEES

259TH ACRS MEETING

Thursday, November 12, 1981

NUCLEAR REACTOR REGULATION

O. P. Choppra
J. P. Joyce
R. W. Stevens
A. Thadani
J. N. Ridgely, DSI
H. Polk, SEB
M. Rubin
W. G. Kennedy
R. Pichumani
R. L. Tedesco
C. G. Tropf
V. Nerses
F. Miraglia
E. F. Goodwin
G. E. Edison
C. E. Rossi, ICSB
B. J. Youngblood, LB 1
R. L. Rothman, GSB
R. B. McMullen, GSB
S. J. Brocoum, GSB
A. K. Ibrahim, GSB
A. Brauner, SAB
R. Codell, HGEB
L. Soffer, SAB
D. R. Muller
H. Krug, AEB
T. Huang, CPB
J. Fairobent, AEB
G. Harrison, CEB
L. Heller, HGEB
D. I. Seris
R. J. Eckenrode
W. J. LeFave
J. B. Hopkins
H. B. Clayton
J. W. Clifford
R. E. Lipinski
R. K. Anand

INSPECTION AND ENFORCEMENT

David B. Matthews

REGION II

Harold E. Bibb
Stephen A. Elrod
H. C. Dance
D. J. Perrotti

REGION III

J. M. Perchel
K. R. Baker

NRC ATTENDEES

259TH ACRS MEETING

Friday, November 13, 1981

NUCLEAR REACTOR REGULATION

A. Thadani, DST
R. L. Tedesco, DOL
M. Thadani, AEB
T. R. Quay, AEB
L. P. Crocker, DHFS
W. S. Hazelton, MTEB
H. C. Garg, EQB
S. B. Burwell, LB 1
W. Russell
M. Ernst
S. Newberry
S. Block
E. Doolittle
S. Diab
R. W. Houston
B. J. Youngblood
J. W. Clifford
W. L. Brooks
H. C. Li
E. F. Goodwin

NUCLEAR MATERIAL AND SAFEGUARDS

D. A. Kers

APPLICANT ATTENDEES

259TH ACRS MEETING

Thursday, November 12, 1981

FLORIDA POWER AND LIGHT COMPANY

J. Sheetz
J. Franklin
R. Gritz
H. D. Johnson
J. H. Barrow
G. J. Boissy
J. W. Williams, Jr.
J. E. Vessely
E. W. Dotson
W. S. Windecker
D. M. Evans
T. C. Grozan
J. E. Sheetz
K. N. Harris
C. L. Fisher
W. B. Derrickson
F. H. Fabor
M. P. Armand
J. Velutter
P. Carier
F. Flugger
B. Pagnozzi
P. L. Fincher
R. R. Jennings
J. G.West
W. F. Brannen
H. F. Buchanan
J. D. Gustin

COMBUSTION ENGINEERING

C. B. Brinkman
D. Whitney
J. C. Moulton
S. E. Ritterbusch
E. H. Kennedy
J. Westhoven
W. Harris
W. Gardner
R. S. Tur
T. R. Tramm

EBASCO

K. P. Twine
M. P. Horrell
M. Brown
V. Oniunas
D. Levins
G. Attarian
E. Z. Zuchman
A. Salvi
G. Martin
M. P. Horrell
R. Sweeney
D. J. Chin

SHAW PITTMAN

J. Silberg

LAWRENCE LIVERMORE LAB

D. Chung

APPLICANT ATTENDEES

259TH ACRS MEETING

Thursday, Novembe. 12, 1981

WESTINGHOUSE ELECTRIC

M. A. Torcaso
W. C. Gagloff
F. X. Thomson
D. L. Cecchett
D. Call
D. Rawlins
J. Murphy
W. L. Luce
H. Julian
J. Mesmeringer
T. Timmons
J. W. Swagger
D. Papp
J. C. Ruh
C. Butterugith
C. R. Tuley
G. Lang
N. Liparulo
R. Mark

Missouri Public Service Comm.

A. S. Cauger
R. M. Fluegge

SNUPPS

F. Schwoerer
J. O. Cermak
N. A. Petrick
R. L. Stright
L. G. Schwoerer

UNION ELECTRIC

R. K. Cothren
A. C. Passnater
N. G. Slaten
D. E. Shafer
J. M. Kaelin
A. P. Neuhalfen
D. W. Capone
R.J. Schukai
M. E. Taylor
D. F. Schnell
M. A. Stiller
S. E. Miltenberger
J. F. McLaughlin
P. Appleby

BECHTEL POWER CORP.

P. Aulard
F. M. Roddy
H. F. Moate
J. H. Smith
C. R. Klee
J. S. Prebula
J. M. Small
K. Lee
D. Grove
Z. Vich
W. Heinmiller
D. C. Gasda
J. M. Small

APPLICANT ATTENDEES

259TH ACRS MEETING

Friday, November 13, 1981

TEXAS UTILITIES

R. R. Parks
D. Woodlan
C. L. Turner
J. Rumsey
R. Werner
C. Feist
H. Schmidt
B. R. Clements
M. R. Blevins
R. E. Kahler
S. Rilyea
D. W. Braswell
R. B. Seidel
R. A. R. A. Jones
J.B. George
J. S. Marshall
J. C. Kugkendill
J. D. Edwards
R. D. Calder
F. W. Madden
A. Vega
B. T. Lancaster
D. H. Wade
W. Stansell

J. Nelson, Quadrex

WESTINGHOUSE

W. C. Gangloff
C. R. Tuley, Jr.
D. Papp
F. Thomson
N. J. Lipanlo
R. Estes
D. Call
C. Buttersonith
H. Julian
J. Buderworth
R. J. Nath
E. Murphy
D. L. Cecchett
T. F. Timmons
J. C. Mesmeringer
D. L. Cecchett
M. A. Torcaso
S. G. Scaglia

GIBBS & HILL

S. Kumar
E. Horovitz
T. Vardaro
A. V. J. Burzi
G. Gisonda
G. Gisonda
E. J. Bond

A-6

PUBLIC ATTENDEES

259TH ACRS MEETING

Thursday, November 12, 1981

J. S. Marshall, Texas Utilities
E. Horovitz, Gibbs & Hill
T. Vardaro, Gibbs & Hill
D. Wade, Texas utilities
C. Feist, Texas Utilities
W. Leyse, Electric Power Research Inst.
D. Green, Kansas Gas & Electric
J. Zudans, NUS
J. R. Provasol, Arizona Public Services
G. P. Rathbun, Kansas Gas & Electric
B. S. Newnan, Cox Newspapers
S. Kumar, Gibbs & Hill
S. Filipour, ARC
R. Terrill, Kansas Gas & Electric Company
T. Tipton, Atomic Industrial Forum
M. A. Bauser, Lowenstein, Newman
J. D. Edwards, Texas Utilities
B. J. Lancaster, Texas Utilities
R. Petrick, WCIX-TV
J. L. Marshal, WCIX-TV
G. Koester, Kansas Gas & Electric Company
H. Gaut, FEMA
L. Liecave, UCLA
F. T. Rhodes, Kansas Gas & Electric

PUBLIC ATTENDEES

259TH ACRS MEETING

Friday, November 13, 1981

T. Tramm, Commonwealth Edison
T. Tipton, Atomic Industrial Forum
W. R. Schmidt, MPR Associates
M. P. Horrell, Ebasco
T. Tipton, Atomic Industrial Forum
R. Leyse, Electric Power Research Inst.
M. D. Patterson, Baltimore Gas & Electric
D. Knuth, KMC
H. Lardner, Morgan Associates
R. A. Vincent, CPCo
R. W. Ganthner, Babcock and Wilcox

APPENDIX II

FUTURE AGENDA

<u>DECEMBER</u>

CESSAR-System 80--final design NSSS (MB/SKB)	2 hrs
Palo Verde Nuclear Plant Units 1, 2, and 3--OL (MB/SKB)	4 hrs
LER Reporting System--ACRS comments regarding proposed NRC Rule 10 CFR 50.72) regarding changes in the LER reporting system	1 hr
Proposed NRC procedure to assign priorities for dealing with unresolved safety issues (MB/RS)	
Public Law 96-567, Nuclear Safety Research, Development, and Demonstration Act of 1980 (WMM/DWM/CPS/RKM/SD/DCF)	

Meeting with NRC Commissioners

- Discuss anticipated changes in the role of the NRC Chairman concerning the RES program (CPS/SD)
- Discuss the Commissioner's response to the ACRS proposed changes in the scheduling and scope of ACRS annual reports on the NRC Safety Research Program (CPS/SD)
- Discuss status of NRC Staff action to evaluate requirements for supporting infrastructure at nuclear plants (DAW/RKM)
- M. Bender shall discuss the Committee's desire to condense project review documentation in such as PSAR's and FSAR's

ACRS comments regarding Task Action Plan A-45, <u>Evaluation of Alternate Decay Heat Removal Systems (DAW/RS)</u>	2 hrs 1/2 hr Sat.

Briefing by NRC Staff of CRBR Staff organization and plans for review (MWC/EI) — Defer

Subcommittee Reports

Subcommittee on Regulatory Activities regarding proposed changes in Regulatory Guides, etc. — 1/4 hr

Report by ACRS Electrical Systems Subcommittee regarding NRC Staff review and reevaluation of requirements for instrumentation to detect inadequate core cooling (WK/RS) — Defer - addressed in St. Lucie letter, Nov. 17, 1981

Future ACRS Activities

Briefing by NRC Staff concerning errors in the seismic design of the Diablo Canyon design reviews (JCM)

Briefing by the NRC Staff to report to the Committee the results of cable performance tests conducted on St. Lucie Unit 1 which were ~~sighted~~ cited in the CP review for St. Lucie Unit 2. This item refers back to a paragraph in a December 12, 1974 letter on St. Lucie Unit 1 in which concern was expressed by the Committee on flooding of dry electrical cables and aging effects on these cables. J. J. Ray will report on experience of the IEEE in this matter

Proposed changes in 10 CFR 50, Appendix K regarding BWR ECCS Evaluation Models (January-February)

12/15/81

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

REVISED

DECEMBER

16 (1:00 pm) & 17 (8:30 am - 1:00 pm)	Class 9 Accidents (Denver, CO) (Beal/Quittschreiber) - Kerr, Bender, Etherington, Okrent, Shewmon, Siess, Ward. Purpose: Review Zion risk study and hydrogen research and rulemaking.
18 & 19	Joint Waste Management & Reactor Radiological Effects (Alderman) - Moeller, Ray, Axtmann. Purpose: To review research program and budget.

JANUARY

5	Human Factors (Major) Ward, Mathis (part-time), Ray. Purpose: To review in more detail and provide comments to the Commissioners on NUREG-0700, NUREG-0801, and NUREG-0835.
5 (p.m.)	Reliability & Probabilistic Assessment (Griesmeyer/Quittschrieber) - Kerr, Bender, Ebersole, Siess, Okrent. Purpose: To review portions of the FY 1983 RES Budget related to reliability and probabilistic assessment.
6	Nuclear Safety Research Program (Duraiswamy)- Siess, Carbon, Kerr, Mark, Okrent, Mathis, Ward. Purpose: To discuss the Draft ACRS Report to Congress on NRC's FY 1983 Safety Research Program.
6 (3:00 pm - 6:00 pm)	Advanced Reactors (Igne) - Carbon, Bender, Mark, Plesset, Lewis, Kerr. Purpose: To review advanced reactor research budget and programs.
7 - 9	261st ACRS Meeting
21 & 22	Advanced Reactors (Argonne, IL) (Igne) - Carbon, Mark, Shewmon, Bender, Plesset*, Kerr*. Purpose: To continue discussion concerning LMFBR safety philosophy and issues and to prepare a report to submit to the ACRS.
22	Fluid Dynamics (Los Angeles) (Boehnert) - Plesset*, Kerr*, Ebersole, Etherington, Mathis. Purpose: To continue review of Mark III Containment modifications and discuss status of USI's on Mark I and II Containments.
23	Joint Electrical Systems and ECCS (Los Angeles) (Savio/Boehnert) - Kerr, Ebersole, Mark, Mathis, Okrent, Plesset, Ray, Etherington. Purpose: To continue review of the NRC- and Industry-sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.

* Note conflict to be resolved.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JANUARY (CONT'D)

28 & 29	Extreme External Phenomena (Savio) - Okrent, Bender, Etherington, Mark, Moeller, Siess. Purpose: To review status of NRC's research program on geology and seismology and the status of research being performed outside of the NRC programs.

FEBRUARY

2 (p.m.) & 3	CRBR (Igne) - Carbon, Bender, Mark, Okrent Siess. Purpose: To review CRBR program status.
3 (8:45 am)	Regulatory Activities (Duraiswamy) Siess, Kerr, Carbon, Ray, Ward. Purpose: To discuss Regulatory Guides and Regulations.
4-6	262nd ACRS Meeting
9 (p.m.)	Simulator Tour (Silver Spring, MD) (Major) - Kerr, Ward, Mathis. Purpose: Visit Singer-Link Corporation.
10	Qualification Program for Safety Related Equipment (Boehnert) - Ray, Ebersole, Kerr (tent.). Purpose: To review the NRC Equipment Qualification Program Plan as outlined in SECY-81-504.
11	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Shewmon, Axtmann, Ray. Purpose: To discuss occupational radiation exposure in BWRs.
12	Joint Metal Components and Waste Management (Igne/Alderman) - Shewmon, Ray, Axtmann. Purpose: To review contractor technical capability and objectives of request for proposal on long-term performance of materials used for high-level waste packaging.
mid-Feb.	Safety Philosophy Technology and Criteria (Griesmeyer/Savio) - Okrent, Bender, Ebersole, Kerr, Mathis, Ray, Ward. Purpose: To review the proposed Systems Interaction Study for the Indian Point Nuclear Power Plant.
mid-Feb.	Watts Bar (Knoxville, TN) (Griesmeyer/Quittschreiber) - Bender, Ebersole, Ward. Purpose: To review application for an operating license.
Late Feb.	Waterford (Beal/Quittschreiber) - Ward, Bender, Carbon, Siess. Purpose: To review Waterford organization, staffing, and training programs.
Late Feb.	Clinton (Decatur, IL tent.) (Savio) - Bender, Axtmann, Kerr, Moeller. Purpose: To review application for OL.

REVISED 12/15/81

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

FEBRUARY

Late Feb.	Zimmer Plant (Boehnert) - Bender, Kerr, Plesset, Shewmon. Purpose: To review QA problems associated with plant construction which resulted in $200,000 fine by NRC/I&E.
Late Feb. early March	Byron Station 1 & 2 (Byron, IL) (Igne) Shewmon, Bender, Mark. Purpose: To review application for OL.

MARCH

3	Babcock & Wilcox (Major) - Ray, Ebersole, Etherington, Okrent, Plesset. Purpose: To explore with B&W changes that have been made to the ICS since the TMI-2, Crystal River 3, and Rancho Seco transients. Other improvements to the plant and plant operations will also be explored.
4-6	263rd ACRS Meeting
March	Joint CRBR & Site Suitability (Igne/Alderman) - Carbon, Moeller, Purpose: To begin site suitability review for CRBR.
Date to Be Determined	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Bender, Kerr, Siess, Mark. Purpose: To review draft Commission Policy Statement on Safety Goals.
Date to Be Determined	Shippingport (Boehnert) - Bender, Carbon, Siess (tent). Purpose: Consider review of extension of LWBR operation from 26,000 EFPH to 30,000 EFPH.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Dec 16 (1:00 to close of business) Dec 17 (8:30 - 1:00)	Class 9 Accidents	(SAVIO) Kerr, Bender, Etherington, Okrent, Shewmon, Siess, Ward. Consultants: T. Theofanous, P. Davis, Z. Zudans

LOCATION: Denver, CO

BACKGROUND:

Purpose:
1. Review the status of the degraded core rulemaking and the NRC policy on the approach to degraded core rulemaking.
2. Review mechanistic aspects of Zion risk assessment.
3. Review latest developments in hydrogen research and rulemaking.

PERTINENT PUBLICATIONS:

1. Zion risk study (portion dealing with core melt assumptions).
2. NRC study on core melt mitigation features (NUREG-0850).
3. Proposed rules and hydrogen.
4. Documents dealing with degraded core rulemaking.
5. Review degraded core research budget in preparation of report to Congress.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
DEC. 18 & 19	Waste Management and Reactor Radiological Effects	(ALDERMAN) Moeller, Ray, Axtmann Cons: H.Parker, Orth, Steindler, Foster F. Parker

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To review research program and budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN. 5	Human Factors	(MAJOR) Ward, Mathis (part-time), Ray Cons: Buck, Debons, Keyserling, Pearson

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: To review in more detail and provide comments to the Commissioners on:

NUREG-0700, "Guidelines for Control Room Design Review."

NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Review."

NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System."

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The above documents have been distributed and are available.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Jan. 5 (P.M.)	Reliability & Probabilistic Assessment	(GRIESMEYER/QUITTSCHREIBE Kerr, Bender, Ebersole, Siess, Okrent

LOCATION: Washington, DC

BACKGROUND:

Purpose: To review portions of the FY-83 Research Budget related to Reliability and Probabilistic Assessment.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
January 6	Nuclear Safety Research Program	(DURAISWAMY) Siess, Carbon, Kerr. Mark, Okrent, Mathis. Ward

LOCATION: Washington, DC

BACKGROUND:

Purpose: To discuss the Draft ACRS Report to the Congress on NRC's FY-1983 Safety Research Program.

PERTINENT PUBLICATIONS:

Draft 2 of the ACRS Report to Congress on the NRC FY 1983 Safety Research Program (NUREG-0864).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN. 6 (3:00 - 6:00 p.m.)	Advanced Reactors	(IGNE) Carbon, Bender, Mark Plesset, Kerr, Lewis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To review Advanced Reactor research budget and programs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-19

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN 21 & 22	Advanced Reactors	(IGNE) Carbon, Mark, Shewmon, Bender, Plesset, Kerr

LOCATION: Argonne, IL

BACKGROUND:

Who proposed action: Subcommittee

Purpose: To continue discussion concerning LMFBR safety philosophy and issues and to prepare a report to submit to the ACRS.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-20

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Los Angeles, CA

BACKGROUND:

Who proposed action: M. Plesset/NRC Staff

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided at a later date.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JAN. 23	Combined ECCS/Electrical Systems Subcommittee	(SAVIO/BOEHNERT) Kerr, Ebersole, Mark, Mathis, Plesset, Okrent, Ray, Etherington

LOCATION: Los Angeles, CA

BACKGROUND:

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

PERTINENT PUBLICATIONS:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

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

PERTINENT PUBLICATIONS:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To review CRBR program status.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A-24

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Feb 3 (8:45 am)	Regulatory Activities	(DURAISWAMY) Siess, Kerr, Carbon, Ray, Ward

LOCATION: Washington, DC

BACKGROUND:

Purpose: To discuss Regulatory Guides and Regulations.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
FEB. 9 (p.m.)	Simulator Tour	(MAJOR) Kerr, Ward, Mathis

LOCATION: Singer-Link Corporation, Silver Spring, MD

BACKGROUND:

Who proposed action: W. Kerr

Purpose: To visit Singer-Link Corporation.

ADDITIONAL DETAILS:

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: J. Ray

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller/P. Shewmon

Purpose: To discuss occupational radiation exposure in BWRs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. P. Shewmon memo to D. Moeller

2. SEC-81-517

A-38

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
mid-FEB.	Safety Philosophy Technology and Criteria	(GRIESMEYER/SAVIO) Okrent, Bender, Ebersole, Kerr, Mathis, Ray, Ward

LOCATION: Washington, DC

BACKGROUND:

Proposed by: NRR and Power Authority of the State of New York

Purpose: To review the proposed systems Interaction Study for the Indian Point Nuclear Power Plant

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposal for the Study (yet to be received)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Commission

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Knoxville, TN

PROPOSED BY: NRR

BACKGROUND:

Purpose: To review the Watts Bar for an OL.

PERTINENT PUBLICATIONS:

Watts Bar SER and Supplement (not yet published)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Late FEB.	Waterford	(Beal/Quittschreiber) Ward, Bender, Carbon, Ray, Siess Cons: Pearson, Binford

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER Supplement scheduled to be issued in Janaury 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Late FEB.	Clinton	(SAVIO) Bender*, Axtmann, Kerr, Moeller

LOCATION: Decatur, IL (tent.)
Site Visit at the Clinton site with a Subcommittee meeting near the site.

BACKGROUND:

Who proposed action:

Purpose: To review application for OL.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Safety Evaluation Report expected to be available by January 25, 1982.

* To be resolved

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Late FEB.	Zimmer Plant	(BOEHNERT) Bender, Kerr, Plesset, Shewmon

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Bender/ACRS

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
End of Feb. or Early March	Byron Station 1 & 2	(IGNE) Shewmon, Bender, Mark Cons: Kassner

LOCATION: Site visit at Byron. Subcommittee meeting nearby.

BACKGROUND:

Who proposed action: NRC Staff & P. Shewmon

Purpose: OL review.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Safety Evaluation Report due 2/07/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: J. Ray

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MARCH	Joint CRBR and Site Suitability	(IGNE/ALDERMAN) Carbon, Moeller Cons:

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To begin site suitability review for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Office of Policy Evaluation

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
TO BE DETERMINED	Shippingport	(BOEHNERT) Bender, Carbon, Siess (tent)

LOCATION: Washington, D.C.

BACKGROUND:

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

Purpose: To consider review of extension of LWBR operation from 26,000 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 AND THEIR AVAILABILITY:

See Above.

LIMITED APPEARANCE STATEMENT

BY

JOETTE LORION

CENTER FOR NUCLEAR RESPONSIBILITY

Prepared for presentation to
the Advisory Committee on Reactor
Safeguards, U.S. Nuclear Regulatory
Commission, In the Matter of
Florida Power and Light Co. St. Lucie
Nuclear Power station, Unit 2
Docket No. 50-389

INTRODUCTION

My name is Joette Lorion. I am research director for The Center for Nuclear Responsibility (CNR) and my business address is 7210 Red Road #208, Miami, Florida 33143. The purpose of my appearance today on behalf of CNR is to ask that the NRC staff and the ACRS not bow to industry pressure to increase the pace of the licensing process, but rather fulfill its mandate to protect the public health and safety by aggressive enforcement of NRC safety regulations and licensing criteria.

It is my understanding that Florida Power and Light Company is asking that St. Lucie Unit #2 be issued an operating license a year before it was scheduled. The reasons for the speed up that have been quoted by FP&L are purely economic. The NRC staff seems to be in agreement with FP&L's plans, despite the fact that the Safety Evaluation Report for St. Lucy #2 contains numerous unresolved safety issues, and numerous incomplete applicant reports and programs. I have come here today on the behalf of CNR in a final attempt to ask the ACRS, an independent advisory body, to address all problem areas that require additional scrutinty before granting Florida Power and Light permission to operate St. Lucie Unit #2. Since you will be acting in an advisory capacity to the Atomic Safety and Licensing Board, it is important that you demand all pertinent facts from both FP&L and the NRC Staff that deal with the extent to which FP&L does not comply with Commission regulations, the aspects of designed that have not been reviewed, and the numerous safety issues that have not been resolved.

DISCUSSION OF ISSUES

Basically, I am asking that the ACRS take the initiative and compel the NRC staff to force FP&L to comply with Commission regulations before St. Lucie #2 is licensed to operate. There are numerous instances in the SAR in which the NRC staff states that FP&L's plans or programs do not comply, but that FP&L will comply after the first refueling outage. Does this not mean that St. Lucie #2 will go into operation without Commission regulations having been met ? How does the NRC justify this "comply later" attitude and how can they assure that the public health and safety will not be adversely affected ?

It is easy to understand why Florida Power & Light is willing to take short cuts to streamline the licensing process. Plagued by nuclear steam generator problems at their Turkey Point Units, FP&L is being forced to repair these units. Since this process involves expensive and lengthy nuclear downtime; FP&L is in need of both the revenue and electricity that St. Lucie 2 will provide. However, and I think you will agree with me, the remedy for Florida Power and Light's problems does not lie in shifting the burden to the public in the form of grave risks to public health and safety. The remedy is to correct known deficiencies before the plant is licensed and put into operation. The cost in time or money that may result to Florida Power and Light for implementing these safety corrections should not be a factor in your decision, since this risk was knowingly assumed by the applicant when they decided to build the plant. Why should you accept weak arguments from a beleaguered utility that are designed to gain premature approval of the design and safety features and specifications of St.Lucie Unit 2 ?

I would also like to emphasize to the ACRS that the Atomic Energy Act quite plainly makes compliance with the Commissions regulations a condition of entitlement to licensing. In memorandum and Order (ALAB-138) in the case of Vermont Yankee Nuclear Power Station, the Atomic Safety and Licensing Appeal Board stated,

> "...neither the applicant nor the staff should be permitted to challange applicable regulations, either directly or indirectly. Thus, those parties should not generally be permitted to seek or justify the licensing of a reactor which does not comply with applicable standards. Nor can they avoid compliance by arguing that, although an applicable regulation is not met, the public health and safety will be protected. For once a regulation is adopted, the standards it embodies represent the Commission definition of what is required to protect public health and safety."

Thus, the ACRS should analyze the NRC staffs decision in the SER that with certain "exemptions" the St. Lucie facility meets commission regulations, and ask if these exemptions are necessary or should be allowed . And I would also ask you, what is so unique about St. Lucie Unit #2 that it demands this extraordinary move on the Commissions part of licensing this unit a year ahead of schedule, while allowing certain safety conditions to be met during the refueling operation one year after start up? Is there some unique design feature that makes this St. Lucie #2 unit more safe than FP&L 's other nuclear plants ? Or is it similar, perhaps, to the Three Mile Island Unit that was rushed on line prematurely to gain a tax advantage for GPU, and one year later had a major accident because safety problems had not been resolved?

It seems to me that the residents of Florida would rather have St. Lucie #2 follow its original licensing schedule if that meant this nuclear unit would operate more safely. The fact that St. Lucie #2 is located on a barrier island with such a large population and no well defined evacuation plan would dictate that safety problems should come under greater scrutinty; not lesser. The fact that under new Commission regulations on power plant siting, St. Lucie would have a very difficult time even being built, requires that the NRC staff, ACRS, and ASLB not compromise on any NRC regulations that could have a direct threa: on public health and safety. A minor site specific problem, such as FP&L's failure to provide stability against liquification of the insitu soils beneath the fill on the slopes North and South of the service water pump intakes, (2.5.5.2.) would be a problem of major importance in a hurricane. Why not resolve it ? Another major problem that should be regarded as site specific in the SAR. is the fact that the FP&L system is extremely vulnerable to electrical blackouts. Infact, according to a Miami Herald article of May 7, 1980, Florida's outage record is the worst in the nation. There have been documented cases over the years of Turkey Point's nuclear units being shut down automatically because of a power failure. (see attatched news articles).

I would like to point out an accident it seems could very possibly happen at St. Lucie #2. This accident is described in NUREG-0651 Evaluation of Steam Generator Tube Rupture Events. Since St. Lucie #2 could have problems with steam generator corrosion and loss of offsite power, it may be important to briefly describe this accident.

The accident involves a large primary to secondary system leak that was caused by a steam generator tube rupture. It took 61 minutes before the operator equalized the primary and secondary pressure and the leak was stopped. The time the primary coolant was leaking through the break was twice as long as described in the FSAR and the amount was 33% more than the FSAR. The radioactive

release to the public did not exceed 10 CFR part 20, but a calculation performed as to the same exact event coupled with offsite power that showed that radiation released would be increased by a factor of 1000. The staff also recommends in this report that future reviews of steam generator tube rupture accident analysis should require a more detailed description of system performance during the event. We wonder if FP&L and Combustion engineering, designers of the steam supply system, have considered double mode accidents involving steam generators failure using reports such as NUREG 0651. And has Combustion Engineering developed capabilities for reevaluation of the Emergency Core Cooling System performance for postulated accidents concurrent with steam generator failure? Has Combustion Engineering complied with the suggestions offered in NUREG o523 Summary of Operating Experience with Recirculating Steam Generators, to provide an analysis of structural integrity for degraded tubes under normal operation and accident conditions involving a main steam line break, loss of coolant accident ? It seems that the fact that St. Lucie Unit #1 is already suffering from SG corrosion would point to the probability that St. Lucie #2 could also suffer from this unresolved safety problem and that in combination with LOOP a major nuclear accident is a possibility that must be considered.

In closing, gentlemen, we would again ask you to review St. Lucie's unresolved safety problems carefully and should you find major questions as to whether or not the licensing of this plant at the present time constitutes a potential threat to the health and safety of the people of this area, please allow sufficient time for these questions to be answered before putting this plant into operation. I am certain that you take your responsibility seriously, I only ask that you take our concerns about the premature licensing of St. Lucie #2 seriously as well. We ask you as scientists, to weigh the scientific evidence and should important questions indicate that the operational license for Florida Power & Light's St. Lucie Unit #2 be either delayed or denied; that you advise the Atomic Safety and Licensing Board not to grant this license until Commission regulation have been met and there is evidence that the public health and safety will be protected.

October 28, 1981

Chairman
Nuclear Regulator Subcommittee
on Safety
Meeting at Holiday Inn,
Century Village
West Palm Beach, Florida

Dear Sir:

As one who has taken an active interest in the St. Lucie #1 and St. Lucie #2 nuclear power plants ever since the first public hearing in May, 1970, I want to express my gratitude for your participation in this hearing on safety features of St. Lucie #2.

I would like to make a generic suggestion, however, in the interest of strengthening safety of nuclear plants: make efforts to obtain more responsible public imput to all official NRC hearings. Two specific suggestions are: site such hearings in the city nearest the power plants; notify by mail organizations or individuals who are known to the agency to have an actively expressed interest in the plants. A review of the history of St. Lucie #1 & #2 should show numerous changes made to the plants which resulted directly from public comments or questions. A number of other changes made later at the direction of AEC or NRC because of problems which occurred at St. Lucie #1 or other sites, might have been made earlier and less expensively if public questions had been taken more seriously and followed up on.

Many local people have through the years become so convinced that NRC hearings are "exercises" that they will no longer attend or take part in any way.

I am not one of them, to the extent that I wish to submit one question at this time, realizing from the just noticed Palm Beach Post news item about the hearing that I should submit 15 copies but hoping for your indulgence of my typewriter's limitations and lack of time to have professionally done. Incidentally, the local paper, Fort Pierce News Tribune, has not carried notice of the hearing.

> Have the safety issues raised in Safety Evaluation of the St. Lucie Plant #1, Docket 50-335, published November 8, 1974, and the Safety Evaluation of St. Lucie Plant #2, published November 7, 1974, been answered or resolved to your satisfaction?

Sincerely,

Betty Lou Wells
1124 Jasmine Avenue
Ft. Pierce, Florida 33450

The following pages has been deleted as:

DELETION 7

Pages: A-47 to A-48

TABLE 1.3-1

PLANT PARAMETER COMPARISON

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Hydraulic and Thermal Design Parameters					
Rated core heat output, MWt	2,560	4.4	3,390	2,815	2,560
Rated core heat output, Btu/hr	8,137 x 10^6	4.4	11,570 x 10^6	9,608 x 10^6	8,373 x 10^6
Heat generated in fuel, %	97.5	4.4	97.5	97.4	97.5
System pressure, nominal, psia	2,250	4.4	2,250	2,250	2,250
System pressure, minimum steady state, psia	2,200	4.4	2,200	2,200	2,200
Hot channel factors,					
Heat flux, F_q	2.57		2.35	2.35	2.85
DNB ratio at nominal conditions	2.64 (CE-1)	4.4	2.07 (CE-1)	2.26 (W-3)	2.30 (W-3)
Coolant flow					
Total flowrate, lb/hr	139.4 x 10^6	4.4	148 x 10^6	120.4 x 10^6	122 x 10^6
Effective flowrate for heat transfer, lb/hr	134.3 x 10^6	4.4	142.8 x 10^6	116.2 x 10^6	117.5 x 10^6
Effective flow area for heat transfer, ft^2	54.7	4.4	54.7	44.6	53.5
Average velocity along fuel rods, ft/sec	15.1	4.4	16.3	16.4	13.6
Average mass velocity, lb/hr-ft^2	2.45 x 10^6	4.4	2.61 x 10^6	2.60 x 10^6	2.20 X 10^6
Coolant temperatures, F					
Nominal inlet	548	4.4	553	553.5	538.9
Design inlet	550	4.4	556	556.5	544
Average rise in vessel	48	4.4	58	58.5	5[illegible]
Average rise in core	50	4.4	60	60.5	56
Average in core	573	4.4	586	583.75	572
Average in vessel	572	4.4	582	582.75	571.5
Nominal outlet of hot channel	622	4.4	642	652	640

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Hydraulic and Thermal Design Parameters (Cont'd)					
Heat transfer at 100% power					
Active heat transfer surface area, ft^2	56,315	4.4	62,000	51,000	48,400
Average heat flux, Btu/hr-ft^2	151,300	4.4	182,400	185,000	176,000
Maximum heat flux, Btu/hr-ft^2	388,800	4.4	428,000	433,800	501,300
Average thermal output, KW/ft (Fuel Rod Only)	4.43	4.4	5.34	5.41	5.94
Maximum thermal output, KW/ft (Fuel Rod Only)	11.4	4.4	12.5	12.7	17
Maximum clad surface temperature at nominal pressure, F	657.0	4.4	657.0	657	657
Fuel center temperature, F maximum at 100% power	2,986	4.4	3,180	3,420	3,890
Core Mechanical Design Parameters					
Fuel assemblies					
Design	CEA	4.2	CEA	CEA	CEA
Rod pitch, in.	0.506	4.2	0.5063	0.5063	0.58
Cross-section dimensions, in.	7.972 x 7.972	4.2	7.972 x 7.972	7.97 x 7.97	7.98 x 7.98
Fuel weight (as UO_2), lb_m	204.4×10^3	4.2	223.9×10^3	183,834	207,200
Total weight, lb_m	282.8×10^3	4.2	314,867	250,208	271,280
Number of grids per assembly	10	4.2	11	12	8
Fuel rods					
Number	49,580	4.2	49,580	40,644	36,896
Outside diameter, in.	0.382	4.2	0.382	0.382	0.44
Diametral gap, in.	0.007	4.2	0.007	0.007	0.0085
Clad thickness, in.	0.025	4.2	0.025	0.025	0.026
Clad material	Zircaloy-4	4.2	Zircaloy-4	Zircaloy	Zircaloy

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Core Mechanical Design Parameters (Cont'd)					
Fuel pellets					
Material	UO_2 sintered	4.2	UO_2 sintered	UO_2 sintered	UO_2 sintered
Diameter, in.	0.325	4.2	0.325	0.325	0.3795
Length, in.	0.390	4.2	0.390	0.390	0.650
Control assemblies					
Neutron absorber	(See Table 4.2-1)	4.2	(See Table 4.2-1)	B_4C/Ag-In-Cd	B_4C/SS
Cladding material	Inconel 625	4.2	Inconel 625	NiCrFe alloy	NiCrFe alloy
Clad thickness	0.035	4.2	0.035	0.035	0.040
Number of assembly, full/part-length	83/8	4.2	83/8	73/8	73/8
Number of rods per assembly	4,5/5	4.2	4,5/5	5	5
Nuclear Design Data					
Structural characteristics					
Core diameter, in. (equivalent)	136	4.2	136	123	136
Core height, in. (active fuel)	136.7	4.2	150	150	136.7
H_2O/U Unit Cell (cold)	1.72	4.2	3.35		1.63
Number of fuel assemblies	217	4.2	217	177	217
UO_2 Rods per assembly, unshimmed/shimmed					
Batch A	236	4.3	236	236	176
Batch B	236/220	4.3	236/220	224	164
Batch C	236/224 or 220	4.3	236/224 or 220	224/234/233	176/164/164
Performance characteristics loading technique	3-batch mixed central zone	4.3	3-batch mixed central zone	3-batch mixed central zone	3-batch mixed central zone
Fuel discharge burnup, MWD/MTU					
Average first cycle	13,187	4.3	12,731	12,500	12,800

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Nuclear Design Data (Cont'd)					
Feed enrichment, wt%					
Region 1	1.79	4.3	1.87	1.93	1.93
Region 2	2.36	4.3	2.38	2.27	2.33
Region 3	2.80	4.3	2.88	2.94	2.82
Control characteristics effective multiplication (beginning of life)					
Cold, no power, clean	1.170	4.3	1.170	1.195	1.170
Hot, no power, clean	1.119	4.3	1.125	1.139	1.134
Hot, full power, Xe equilibrium	1.070	4.3	1.067	1.082	1.078
Control Assemblies					
Total rod worth (hot), %	11.16 (EOC)	4.3	11.35	12.3	11.0
Boron concentrations for criticality:					
Zero power no rods inserted, clean, ppm Old/Hot	901/809	4.3	899/832	1011/1001	945/935
At power with no rods inserted, clean/equilibrium xenon, ppm	715/457	4.3	719/452	881/611	820/590
Kinetic characteristics, range over life					
Moderator temperature coefficient, $\Delta\rho/F$	See Table 4.3-4	4.3	See Table 4.3-4	-0.3×10^{-4} to -2.5×10^{-4}	-0.4×10^{-4} to -2.1×10^{-4}
Moderator pressure Coefficient, $\Delta\rho$/psi	$+0.6 \times 10^{-6}$	4.3	$+0.7 \times 10^{-6}$	$+0.06 \times 10^{-6}$ to $+2.6 \times 10^{-6}$	$+0.49 \times 10^{-6}$ to $+2.55 \times 10^{-6}$
Moderator void coefficient, $\Delta\rho$/% Void	-0.22×10^{-3}	4.3	-0.36×10^{-3}	-0.03×10^{-3} to -1.22×10^{-3}	-0.26×10^{-3} to -1.35×10^{-3}
Doppler coefficient, $\Delta\rho/F$	See Figure 4.3-34	4.3	1.18 x 10 to 1.28×10^{-5}	-1.18×10^{-5} to -1.78×10^{-5}	-1.45×10^{-5} to -1.07×10^{-5}

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Coolant System					
Operating pressure, psig	2,235	5.1	2,235	2,235	2,235
Reactor inlet temperature, F	548	5.1	553	553.5	539.7
Reactor outlet temperature, F	597.5	5.1	611.2	612.5	595.7
Number of loops	2	5.1	2	2	2
Design pressure, psig	2,485	5.1	2,485	2,485	2,485
Design Temperature, F	650	5.1	650	650	650
Hydrostatic test pressure (cold), psig	3,110	5.1	3,110	3,110	3,110
Principal Design Parameters of the Reactor Vessel					
Material	See Table 5.2-3	5.2	See Table 5.2-2	SA-533, Grade B, Class I, low alloy steel, internally clad with Type 304 austenitic SS	SA-533, Grade B, Class I, low alloy steel, internally clad with Type 304 austenitic SS
Design pressure, psig	2,485	4.4	2,485	2,485	2,485
Design temperature, F	650	4.4	650	650	650
Operating pressure, psig	2,235	4.4	2,235	2,235	2,235
Inside diameter of shell, in.	172	5.3	172	157	172
Outside diameter across nozzles, in.	253	5.3	253	238	253
Overall height of vessel and enclosure head, ft-in. to top of CEDM nozzle	41-10-3/8	5.3	43-6-1/2	43-4-1/6	41-11-3/4
Minimum clad thickness, in.	1/8	5.3	1/8	1/8	5/16
Principal Design Parameters of the Steam Generators					
Number of Units	2	5.4	2	2	2

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Steam Generators (Cont'd)					
Type	Vertical U-tube with integral moisture separator	5.4	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator
Tube material	NiCrFe alloy	5.2	NiCrFe alloy	NiCrFe alloy	NiCrFe alloy
Shell material	SA-533 GR A&B, Class 1 and SA 516, Gr. 70	5.2	SA-533 Gr. B Class 1 and SA-516, Gr.70	SA-533 Gr. B Class 1 and SA-516, Gr. 70	SA-533 Gr. B Class 1 and SA-516, Gr. 70
Tube side design pressure, psig	2,485	5.4	2,485	2,485	2,485
Tube side design temperature, F	650	5.4	650	650	650
Tube side design flow, lb/hr	61×10^6	5.4	74×10^6	60.2×10^6	61×10^6
Shell side design pressure, psia	1,000	5.4	1,100	1,100	1,000
Shell side design temperature, F	550	5.4	560	560	550
Operating pressure, tube side, nominal, psig	2,235	5.4	2,235	2,235	2,235
Operating pressure, shell side, maximum, psig	885		985	985	885
Maximum moisture at outlet at full load, %	0.2	5.4	0.2	0.2	0.2
Hydrostatic test pressure, tube side (cold) psig	3,110		3,110	3,110	3,110
Steam pressure, at full power, psia	815	5.4	900	900	815
Steam temperature, at full power, F	520.3	5.4	532	531.95	520.3
Principal Design Parameters of the Reactor Coolant Pumps					
Number of units	4	5.4	4	4	4
Type	Vertical, single stage centrifugal with bottom suction and horizontal discharge		Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design pressure, psig	2,485	5.4	2,485	2,485	2,485

TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Principal Design Parameters of the Reactor Coolant Pumps (Cont'd)					
Design temperature, F	650	5.4	650	650	650
Operating pressure, nominal psig	2,235	5.4	2,235	2,235	2,235
Suction temperature, F	540	5.4	553	553.5	540
Design capacity, gal/min	81,200	5.4	[illegible],000	80,000	8[illegible],000
Design head, ft	310	5.4	310	275	25[illegible]
Hydrostatic test pressure (cold), psig	3,1[illegible]		3,[illegible]	3,110	3,110
Motor type	AC induction, single speed		AC induction, single speed	AC induction, single speed	AC induction, single speed
Motor rating, hp	6,500		9,700	[illegible]	6,500
Principal Design Parameters of the Reactor Coolant Piping					
Material	See Table 5.2-3		SA-516, Gr 70 with nominal 7/32 SS clad	SA-516, Gr 70 with nominal 3/16 SS clad	SA-516, Gr 70 with nominal 7/32 SS clad
Hot leg ID, in.	42	5.4	42	42	42
Cold leg ID, in.	30	5.4	30	30	30
Between pump and steam generator ID, in.	30	5.4	30	30	30
Engineered Safety Features					
High pressure safety injection pumps	2	6.3	3	3	3
Low pressure safety injection pumps	2	6.3	2	2	2
Safety injection tanks, number	4	6.3	4	4	4
Containment spray pumps	2	6.2	2	2	2
Containment fan coolers units	4	6.2	4	4	4
Air flow capacity, each at emergency conditions, ft^3/min	40,000	6.2	31,000	[illegible]	35,800

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Engineered Safety Features (Cont'd)					
Emergency power Diesel-generator unit	2	8.3	4 (for two units)	2	2
Containment System Parameters					
Type					
	Steel containment vessel with cylindrical shell, hemispherical dome and ellipsoidal bottom - ASME Code, Section III, Class MC, surrounded by reinforced concrete Shield Building.	3.8.2	Steel-lined prestressed post tensioned concrete cylinder, curve dome roof.	Steel-lined prestressed post tensioned concrete cylinder, curved dome roof.	Steel containment vessel with cylindrical shell, hemispherical dome and ellipsoidal bottom - ASME Code, Section III, Class B, surrounded by reinforced concrete Shield Building.
Inside Diameter, ft.	140	3.8	150	116	140
Height, ft.	232	3.8	172	207	232
Free volume, ft^3	2,500,000	6.2	2,335,000	1,780,000	2,500,000
Reference accident Pressure, psig	44	3.8	60	54	44
Steel Thickness, in.			-	-	-
Vertical Wall	1.92	3.8	Not Applicable	Not Applicable	1.91
Hemispherical Head	0.96		Not Applicable	Not Applicable	0.95
Knuckles	2.125		Not Applicable	Not Applicable	225
Concrete Thickness, ft.					
Vertical Wall	Not Applicable	3.8	4 1/3	3 3/4	Not Applicable
Dome	Not Applicable		3 3/4	3 1/4	Not Applicable
Design Parameters - Shield Building		3.8	Not Applicable	Not Applicable	
Inside Diameter, ft.	148				148
Height, ft. (top of foundation to top of dome)	230.5				230.5
Concrete Thickness, ft.					
Vertical Wall	3				3
Dome	2.5				2.5

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
Containment Leak Prevention and Mitigation Systems	Leak-tight penetration, Automatic isolation where required.	6.2	Leak-tight penetration, and continuous steel liner. Automatic isolation where required.	Leak-tight penetration, and continuous steel liner. Automatic isolation where required.	Leak-tight penetration, Automatic isolation where required.
Gaseous Effluent Purge	Discharge through vent.	6.2	Discharge through vent.	Discharge through vent.	Discharge through vent.
RADIOACTIVE WASTE MANAGEMENT SYSTEM					
Liquid Waste Processing Systems					
Reactor Coolant Waste Holdup Tank (BMS)		11.2			
Number	4		1/2	4	4
Capacity (Gal.), each	40,000		6,000/25,000	51,270	40,000
Concentrators					
Number	1		1 (For 2 units)	1	1
Capacity (gpm)	20		50	20	2
Gaseous Waste Processing Systems					
Waste Gas Decay Tank		11.3			
Number	3		6 (For 2 units)	3	3
Capacity (ft^3), each	138		500	300	144
Pressure (psig)	190		150	380	190
Hold-up Time (days)	25		30	30	30
ELECTRIC SYSTEMS					
Number of Offsite Circuits	3	8.1	8	3	3
Number of Incoming Lines to Startup Transformers	2	8.2	2	2	2
Number of Startup Transformers	2	8.2	4	1+1(shared)	2
Number of Main Unit Transformers (Three Phase)	2	8.2	1	3 (single phase)	2
Number of 4.16 KV Engineered Safety Features System Buses	3	8.3	3	2	3

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TABLE 1.3-1 (Cont'd)

Item	St. Lucie Unit 2	Reference Section	San Onofre Units 2 and 3	ANO-2	St. Lucie Unit 1
ELECTRIC SYSTEMS (Cont'd)					
Number of 480V Engineered Safety Features System Buses	3	8.3	3	2	4
Number of 120V Safety Related Vital Buses	4	8.3	4	4	3
Number of Standby Diesel Generators	2	8.3	2	2	4
Diesel Generator Rating (KW)	3685	8.3	4700	2850	3500
INSTRUMENTATION SYSTEMS*					
Reactor Protective System	7.2	7.2	7.2	7.2	7.2
Reactor and Reactor Coolant System	7.7.1.1 7.6.1	7.7.1.1 7.1.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2	7.7.1.1 7.7.1.2
Steam and Feedwater Control System	7.7.1.1	7.7.1.3	7.7.1.3	7.7.1.3	7.7.1.3
Nuclear Instrumentation	7.2.1.1 7.7.1.1	7.2.1.1	7.2.1.1	7.2.1.1	7.2.1.1
Non-Nuclear Process Instrumentation	7.7.1.1 7.5.1	7.5.1.5	7.5.1.5	7.5.1.5	7.5.1.5
CEA Position Instrumentation	7.7.1.1	7.5.1.3	7.5.1.3	7.5.1.3	7.5.1.3

* This section is not suited for tabular description. SAR section numbers have been included for the location of the detailed description of each system.

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1.7 Summary of Outstanding Issues

Section 18 is reserved for the report by ACRS to be issued following its review of the St. Lucie 2 application and this SER. The ACRS report is normally included in a supplement to the SER.

As a result of NRC review of the safety aspects of the St. Lucie 2 application, a number of items remain outstanding at the time of issuance of this report. Since the staff has not completed its review and reached final positions in these areas, NRC considers these issues to be open. The review of these items will be completed before issuing an OL and will be reported in supplements to this report. The open items, with appropriate references to subsections of this report, are summarized below.

(1) Stability of Slopes (2.5.5)

(2) Turbine missiles (3.5.1.3, 3.5.3)

(3) Seismic Displacement of Category I pipes (3.7.2, 3.7.3)

(4) Pump and Valve Operability Assurance (3.9.3.2)

(5) Seismic qualifications (3.10)

(6) Environmental qualifications (3.11)

(7) Seismic and LOCA loads (4.2.3.3)

(8) Matrix Power Supply Test Results (7.1.3, 7.2.5)

(9) Fire protection (7.4, 7.5, 7.7, 8.3.3, 9.5.1)

(10) Starting voltage for 460 V ESF motor (8.3.1.1)

(11) Station electric distribution system voltages (8.4.6)

(12) Fuel Handling System (light loads) (9.1.4)

(13) Operator Training (13.2.1)

(14) Emergency planning (13.3)

(15) Operating and Maintenance Procedures (13.5.2)

(16) ATWS Procedures (15.10.6)

(17) Station Blackout (ALAB 603) (App. C)

(18) TMI issues (Section 22)

Emergency Operating Procedures (I.C.1, I.C.7, I.C.8)
Control Room Design (I.D.1)
Inadequate Core Cooling Instrumentation (II.F.2)
Degraded Core Training (II.B.4)

1.11 Special Plant Features

(1) Automatic Auxiliary Feedwater System

(2) Containment and Shield Building Design

St. Lucie 2 possesses an advanced containment design which, in conjunction with a hold-up, dilution and multiple pass filtration system, significantly reduces off-site doses in the event of postulated accidents.

The design embodies a free standing steel containment vessel within a separate reinforced concrete shield building. There is an annulus between these two structures in which are supply and return ring ducts. To these ducts are connected two independent and safety grade trains of air handling and filtration equipment. This system of air handling and filtration equipment is known as the Shield Building Ventilation System (SBVS).

The annulus is maintained at negative pressure relative to atmospheric pressure during normal operation and this pressure remains negative over the course of an accident. As a result, any leakage in the shield building structure causes atmospheric air to be drawn into the annulus rather than leakage of contaminated annulus air to the atmosphere.

Following an accident, there is an initial draw-down period of single pass filtration followed by a filtered recirculation mode during which there is a filtered and diluted release to the atmosphere.

(3) Condensate Storage Pool and Refueling Water Storage Pool

(4) Plant Computer

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

December 12, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ST. LUCIE PLANT UNIT NO. 2

Dear Dr. Ray:

At its 176th meeting, December 5-7, 1974, the Advisory Committee on Reactor Safeguards completed its review of the application of the Florida Power and Light Company for authorization to construct a second nuclear power unit at its Hutchinson Island site in St. Lucie County, Florida. Members of the Committee visited the site on May 19, 1974; and a Subcommittee meeting was held in West Palm Beach, Florida, on that date. A second Subcommittee meeting was held in Washington, D. C. on November 13, 1974. During its review, the Committee had the benefit of discussions with the Applicant, Combustion Engineering, Inc., Ebasco Services, Inc., the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported on the construction permit application of St. Lucie 1 (Hutchinson Island) on March 12, 1970.

The St. Lucie Plant Unit No. 2 will be located next to St. Lucie Unit No. 1 on a tract of land of approximately 1100 acres, about half way between the towns of Fort Pierce and Stuart on the east coast of Florida. About 1000 people live within a five mile radius of the site. The nearest population center is Fort Pierce (population about 34,000), which is eight miles to the north. However, some buildup of population on the island is probable in the coming years, and the plant and its engineered safety features will be designed on the basis of a low population zone distance of 1 mile.

The plant site on Hutchinson Island is underlain by sand to a depth of several hundred feet. To provide satisfactory bearing and settlement characteristics and resistance to liquifaction, the area of most seismic Category I structures was dewatered, excavated to minus 60 feet (MSL), and filled with compacted soils to form a 30-foot-thick base.

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Earthquake-induced liquefaction of banks of the cooling water canals or of the soils under a non-seismic Class 1 structure such as the St. Lucie Unit 1 switchyard represents a potential problem for the continued reliability of shutdown cooling. One important aspect of this matter relates to the potential for blockage of the inlets for the cooling water system and possibly to the presence of turbidity and particles in the cooling water. The Applicant and the Staff concur that a practical engineering solution exists for any regions which appear to be subject to liquefaction after the current tests are completed and evaluated. The Committee recommends that a conservative approach be taken in assuring integrity of the ultimate heat removal capability. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The proposed pressurized water reactor has a design power level of 2570 MW(t). The St. Lucie Plant Unit No. 2 design duplicates most of the principal features of Unit No. 1; the use of 16x16 fuel in Unit 2 is a principal difference between the two units. The containment system consists of a steel vessel enclosed within a reinforced concrete building, with the annular space maintained at a slightly negative pressure and exhausted through filters. The Applicant has stated that the containment and other structures and systems important to safety will be designed to meet the same tornado design criteria as have been used for other recently reviewed plants, and that protection of vital components will be provided against the probable maximum hurricane-induced flood and runup level as estimated by National Oceanic and Atmospheric Administration and Corps of Engineers methodology.

The St. Lucie Plant Unit No. 2 is the first to propose use of the Combustion Engineering (CE) 16x16 fuel assembly at the construction permit stage. However, some previously reviewed plants employing CE nuclear steam-supply systems are converting from 14x14 fuel to 16x16 fuel during the construction stage and should operate prior to St. Lucie Unit No. 2. Mechanical tests, fuel tests and other research and development are underway. Neither the Regulatory Staff nor the ACRS have completed their review of the new core design. The Committee wishes to be kept informed concerning the results of the various on-going experimental and analytical programs and of any design changes which may be proposed in the future.

An evaluation of the compliance of St. Lucie 2 with 10 CFR 50.46 remains to be performed; however, calculated peak clad temperatures well below the limit are anticipated by the Applicant and the Regulatory Staff.

The ATWS evaluation, including any need for design modifications, remains to be submitted by the Applicant and evaluated by the Regulatory Staff. The Committee wishes to be kept informed.

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St. Lucie Unit No. 2 has some reactor vessel and core design features different from other Combustion Engineering reactors. The Regulatory Staff plans to require an instrumented reactor internals vibration program appropriate to a prototype plant unless the Applicant can provide test results for other plants which clearly substantiate the St. Lucie Unit No. 2 analytical vibration response model. The Committee concurs.

The adequacy of protection against flooding of the ECCS pump room is under study. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Means of qualification of the electric cables from the diesel generators for operation under conditions of temporary tunnel flooding are under review. A different design approach represents a possible alternative for this important function. The Committee recommends that the Applicant and the Staff continue to study this matter.

The Regulatory Staff has proposed that the Applicant upgrade specific pressure systems to seismic Category I and Quality Group C in accordance with interpretations of Regulatory Guides 1.26 and 1.29. Included systems are the letdown loop of the chemical and volume control system, the component cooling lines which service the letdown heat exchanger and the reactor coolant pumps, and the fuel pool makeup system. The Applicant believes that alternate flow paths exist where a safety function must be met and that there is no requirement to upgrade to seismic Category I and Quality Group C in components not necessary to safety. The Committee recommends that the safety significance of these systems be reassessed by the Applicant and by the Staff and the matter resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

The matter of the generation of turbine missiles and their probable effects on reactor safety is under review, including the possible need of design features to reduce the probability or mitigate the consequences. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

Generic problems relating to large water reactors have been identified by the Regulatory Staff and the ACRS and discussed in the Committee's report dated February 13, 1974. These problems should be dealt with expeditiously and appropriately by the Regulatory Staff and the Applicant.

The Committee believes that the above items can be resolved during construction and that, if due consideration is given to these items, St. Lucie Unit No. 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,

/s/

W. R. Stratton
Chairman

References attached

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

June 10, 1975

Mr. William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REPORT ON ST. LUCIE PLANT, UNIT No. 1

Dear Mr. Anders:

At its 182nd meeting, June 5-7, 1975, the Advisory Committee on Reactor Safeguards completed its review of the application of the Florida Power and Light Company for authorization to operate the St. Lucie Plant, Unit No. 1. The project was previously considered at Subcommittee meetings at West Palm Beach, Florida on May 16, 1974; in Washington, D. C. on November 12-13, 1974, and on June 4, 1975. The facility was toured on May 16, 1974. In its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Combustion Engineering, Inc., Ebasco Services, Inc. and the NRC Staff. The Committee reported on the construction permit application of St. Lucie Plant, Unit No. 1 (Hutchinson Island), on March 12, 1970, and on the construction permit application of St. Lucie Plant, Unit No. 2, on December 12, 1974.

The St. Lucie Plant, Unit No. 1, is located on Hutchinson Island on a tract of land of approximately 1100 acres, about half way between Fort Pierce and Stuart on the east coast of Florida. About 1000 people live within a five-mile radius of the site, the originally proposed low population zone (LPZ). The minimum exclusion distance is 5100 feet. The nearest population center is Fort Pierce (1970 population about 30,000), which is eight miles to the northwest. However, some buildup of population on the island is probable in the coming years, and the plant and its engineered safety features are being modified to meet an LPZ radius of 1 mile.

The plant site is underlain by sand to a depth of several hundred feet. To provide satisfactory bearing and settlement characteristics and resistance to liquefaction, the area of most seismic Category I structures was dewatered, excavated to minus 60 feet (MSL), and filled with compacted soils to form a 30-foot-thick base.

(See Hutchinson Island for CP report)

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Earthquake-induced liquefaction of the banks of the cooling water canals or under the dam to Big Mud Creek, which provides a seismic Class 1 source of water for the ultimate heat sink, represents a potential problem for the continued reliability of shutdown cooling. The Applicant and the NRC Staff differ in their conclusions regarding a prudent interpretation of the existing data with regard to the potential for liquefaction. The Committee agrees with the Staff that unless additional information by the Applicant establishes that unacceptable soil movements cannot occur, appropriate remedial measures should be taken. This matter should be resolved in a manner satisfactory to the NRC Staff.

Questions related to the potential effects of a stalled hurricane on the integrity of safety features are currently under review. This matter should be resolved in a manner satisfactory to the NRC Staff.

Additional information and evaluation thereof is required with regard to the potential effects of tornado-induced missiles on some engineered safety features. This matter should be resolved in a manner satisfactory to the NRC Staff.

The St. Lucie Plant, Unit No. 1, includes a pressurized water reactor similar to that currently employed at the Calvert Cliffs and Millstone 2 plants. The current application requests an operating license of 2560 MWt; the power level requested in the construction permit application was 2440 MWt.

Several changes have been made in the Combustion Engineering ECCS evaluation model to bring it into conformance with the Commission Criteria per 10 CFR 50, Appendix K. A partial analysis (a break in the pump discharge leg) has been made using the new model; hot leg and suction leg analyses remain to be evaluated, but the Applicant and the NRC Staff expect the pump discharge leg break to be limiting. This analysis leads to a maximum permitted linear heat generation rate of 14.6 kw/ft. A relatively low peaking factor is required to achieve this limit and the Applicant proposes to use both in-core and ex-core instrumentation in order to assure adequate accuracy of measurement of core power distributions.

The Committee believes that the proposed monitoring methods may be acceptable, but that an augmented startup program be employed, and that satisfactory experience at steady state, 100% power and during transients at less than full power should be obtained, reviewed, and evaluated by the NRC Staff prior to operating at full power in a system-load-follow mode.

A question has arisen concerning loads on the vessel support structure for certain postulated loss-of-coolant accidents in pressurized water reactors. This matter should be resolved in a manner satisfactory to the NRC Staff.

Potentially damaging water hammer has been observed in the feed water inlet piping of some PWR steam generators. Corrective measures are planned upon completion of studies and experimental investigation of the phenomenon. The adequacy of the corrective measures should be experimentally verified to the satisfaction of the NRC Staff. The Committee wishes to be kept informed.

The analysis of Anticipated Transients Without Scram is incomplete for the St. Lucie Plant, Unit No. 1. The Committee recommends that a schedule for submission of information and for any modifications, if necessary, be prepared, and that this matter be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Some questions remain with respect to the handling of heavy loads over the fuel storage pool. This matter should be resolved in a manner satisfactory to the NRC Staff.

Means of qualification of the electric cables from the diesel generators for operation under various environmental conditions are still under review. This matter should be resolved in a manner satisfactory to the NRC Staff.

Suitable instrumentation to follow the course of an accident has been generically identified as an important feature needed to assist operating personnel in diagnosing unexpected events. The NRC Staff should initiate prompt action to clarify the essential requirements for this instrumentation including information to be monitored, environmental conditions under which it must operate, location and type of display, relationship to normally used instrumentation and methods of assuring functional effectiveness at the time of need. Arrangements should be made to incorporate the required instrumentation in all plants licensed for construction. Where possible the necessary equipment should also be provided on licensed operating power plants. The Committee wishes to be kept informed.

The Applicant is making progress in arrangements for emergency procedures to be followed in case of an accidental release of radioactive materials from the plant. Yet to be confirmed, however, are plans of the state agencies whose actions would be essential in dealing with the population in case of some such events. The Committee recommends that the applicant and the NRC Staff continue to collaborate with the State in moving ahead to complete development of an emergency response plan and that the adequacy of arrangements for implementing such a plan be confirmed prior to initial operation of the plant.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the St. Lucie Plant, Unit No. 1, can be operated at power levels up to 2560 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

WKerr
W. Kerr
Chairman

ST. LUCIE 1

During eddy current testing of steam generator tubes at St. Lucie Unit 1, Florida Power & Light Company (FPL) found a significant number of defective tubes (wall thickness reduced by greater than 40%). The degradation was located in the U-shaped portions of the center tubes which differs from previous expereicne with CE steam generators. FPL is performing further inspections.

SITE & PLANT DESCRIPTION

PROJECT DESCRIPTION

- COMBUSTION ENGINEERING – PRESSURIZED WATER REACTOR NUCLEAR STEAM SUPPLY SYSTEM
- WESTINGHOUSE TURBINE GENERATOR
- ARCHITECT/ENGINEER – EBASCO SERVICES INC.
- CONSTRUCTOR – FLORIDA POWER & LIGHT CO./ EBASCO SERVICES INC. – INTEGRATED ORGANIZATION
- LOCATION: HUTCHINSON ISLAND, ST LUCIE COUNTY, FLORIDA
- CORE LOAD DATE – OCTOBER 28, 1982

SELECTED QUANTITY STATUS

COMMODITY	FORECAST PERCENT COMPLETE BY INDUSTRY MODEL	PSL NO 2 ACTUAL PERCENT COMPLETE AS OF 9/30/81	TOTAL PROJECT FORECASTED QUANTITIES AS OF 9/30/81
TERMINATIONS	28.5	31.2	135,955
CABLE	50.0	53.8	3,861,000
SMALL BORE PIPE	55.5	74.1	91,651
CABLE TRAY	86.0	96.5	39,082
CONDUIT	66.5	81.6	335,419
LARGE BORE PIPE	73.0	92.2	78,137
EMBEDS	93.5	95.3	3,915,564
FORMWORK	94.5	95.5	1,652,005
REBAR	96.0	98.5	28,038,787
CONCRETE	98.5	97.5	136,537

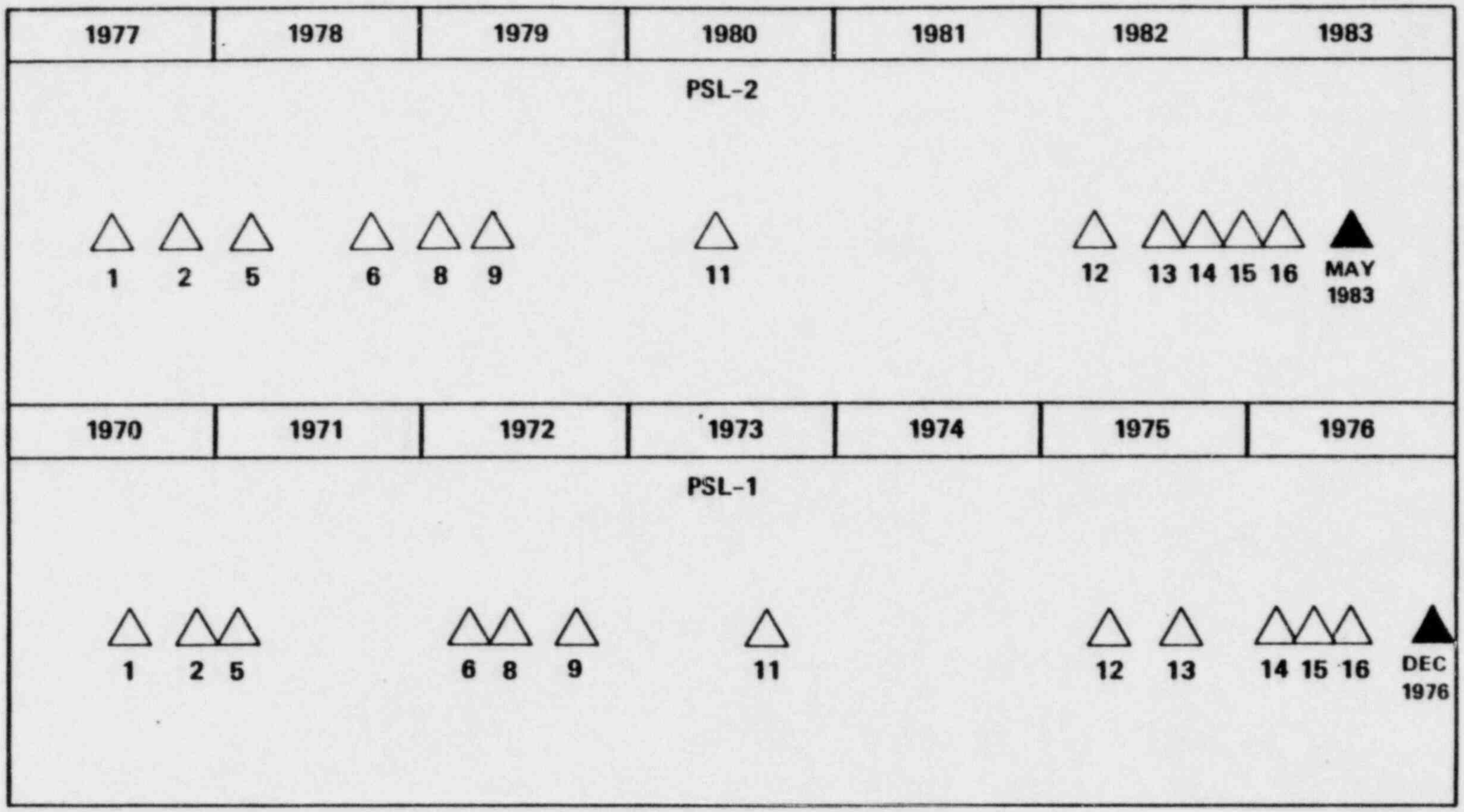
COMPLETED MILESTONE ANALYSIS

SCHEDULED	ITEM	ACTUAL
05/02/77	CONSTRUCTION PERMIT	05/02/77
07/06/77	COMPLETE RCB BASEMAT	07/07/77
10/25/77	COMPLETE INTAKE BASEMAT	10/27/77
01/04/78	COMPLETE TGB BASEMAT	11/29/77
01/18/78	START RCB LINER	12/21/77
03/01/78	COMPLETE RAB BASEMAT	03/15/78
12/06/78	POST WELD HEAT TREATMENT	09/20/78
01/17/79	START RCB INTERNAL CONCRETE	11/03/78
05/09/79	COMPLETE FHB BASEMAT	07/20/79
06/18/80	SET NSSS COMPONENTS	06/26/80
09/23/80	COMPLETE RCB OPERATING FLOOR AT 62.0'	10/17/80
09/26/80	SET CONTAINMENT VESSEL DOME	10/04/80
12/15/80	COMPLETE RAB EXTERIOR CONCRETE	12/18/80
03/14/81	COMPLETE RC LOOP LARGE BORE PIPING	02/06/81
04/28/81	COMPLETE REFUELING WATER STORAGE TANK	04/30/81
09/06/81	COMPLETE EXTERIOR SHIELD WALL DOME CONCRETE	08/11/81
09/25/81	INTAKE TURNOVER (CTO)	09/23/81

REMAINING MILESTONES

ITEM	SCHEDULE
COMPONENT COOLING WATER TURNOVER (CTO)	10/11/81
PLACE TURBINE-GENERATOR ON TURNING GEAR	12/15/81
COMPLETE OCEAN DISCHARGE PIPE	12/25/81
DIESEL GENERATOR INITIAL RUN (CTO)	01/13/82
SECONDARY HYDRO	01/21/82
COMMENCE COLD HYDRO	03/17/82
START HOT OPERATIONS	07/03/82
COMMENCE CORE LOAD	10/28/82

UNIT #1 VS. UNIT #2 SCHEDULE MILESTONES



1 △ START RCB BASE MAT CONC.

2 △ START INTAKE STRUCTURE BASE MAT CONC.

5 △ START RAB BASE MAT CONCRETE

6 △ COMP ERECT & TEST STEEL CU TO EL 149'

8 △ START RCB INTERNAL CONCRETE

9 △ START FHB BASE MAT CONCRETE

11 △ START SETTING NSSS MAJOR EQUIPMENT

12 △ COMMENCE COLD HYDRO

13 △ START HOT OPS #1

14 △ COMMENCE CORE LOAD

15 △ START CRIT PERFORMANCE TESTS

16 △ COMMENCE POWER ESCALLATION

▲ COMMERCIAL OPERATION

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UTILITY TECHNICAL CAPABILITY AND ORGANIZATION

J. Williams

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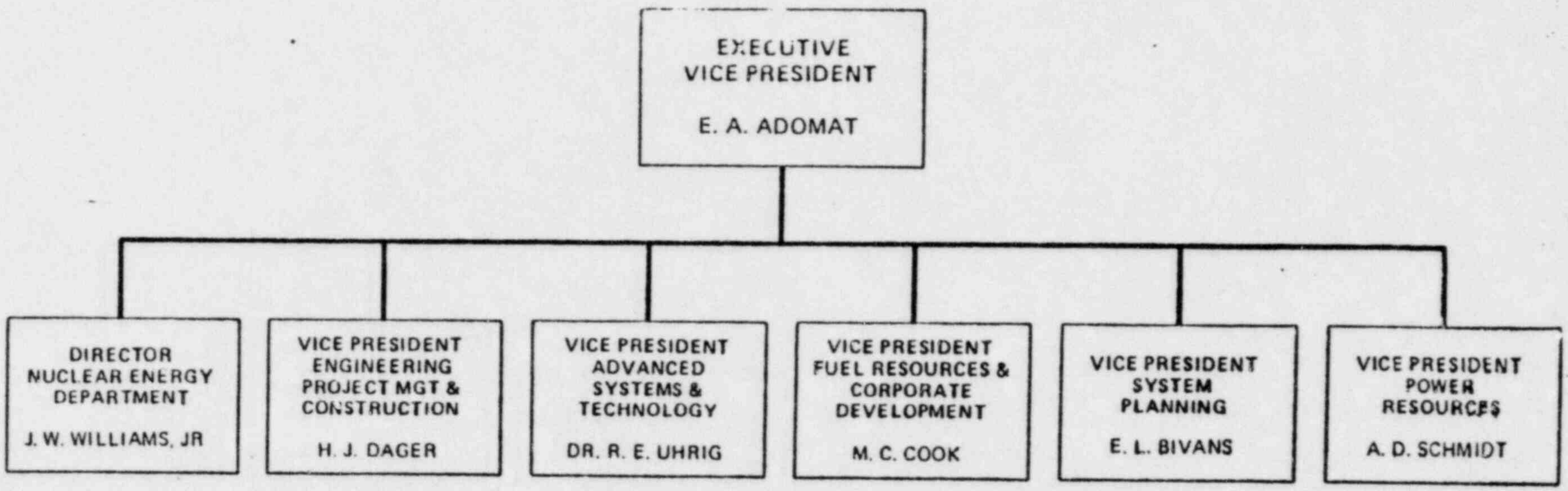
ORGANIZATION CHART



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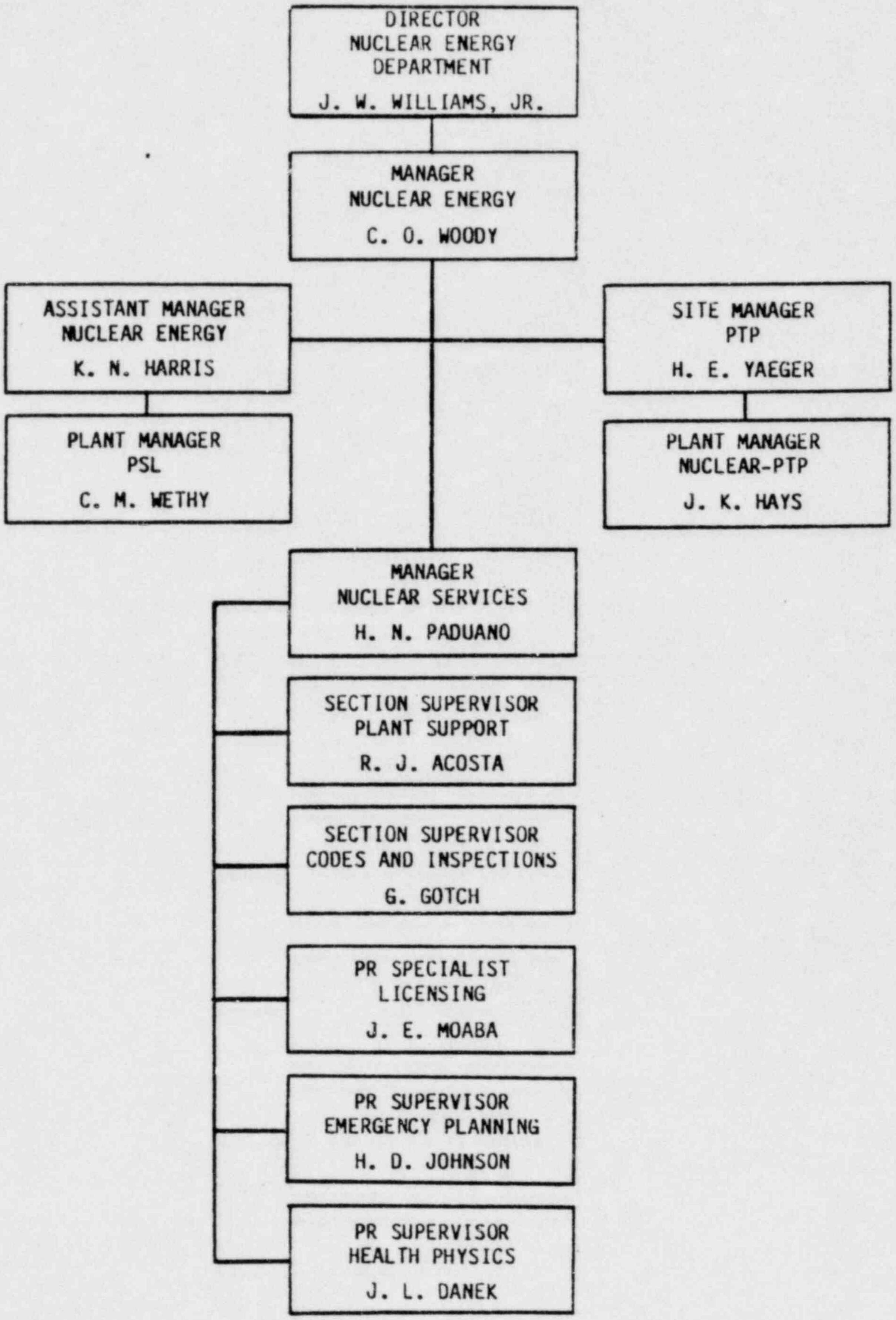
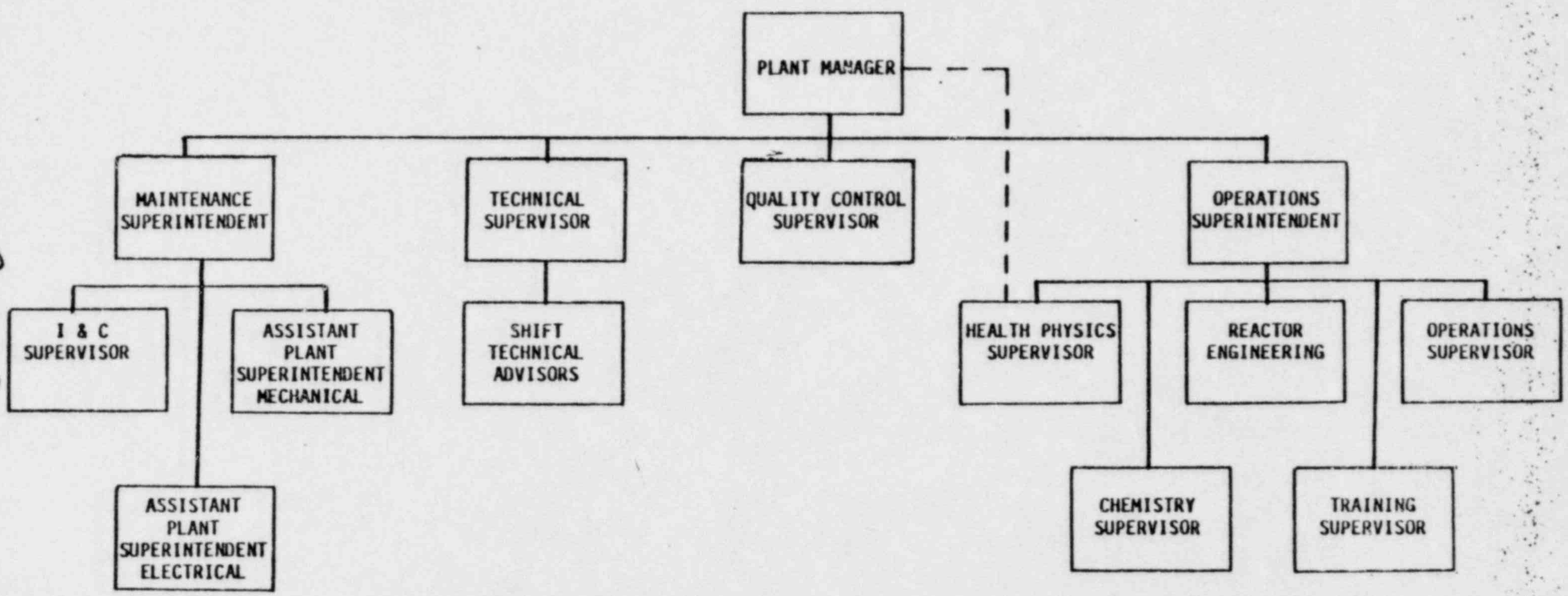
ORGANIZATION CHART



ABBREVIATED PLANT ORGANIZATION



COMPANY NUCLEAR REVIEW BOARD
Members and Alternates

	ADVANCED SYSTEMS AND TECHNOLOGY	POWER PLANT ENGINEERING	NUCLEAR ENERGY/POWER RESOURCES NUCLEAR	NUCLEAR FUELS
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NOTES: NUMBERS DENOTE THE LINE OF SUCCESSION OF THE CHAIRMAN POSITION FOR THE CNRB UNLESS OTHERWISE DESIGNATED. ALTERNATES MAY SUBSTITUTE FOR ANY MEMBER FROM THE SAME DIVISION.

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ST.LUCIE
FACILITY REVIEW GROUP

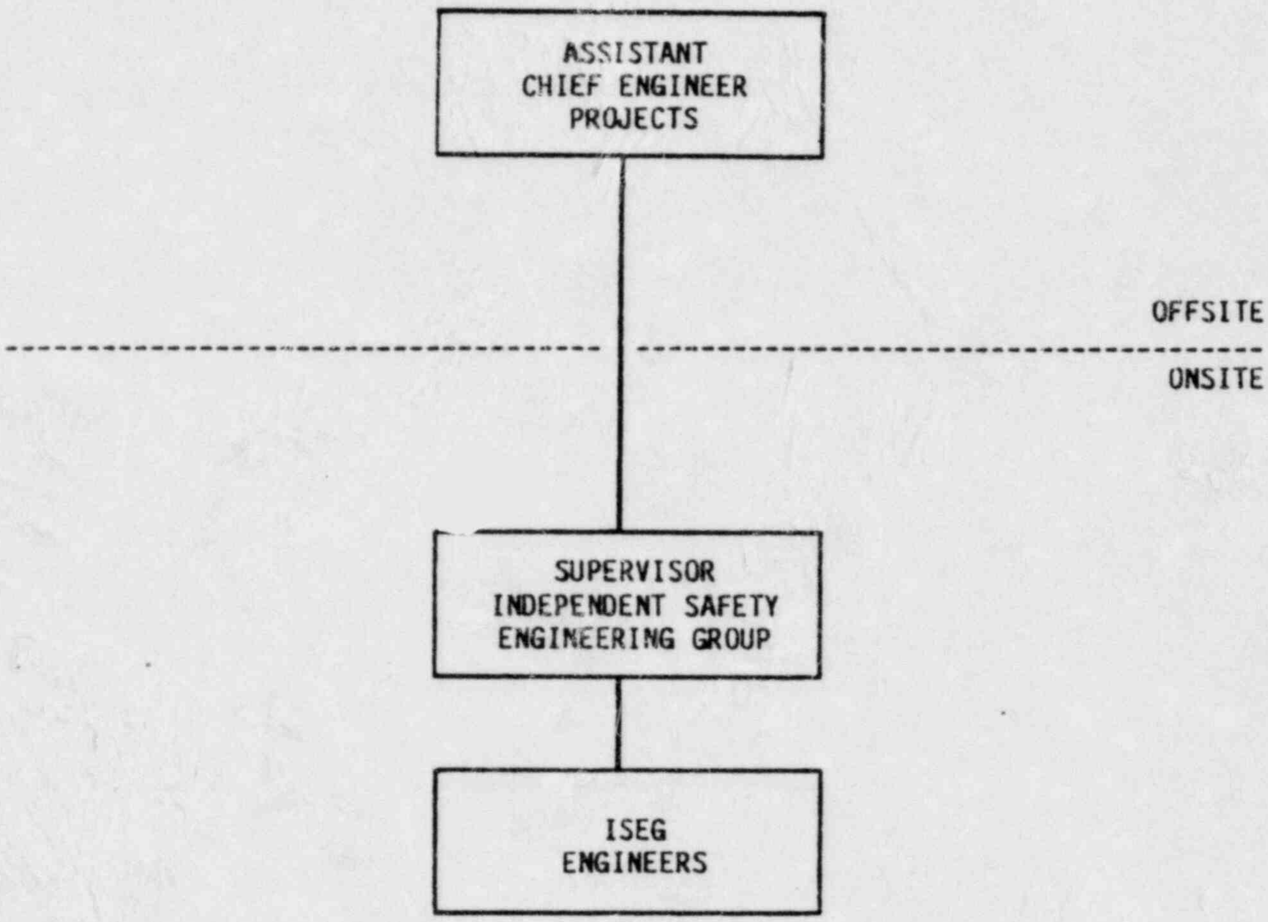
MEMBERS:

- PLANT MANAGER – CHAIRMAN
- OPERATIONS SUPERINTENDENT
- OPERATIONS SUPERVISOR
- MAINTENANCE SUPERINTENDENT
- INSTRUMENT AND CONTROL SUPERVISOR
- REACTOR SUPERVISOR
- HEALTH PHYSICS SUPERVISOR
- TECHNICAL SUPERVISOR
- CHEMISTRY SUPERVISOR
- QUALITY CONTROL SUPERVISOR
- ASSISTANT PLANT SUPERINTENDENT – MECHANICAL
- ASSISTANT PLANT SUPERINTENDENT – ELECTRICAL

ALTERNATE MEMBERS:

- TRAINING SUPERVISOR
- NUCLEAR PLANT SUPERVISORS
- OUTAGE COORDINATOR
- OTHER SENIOR INDIVIDUALS PRE-DESIGNATED BY FRG MEMBERS

INDEPENDENT SAFETY ENGINEERING GROUP



INSTRUMENTATION TO FOLLOW THE COURSE OF A SERIOUS ACCIDENT (INCLUDING INADEQUATE CORE COOLING (ICC))

DEVELOPEMENTAL HISTORY OF ACCIDENT MONITORING INSTRUMENTATION FOR ST. LUCIE - UNIT 2

- SAFETY RELATED DISPLAY INSTRUMENTATION
  (SRP SECTION 7.5)
- UNIQUE IDENTIFICATION OF POST ACCIDENT MONITORING INSTRUMENTATION (PAM)
  (BTP EICSB 23)
- TMI IMPACT OF ADDITIONAL MONITORING CAPABILITY
  (NUREG 0737)
- REGULATORY GUIDE 1.97 REVISION 2

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FPL COMMITMENTS
TO
R.G. - 1.97 REV 2

- FPL INTERPRETATION OF RG-1.97 REQUIREMENTS
- ASSESSMENT OF VARIABLE TYPES
  - B,C,D,E,
  - A

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SCHEDULE
FOR INSTALLATION

PRESENTLY INSTALLED 64%

PLANNED INSTALLATION BY FUEL LOAD 95 %

PLANNED INSTALLATION BY JUNE 1983 100%

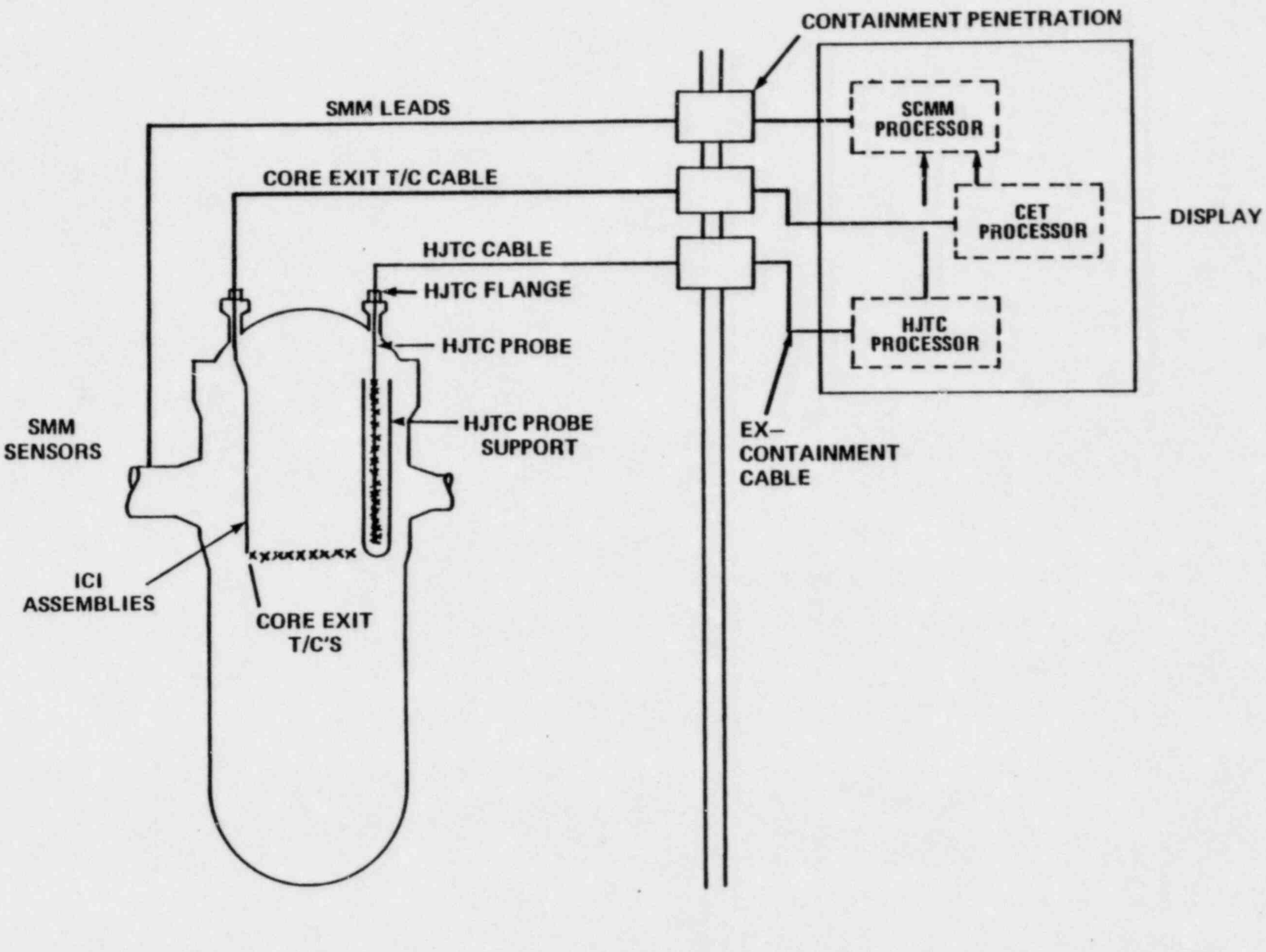
INADEQUATE CORE COOLING MONITORING INSTRUMENTATION (ICC)

A-87

MAJOR FUNCTIONAL COMPONENTS OF THE CE ICC MONITORING INSTRUMENTATION SYSTEM

- SUBCOOLED MARGIN MONITOR (SMM)
- REACTOR VESSEL LEVEL (HJTC)
- CORE EXIT TEMPERATURE (CET)

ICC HARDWARE SUMMARY



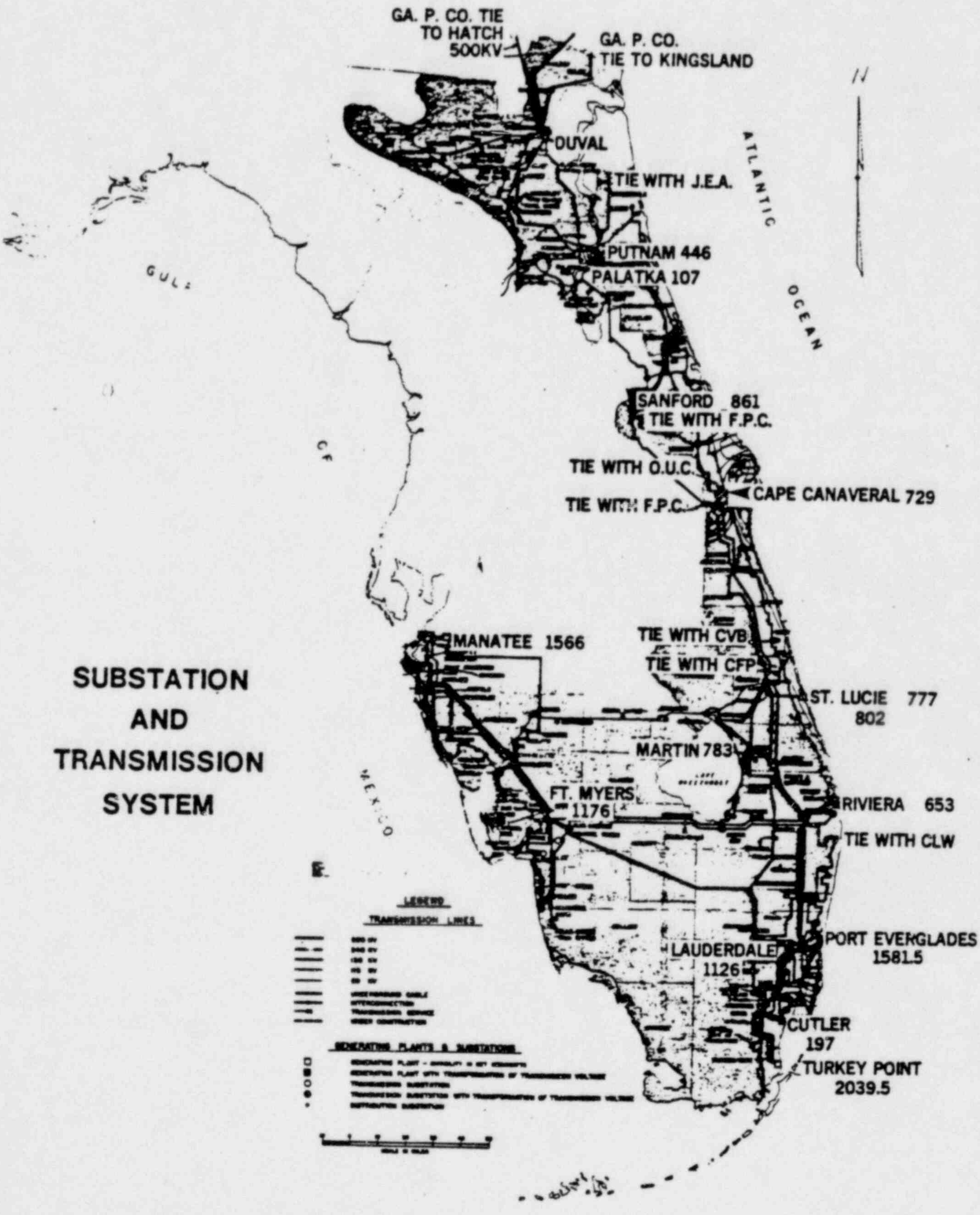
AC AND DC POWER RELIABILITY (INCLUDING STATION BLACKOUT)

A-90

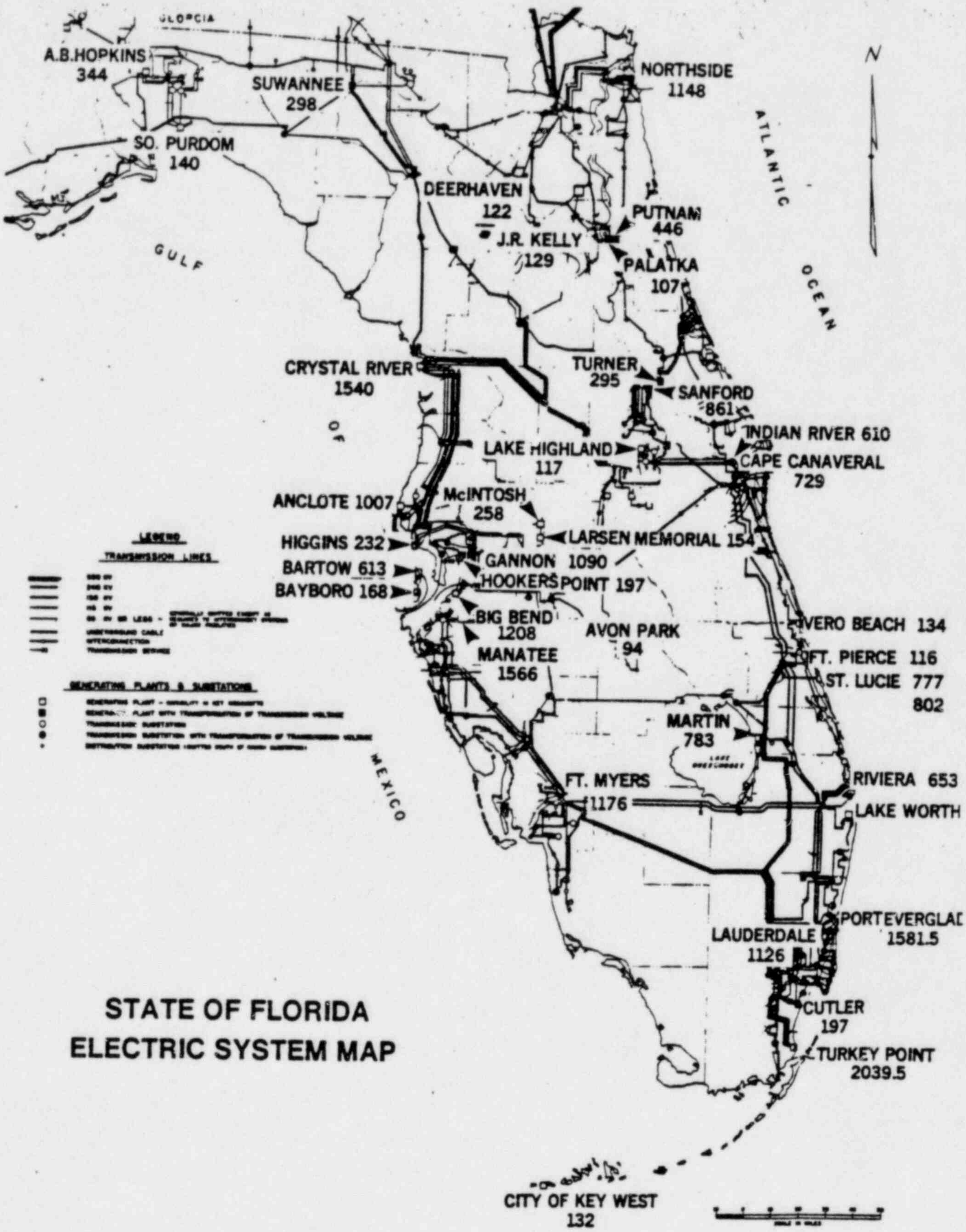
STATION ELECTRIC POWER RELIABILITY

- THE FPL POWER SUPPLY SYSTEM
- TRANSMISSION FACILITIES SUPPLYING THE ST. LUCIE PLANT
- ON SITE AC AND DC POWER SYSTEM
- STATION BLACKOUT

SUBSTATION
AND
TRANSMISSION
SYSTEM

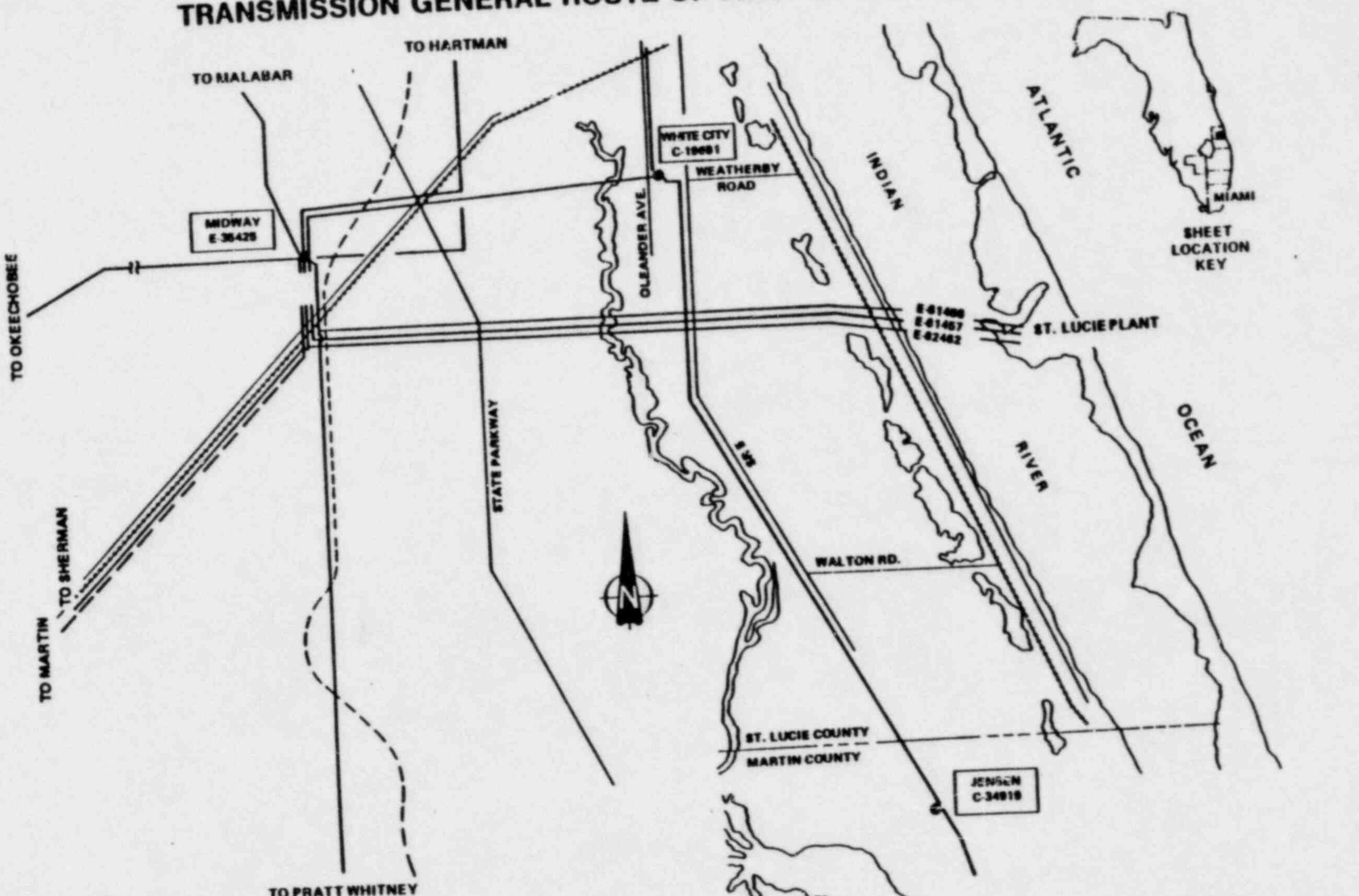


A-92

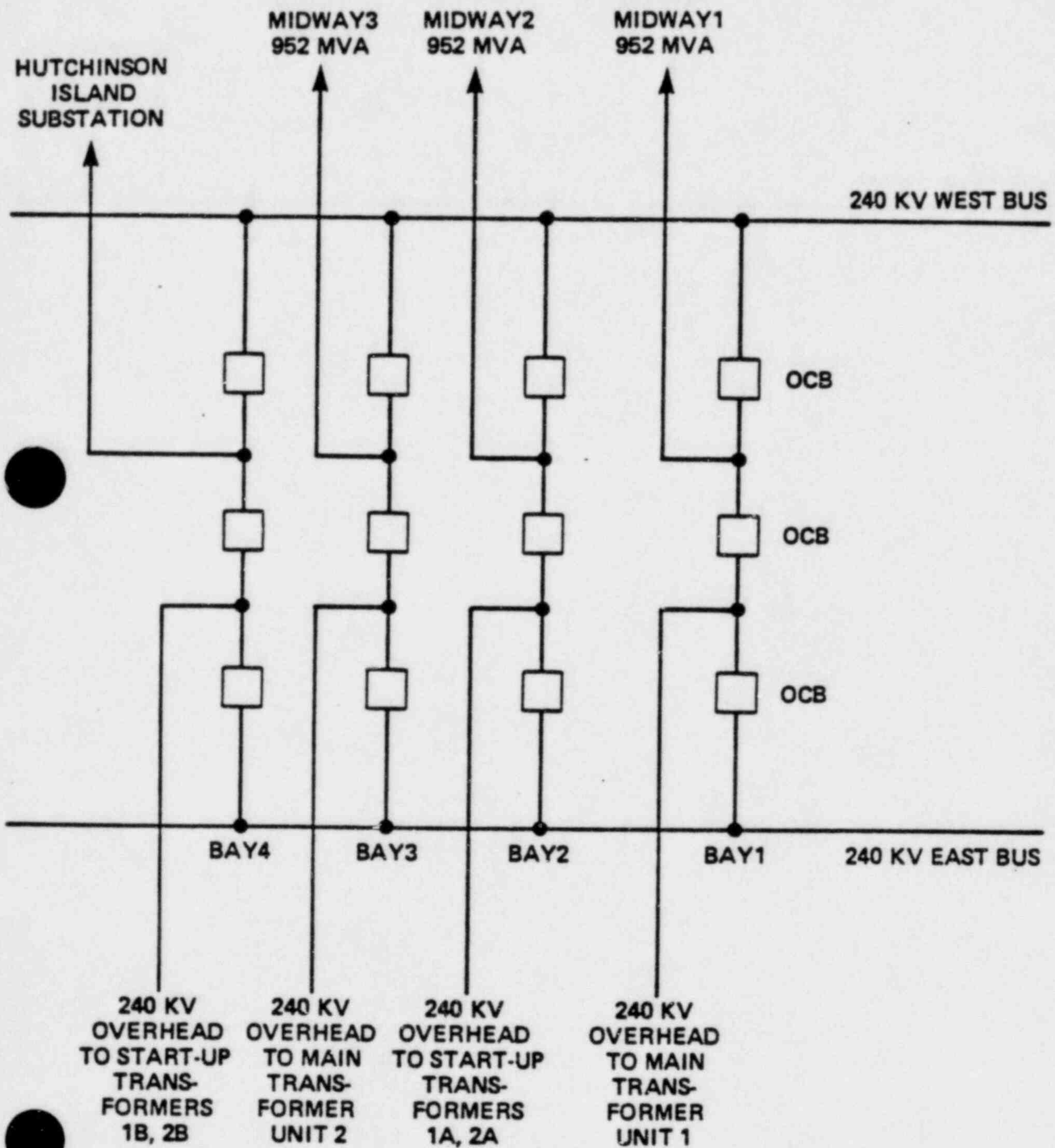


STATE OF FLORIDA
ELECTRIC SYSTEM MAP

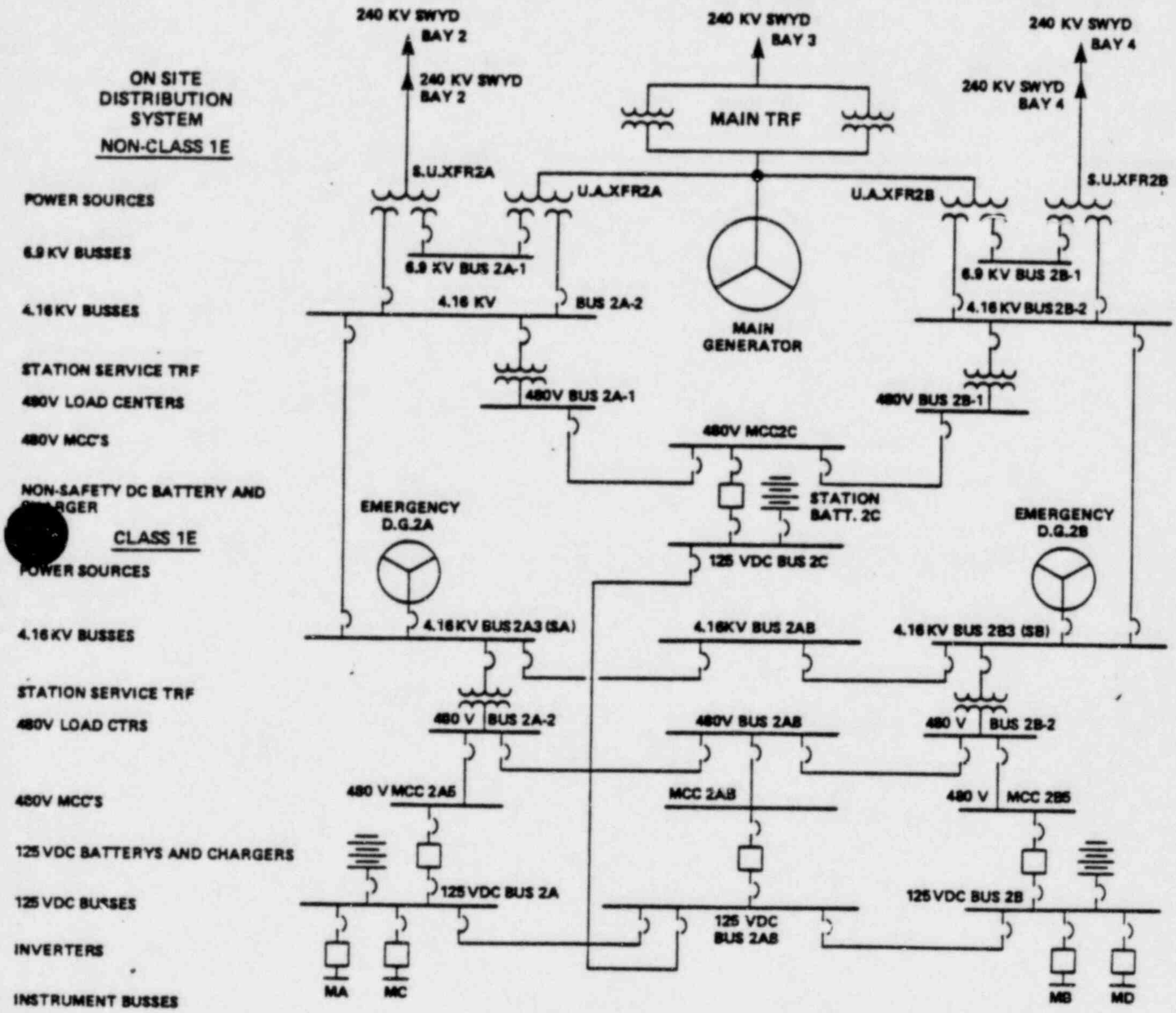
TRANSMISSION GENERAL ROUTE OF LINE DRAWINGS KEY MAP



ST. LUCIE 240KV SWITCHYARD



ONSITE DISTRIBUTION SYSTEM



TOTAL LOSS OF AC POWER (STATION BLACKOUT)

EVENT

- LOSS OF OFF SITE POWER
- FAILURE OF BOTH DIESEL GENERATORS

ANALYSIS RESULTS

- SUBCOOLED NATURAL CIRCULATION IS MAINTAINED FOR A MINIMUM OF 4 HOURS
- DECAY HEAT REMOVAL CAPABILITY MAINTAINED IN EXCESS OF 4 HOURS
- BATTERY CAPACITY SUFFICIENT TO OPERATE REQUIRED EQUIPMENT IN EXCESS OF 4 HOURS WITH SELECTIVE LOAD REDUCTION

FISHER

A-98

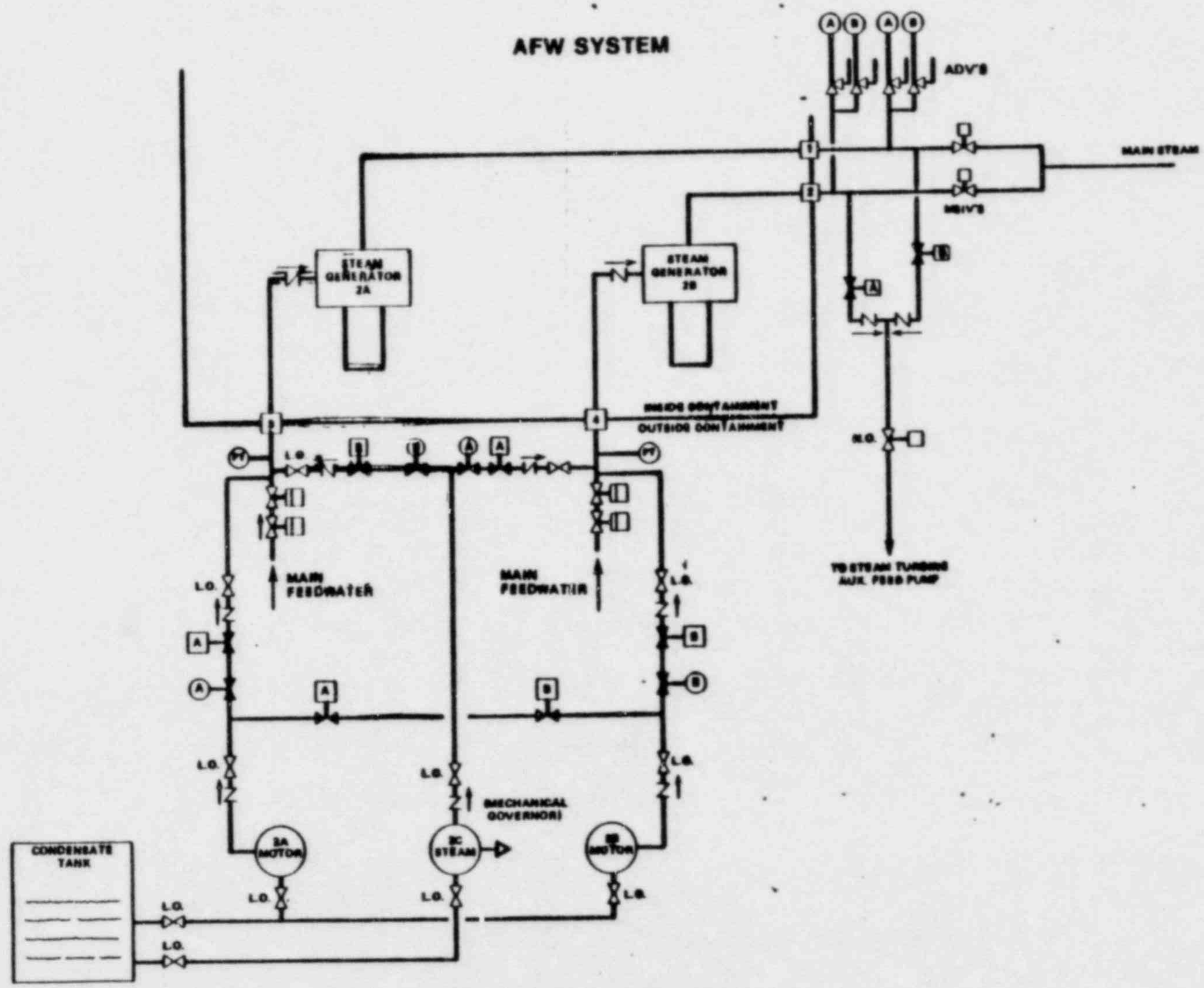
GDC 19 (SHUTDOWN FROM OUTSIDE OF CONTROL ROOM)

FOUR MAIN FUNCTIONS MUST BE ACCOMPLISHED IN A COOLDOWN TO COLD SHUTDOWN CONDITIONS

- REACTOR COOLANT CIRCULATION
- DECAY HEAT REMOVAL
- BORATION AND MAKEUP
- DEPRESSURIZATION

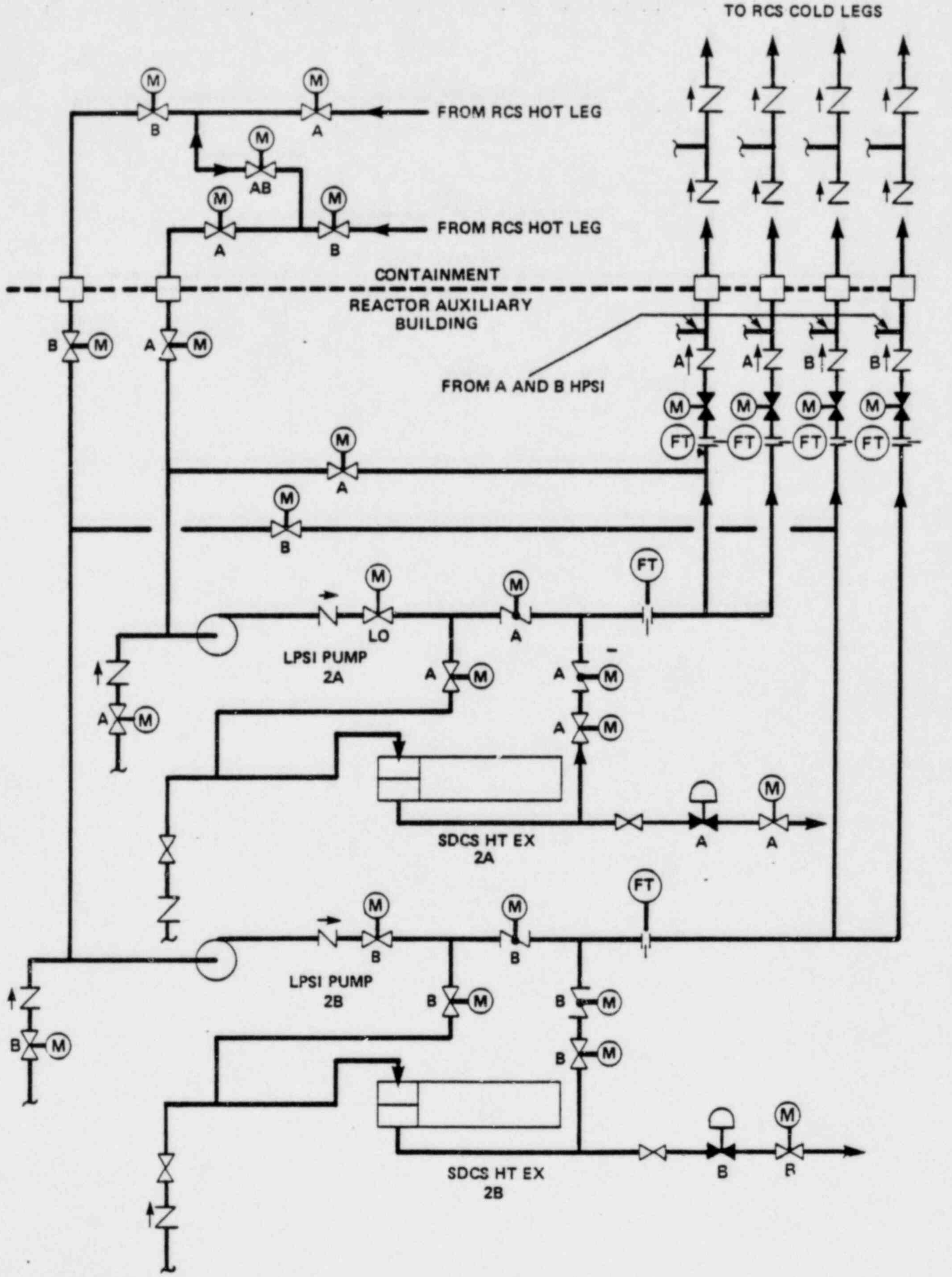
A-99

AFW SYSTEM



ADV'S
MAIN STEAM
MSIV'S
STEAM GENERATOR 2A
STEAM GENERATOR 2B
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
N.O.
TO STEAM TURBINE AUX. FEED PUMP
MAIN FEEDWATER
MAIN FEEDWATER
L.O.
MECHANICAL GOVERNOR
2A MOTOR
2C STEAM
2B MOTOR
CONDENSATE TANK

SHUTDOWN COOLING



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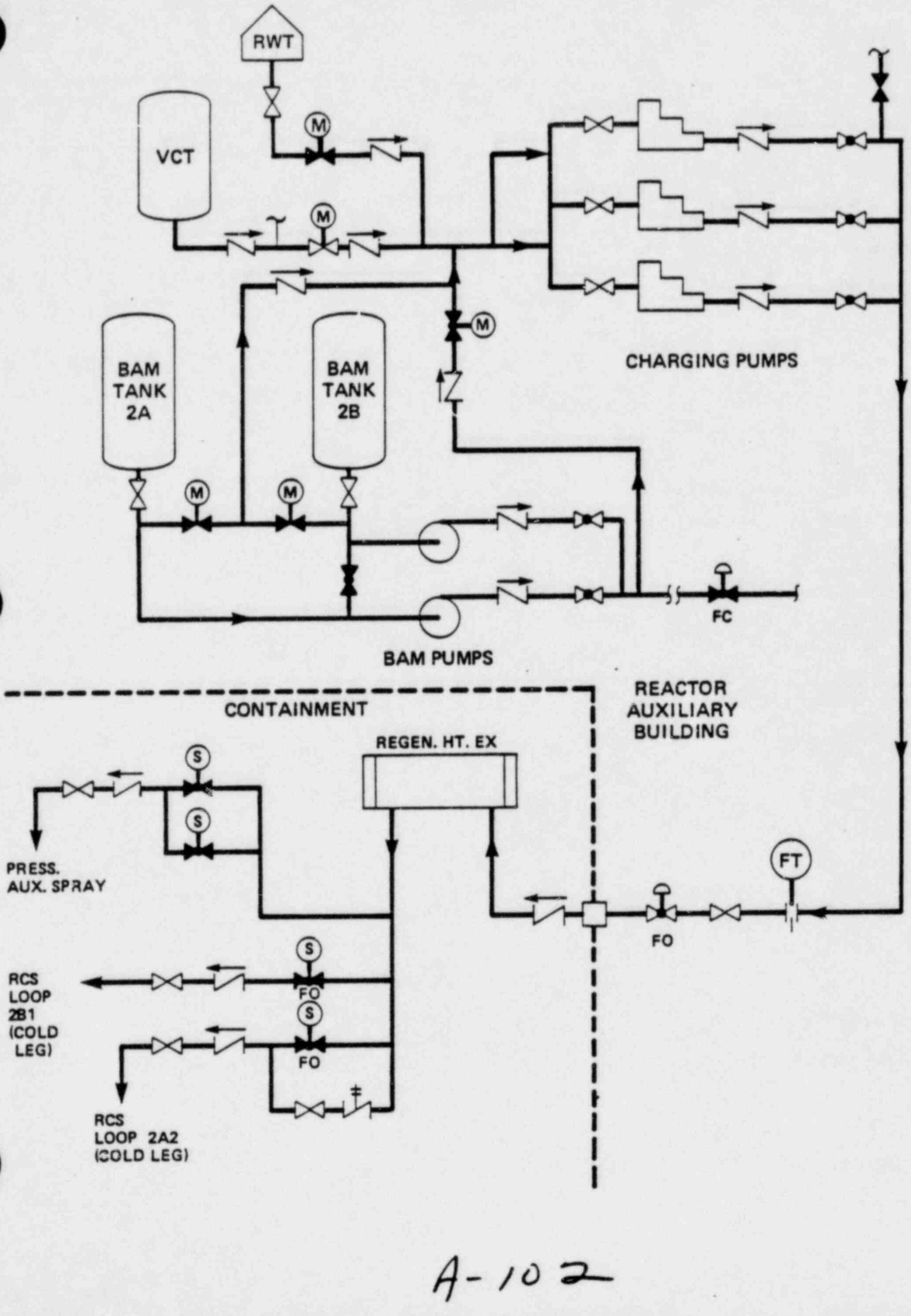
BORATION MAKEUP AND DEPRESSURIZATION

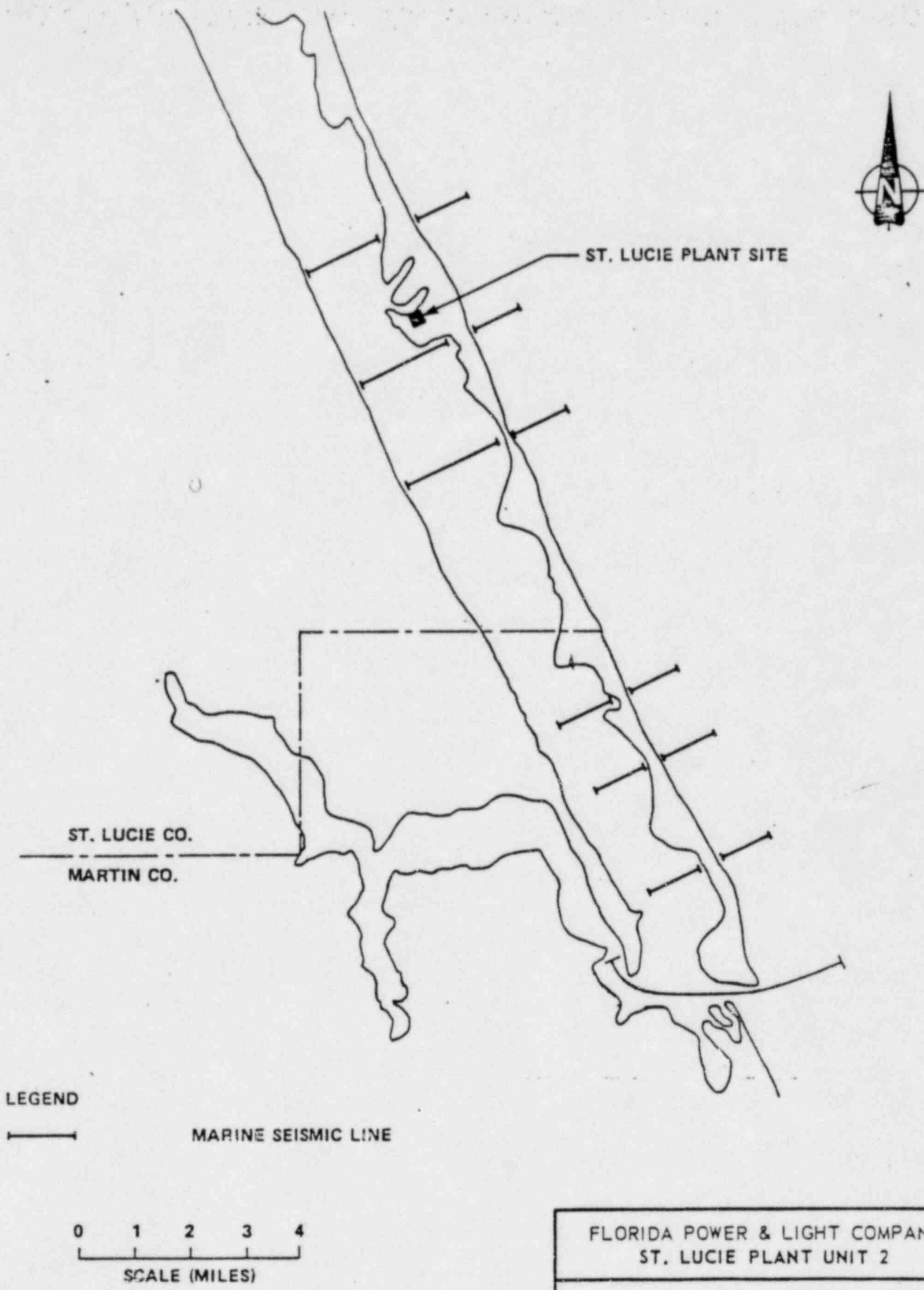


CONTROLS

- SIAS BLOCK CONTROLS (BOTH TRAINS)
- AUXILIARY FEEDWATER (AFW) PUMP START/STOP (ALL PUMPS)
- AUXILIARY FEEDWATER (AFW) VALVE OPEN/CLOSE (ALL PUMPS)
- ATMOSPHERIC DUMP VALVE (ADV) CONTROL (ALL FOUR VALVES)
- CHARGING PUMP START/STOP (ALL PUMP)
- AUXILIARY SPRAY VALVES OPEN/CLOSE (BOTH VALVES)
- LETDOWN ISOLATION VALVES OPEN/CLOSE (ALL FOUR VALVES)
- CHARGING LINE ISOLATION VALVES OPEN/CLOSE (BOTH VALVES)

HOT SHUTDOWN PANEL INSTRUMENTS

- STEAM GENERATOR 2A WATER LEVEL AND PRESSURE
- STEAM GENERATOR 2B WATER LEVEL AND PRESSURE
- A & B TRAIN PRESSURIZER WATER LEVEL AND PRESSURE
- A & B TRAIN PRESSURE INDICATING CONTROLLER FOR ADV'S
- A & B TRAIN RCS FOR COLD LEG TEMPERATURE
- A & B TRAIN SDCS TEMPERATURE AND FLOW
- A & B TRAIN DIESEL GENERATOR VOLTAGE AND POWER
- TURBINE DRIVEN AFW PUMP HAND INDICATING CONTROLLER



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

LOCATION MAP OF PROPOSED
MARINE SEISMIC LINES
FIGURE 2.5-27a

A-105

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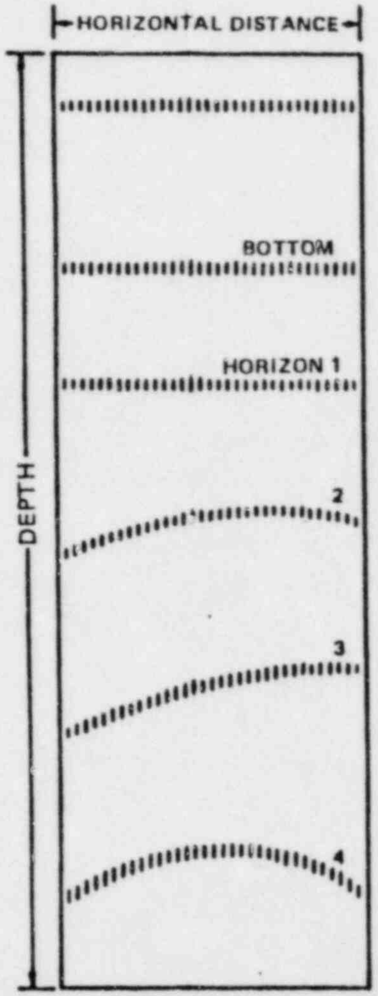
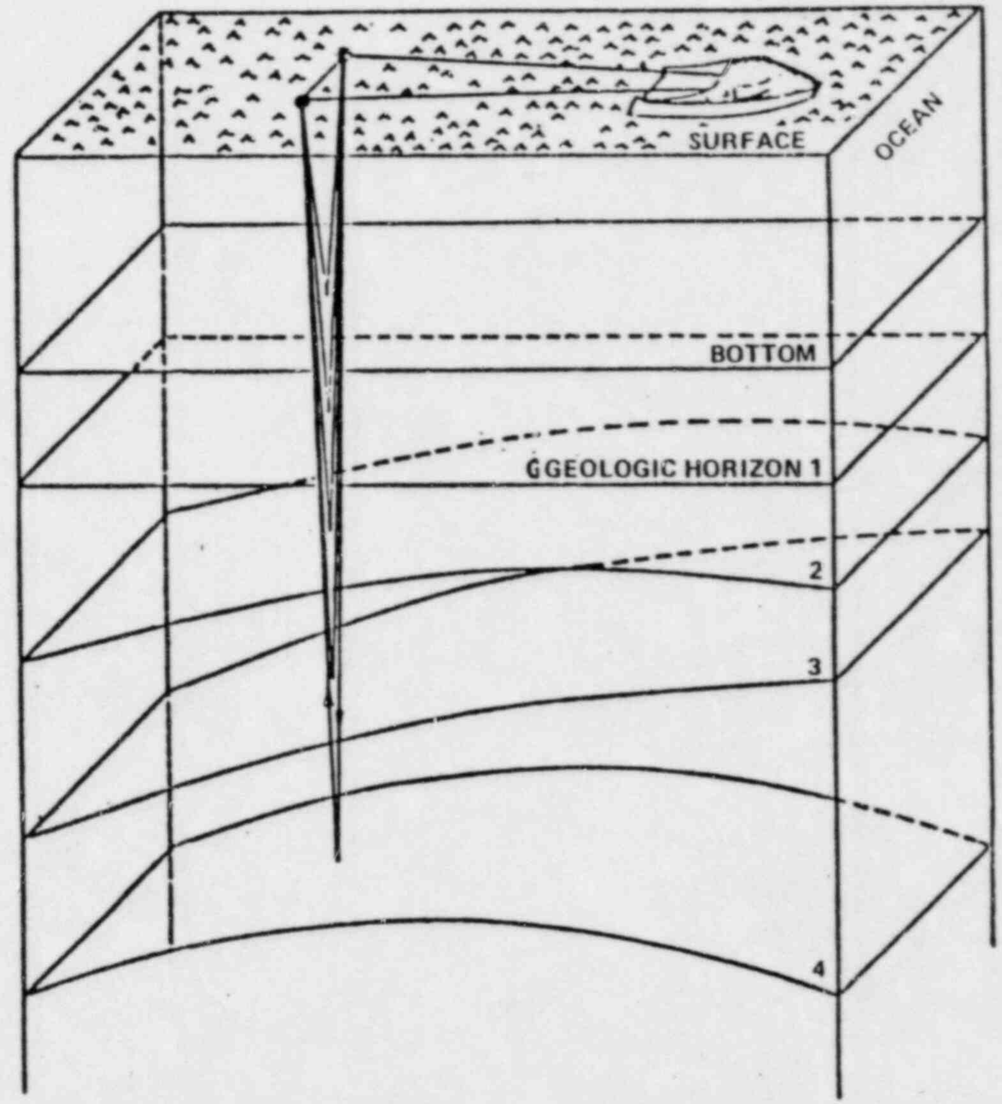
SURVEY RECORD



SURVEY OPERATION



CONTINUOUS SEISMIC REFLECTION PROFILING

ST. LUCIE 2
RECENT LICENSING MILESTONES

- 02/17/81 FSAR DOCKETED
- 02/17/81 ER DOCKETED
- 08/07/81 ER INPUT TO PROJECT MANAGER
- 09/11/81 SER INPUT TO PROJECT MANAGER
- 10/09/81 SAFETY EVALUATION REPORT (SER) ISSUED
- 10/16/81 DRAFT ENVIRONMENTAL STATEMENT (DES) ISSUED
- 10/30/81 ACRS SUBCOMMITTEE MEETING ON ST. LUCIE 2

CONTACT:
V. NERSES
(301) 492-7318

ST. LUCIE 2

PROJECTED LICENSING MILESTONES

- o 11/12/81 FULL ACRS MEETING ON ST. LUCIE 2
- o 11/27/81 SUPPLEMENT TO SER ISSUED (CLOSED OUT OPEN ITEMS, ADDRESS ACRS CONCERNS)
- o 12/~~01~~ 14/81 CLOSING DATE FOR COMMENTS ON DES
- o 01/15/8~~1~~ 2 FINAL ENVIRONMENTAL STATEMENT (FES) ISSUED
- o 04/30/82* ATOMIC SAFETY AND LICENSING BOARD (ASLB) HEARING
- o 09/82* ASLB LICENSING DECISION
- o 10/82 COMMISSION LICENSING DECISION
- o 10/82 FUEL LOAD

*SCHEDULED PRIOR TO INFORMATION THAT THESE ARE NO HEALTH, ENVIRONMENTAL AND SAFETY CONTENTIONS.

CONTACT:
V. NERSES
(301) 492-7318

ST. LUCIE 2

PROJECTED SER OPEN ITEMS AT TIME SSER ISSUED (11/27/81)

ITEM	NEXT ACTION	DATE OF NEXT ACTION	ESTIMATED CLOSE OUT DATE
(1) PUMP AND VALVE OPERABILITY ASSURANCE	FP&L	11/30/81	1st QTR. '82
(2) SEISMIC QUALIFICATIONS	FP&L	11/30/81	1st QTR. '82
(3) ENVIRONMENTAL QUALIFICATIONS	FP&L	11/30/81	1st QTR. '82
(4) SEISMIC AND LOCA LOADS	FP&L	02/82	04/82
(5) FUEL HANDLING SYSTEM (LIGHT LOADS)	FP&L	11/30/81	01/82
(6) OPERATING AND MAINTENANCE PROCEDURES	FP&L	11/16/81	12/18/81
(7) ATWS PROCEDURES	FP&L	11/16/81	12/18/81
(8) EMERGENCY OPERATING PROCEDURES (I.C.1, I.C.7, I.C.8)	FP&L	11/16/81	12/18/81

ST. LUCIE 2

SER OPEN ITEMS

ITEM		NEXT ACTION	DATE OF NEXT ACTION	ESTIMATED CLOSE OUT DATE
* (1)	STABILITY OF SLOPES	FP&L	10/28/81	11/27/81
* (2)	TURBINE MISSILES	STAFF	11/16/81	11/27/81
* (3)	SEISMIC DISPLACEMENT OF CATEGORY I PIPES	"	11/16/81	11/27/81
(4)	PUMP AND VALVE OPERABILITY ASSURANCE	FP&L	11/30/81	1st QTR. OF '82
(5)	SEISMIC QUALIFICATIONS	"	"	"
(6)	ENVIRONMENTAL QUALIFICATIONS	"	"	"
(7)	SEISMIC AND LOCA LOADS	"	02/82	04/82
(8)	MATRIX POWER SUPPLY TEST RESULTS	STAFF	11/16/81	11/27/81
(9)	FIRE PROTECTION	FP&L	1/82	1st QTR. OF '82
*(10)	STARTING VOLTAGE FOR 460 V ESF MOTOR	STAFF	11/16/81	11/27/81
*(11)	STATION ELECTRIC DISTRIBUTION SYSTEM VOLTAGES	"	"	"
(12)	FUEL HANDLING SYSTEM (LIGHT LOADS)	FP&L	11/30/81	12/82
*(13)	OPERATOR TRAINING	STAFF	11/16/81	11/27/81
*(14)	EMERGENCY PLANNING	FP&L	11/23/81	11/27/81

*RESOLUTION OF THESE ITEMS WILL BE WRITTEN UP IN THE NEXT SSER

CONTACT:
V. NERSES
(301) 492-7318

ITEM		NEXT ACTION	DATE OF NEXT ACTION	ESTIMATED CLOSE OUT DATE
(15)	OPERATING AND MAINTENANCE PROCEDURES	FP&L	11/16/81	12/18/81
(16)	ATWS PROCEDURS	"	"	"
*(17)	STATION BLACKOUT (ALAB 603)	STAFF	11/16/81	11/27/81
(18)	TMI ISSUES			
	EMERGENCY OPERATING PROCEDURES (I.C.1, I.C.7, I.C.8)	FP&L	11/16/81	12/18/81
	*CONTROL ROOM DESIGN (I.D.1)	STAFF	11/16/81	11/27/81
	*INDADEQUATE CORE COOLING INSTRUMENTATION (II.F.2)	FP&L	10/28/81	"
	*DEGRADED CORE TRAINING (II.B.4)	STAFF	11/16/81	"

*RESOLUTION OF THESE ITEMS WILL BE WRITTEN UP IN THE NEXT SSER.

CONTACT:
V. NERSES
(301) 492-7318

OPERATOR SELECTION PROCESS
CRITERIA FOR SELECTION

- STANDARDIZED SCREENING TEST SCORE
- PSYCHOLOGICAL REVIEW
- INTERVIEW BY PLANT MANAGEMENT
- FINAL REVIEW BY PLANT MANAGEMENT

APPENDIX IX
OPERATOR SELECTION PROCESS: CRITERIA FOR SELECTION

A-112

LICENSED OPERATOR TRAINING

- UNIT I LICENSING PROGRAM
- OPERATIONS EXPERIENCE PHASE
- UNIT I/2 DIFFERENCES TRAINING
- SIMULATOR EXAMINATION
- PRE-LICENSE SCREENING EXAM

A-113

LICENSED OPERATOR REQUALIFICATION

- IMPLEMENTS 10 CFR 55, APPENDIX A
- PERIODIC SYSTEM & THEORY LECTURES & EXAMS
- REVIEW OF PLANT & PROCEDURE CHANGES
- REVIEW OF SIGNIFICANT EVENTS
- SIMULATOR REFRESHER TRAINING
- ANNUAL WRITTEN & ORAL EXAMS

A-114

UNIT I LICENSING PROGRAM

- INITIAL SCREENING EXAMINATION
- BASIC SCIENCES
- NUCLEAR FUNDAMENTALS
- REACTOR PHYSICS
- PLANT SYSTEMS
- RADIATION PROTECTION
- TRANSIENT / ACCIDENT ANALYSIS
- OPERATIONS PRACTICE
- SIMULATOR TRAINING
- COURSE REVIEW
- PRE-LICENSE SCREENING EXAM

A-115

UNIT 1/2 DIFFERENCES TRAINING

- WEEKS 1 THRU 5 - UNIT 1 & 2 SYSTEM COMPARISONS AND DIFFERENCES TRAINING
- WEEKS 6 AND 7 - REACTOR THEORY REVIEW
  THERMODYNAMICS
  HEALTH PHYSICS
- WEEKS 8 AND 9 - PROCEDURES
  TRANSIENT & ACCIDENT ANALYSIS
- WEEKS 10 THRU 12 - SYSTEM TRACING
  REVIEW & EXAMINATION

A-116

LICENSE CANDIDATE SIMULATOR PROGRAM

- 3 - 5 WEEKS TRAINING
- 50% BOARD OPERATION
- 50% CLASSROOM
- NORMAL OPERATIONS PRACTICE
- EMERGENCY OPERATIONS PRACTICE
- STARTUP PRACTICE AND CERTIFICATIONS
- NRC ADMINISTERED OPERATING EXAMINATION

A-117

LICENSED REQUALIFICATION SIMULATOR PROGRAM

- ONE WEEK PER YEAR
- 50% BOARD OPERATION
- 50% CLASSROOM
- NORMAL OPERATIONS
- EMERGENCY OPERATIONS
- NRC ADMINISTERED OPERATING EXAMINATION

A-118

J. Barrow

EMERGENCY OPERATING PROCEDURES (EOP)

EMERGENCY OPERATING PROCEDURES

A-120

- BACKGROUND REQUIREMENTS
- EOP OBJECTIVES
- EOP HIERARCHY OF PRIORITIES
- EOP FORMAT & CONTENT
- TRAINING

BACKGROUND REQUIREMENTS

- THREE MILE ISLAND (TMI)
- NUREG 0737 - TMI REQUIREMENTS
- PARTS I.C.I. & I.C.7 GUIDANCE FOR EOP
- COMBUSTION ENGINEERING OWNERS GROUP SYSTEM GUIDELINES
- NUREG 0799 - EOP CONTENT & FORMAT

A-121

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EOP OBJECTIVES

- REACTIVITY CONTROL
- REACTOR COOLANT SYSTEM (RCS) INVENTORY CONTROL
- REACTOR COOLANT SYSTEM (RCS) PRESSURE CONTROL
- CORE HEAT REMOVAL
- REACTOR COOLANT SYSTEM (RCS) HEAT REMOVAL
- CONTAINMENT INTEGRITY
- RADIOACTIVE RELEASE CONTROL

EOP HIERARCHY OF PRIORITIES

- SYMPTOMS
- PREVENTIVE ACTION
- IMMEDIATE ACTION
- MULTIPLE FAILURES
- SUBSEQUENT ACTION

EOP FORMAT & CONTENT

- SPECIFIC SECTIONS OF PROCEDURE
- PROCEDURE CONSISTENCY
- AVAILABILITY & ACCESSIBILITY
- PRESENTATION OF INFORMATION
- SPECIFIC REQUIREMENTS
- BALANCE

A-124

TRAINING

- TECHNICAL
- CLASSROOM
- PLANT WALKTHROUGHS
- SIMULATOR
- DRILLS

INSTRUMENTATION TO FOLLOW THE COURSE OF A SERIOUS ACCIDENT (INCLUDING INADEQUATE CORE COOLING (ICC))

DEVELOPEMENTAL HISTORY OF ACCIDENT MONITORING INSTRUMENTATION FOR ST. LUCIE - UNIT 2

- SAFETY RELATED DISPLAY INSTRUMENTATION (SRP SECTION 7.5)
- UNIQUE IDENTIFICATION OF POST ACCIDENT MONITORING INSTRUMENTATION (PAM) (BTP EICSB 23)
- TMI IMPACT OF ADDITIONAL MONITORING CAPABILITY (NUREG 0737)
- REGULATORY GUIDE 1.97 REVISION 2

FPL COMMITMENTS
TO
R.G. - 1.97 REV 2

- FPL INTERPRETATION OF RG-1.97 REQUIREMENTS
- ASSESSMENT OF VARIABLE TYPES
  - B,C,D,E,
  - A

SCHEDULE
FOR INSTALLATION

PRESENTLY INSTALLED 64%

PLANNED INSTALLATION BY FUEL LOAD 95 %

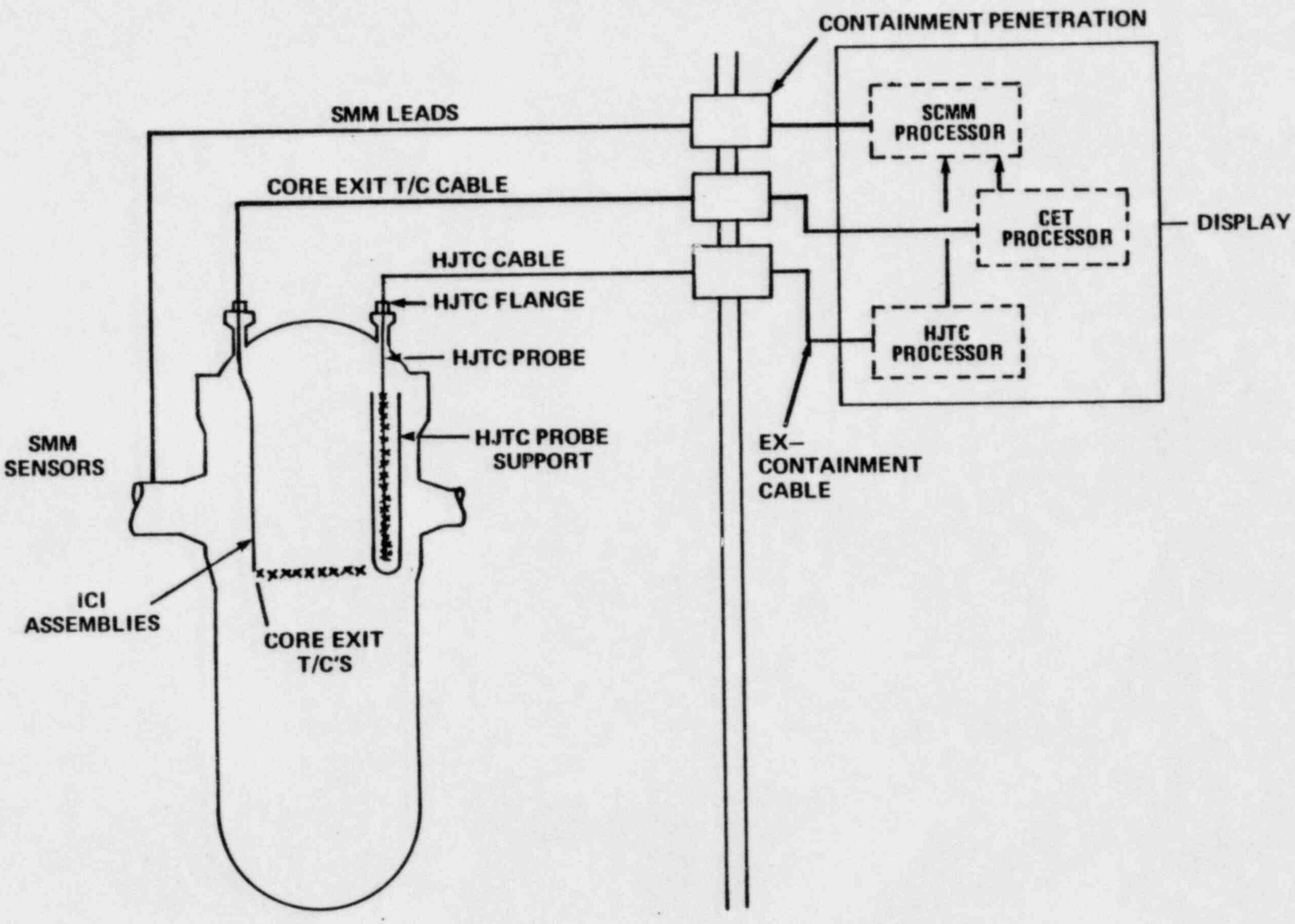
PLANNED INSTALLATION BY JUNE 1983 100%

INADEQUATE CORE COOLING MONITORING INSTRUMENTATION (ICC)

MAJOR FUNCTIONAL COMPONENTS OF THE CE ICC MONITORING INSTRUMENTATION SYSTEM

- SUBCOOLED MARGIN MONITOR (SMM)
- REACTOR VESSEL LEVEL (HJTC)
- CORE EXIT TEMPERATURE (CET)

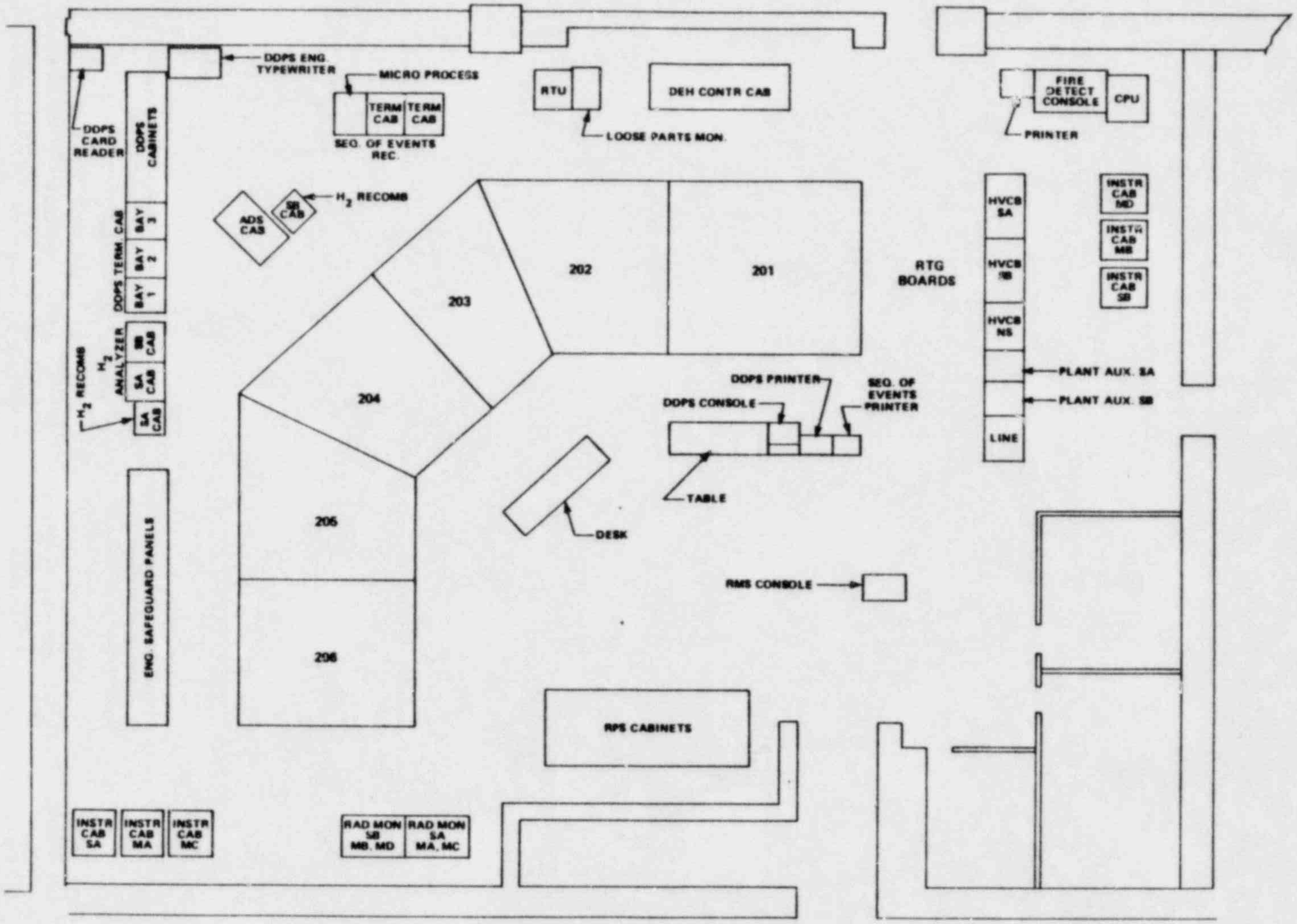
ICC HARDWARE SUMMARY



A-132

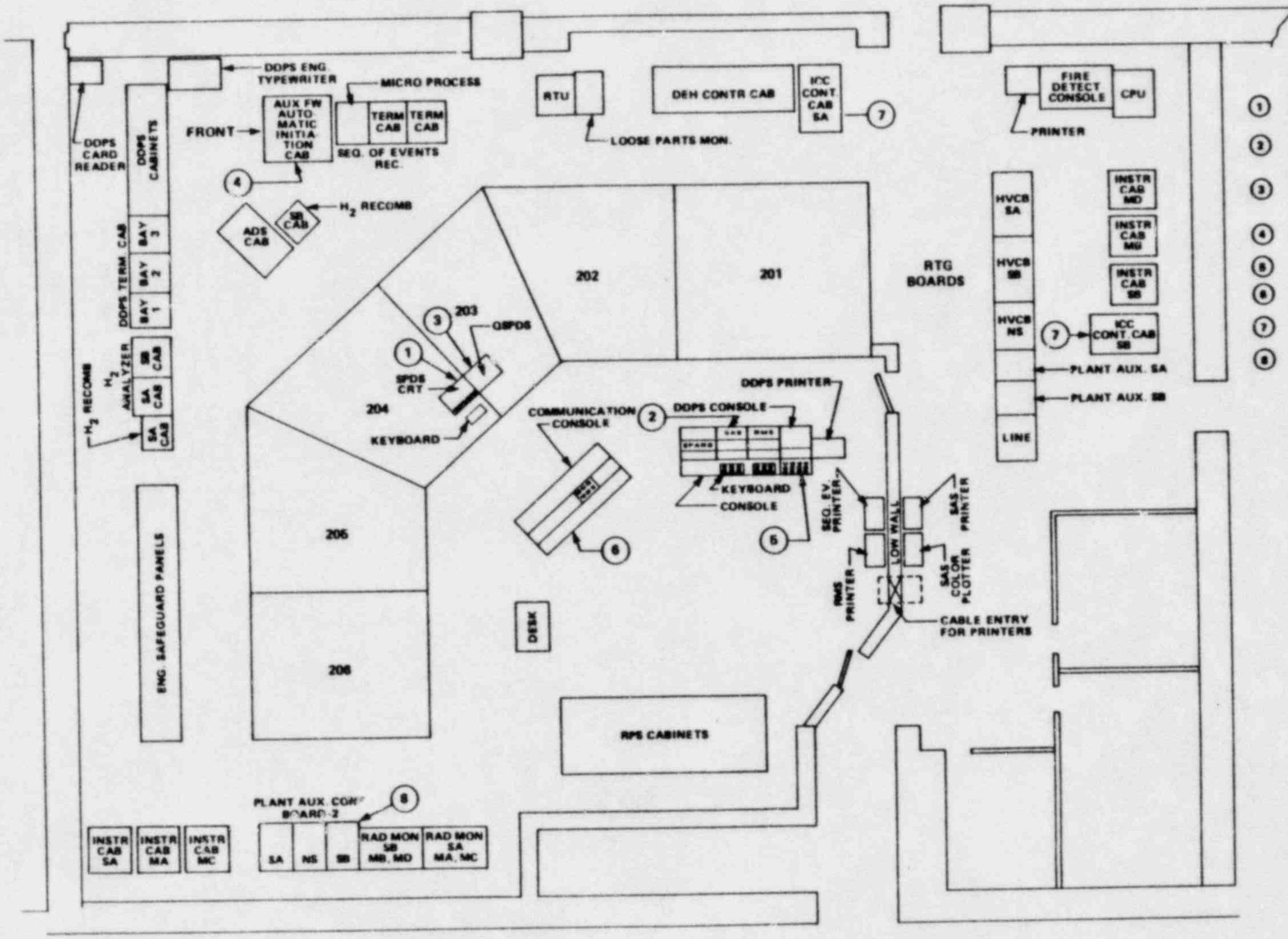
CONTROL ROOM DESIGN
CHANGES AS
A RESULT OF TMI.

St. Lucie No. 2
CONTROL ROOM AS DESIGNED BEFORE TMI



A-134

St. Lucie No. 2
CONTROL ROOM MODIFICATIONS AFTER TMI



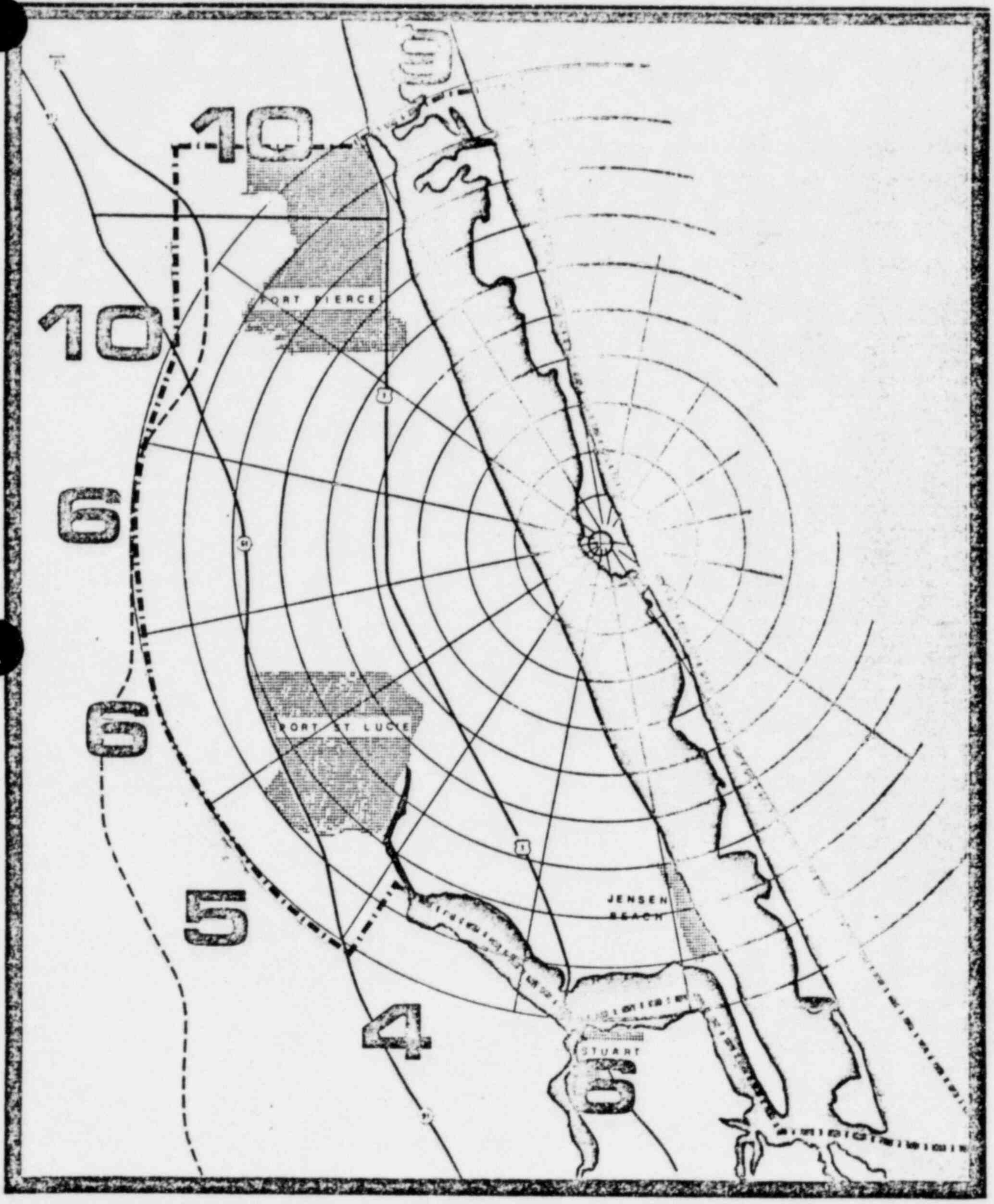
1. SAFETY PARAMETER DISPLAY SYSTEM (SPDS) & KEYBOARD
2. SAFETY ASSESSMENT SYSTEM (SAS) CRT & KEYBOARD
3. QUALIFIED SAFETY PARAMETER DISPLAY SYSTEM (QSPDS) CRT & PLASMA INDICATORS
4. AUXILIARY FEEDWATER AUTOMATIC INITIATION SYSTEM (AFAS) CABINET
5. OPERATOR'S CONSOLE
6. COMMUNICATION CONSOLE
7. INADEQUATE CORE COOLING (ICC) DETECTION COMPUTER CABINETS
8. PLANT AUX CONTROL BOARD

DR. HOWARD D. JOHNSON
EMERGENCY PLANNING SUPERVISOR
NUCLEAR ENERGY DEPARTMENT
FLORIDA POWER AND LIGHT COMPANY

EMERGENCY PLANNING

- CHANGES RESULTING FROM POPULATION INCREASES
- BASIS FOR ACCEPTABLE EMERGENCY PLANNING
- WIND DIRECTION AND RELATION TO EVACUATION

ST. LUCIE PLANT



A-137

POPULATION CONSIDERATIONS IN SITING

- 10 CFR PART 100
  - Exclusion area - no residents
  - LPZ - relatively low population density
  - Distance to nearest population center must be at least 1 1/3 X LPZ
  - Ex. Area & LPZ not fixed in size, but must meet dose guidelines of Part 100
- REG. GUIDE 4.7 (OCT. 1975)
  - Present and projected population (to end-of-life) examined
  - If population density at plant startup exceeds 500/mi^2 out to 30 miles, OR if density at end-oflife exceeds 1000/mi^2 out to 30 miles, applicant should examine alternative sites with low population
  - Pop. values are "trip" levels, not upper bounds of acceptability.
  - Pop. review is made at CP stage in context of alternative sites.

POPULATION COMPARISON

(PEOPLE/MILE^2)

DIST. (MILES	INDIAN PT. (1970)	NEWBOLD IS. (1970)	ST. LUCIE 1983	1990	2000	2010	2020
0-2	740	240	39	72	166	196	196
0-5	664	1350	159	310	545	762	1064
0-10	689	1720	312	387	519	711	1009
0-20	700	1360	156	177	245	321	437
0-30	1400	1530	95	118	156	195	252

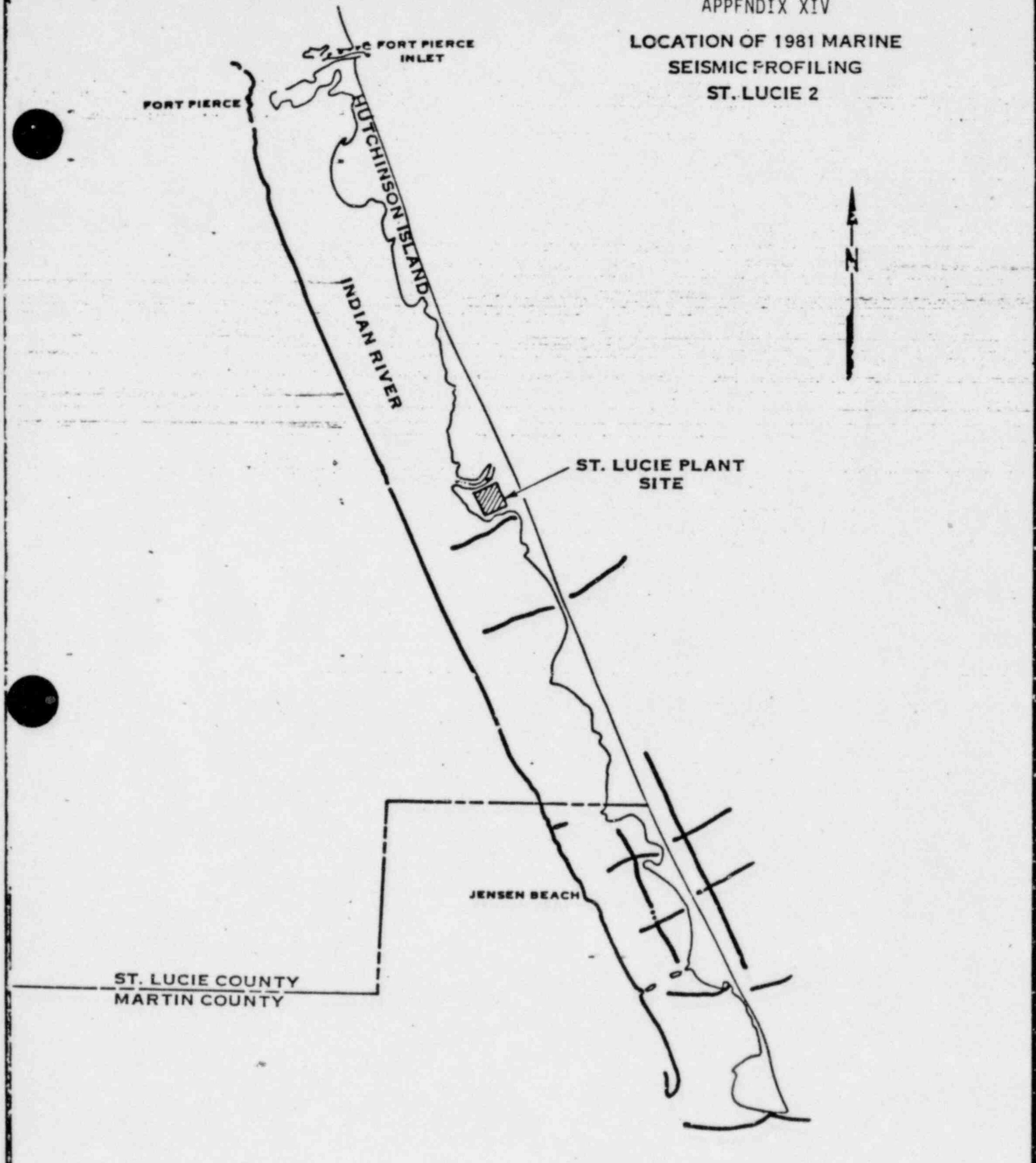
CONCLUSIONS

- PRESENT POPULATION DENSITY FOR ST. LUCIE IS TYPICAL OF OTHER NUCLEAR POWER PLANTS.
- END-OF-LIFE DENSITY SLIGHTLY EXCEEDS REG. GUIDE 4.7 "TRIP" LEVELS.
- END-OF-LIFE POPULATION PROJECTED TO BE ABOVE AVERAGE, BUT NOT BEYOND RANGE OF OTHER PLANTS AT THAT TIME.

A-1/40

APPENDIX XIV

LOCATION OF 1981 MARINE
SEISMIC PROFILING
ST. LUCIE 2



FRANK MIRAGLIA
492-7243
11/09/81

BRIEFING OUTLINE

- BACKGROUND
- ERRORS DETECTED TO DATE
- PG&E PROPOSED REVERIFICATION PROGRAM
- TENTATIVE STAFF CONCLUSIONS
- PG&E'S REVERIFICATION PROGRAM PROPOSED BY STAFF
- STAFF ACTIONS
- SCHEDULE

A-14/2

FRANK MIRAGLIA
492-7243
11/09/81

o BACKGROUND

- SEISMIC DESIGN ERRORS DETECTED IN SEPTEMBER

- FUEL LOADING OF PLANT DEFERRED

- HQ'S MEETING WITH PG&E ON OCTOBER 9
    - INITIAL STAFF REQUIREMENTS
        . TECHNICAL REPORT ON REANALYSIS AND MODIFICATION
        . REVERIFICATION OF SEISMIC DESIGN OF ALL SAFETY-RELATED SYSTEMS
        . PRIORITY REVERIFICATION OF SEISMIC DESIGN INVOLVING URS/BLUME

- PRELIMINARY TECHNICAL AUDIT OCTOBER 14-16

- REGION V INSPECTION ACTIVITIES OCTOBER 14-23

- HQ'S MEETING WITH PG&E ON NOVEMBER 3

FRANK MIRAGLIA
492-7423
11/09/81

- ERRORS DETECTED TO DATE
  - INAPPROPRIATE APPLICATION OF CONTAINMENT ANNULUS "DIAGRAM"
  - INCORRECT DISTRIBUTION WITHIN PG&E OF CORRECT SEISMIC RESPONSE SPECTRA
  - INCORRECT WEIGHT AND WEIGHT DISTRIBUTIONS FOR EQUIPMENT, COMPONENTS AND PIPING IN CONTAINMENT ANNULUS.
  - ERRORS, UNRELATED TO ABOVE, DETECTED DURING RE-ANALYSIS EFFORT
  - ERRORS, DETECTED DURING REVERIFICATION EFFORT

A-14/c

FRANK MIRAGLIA
492-7243
11/09/81

PG&E'S REVERIFICATION PROGRAM

- o COMPLETE REVERIFICATION OF SEISMIC DESIGN OF SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS BY AN INDEPENDENT CONSULTANT RETAINED BY PG&E

  - DRAFT PROGRAM OUTLINE PROVIDED TO ALL PARTIES (BN-81-38)

  - EFFORT INCLUDES REVIEW OF ALL URS/BLUME PG&E INTERFACES ON A PRIORITY BASIS

- o INTERNAL QA/QC REVIEW BY PG&E OF ALL PRE-1978 SERVICE CONTRACTS RELATED TO SEISMIC DESIGN OF SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS

  - REVERIFICATION TO BE DONE IN AREAS FOUND TO BE DEFICIENT

A-141d

FRANK MIRAGLIA
492-7243
11/09/81

TENTATIVE STAFF CONCLUSIONS

o LACK OF RIGOR AND FORMALITY IN DESIGN CONTROL:

- THE PG&E QA SYSTEM DID NOT APPEAR TO HAVE ESTABLISHED A FORMAL INTERFACE WITH AND QA CONTROLS ON URS/BLUME

- THE PG&E QA SYSTEM DID NOT APPEAR TO ADEQUATELY CONTROL & DISTRIBUTE DESIGN INFORMATION

- THE PG&E QA SYSTEM DID NOT APPEAR TO HAVE ESTABLISHED A FORMAL INTERFACE AND QA CONTROLS ON OTHER SERVICE CONTRACTORS.

A-14/e

FRANK MIRAGLIA
492-7243
11/09/81

PG&E REVERIFICATION PROGRAM PROPOSED BY
STAFF (SUBJECT OF 50.54F LETTER)

- o PRIOR TO FUEL LOAD AND OPERATION UP TO 5% POWER
  - INDEPENDENT AND COMPLETE REVERIFICATION OF ALL SAFETY-RELATED ACTIVITIES PERFORMED BY URS/BLUME
  - IMPACT OF ALL ERRORS DETECTED ON FINAL DESIGN EVALUATED AND REANALYSIS DOCUMENTED IN TECHNICAL REPORT
  - ALL APPLICABLE MODIFICATIONS TO FACILITY DESIGN RESULTING FROM REANALYSIS AND REVERIFICATION EFFORTS BE COMPLETED

- o PRIOR TO EXCEEDING 5% POWER
  - INDEPENDENT AND COMPLETE DESIGN REVIEW AND REVERIFICATION OF ALL OTHER PRE-JUNE 1978 SERVICE-RELATED CONTRACTS
  - INDEPENDENT AND SAMPLING REVIEW AND REVERIFICATION OF PG&E INTERNAL DESIGN ACTIVITIES
  - INDEPENDENT AND SAMPLING REVIEW AND REVERIFICATION OF POST-1978 SERVICE CONTRACTS
  - ALL APPLICABLE MODIFICATIONS TO FACILITY DESIGN RESULTING FROM REANALYSIS AND REVERIFICATION EFFORTS BE COMPLETED

A-14/f

FRANK MIRAGLIA
492-7243
11/09/81

PG&E REVERIFICATION PROGRAM PROPOSED BY STAFF
(SUBJECT OF 50.54F LETTER) CONTINUED

- PG&E TO PROVIDE:
  - DETAILED PROGRAM PLAN FOR CONDUCT OF INDEPENDENT DESIGN REVERIFICATION PROGRAMS
  - DESCRIPTION OF QUALIFICATION OF PERSONNEL AND CONTRACTORS PERFORMING INDEPENDENT DESIGN REVERIFICATION PROGRAMS

9-14/9

FRANK MIRAGLIA
492-7243
11/09/81

- o STAFF ACTIONS

  - ISSUE 50.54F LETTER REQUIRING REVERIFICATION PROGRAM BY PG&E
  - MONITOR PG&E'S REVERIFICATION EFFORTS (BI-WEEKLY REPORTS FROM PG&E)
  - REVIEW PG&E'S REVERIFICATION PROGRAM PLAN
  - REVIEW QUALIFICATIONS OF PERSONNEL/CONTRACTORS CONDUCTING REVERIFICATION PROGRAMS
  - REVIEW AND AUDIT PG&E'S REVERIFICATION REPORTS
  - PERFORM TECHNICAL REVIEW OF PG&E RE-ANALYSIS
  - PERFORM SITE INSPECTIONS TO VERIFY MODIFICATIONS

A-141h

FRANK MIRAGLIA
492-7423
11/09/81

SCHEDULE

- TECHNICAL REPORT
- REVERIFICATION OF URS/BLUME - PG&E INTERFACES
- REVERIFICATION OF PROGRAM PLAN (RESTRUCTURED PER STAFF REQUIREMENTS)
- REVERIFICATION REPORTS
  - PRE-1978 SERVICE CONTRACTS
  - INTERNAL PG&E DESIGN REVIEW
  - POST-1978 SERVICE CONTRACTS

A-141i

The following pages has been deleted as:

DELETION 1

Pages: A-142 to A-144

ATTACHMENT 1

FRANK MIRAGLIA
492-7243
11/09/81

NOVEMBER 9, 1981

COMMISSION BRIEFING
ON
DIABLO CANYON

A-145

FRANK MIRAGLIA
492-7243
11/09/81

BRIEFING OUTLINE

- BACKGROUND
- ERRORS DETECTED TO DATE
- PG&E PROPOSED REVERIFICATION PROGRAM
- TENTATIVE STAFF CONCLUSIONS
- PG&E'S REVERIFICATION PROGRAM PROPOSED BY STAFF
- STAFF ACTIONS
- SCHEDULE

FRANK MIRAGLIA
492-7243
11/09/81

- o BACKGROUND
  - SEISMIC DESIGN ERRORS DETECTED IN SEPTEMBER
  - FUEL LOADING OF PLANT DEFERRED
  - HQ'S MEETING WITH PG&E ON OCTOBER 9
    - INITIAL STAFF REQUIREMENTS
      - . TECHNICAL REPORT ON REANALYSIS AND MODIFICATION
      - . REVERIFICATION OF SEISMIC DESIGN OF ALL SAFETY-RELATED SYSTEMS
      - . PRIORITY REVERIFICATION OF SEISMIC DESIGN INVOLVING URS/BLUME
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  - REGION V INSPECTION ACTIVITIES OCTOBER 14-23
  - HQ'S MEETING WITH PG&E ON NOVEMBER 3

FRANK MIRAGLIA
492-7423
11/09/81

- ERRORS DETECTED TO DATE
  - INAPPROPRIATE APPLICATION OF CONTAINMENT ANNULUS "DIAGRAM"
  - INCORRECT DISTRIBUTION WITHIN PG&E OF CORRECT SEISMIC RESPONSE SPECTRA
  - INCORRECT WEIGHT AND WEIGHT DISTRIBUTIONS FOR EQUIPMENT, COMPONENTS AND PIPING IN CONTAINMENT ANNULUS.
  - ERRORS, UNRELATED TO ABOVE, DETECTED DURING RE-ANALYSIS EFFORT
  - ERRORS, DETECTED DURING REVERIFICATION EFFORT

A-148

FRANK MIRAGLIA
492-7243
11/09/81

- ERRORS DETECTED DURING RE-ANALYSIS EFFORT
  - SINGLE ANALYSIS TO MODEL TWO PARALLEL LINES FOUND APPLICABLE TO ONE ONLY
  - TWO SMALL BORE PIPING SNUBBERS REQUIRED BY ANALYSIS BUT NOT DESIGNED
  - TWO SUPPORTS DESIGNED WITH INSUFFICIENT GAPS FOR THERMAL MOVEMENT
  - ONE SUPPORT NOT RIGID IN RESTRAINED DIRECTION
  - ONE NON-SAFETY RELATED PIPE SUPPORT NOT QUALIFIED AS REQUIRED BY SYSTEM INTERACTION STUDY

FRANK MIRAGLIA
492-7243
11/09/81

o ERRORS DETECTED DURING REVERIFICATION EFFORT

- DISCREPANCY IN SPECTRA USED IN REGENERATIVE HEAT EXCHANGER

- BUILDING MASSES FOR AUXILIARY BUILDING

- CONTROL ROOM DESIGN TWO MODELS USED

- CONDUIT SUPPORTS MISAPPLICATION OF SPECTRA

- HVAC EQUIPMENT MISAPPLICATION OF SPECTRA

FRANK MIRAGLIA
492-7243
11/09/81

PG&E'S REVERIFICATION PROGRAM

- o COMPLETE REVERIFICATION OF SEISMIC DESIGN OF SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS BY AN INDEPENDENT CONSULTANT RETAINED BY PG&E

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  - EFFORT INCLUDES REVIEW OF ALL URS/BLUME PG&E INTERFACES ON A PRIORITY BASIS

- o INTERNAL QA/QC REVIEW BY PG&E OF ALL PRE-1978 SERVICE CONTRACTS RELATED TO SEISMIC DESIGN OF SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS

  - REVERIFICATION TO BE DONE IN AREAS FOUND TO BE DEFICIENT

FRANK MIRAGLIA
492-7243
11/09/81

TENTATIVE STAFF CONCLUSIONS

o LACK OF RIGOR AND FORMALITY IN DESIGN CONTROL:

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FRANK MIRAGLIA
492-7243
11/09/81

PG&E REVERIFICATION PROGRAM PROPOSED BY
STAFF (SUBJECT OF 50.54F LETTER)

- PRIOR TO FUEL LOAD AND OPERATION UP TO 5% POWER
  - INDEPENDENT AND COMPLETE REVERIFICATION OF ALL SAFETY-RELATED ACTIVITIES PERFORMED BY URS/BLUME
  - IMPACT OF ALL ERRORS DETECTED ON FINAL DESIGN EVALUATED AND REANALYSIS DOCUMENTED IN TECHNICAL REPORT
  - ALL APPLICABLE MODIFICATIONS TO FACILITY DESIGN RESULTING FROM REANALYSIS AND REVERIFICATION EFFORTS BE COMPLETED

- PRIOR TO EXCEEDING 5% POWER
  - INDEPENDENT AND COMPLETE DESIGN REVIEW AND REVERIFICATION OF ALL OTHER PRE-JUNE 1978 SERVICE-RELATED CONTRACTS
  - INDEPENDENT AND SAMPLING REVIEW AND REVERIFICATION OF PG&E INTERNAL DESIGN ACTIVITIES
  - INDEPENDENT AND SAMPLING REVIEW AND REVERIFICATION OF POST-1978 SERVICE CONTRACTS
  - ALL APPLICABLE MODIFICATIONS TO FACILITY DESIGN RESULTING FROM REANALYSIS AND REVERIFICATION EFFORTS BE COMPLETED

FRANK MIRAGLIA
492-7243
11/09/81

PG&E REVERIFICATION PROGRAM PROPOSED BY STAFF
(SUBJECT OF 50.54F LETTER) CONTINUED

- PG&E TO PROVIDE:
  - DETAILED PROGRAM PLAN FOR CONDUCT OF INDEPENDENT DESIGN REVERIFICATION PROGRAMS
  - DESCRIPTION OF QUALIFICATION OF PERSONNEL AND CONTRACTORS PERFORMING INDEPENDENT DESIGN REVERIFICATION PROGRAMS

FRANK MIRAGLIA
492-7243
11/09/81

- STAFF ACTIONS

  - ISSUE 50.54F LETTER REQUIRING REVERIFICATION PROGRAM BY PG&E
  - MONITOR PG&E'S REVERIFICATION EFFORTS (BI-WEEKLY REPORTS FROM PG&E)
  - REVIEW PG&E'S REVERIFICATION PROGRAM PLAN
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  - REVIEW AND AUDIT PG&E'S REVERIFICATION REPORTS
  - PERFORM TECHNICAL REVIEW OF PG&E RE-ANALYSIS
  - PERFORM SITE INSPECTIONS TO VERIFY MODIFICATIONS

FRANK MIRAGLIA
492-7423
11/09/81

SCHEDULE

- TECHNICAL REPORT –
- REVERIFICATION OF URS/BLUME - PG&E INTERFACES and [illegible]
- REVERIFICATION OF PROGRAM PLAN (RESTRUCTURED PER STAFF REQUIREMENTS)
- REVERIFICATION REPORTS
  - PRE-1978 SERVICE CONTRACTS
  - INTERNAL PG&E DESIGN REVIEW
  - POST-1978 SERVICE CONTRACTS

ATTACHMENT 2

November 6, 1981

SECY-81-636

POLICY ISSUE
(Notation Vote)

FOR: The Commissioners

FROM: William J. Dircks, Executive Director for Operations

SUBJECT: RE-EVALUATION OF SEISMIC DESIGN OF CONTAINMENT ANNULUS EQUIPMENT OF DIABLO CANYON UNIT 1 AND RELATED DESIGN REVERIFICATION PROGRAM

PURPOSE: To obtain Commission approval for transmittal of the attached 10 CFR 50.44(f) letter to the Pacific Gas & Electric Company

DISCUSSION: In late September 1981 errors in the seismic design of containment annulus equipment of Diablo Canyon Unit 1 and were detected by Pacific Gas and Electric Company (PG&E) and reported to the NRC. As a result PG&E has initiated a reanalysis effort of the seismic design of containment annulus equipment. Meetings on this matter were held with PG&E in Bethesda, Maryland, on October 9 and November 3, 1981. A preliminary audit of PG&E's reanalysis efforts were conducted by the staff in San Francisco on October 14 through 16, 1981. In addition, recent inspections were conducted by NRC Region V at PG&E and its seismic consultant URS/Blume. As a result of these activities,

Contact:
Frank J. Miraglia, NRR
X27243

the staff has developed a proposed design reverification program to be conducted by PG&E. The principal elements of this program were discussed with PG&E at the November 3, 1981 meeting. The attached letter provides PG&E with details of the proposed design reverification program and formally requests PG&E to provide the information resulting from the conduct of this program.

It should also be noted that as a result of these recent activities the Governor of California has provided the Commission his views on the matter in a letter to the Chairman dated October 30, 1981.

RECOMMENDATION: It is recommended that the Commission approve the transmittal of the attached letter to PG&E.

William J. Dircks,
Executive Director for Operations

Enclosure

cc: SECY
OPE
OGC
OCA

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Monday, November 23, 1981.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT November 16, 1981, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DISTRIBUTION
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Commission Staff Offices
Exec Dir for Operations
Exec Legal Director
ACRS
ASLBP
Secretariat

ENCLOSURE

A-159

Mr. Malcolm H. Furbush
Vice President - General Counsel
Pacific Gas & Electric Company
P. O. Box 7442
San Francisco, California 94106

Dear Mr. Furbush:

Subject: 10 CFR 50.54(f) Request for Statements Concerning the Re-evaluation of Seismic Design of Containment Annulus Equipment of Diablo Canyon Unit 1 and Related Design Reverification Program

In late September 1981 errors in the seismic design of equipment in the containment annulus in the Diablo Canyon Unit 1 were detected by Pacific Gas and Electric (PG&E) and reported to the NRC. As a result PG&E has initiated a reanalysis effort of the seismic design of this equipment. To date the following errors have been detected:

- o Inappropriate application of a containment annulus "diagram"
- o Incorrect distribution within PG&E of correct seismic response spectra
- o Incorrect weight and weight distribution of various equipment, components and piping at different elevations in containment annulus area.

At the October 9, 1981 meeting between PG&E and the NRC in Bethesda, Maryland, on this subject, PG&E was requested to provide the NRC the following reports:

(1) A technical report that discusses the re-analysis of the seismic design of the structures, systems and components in the containment annulus of Diablo Canyon Unit 1. This technical report would provide the basis for concluding that the proposed modifications to the design of affected equipment and components in the containment annulus would assure that the original acceptance criteria for the facility have been met.

(2) A reverification of the seismic design of all safety-related seismic design activities performed under the PG&E-URS/Blume contract as they relate to the Hosgri reanalysis.

(3) A reverification of the seismic design of all safety-related structures, systems and components. A program plan for this reverification effort was to be provided for NRC staff review at an early date.

Based upon recent NRC Region V inspections conducted at PG&E and URS/Blume Offices in San Francisco, the NRC has identified the following Quality Assurance program weaknesses related to these errors and to the performance of the Diablo Canyon Unit 1 facility design and the implementation by PG&E of applicable criteria of Appendix B of 10 CFR Part 50.

o the PG&E Quality Assurance Program did not appear to effectively excercise control over the review and approval of information passed to and received from URS/Blume

o the PG&E Quality Assurance Program did not appear adequately to control the distribution of design information within affected PG&E design groups

o the PG&E Quality Assurance Program did not appear to define and implement adequate Quality Assurance controls on other service related contracts.

As a result of the above and our discussions with you at our November 3, 1981 meeting, on this subject, you are requested to submit written statements signed under oath or affirmation, in accordance with 10 CFR 50.54(f) of the Commission's regulations, to enable the Commission to determine whether or not your license should be modified, suspended or revoked. Specifically, you are requested to submit the following information to the NRC on the schedules indicated below:

(1) At least 30 days prior to the date you plan to proceed with fuel loading and operations up to 5% power, provide the following information:

(a) The results of an independent and complete design review and verification program of all safety-related activities performed under the PG&E-URS/Blume contract as they relate to the Hosgri reanalysis. This information should address the development, accuracy, transmittal, and use of information both within PG&E and within URS/Blume, as well as the transmittal of information between PG&E and URS/Blume.

(b) A technical report that fully assesses the basic cause of all errors identified and their impact upon the final design of the facility.

(c) A schedule for completing any modifications to the facility that are required as a result of the design review described in (1)(a) above. For modifications that will not be completed prior to fuel load, the bases for proceeding shall be provided.

(2) At least 30 days prior to the date you plan to proceed with operations above 5% power, provide the following information: A-162

(a) The results of an independent and complete design review and verification program of all safety related structures, systems, and components that received design input information or data developed by PG&E service-related contractors prior to June 1, 1978. This review should address all activities involved in the development, accuracy, transmittal, and use of safety-related information, both within the PG&E organization and within each contractor's organization, as well as the transmittal of information between PG&E and each contractor.

Information to be provided as a result of this design review program should include the following:

(i) Your review of overall quality assurance procedures used for all safety-related service contractors (pre-1978), with particular emphasis on the applicable criteria of Appendix B. of 10 CFR Part 50. Areas where weaknesses exist should be identified. Steps taken by PG&E to correct these weaknesses, if any, also should be identified.

(ii) The development of a network for the design chain for the safety-related structures, systems, and components involved. This should include all interfaces you have identified where design information was transmitted between PG&E design groups and contractors.

A/63

(iii) Your review of the implementation of design verification procedures used by and for:

- o PG&E internal design groups
- o contractors
- o transmittal of information between PG&E and contractors
- o transmittal of information within PG&E design groups

In addition the information should be sufficiently complete to enable a determination as to whether or not the procedures conform to Appendix B quality assurance requirements and whether or not specific areas of weakness in the design process have been identified.

(iv) The criteria for the selection of a sample of safety-related structures, systems and components for reverifying the design process. The sampling criteria provided should be based on verifying the design in the areas of identified weaknesses from the procedure review discussed in (2)(a)(i) through (iii) above. Criteria for expanding the sample size when problems are identified should also be provided.

(v) The development of conclusions on the effectiveness of this design verification program, the significance of design errors found, and their impact on facility design.

(vi) A schedule for completing any modifications to the facility that are required as a result of the design review program described in (2)(a) above. For modifications that will not be completed prior to operation above 5% power, the basis for proceeding shall be provided.

(b) The results of an independent design sampling review and verification program of PG&E internal design activities that have been performed on Diablo Canyon Unit 1 related to the development of the design of safety related structures, system, and components. The extent of the information provided related to this program should be that which is necessary to confirm that the PG&E quality assurance controls described in their QA Manual and associated procedures since 1970, have been fully and effectively implemented.

This design review program should include the following:

(i) Your review of overall quality assurance procedures used in the area of design verification for safety-related structures, systems and components. Material provided related to your review should include areas where weaknesses may exist. A network for the design chain for the safety-related structures, systems and components should be identified, including all interfaces where design information was transmitted between PG&E design groups. You should provide information concerning your review of the implementation of the design verification program within the various design groups, including the specific procedures for verification and transmittal of design information internal to PG&E.

(ii) The criteria for the selection of a sample of safety-related structures, systems and components to verify the implementation of QA controls on the design process. The statements to be provided concerning this sampling criteria should be based on verifying the design in the areas of identified weaknesses from the procedure review discussed in (2)(b)(i) above. Criteria for expanding sample size should also be established as discussed in (2)(a)(iv) above.

(iii) the development of conclusions on the effectiveness of the design verfication program, the significance of design errors found, and their impact on facility design.

(iv) Statements to be provided should include your schedule for completing any modifications to the facility that are required as a result of the design review described in (2)(b) above. For modifications not to be completed before operation above 5% power, the bases for proceeding shall be provided. A-166

(c) You also should submit information concerning the the results of an independent design sampling review and verification of all PG&E service related contractor work that was completed subsequent to January 1, 1978 and that has been or will be used as input into the the design of safety related structures, systems, and components. The extent of this information should be that which is necessary to confirm that the contractor and PG&E quality assurance controls and procedures that were in effect during this time period were fully and effectively implemented.

Statements concerning this design review program should address all interface activities associated with the work, both internal to the contractor and within PG&E. The statements should also include the items discussed in (2)(b)(i) through (iii) above, and a schedule for completing any modifications to the facility that are required as a result of the design review program described in (2)(c) above. For modifications that will not be completed prior to operation above 5% power, the bases for proceeding shall be provided.

(3) At least 60 days prior to the date you plan to proceed with operations above 5% power provide:

(a) Information concerning a detailed program plan for conducting the design reverification review programs discussed in (2)(a) through (c) above. This plan should provide the bases for the service-related contracts and safety-related structures, systems and components selected for review, the selection of sampling plans to be used, and the criteria for modifying sample sizes.

(b) A description of the qualifications of the personnel and contractors who are to perform the independent design reviews discussed in (2)(a) through (c) above.

(4) Starting on Friday November 13, 1981, a semi-monthly status report on the second and fourth Friday of each month, on the ongoing reanalysis and reverification efforts being conducted by PG&E.

In the interest of efficient evaluation of your submittals, we request that you submit, as soon as practicable, a response to the request for additional information that was enclosed in the Staff's Meeting Summary dated October 19, 1981. of the October 14-16, Meetings with PG&E.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

ATTACHMENT 3

State of California

GOVERNOR'S OFFICE
SACRAMENTO 95814

EDMUND G. BROWN JR.
GOVERNOR

(916)445-2843

October 30, 1981

Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Palladino:

I am writing in reference to the recent disclosures of design, construction, and quality assurance errors at the Diablo Canyon nuclear power plant. I ask that you and your fellow commissioners immediately order an independent audit of the earthquake protection and other safety-related features of this plant.

Your prompt action in this regard would serve not only the welfare of California and its citizens, but it would also be in the best interests of the NRC. PG&E and the NRC staff have in the past repeatedly asserted that Diablo Canyon met all requisite safety standards. The disclosures of serious errors by PG&E at Diablo -- disclosures made within the very week that you licensed operation of the plant -- have undermined the NRC's credibility. As a consequence, I submit that the public will simply not believe the results of any audit performed by PG&E or the NRC. Indeed, for such an investigation to be received with any credibility, it must be performed by a team of truly independent experts who have no stake, real or apparent, in the outcome.

Accordingly, I am enclosing with this letter a workable proposal for an independent audit that focuses on the quality of the actual design and construction of the Diablo Canyon plant.

California's citizens have every reason to expect that the NRC now take swift and decisive regulatory action to protect their health and safety. It is clear that the Commission made a mistake last month in licensing Diablo Canyon. I ask that you take action to rectify that mistake, and that you order an independent audit and revoke the PG&E license for low power testing.

Sincerely,

Edm G Brown Jr.

EDMUND G. BROWN JR.
Governor

Enclosure

cc: Commissioners

A-169

October 29, 1981

MEMORANDUM

TO: Dr. Harold Denton

FROM: Counsel to Governor Brown

RE: Proposed Diablo Canyon Quality Verification Program

Since late September 1981, a number of serious errors in seismic design have been discovered at Diablo Canyon. These errors have primarily involved problems in the development, distribution, and use of design data by PG&E and its engineering services subcontractors. These errors were discussed at Commission and Staff meetings in Washington and at recent meetings with PG&E in San Francisco, California. As a result of these discussions and investigations, it is now clear that each error involved a failure of PG&E to implement properly the 18 quality assurance criteria of 10 C.F.R. Part 50, Appendix B.

Because PG&E failed to implement a Quality Assurance/ Quality Control ("QA/QC") 1/ program which satisfies Part 50, Appendix B, and because this failure led to serious errors, 2/ there is now substantial uncertainty in the actual quality level achieved in design and construction of safety-related structures, systems, and components at Diablo Canyon. This uncertainty is heightened by the Staff's forthright statement, made in light of PG&E's QA/QC deficiencies and widely reported in the press,

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1/ "Quality Assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes "Quality Control," which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

2/ The seriousness of these errors cannot be disputed. Indeed, Dr. Harold Denton stated that the low power license would not have been issued if these erros had been known to the Staff before the NRC issued the license. See Oct. 9 Meeting Transcript, p. 117.

that further analyses by experts will doubtless reveal further errors. 3/

The substantial uncertainty which surrounds the actual quality level achieved in design and construction of Diablo Canyon is clearly unacceptable under the licensing standards of the NRC's regulations. Governor Brown proposes, therefore, that the NRC immediately order that an independent quality verification program be undertaken and completed, prior to fuel loading at Diablo Canyon. 4/

The independent audit program proposed herein is not unique. Indeed, it is similar in concept to the "outgoing product quality" audit programs now used by the nation's nuclear equipment manufacturers. In such an audit, an "outgoing product quality" index is obtained by reconducting the acceptance tests and inspections on an equipment item that was previously accepted by QA as ready for shipment. The index is a useful management tool to confirm independently that the desired level of quality is actually being achieved.

The following general guidelines are suggested for an independent physical reinspection and design review of the Diablo Canyon QA/QC program for design, construction, and operation:

1. Selection of an Independent Auditor:

   After consultation with all parties in the Diablo Canyon proceeding, the NRC should select an experienced QA/QC consultant to conduct the review. The consultant must not be an employee or contractor of PG&E, Westinghouse, or any other contractor having direct responsibility for the Diablo Canyon QA/QC program. Attached hereto for your consideration is a list of firms which appear to have the technical capability to conduct the type of review described herein. These firms, of course, must be screened to assure that they have no real or apparent conflict of interest in this matter.

3/ For example, the Wall Street Journal on October 26, 1981, quoted a Staff spokesman as stating: "Obviously, if one engineer can find a problem by accident, it is reasonable to assume that an army of engineers second-guessing everything can find many more."

4/ This proposal is separate from the need for the NRC to take straightforward licensing-related action that addresses the fact that the NRC mistakenly issued the low power license to PG&E.

2. Steering Committee:

The independent auditor should perform its services pursuant to the oversight of a Steering Committee. This Committee should be composed of four persons, each with expertise on quality assurance matters. One each should be selected by PG&E, the Staff, the Governor, and the Joint Intervenors.

3. Scope of Design Review and Physical Inspection:

The assessment of the Diablo Canyon QA/QC program by the independent auditor should include:

(a) A review of QA/QC design records for, and a physical reinspection of, one electrical system (the reactor protection system is suggested) and two mechanical systems, including the structural supports, chosen from among the emergency core cooling systems (the safety injection and the decay heat removal systems are suggested). This review should cover:

(i) the designation of safety-related items to determine whether the systems, structures, and components have been properly classified;

(ii) the design verification records to assure the adequacy of design criteria bases, the adequacy of design implementation, including the internal and external transmittal, distribution, and use of design data, and the consistency between the design documents and the FSAR commitments;

(iii) training and qualification records for design, construction, and inspection personnel;

(iv) records concerning the identification and control of installed material, parts and components;

(v) records concerning the control of special construction processes;

(vi) records concerning the adequacy of disposition of non-conformances;

(vii) records of corrective action measures and timely closeouts;

(viii) PG&E audit findings, follow-ups, and resolutions;

(ix) equipment qualification records;

(x) drawing change control procedures including implementation for field changes;

(xi) comparison of "as-built" drawings to actual plant configuration;

(xii) receiving inspection and test results;

(xiii) material certifications;

(xiv) concrete strength where applicable;

(xv) visual inspection of the systems, including welds;

(xvi) cable identification and separation;

(xvii) control panel functional test results;

(xviii) verification of the torque of bolts;

(xix) non-destructive test record interpretations;

(xx) the program for control of materials, parts and components for non-safety grade portions of the systems; and

(xxi) a determination of the adequacy of the PG&E and major contractors' QA/QC programs and their implementation based on all the above.

(b) A comparison of the PG&E design and construction QA/QC program to NRC Regulatory Guides cited in the FSAR related to QA/QC activities.

(c) A review of PG&E's operational QA/QC program, including:

(i) the qualification of the Diablo Canyon QA/QC staff;

(ii) the availability of QC personnel on off-shifts;

(iii) the availability of "as-built" drawings;

(iv) the selection of replacement materials and parts for safety-related items;

(v) the applicability of the QA/QC program to replacement electrical and instrumentation components, modules, and equipment;

(vi) the handling and installation of replacement parts and materials for safety-related items;

(vii) the program for procurement of non-safety related replacement materials and parts; and

(viii) a comparison of the PG&E operation QA/QC program to NRC Regulatory Guides cited in the FSAR related to QA/QC activities.

The Governor's consultants and counsel are prepared to discuss with the Staff further details in pursuit of the foregoing proposal. We believe that a satisfactory investigation of the errors at Diablo Canyon calls for a cooperative Federal-State working relationship between the NRC and California. Both levels of government have vital interests that are at stake in bringing about a sound resolution of this matter.

LIST OF SEVERAL POTENTIAL CONSULTANTS TO PERFORM INDEPENDENT QUALITY VERIFICATION REVIEW OF DIABLO CANYON*

1. Energy, Inc.
   P. O. Box 736
   Idaho Falls, Idaho 83401

2. Gilbert Associates, Inc.
   P. O. Box 1498
   Reading, Pennsylvania 19603

3. Management Analysis Company
   11100 Roselle Street
   San Diego, California 92121

4. Project Assistance Corp.
   1 Whale Row
   New London, Connecticut 06320

5. Quadrex Corporation
   1700 Dell Avenue
   Campbell, California 95008

6. Technodyne Engineering Co.
   333 Market Street
   Suite 2735
   San Francisco, California 94105

7. Teledine Engineering Corp.
   303 Bear Hill Road
   Watham, Massachusetts 02154

8. Torrey Pines
   (A Division of General Atomic Co.)
   P. O. Box 81608
   San Diego, California 92138

9. Universal Testing Laboratories, Inc.
   579 Pompton Avenue
   Cedar Grove, New Jersey 07009

* These are intended as suggestions only, not endorsements.

State of California

GOVERNOR'S OFFICE
SACRAMENTO 95814

EDMUND G. BROWN JR
GOVERNOR

ATTACHMENT 4

916/445-1915

November 7, 1981

Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Pacific Gas and Electric Co.
(Diablo Canyon Nuclear Power Plant,
Units 1 and 2)
Docket Nos. 50-275, 50-323

Dear Chairman Palladino:

This letter is in furtherance of Governor Brown's letter to you dated October 30, 1981, in which the Governor asked the Commission to order a truly independent audit of the safety features of the Diablo Canyon Plant before the Commission permits any operation of the plant.

Attached is a list of thirteen separate seismic design and construction errors at Diablo Canyon that have been discovered since September 21, 1981. These errors, which involve a large number of systems, components, and equipment critical to safe operation of the plant, demonstrate a serious and widespread breakdown of the quality assurance program at Diablo Canyon. The errors are particularly significant because they were overlooked by PG&E and NRC inspectors for four years, during which time both PG&E and the NRC repeatedly gave assurances that the seismic design and safety features of the plant were being analyzed with the most careful and detailed attention.

The NRC's regulations, specifically the Quality Assurance requirements of 10 C.F.R. Part 50, Appendix B, are designed to ensure compliance with the NRC's technical requirements and, thereby, to detect the very types of errors that were overlooked by PG&E.*/ PG&E did not detect these errors because PG&E did not comply with Appendix B. In short, PG&E violated the NRC's regulations.

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*/ The NRC Staff pointed out to PG&E on October 9 that errors of the type discovered at Diablo Canyon should have been detected if Appendix B had been properly implemented by PG&E. See October 9 transcript, p. 87. Also, PG&E's Mr. Maneatis, Senior Vice President, stated on November 3 that PG&E's error was "the result of failure to follow established practice and represented a clear violation of our quality assurance program." Transcript, p. 131. Finally, on October 9, Dr. Denton stated that had the Staff known of PG&E's errors, the Staff would not have recommended issuance of the low power license.

11/9..To OGC to Prepare Response for Signature of Chairman and Comm Review....
Date due : Nov 23...Cpys to: Chm,Cmrs,EDO, RF, SECY..81-2313 A-176

Substantial uncertainty now exists concerning the actual degree of quality achieved at Diablo Canyon. The Staff has recognized, to some degree, this uncertainty. As a result, on November 3, the Staff directed PG&E to conduct an expanded audit of Diablo Canyon safety systems. However, the Staff's directive does not reach far enough. It does not establish the basis for a truly independent audit of the existing and potential errors at Diablo Canyon by outside experts who have no real or apparent interest in the results of their audit.

At the November 3 meeting with PG&E, the Staff did not direct an audit of PG&E's errors by independent experts in accordance with the Governor's request. Instead, PG&E was permitted to perform an audit by a consultant selected exclusively and unilaterally by PG&E. The Governor pointed out in his October 30 letter that an audit by PG&E of the very errors which PG&E itself committed and overlooked for four years simply would not be credible.

If the Commission authorizes the kind of audit permitted on November 3 by the Staff, the credibility of the audit itself and the credibility of the Commission will be undercut. We remind the Commission that any hope for public confidence in the Commission's Diablo Canyon determinations has been shattered by the recent post-licensing disclosures of errors at Diablo Canyon. We submit that a truly independent audit ordered by the Commission is the only means by which the NRC can recapture any degree of credibility.

The importance of quality assurance at nuclear power plants was recently emphasized by the Commission in the NRC's 1980 Annual Report.

> The application of disciplined engineering practices and thorough management and programmatic controls to the design, fabrication, construction, and operation of nuclear power plants is essential to the protection of public health and safety and of the environment. Quality Assurance (QA) provides this necessary discipline and control. Through a QA program that meets NRC requirements, all organizations performing work that is ultimately related to the safety of plant operation are required to conduct that work in a preplanned and documented manner; to independently verify the adequacy of completed work; to provide records that will confirm the acceptability of work and manufactured items; and to assure that all individuals involved with the work are properly trained and qualified to carry out their responsibilities. (p. 79)

These words have been put to a critical test by the multiple QA errors at Diablo Canyon. If the public is to believe that the Commission is genuinely serious about QA, then Diablo Canyon must not be permitted to operate until a truly independent audit is completed and full compliance with Appendix B is demonstrated.

Accordingly, in furtherance of the Governor's October 30 letter, we hereby ask that the Commission:

1. Order that an audit of the errors at Diablo Canyon be performed by outside experts who are independent of PG&E. At a minimum, these outside experts should be persons who have not worked for PG&E or on the Diablo Canyon project. (PG&E's current auditor, Dr. Cloud -- who was approved by the Staff on November 3 --has previously worked on Diablo Canyon.) Moreover, the outside experts should not be selected unilaterally by PG&E; they should be acceptable to all parties in the Diablo Canyon proceeding. The final selection of the independent auditor should be approved by the Commissioners, following the Staff's consultations with all parties. We have already submitted to the Staff a suggested list of nine possible independent auditors. Surely, independent firms which are acceptable to all parties can be found.

2. Order that Diablo Canyon shall not be permitted to operate until the entire audit is completed. On November 3, the staff indicated it would permit low power operation before completion of the audit. However, there is no technical or legal basis for quality assurance differentials between operation at low power or greater power. Put briefly, Diablo Canyon should not operate at any power level unless it complies with the NRC's technical requirements and regulations.

3. Order that the staff convene a working session which leads to selection of outside experts, acceptable to all parties, who will perform a truly independent audit. At such a working session, any and all parts of our October 30 proposal could be discussed, and appropriate modifications to that proposal could be evaluated by all of the parties in a cooperative atmosphere.

We note, on the basis of discussions with the Staff, that it appears that our proposal for a Steering Committee has been misunderstood. We understand that the Staff apparently regards our proposal as unacceptable because it appears to intrude upon the NRC's regulatory authority. We are sensitive to the Staff's concerns and thus wish to clarify that our proposal was intended primarly as a point of departure for discussion. We believe that this is a subject that surely can be resolved to the satisfaction of all parties at the working session.

Our intention so far has been to approach the independent audit as being an issue above and beyond the on-going adversarial proceedings. We cannot conceive how the Governor's proposal for a truly independent audit could adversely affect the interests of any party to this proceeding. In our view, PG&E should welcome such an audit, and the Commission should sieze it as a means to shore-up the damage suffered by the NRC in mistakenly licensing Diablo Canyon as a facility safe to operate.

We bring to the Commissioners' attention the fact that the following governmental bodies have already called for an independent audit: City Council of San Luis Obispo; City Council of Santa Barbara; City Council of Pismo Beach; City Council of Thousand Oaks; Board of Supervisors of Ventura County; Board of Supervisors of Santa Barbara County. In addition, several major California newspapers have editorialized in support of the Governor's proposal for an independent audit.

We believe that it would be a profound mistake for the Commission to permit an audit of PG&E's errors under the terms directed by the Staff on November 3. In actuality, such an audit amounts to a business-as-usual approach that in no way befits the extraordinary recent disclosures at Diablo Canyon. We reiterate our interest and availability to work cooperatively with the NRC in establishing the framework for an independent audit of the quality of Diablo Canyon. We ask that the Commission now take action that results in a truly independent audit worthy of belief by the affected citizens of California.

Sincerely,

Byron S. Georgiou

Byron S. Georgiou
Legal Affairs Secretary

cc: Commissioners
Service List

QUALITY ASSURANCE ERRORS REVEALED SINCE SEPTEMBER 21, 1981

In the six weeks since September 21, 1981, 13 serious errors in seismic design have been discovered at Diablo Canyon. These errors have primarily involved problems in the development, distribution, and use of design data by PG&E and its engineering services subcontractors. These errors were discussed at Commission and Staff meetings in Washington, D.C. and Bethesda, Maryland and at a series of meetings during October, 1981, with PG&E in San Francisco, California. As a result of these discussions and investigations, it is now clear that each error involved a failure of PG&E to implement properly a number of the 18 quality assurance criteria of 10 C.F.R. Part 50, Appendix B. The errors are described below:

a. Error 1 - Opposite Hand Design

On September 28, 1981, PG&E reported that a diagram error had been found in a portion of the seismic qualification of the Diablo Canyon Unit 1 Nuclear Power Plant (DCPP-1). This error resulted in an incorrect application of the seismic floor response spectra in the crane wall-containment shell annulus of the Unit 1 Containment Building. The error was that the diagram used to locate Vertical Seismic Floor Response (VSFR) spectra for the Unit 1 containment annulus was erroneous. The diagram was applicable to Unit 2 but was identified as being that of Unit 1. Since the Units are opposite hand, this resulted in an incorrect orientation of VSFR spectra for Unit 1 component and system design. The origin of the error was in the PG&E transmittal, to a subcontractor (John A. Blume and Associates), of an unverified, handwritten sketch of the Unit 2 opposite hand geometry in place of the Unit 1 geometry. 1/ (Also see Error 3).

b. Error 2 - Document Distribution

At the October 9 meeting between the NRC Staff and PG&E, PG&E disclosed that the Seismic Category I electrical cable trays and conduit supports had been qualified to design response spectra which had been superseded. The error was caused by PG&E's failure to distribute the latest revised spectra to the responsible engineer. 2/

1/ LER 81-002/01T-0, October 12, 1981.

2/ October 9 meeting transcript, pp. 105-07.

c. Error 3 - Incorrect Weights

On October 22, 1981 inspectors from the NRC's Office of Inspection and Enforcement determined that, in addition to the improper application of the diagram as reported to the NRC by PG&E on September 28, 1981, the weights listed on the diagram and used as an imput to John A. Blume and Associates for their development of response spectra, could not be verified as being accurate. PG&E representatives recalculated the weights, using current as-built drawings, and determined the new weights to be different.3/

PG&E concluded that the "substantial" weight variations resulted from three principal causes:

(i) The large bore piping equipment weights were not associated with the correct frames because the Unit 1 piping orientation was used in conjunction with the Unit 2 frame orientation.

(ii) PG&E's current calculations include additional contributors to the total weight: e.g., conduit and cable trays and steel grating, which were considered to be insignificant in the 1977 analysis.

(iii) A more detailed calculation of large bore piping weights, piping support weights and equipment weights.

d. Error 4 - Containment Spray System Pipe Supports

On September 18, 1981, the NRC's resident inspector was notified by telephone of a deficiency reportable under 10 CFR 50.55(e). 4/ The report from PG&E addressed deficiencies in the design of the containment spray system pipe supports located within containment.

The following four deficiencies were identified:

(i) An incorrect thermal analysis was used for hanger loads.

3/ PNO-V-81-59, Preliminary NRC Notification of Event or Unusual Occurrence, October 26, 1981.

4/ Letter from Crane to Engelken, October 19, 1981. Inexplicably, this error was not brought to the NRC's Commissioners' attention on September 21, 1981.

(ii) This analysis was performed with one snubber modeled as a rigid member.

(iii) The variable spring settings for the pipe supports were improperly set, based on a deadload analysis which assumed, incorrectly, that the pipes contained water.

(iv) In designing a pipe anchor, the loads from only one side were used.

The root cause of the preceding series of errors has not yet been explained by PG&E.

e. Error 5 - Wrong Spectra

During the period of October 14 through 16, 1981, representatives of the NRC staff and their consultants from Brookhaven National Laboratories met with the PG&E staff in San Francisco. During the meeting, piping problem (PG&E #6-11) was reviewed. PG&E initially asserted that this problem did not require reevaluation as a result of the opposite hand error. However, it was subsequently determined that the original PG&E calculation used erroneous spectra input and hence required reanalysis with the appropriate spectra. 5/ The cause of the error has not yet been identified by PG&E.

f. Errors 6 to 10 - Additional Design Errors

At the November 3 meeting between PG&E and the NRC, PG&E disclosed that during its internal review undertaken as a result of the diagram frame orientation error, it has identified five additional design errors requiring plant modifications from causes not related to the diagram error. These design deficiencies are:

(i) In a single case, parallel piping lines which were qualified and designed from a single analysis actually require two analyses to properly model both configurations.

5/ NRC Meeting Summary for October 14-16, 1981, Discussions and Preliminary Audit of Seismic Analysis for Equipment and Components in Diablo Canyon, Unit 1 Containment Annulus, p. 4.

(ii) In two cases, a small bore piping snubber required by analysis was not designed.

(iii) Two supports contained gaps insufficient for thermal movement.

(iv) A center support was not rigid in the restrained direction.

(v) A single support on a nonsafety-related pipe had not been qualified to Hosgri loads as required to prevent interaction with an adjacent safety-related pipe. 6/

The cause of four of the five design errors has not yet been determined by PG&E. The first error resulted from an error in judgment. The engineer looked at the two parallel-component cooling lines which run around the annulus, picked what he thought was the worst case, analyzed that worst case to determine pipe stressing and support loadings, and based on that the supports were designed for both lines. PG&E subsequently determined that the engineer had not used a conservative approach in that the engineer did not pick the worst case. Rather, the line he did not use as a model for the analysis had a longer riser section, and so it was necessary to perform a separate model for that other line. 7/

g. Error 11 - Incorrect Vertical Spectra

A discrepancy in the spectra that were applied to the regenerative heat exchanger was discovered during Dr. Cloud's review. The error was that the engineer responsible for qualifying this equipment used two-thirds of the horizontal Hosgri acceleration filtered for the tau effect. In fact he was supposed to have used two-thirds of the unfiltered spectra. The underlying cause of the error has not yet been determined by PG&E. 8/

6/ November 3 meeting transcript, pp. 138-140.

7/ November 3 meeting transcript, p. 142.

8/ November 3 meeting transcript, p. 201.

h. Error 12 - Misapplications of Hosgri Spectra

Error 12 involved misapplication of the Hosgri spectra. Electrical raceway and conduit supports are unistrut type supports, are all Class 1 equipment, and are all laterally braced. The PG&E seismic analysis is based on an enveloping procedure using static analysis. In this analysis, which contains a large number of configurations, the largest weight that a particular configuration is considered to be able to have applied to it is determined, and the highest acceleration the support can experience owing to its location in the building, is also determined. Then, with those two inputs, the first mode frequency of the supports is calculated, and the corresponding acceleration level is taken from the response spectra and the stress analysis is conducted. The misapplication errors were basically of two kinds. First, the analyst selected the wrong number off the response spectra curve; and second, in some cases the engineer apparently used one of the Hosgri spectra from a different location in the building. As before, the cause of the errors has not yet been determined by PG&E.9/

i. Error 13 - Further Spectra Misapplication

For the heating and ventilating system components, Dr. Cloud reviewed the seismic input for the fans and dampers. He found one instance where the Hosgri spectra were misapplied. Once again, the manner in which this analysis was conducted is very similar to that for the conduit supports (Error 12). The engineer confirmed that the equipment was rigid, and then went to the zero portion of the response spectra curve and selected the wrong value for the acceleration level. In this case, PG&E believes that the engineer used a spectra from a different location of the building. The cause of the error has not been determined by PG&E. 10/

9/ November 3 meeting transcript, pp. 204-205.

10/ November 3 meeting transcript, p. 206.

APPFNDIX XVI
SUBCOMMITTEE ON CALLAWAY PLANT
UNIT 1

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

November 10, 1981

MEMORANDUM FOR: M. Carbon Chairman,
ACRS Subcommittee on Callaway Plant Unit No. 1

FROM: R. Major, Staff Engineer R Major

SUBJECT: SUBCOMMITTEE ON CALLAWAY PLANT UNNIT NO. 1 MEETING OF NOVEMBER 4 & 5, 1981

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

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
J. Arnold
W. Lipinski
Z. Zudans
E. Case, NRR
E. Goodwin, NRR
G. Edison, NRR
J. Youngblood, NRR

11/10/81

PROPOSED SUMMARY
OF THE
NOVEMBER 4 & 5, 1981 MEETING OF THE SUBCOMMITTEE
ON CALLAWAY PLANT, UNIT NO. 1

PURPOSE:

The purpose of the meeting was to review the application of Union Electric Company for a license to operate the Callaway Plant, Unit No. 1. Prior to the meeting the Subcommittee and its consultants toured the Callaway Plant.

ATTENDEES:

ACRS

M. Carbon, Chairman
C. Mark, Member
J. Ray, Member
J. Arnold, Consultant
W. Lipinski, Consultant
Z. Zudans, Consultant
R. Major, Designated Federal Employee

WESTINGHOUSE ELECTRIC CORP.

J. Irons
C. Tuley
J. Swogger
J. Greshain
D. Paddleford
G. Lung
W. Luce
T. Timmons
G. Butterworth
D. Raulins
R. Cothren

SNUPPS STAFF (NUCLEAR PROJECTS INC.)

F. Schwoerer
J. Cermaks
W. Reilly
N. Petrick
R. Stright

NRC STAFF

J. Youngblood
G. Edison

UNION ELECTRIC CO.

A. Passwater
D. Schnell
D. Capone
D. Shafer
G. Pendergraff
M. Taylor
A. Neuhalfen
G. Hughes
F. Field
R. Leuther
F. Sempter
R. Wilks
T. McFarland
J. Kaelin
J. Watson
R. McAleenan
R. McLaughlin
E. Dille
S. Miltenberger
R. Dettenmeier
P. Appleby

MISSOURI DISASTER
PLANNING & OPERATIONS OFFICE

W. Johnson
E. Durham
J. Houston

ATTENDEES (CONT'D)

KCPL

J. Miller
D. Crawford

KG&E

G. Koester
G. Rathbun
F. Rhodes
J. Bailey
T. Keenan
D. Green

BECHTEL POWER CORP.

K. Lee
D. Grove
F. Roddy
P. Ward
J. Prebula
D. Gasda
J. Smith

OTHERS

R. Fluegge - Missouri Public Service Commission
A. Canger - Missouri Public Service Commission
J. Provasoli - Arizona Public Service
D. Bullmann - Dames & Moore
G. Labelle - KOMV-TV News
M. Nahrstedt - Columbia Missourian
T. Plunkett - Illinois Power Company
M. Reilly - (Self)
K. Drey - (Self)

PLANT TOUR:

Members of the Subcommittee (M. Carbon, C. Mark, J. Ray) and consultants (J. Arnold, W. Lipinski, Z. Zudans) toured the Callaway Plant with representatives from Union Electric Company and their contractors on the morning of November 4, 1981. The tour lasted approximately four hours and covered the majority of structures at the Callaway Plant.

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

1. Gordon Edison, NRC Staff Licensing Project Manager, noted the basis for the review was the SNUPPS (Standardized Nuclear Unit Power Plant

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

System) FSAR through Revision 7 and the Callaway Plant, Unit No. 1 Site Addendum through Revision 4. There are no dissenting NRC Staff technical opinions on this case.

2. D. Schnell, Vice President-Nuclear, made an introductory presentation and briefly described the site, plant, and gave the remainder of the construction schedule. Callaway is 80 miles west of the metropolitan St. Louis, Missouri area. It is 10 miles northwest of Fulton, Missouri (pop. 12,000); 25 miles west-southwest of Jefferson City, Missouri (pop. 32,000); and 30 miles west-northwest of Columbia, Missouri (pop. 59,000). The site is located on an 8 square mile plateau, 325 feet above the Missouri River flood plain and about 5 miles north of the river. The plant site is 3,188 acres. Concerning development around the site, there are no airports, industry, or military facilities.

   The Callaway Plant employs the SNUPPS concept. The power block is duplicated for a number of plants. (Since Callaway 2 was cancelled in October 1981 due to financial consideration, only the Wolf Creek Generating Station remains from the original SNUPPS group of plants.) The SNUPPS design envelope was developed by use of the most restrictive site conditions imposed by any one of the 4 original sites. The standard portions (Reactor Building, Turbine Building, Fuel Building, etc.) were designed to one standard. The plants use identical equipment and systems.

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

The NSSS for Callaway is a Westinghouse RESAR-3. (This is very similar to Comanche Peak and McGuire.) The reactor core is designed for an output of 3,411 MWt. When the reactor coolant pump input of 14 MWt is added to the core output, the warranted nuclear steam supply system output is 3,425 MWt, which is defined as the rated power in the license application. The engineered safety features are designed for a core power of 3,565 MWt. An additional 2 percent conservatism is added for some analyses to give a maximum accident analysis power of 3,635 MWt.

The turbine generator is supplied by GE. The unit rating is 1,120 MWt net output.

Normal cooling is from a closed cycle, natural draft tower. Backup cooling is from an excavated retention pond and mechanical draft cooling tower which are both seismic Category I structures.

The expected date for fuel load is June, 1983. This will be 86 months after issuance of the CP.

3. Organization and Management structures of Union Electric Company (UE) was reviewed. Callaway will be the first nuclear power plant in the state of Missouri. It is Union Electric's first nuclear unit. The management organization of the Callaway Plant is composed mainly of long-time UE personnel. Many have been with the Callaway project since its inception. There was a lack of commercial nuclear experience held by UE management. However, management did possess formal training in

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

nuclear fundamental through universities and from Westinghouse. Management has participated in nuclear industry organizations such as EPRI, EEI, and INPO. There was also some transfer of experience by participation in the SNUPPS organization which through 80% of the design phase, included utilities with operating nuclear plants. Members of management have spent time at other operating facilities as observers, and more participation at operating facilities is planned for operations management and operators. The start-up organization has had extensive experience. The NRC Staff will condition the license so that at least one operator per shift for the first year will have had commercial nuclear experience. The Staff will address the issue of commercial experience necessary for operation of the Plant at the full Committee meeting on November 12, 1981. Union Electric will describe how the transfer of experience will take place from the start-up organization to the UE operating staff during the first year of commercial operation.

4. The SNUPPS organization (Nuclear Porjects, Inc.) serves as an interface between the utilities and prime contractors (NSSS-Westinghouse, A/E-Bechtel, and Turbine-Generator- General Electric). The SNUPPS representatives felt there is a safety advantage in the SNUPPS concept from cost-sharing. A pool of additional nuclear expertise is maintained through the SNUPPS organization and experience exchange among units.

5. Callaway will have three special safety oriented groups. An Onsite Review Committee will be chaired by the plant superintendent. Members include the QA supervisor. The Nuclear Safety Review Board will

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

be chaired by the general manager of engineering. Members include the QA manager and plant superintendent. The Independent Safety Engineering Group/Shift Technical Advisors (ISEG/STA) group is based onsite. It reports offsite to the manager-nuclear engineering.

6. There will be Shift Technical Advisors (STAs) in the ISEG. All 7 will be scheduled for rotating shift duty. If required, the group will be expanded to 10. The duties of an STA are to independently observe plant status and advise shift supervision of conditions that could compromise plant safety. They will review and evaluate operating maintenance experience to improve plant safety. STAs will be responsible for the dissemination of of information to appropriate utility staff. They will review and evaluate safety-related matters assigned by corporate management or required by regulatory requirements. STAs will also report offsite to management on the overall quality of plant operations.

   ISEG/STA qualifications include an engienering or related science degree. They will have 2 years of nuclear-related experience and will have been onsite for 6 months. STAs will complete supplemental college level courses in accordance with INPO guidelines. Five STAs are currently in training.

7. The nature of the Safety Parameter Display System has not been finalized. There is still not a firm understanding of what will be required, especially concerning seismic design and necessary backup to a CRT, computer-driven system.

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

8. The startup organization was reviewed. Years of commercial nuclear experience and years of total power plant experience for startup engineers and startup management were covered. The amount of nuclear experience in the Westinghouse advisory group averages about 5 years. There is a spread of from 5 to 9 years nuclear experience in the Westinghouse Advisory Committee.

9. The operating organization for Callaway is currently 80% staffed. There will be a rotating 6-shift approach to operating staffing with one shift in training full-time, one week out of six. The 12-shift supervisors (SROs) are taking tours at other operating nuclear plants and attending commercial nuclear station experience from 6 weeks to 1 year with the average being 2 months.

10. The selection criteria of operators, as well as, training of operators for normal and off-normal situations was covered. Selection criteria for operating personnel and technicians includes a review of experience, testing in basic science and math, a psychological test series, physical examination, and an interview. The experience levels in training department were presented. Currently 20 out of 22 positions in the training department are filled. The operations training program consists of a refresher course in math and science. Phase I of the program teaches theory in nuclear fundamentals, thermodynamics, fluids and heat transfer; health physics and chemistry; and instrumentation. A research reactor is used as a teaching aid in startup operations, HP controls, and for

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

lab demonstrations. Phase II of the program consists of plant systems training and observation training at the Zion plant. Another training phase includes simulator training. The simulator is used for normal operations, transient/casualties and cold license certification exams. The final phase is onsite training, including systems, procedures, checkouts, and review.

A Callaway-specific simulator will be installed in the Callaway training center which is adjacent to the plant by February 1982. Most of the training for the plant staff will be conducted onsite.

11. In the area of emergency planning, a plan has been submitted by the Applicant. This plan is for the Callaway onsite and corporate activities only. Offsite state and local entities within the plume emergency planning zone have not yet been included. State and local plans must be submitted to NRC prior to fuel load. These plans will also be submitted to FEMA and a finding will be made relevant to the state of offsite emergency preparedness. No operating license will be issued unless NRC finds the state of onsite and offsite emergency preparedness is acceptable. Prior to fuel load, the Applicant must complete successfully, a full-scale exercise with state and local officials. This joint exercise, observed by FEMA and NRC, must be an integrated emergency exercise which will include a test of the capability of the basic elements existing within all emergency preparedness plans and organization. The full-scale exercise is currently scheduled for December 1982.

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MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

In the presentation on emergency planning, UE covered the four standard emergency classes (unusual event, alert, site emergency, and general emergency) that have been established as well as notification methods and procedures. Also covered was emergency training and drills and exercises for an emergency. A description of and manning for emergency support facilities was covered. It was noted that, in general, co-ordination with state and local organizations has been good.

12. Callaway Plant AC/DC power reliability was discussed. Covered in the discussion was the UE transmission system and stability; the Callaway AC distribution and AC/DC systems and loss of AC power to Callaway. The load flow and transient stability of the Callaway Plant was described. The grid will provide uninterrupted power to the Callaway 345 KV switchyard for a number of 3-phase fault conditions on the transmission lines between the plant and grid as well as a full load trip of Callaway. To have a total loss of AC power at Callaway, all three 345 KV transmission lines would have to be lost and both emergency diesel generators would have to fail to start. It was noted since the 345 KV transmission system has been installed, there has never been a total loss of power. If all AC was lost at the plant, still remaining would be 4 safety battery systems, 4 120 VAC vital buses, and 8 nonsafety battery systems. On loss of AC power, the operator would lose the ability to add make-up water to the RCS mass inventory. Operators would isolate the system and check that PORVs are closed. With cooldown and

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

normal seal leak there would be 100 hours before the water level reached the top of the core. Vital battery systems will operate 7-14 hours depending on the battery system and given the operator sheds some of the load. The vital batteries will operate for 200 minutes at full capacity. A management commitment has been made to provide power back to the Callaway Plant on a priority basis following a blackout.

13. The reactor vessel level instrumentation system was described. It uses two sets of two ΔP cells. Each set covers a range from the bottom of the RV to the top. In each set, one cell performs in a narrow pressure range (natural circulation). The other wide range cell detects level with any combination of operating RCPs. The system meets NRC requirements.

14. Callaway will have a thermocouple monitoring system which meets NRC requirements. The primary system measures all thermocouples. The primary system is electrically independent and has a Class IE power source up to an isolator; hardware and display beyound isolator is non-Class IE. A backup system has two channels. Each channel monitors 25 core outlet thermocouples. The entire backup system is Call IE.

15. A core cooling monitor (saturation margin) will be installed at Callaway, which meets NRC requirements. It will be a Class IE system with redundant channels. Saturation margin is determined from the lowest of three pressure signals and core outlet thermocouples and hot and cold leg RTDs.

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

16. Open items contained in the Callaway SER were discussed. There are 11 non-TMI outstanding issues and 5 TMI-related issues. Many of these issues remain open due to ongoing analyses by the Applicant that have yet to be submitted to the Staff for review or items remain open pending the completion of Staff review or an onsite audit. On the whole, there are not items of serious contention between UE and the Staff and resolution of the open items appears to be progressing orderly. These items will be presented one-by-one during the full Committee review. A list of the open items is attached.

17. There are 39 confirmatory issues listed in the SER (27 non-TMI and 12 TMI issues). Confirmatory issues are items which have essentially been resolved to the Staff's satisfaction, but for which certain confirmatory information has not yet been provided by the applicant. In these instances, the applicant has committed to provide the confirmatory information. A list of these items is attached.

18. The SER lists 20 license conditions. These license conditions may be desirable to ensure that Staff requirements are met during plant operation. The license condition may be in the form of a condition in the body of the operating licese, or a limiting condition for operation in the Technical Specifications appended to the license. However, the Staff expected most to be implemented prior to licensing and therefore will not become license conditions. The list of 20 license conditions is attached.

MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

19. Emergency procedures are being written for Callaway using Westinghouse Owners Group guidelines. There are two sets of procedures. Optimal recovery guidelines are event oriented procedures such as for reactor trip or safety injection or loss of reactor coolant. These are coupled with emergency contingency actions such as for ATWS or loss of all AC power. Also under development is a function restoration guideline set which are symptom-based procedures which include response to RCS over-pressurization, response to inadequate core cooling, and response to saturated core cooling conditions. Functional restoration guidelines are used in conjunction with a series flow diagrams (6) that can be called up on the SPDS. A particular flow diagram is selected based on plant indications, which should then lead the operator to the appropriate emergency procedures and optimal recovery path.

20. The control room design was presented. The design was basically finalized in 1978. Currently, there is still some uncertainty as to where to locate the SPDS. Because of a large number of systems and items not available for evaluation during a initial review, the Staff will further review the design after construction and installation are closer to completion.

21. Union Electric presented their systems interaction review. In addition to the initial design consideration, four additional activities address the concern of systems interaction. The four activities include a hazards analysis (fires, missiles, earthquake-induced failures, etc.), control systems failures, environmental impacts on control systems, and heavy loads analysis.

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MEETING HIGHLIGHTS, REQUESTS, AND AGREEMENTS:

22. Hydrogen control at the Callaway Plant was covered. Callaway will have redundant recombiners, redundant hydrogen mixing system, redundant hydrogen monitoring subsystem, and a backup hydrogen purge system. Callaway uses a large dry containment with a volume of 2.5×10^6 ft^3. For a hypothetical case of 75% metal-water reaction, results indicate 12.5% hydrogen by volume in containment. A constant volume and adiabatic deflagration of hydrogen yields a pressure increase of 60 psi (75 psia), the design pressure of the containment.

    The Staff was asked to present the current status of the hydrogen generation rate assumed for various plants at the full ACRS review. The Staff noted they were not overly concerned about hydrogen control in a large, dry containment.

23. The Applicant discussed the capabilities of the decay heat removal systems. Also covered was the use of feed-and-bleed cooling. It was explained that the two PORVs at Callaway are fully qualified and operate off the DC power supply.

FUTURE MEETINGS:

The full ACRS is scheduled to review Union Electric Company's application for a license to operate the Callaway Plant, Unit No. 1 during the 259th ACRS meeting on Thursday, November 12, 1981 from 1:45 p.m. until 5:45 p.m.

CALLAWAY UNIT 1
ACRS FULL COMMITTEE MEETING

November 12, 1981

ACRS FULL COMMITTEE MEETING
NOVEMBER 12, 1981
CALLAWAY UNIT 1

November 12, 1981		Name
1:45pm	Subcommittee Report	Dr. M. Carbon-ACRS
2:05pm	Management Organization and Experience	D. F. Schnell, Vice President-Nuclear
2:40pm	Startup Organization	J. F. McLaughlin, Assistant to the Vice President
		J. N. Kaelin, Superintendent-Startup
	Operations Staffing Feedback of Operating Experience	M. A. Stiller, Plant Superintendent
	Training	P. T. Appleby, Superintendent-Training
3:05pm	SER Open Issues	R. L. Stright, Licensing Manager-SNUPPS
3:50pm	Emergency Planning	N. G. Slaten, Supervising Engineer Environmental
4:10pm	Control Room Design	M. E. Taylor, Assistant Superintendent of Operations
4:30pm	Operating Procedures A) Implementation B) Instrumentation	A. P. Neuhalfen, Superintendent of Operations
5:20pm	Decay Heat Removal A) Capability and Reliability B) Bleed and Feed Operation	F. Schwoerer, Technical Director-SNUPPS

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MANAGEMENT ORGANIZATION AND EXPERIENCE

D. F. Schnell, Vice President-Nuclear

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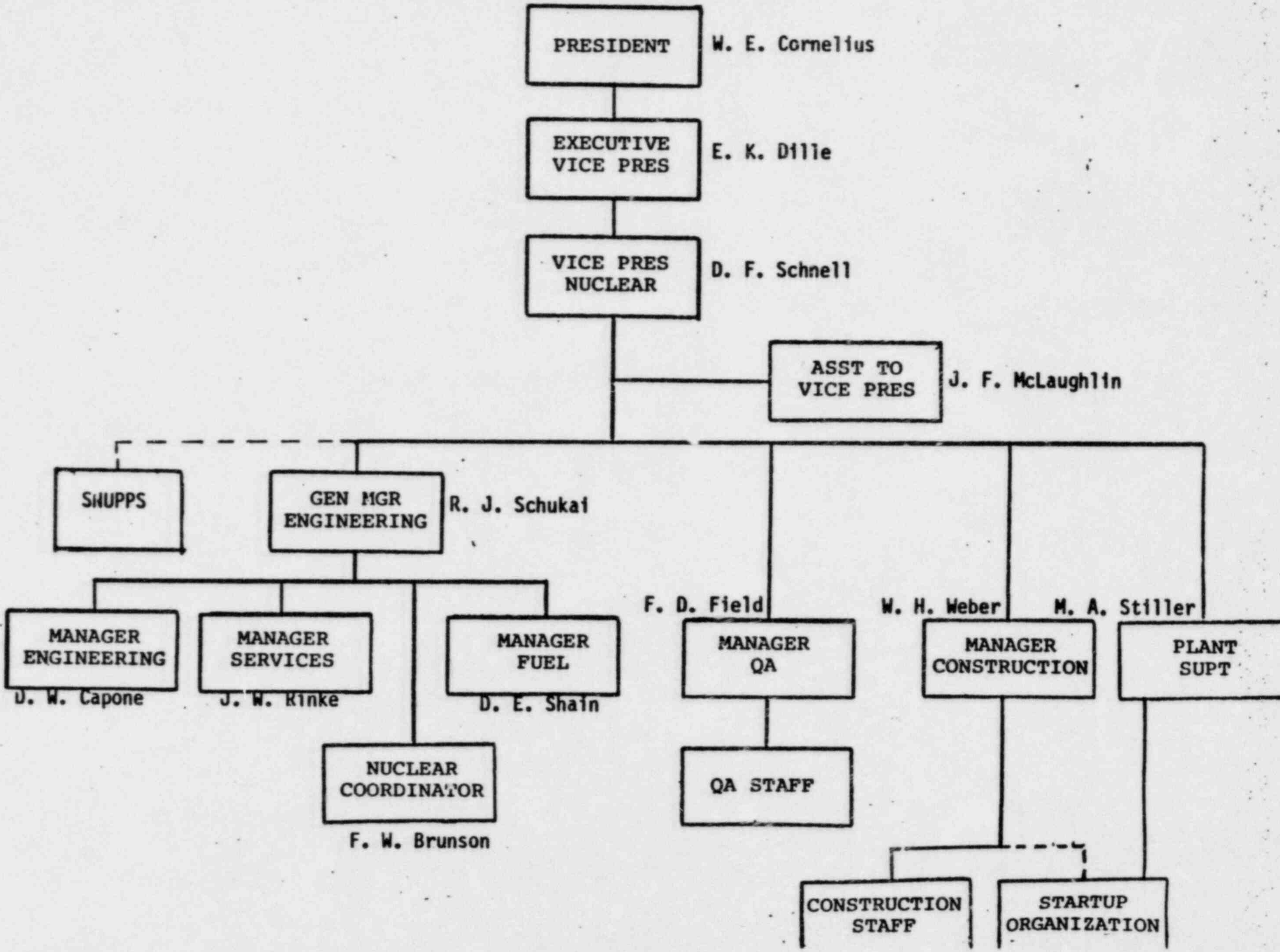
NUCLEAR FUNCTION ORGANIZATION



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TECHNICAL SUPPORT AVAILABLE FOR CALLAWAY

Engineers, Technical and Professional Employees:

- o Home Office:

Nuclear Function	60	(19)
Engineering & Construction	184	

- o Site Organizations:

Operations Staff	249	
Construction Staff	17	
Startup Organization	12	
Consultant Personnel	190	

- o SNUPPS 13

Related Organizations:

- o Bechtel
- o Westinghouse
- o KGE
- o INPO

CALLAWAY PROJECT MANAGEMENT

Position	Individual	Degree	Total Experience	Callaway Experience
Vice President Nuclear	D. F. Schnell	BS, MSME	25	10
Asst. to the Vice President	J. F. McLaughlin	BSME	33	2
General Manager Nuclear Engineering	R. J. Schukai	BS, MSEE	21	6
Manager Quality Assurance	F. D. Field	BSEE	29	9
Manager Nuclear Engineering	D. W. Capone	BSME	24	8
Plant Superintendent	M. A. Stiller	BS, MSME	28	7
Manager Nuclear Construction	W. H. Weber	BSME, MSAM	33	7
Manager Nuclear Fuel	D. E. Shain	BSEE, MBA	26	7
Manager Nuclear Service	J. W. Rinke	BS Math	15	1
Nuclear Development Coordinator	F. W. Brunson	MSME	32	5

All management personnel are registered Professional Engineers

UE NUCLEAR SUPPORT PERSONNEL*

Nuclear Engineering	Degreed Engineers	Average Total Experience	Average Callaway Experience
Nuclear	6	7	4
Electrical	4	27	6
Mechanical	8	14	3
Civil	2	19	7
Nuclear Construction			
Mechanical	7	7	1
Civil	9	14	2
QA and Other Home Office			
Nuclear	2	2	2
Electrical	5	9	4
Mechanical	13	10	4
Civil	5	6	3

*Excludes Startup & Operations

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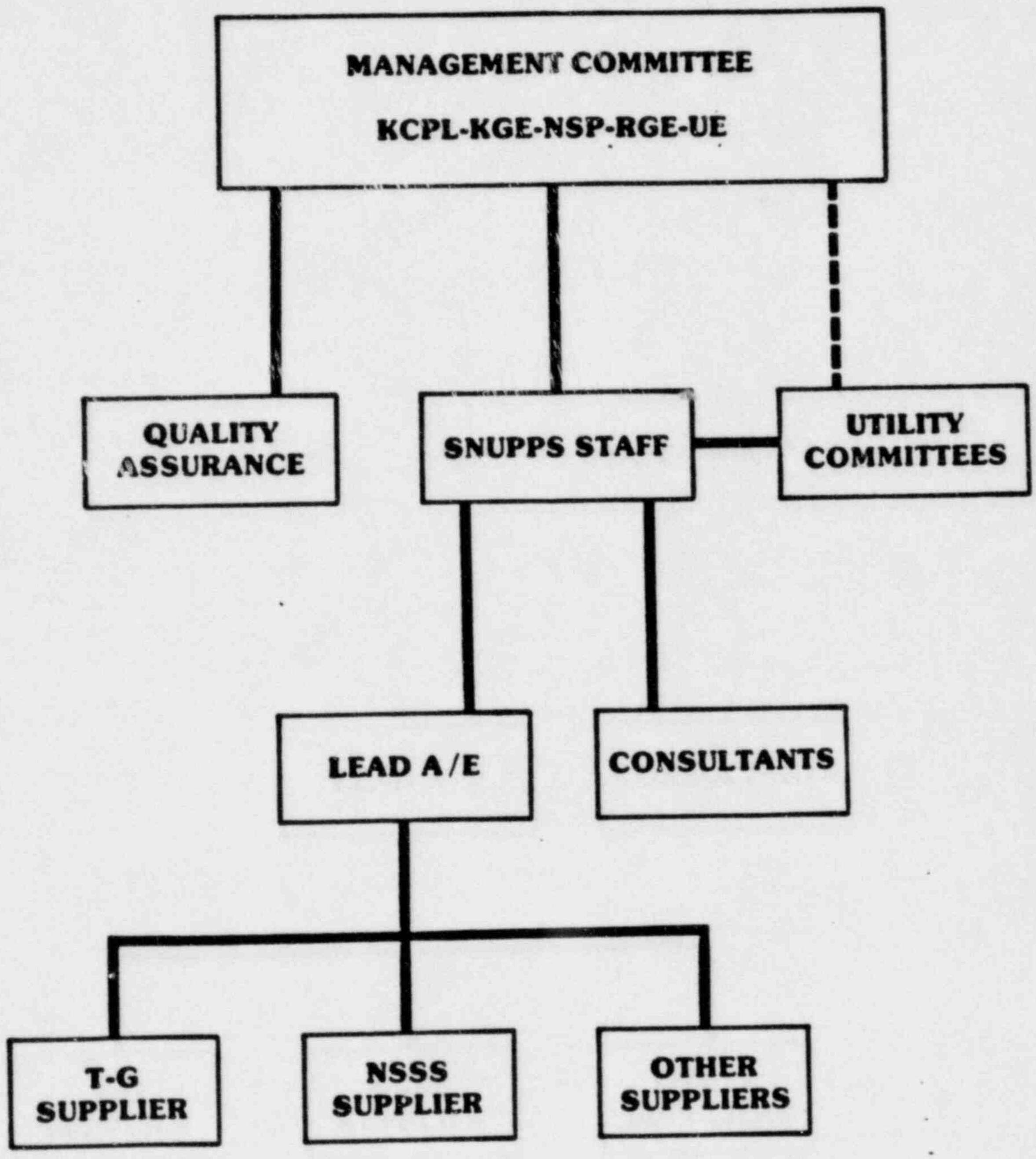
MANAGEMENT RELATIONSHIPS
(POWER BLOCK)



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SNUPPS STAFF

POSITION	DEGREE	YEARS EXPERIENCE: TOTAL PROFESSIONAL	TOTAL NUCLEAR	CALLAWAY	OTHER COMM. NUCLEAR	NON-COMM. NUCLEAR
EXECUTIVE DIR. N. A. PETRICK	BS, MSME ORSCMT, PE	37	32	8	7	17
TECHNICAL DIR. F. SCHWOERER	BS, MSME PE	37	24	7	10	7
MGR. NUCL. SAFETY J. O. CERMAK	BS, MSNE PHDNE, PE	20	20	1	18	1
MGR. TECH. SERV. E. F. BECKETT	CE, MSNE JD, PE	29	21	6	7	8
MGR. QA. S. J. SEIKEN	BS ENG. MSME	26	25	7	0	18
MGR. LICENSING R. L. STRIGHT	B.S. ENG MBA, PE	14	14	3	3	8
SITE REP R. D. BROWN	B.S. CHE PE	32	26	4	2	20
SITE REP W. R. REILLY	BS, MSNE MSMGMT, PE	29	16	5	0	11
MGR. ADMIN W. W. BALDWIN	BSME	31	7	7	0	0
MGR. MECH. EQUIP. C. W. HULTMAN	BSME	26	23	4	0	19
ENGINEER R. P. WHITE	BSEE MSNE	12	12	6	5	1
ENGINEER J. H. RILEY	BS AERO MBA	9	5	0	0	5
ENGINEER D. J. KLEIN	BSME	4	4	1	3	0
13	TOTAL	306	229	59	55	115
	AVERAGE	23.5	17.6	4.5	4.2	8.8

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SPECIAL SAFETY-ORIENTED GROUPS

NUCLEAR SAFETY REVIEW BOARD

CHAIRMAN: General Manager - Engineering

Members Include QA Manager, Plant Superintendent

ON-SITE REVIEW COMMITTEE

CHAIRMAN: Plant Superintendent

Members Include QA Supervisor

INDEPENDENT SAFETY ENGINEERING GROUP
SHIFT TECHNICAL ADVISORS

Group is based at site

Reports Off-Site to Manager - Nuclear Engineering

SUMMARY

- o Dedicated, fully-integrated project management organization since 1976

- o Participation in first multi-utility plant standardization project

- o Depth of experience

  - Management and technical staff

  - SNUPPS

- o Active participation in nuclear industry organizations

  - AEIC Power Generation Committee

  - EPRI Nuclear Divisional Committee

  - EEI Nuclear Operations Committee
    Quality Assurance Subcommittee

  - INPO Assistance Agreements

  - Westinghouse Owners Group

STARTUP ORGANIZATION

J. F. McLaughlin, Assistant to the Vice President

&

J. N. Kaelin, Superintendent-Startup

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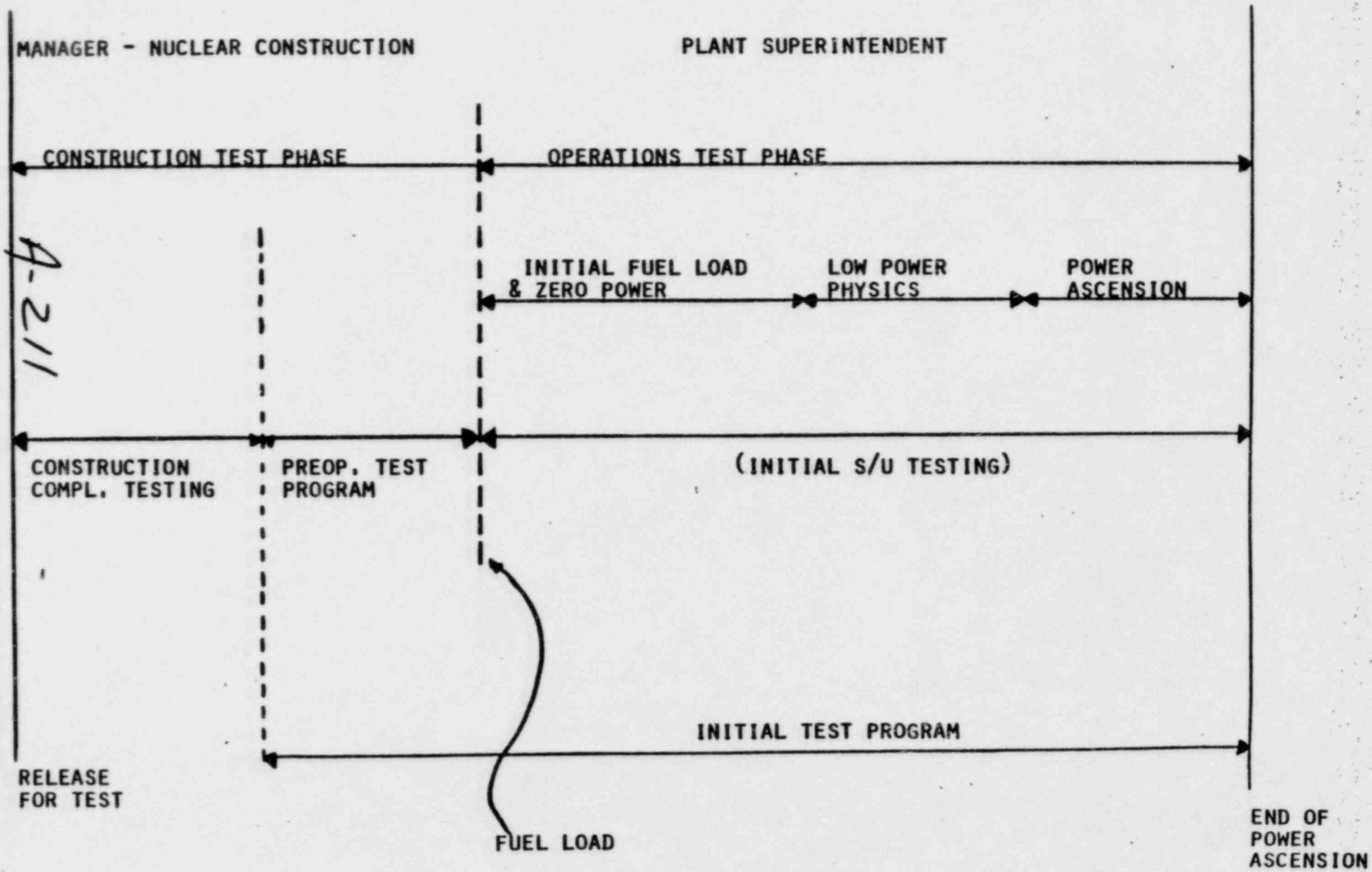
CALLAWAY TEST PROGRAM (FSAR 14.2 - SNUPPS-C)



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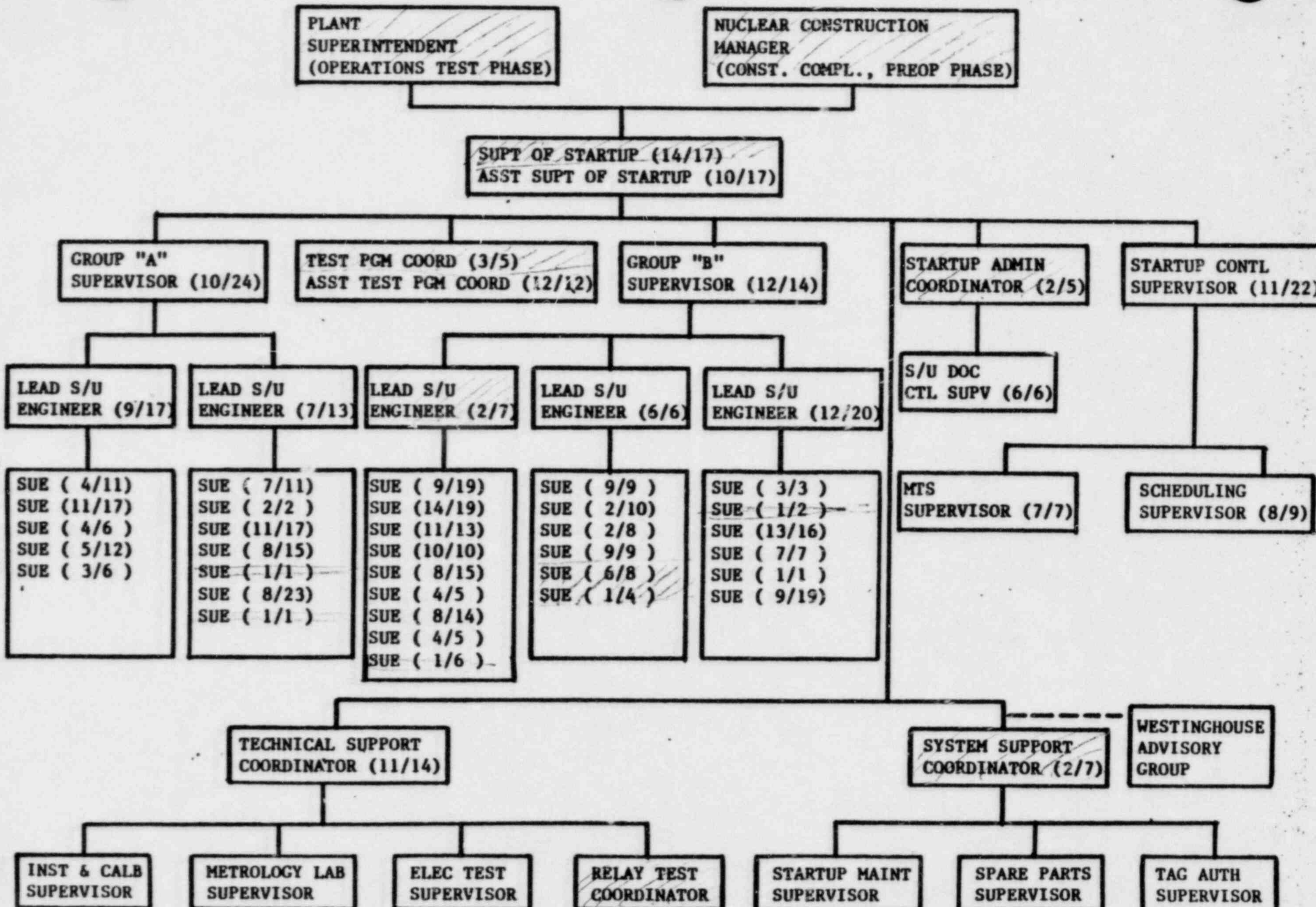

SUE = Startup Engineer
(X/Y) = (Years Commercial Nuclear Experience/Years Total Power Plant Experience)

UE EMPLOYEE

STARTUP MANAGEMENT

POSITION	INDIVIDUAL	DEGREE	TOTAL EXPERIENCE	NUCLEAR COMMERCIAL EXPERIENCE	NUCLEAR NAVY	OTHER RELATED EXPERIENCE
SOS	J. N. Kaelin	BS Chem Eng.	21	14	3	4
Asst. SOS	E. H. Smith	BSEE	17	10	7	
Supv. Group A	R. J. Daly	--	30	10	14	6
Supv. Group B	D. B. Brimmer	A. A.	15	12		2
Test Program Coordinator	W. H. Stahl	BSME	8	3		5
Startup Control Supervisor	R. S. Neal	--	30	11	11	8
Engineering Coordinator	R. K. Cothren	MSNE	19	12	4	3
System Support Coordinator	W. Sheppard	BSEE	7	2		5
Technical Support Coordinator	T. W. Holcomb	--	14	11		3
Startup Admin. Coordinator	H. J. Timms	BSEE	5	2		3

OPERATIONS STAFFING

&

FEEDBACK OF OPERATING EXPERIENCE

M. A. Stiller, Plant Superintendent

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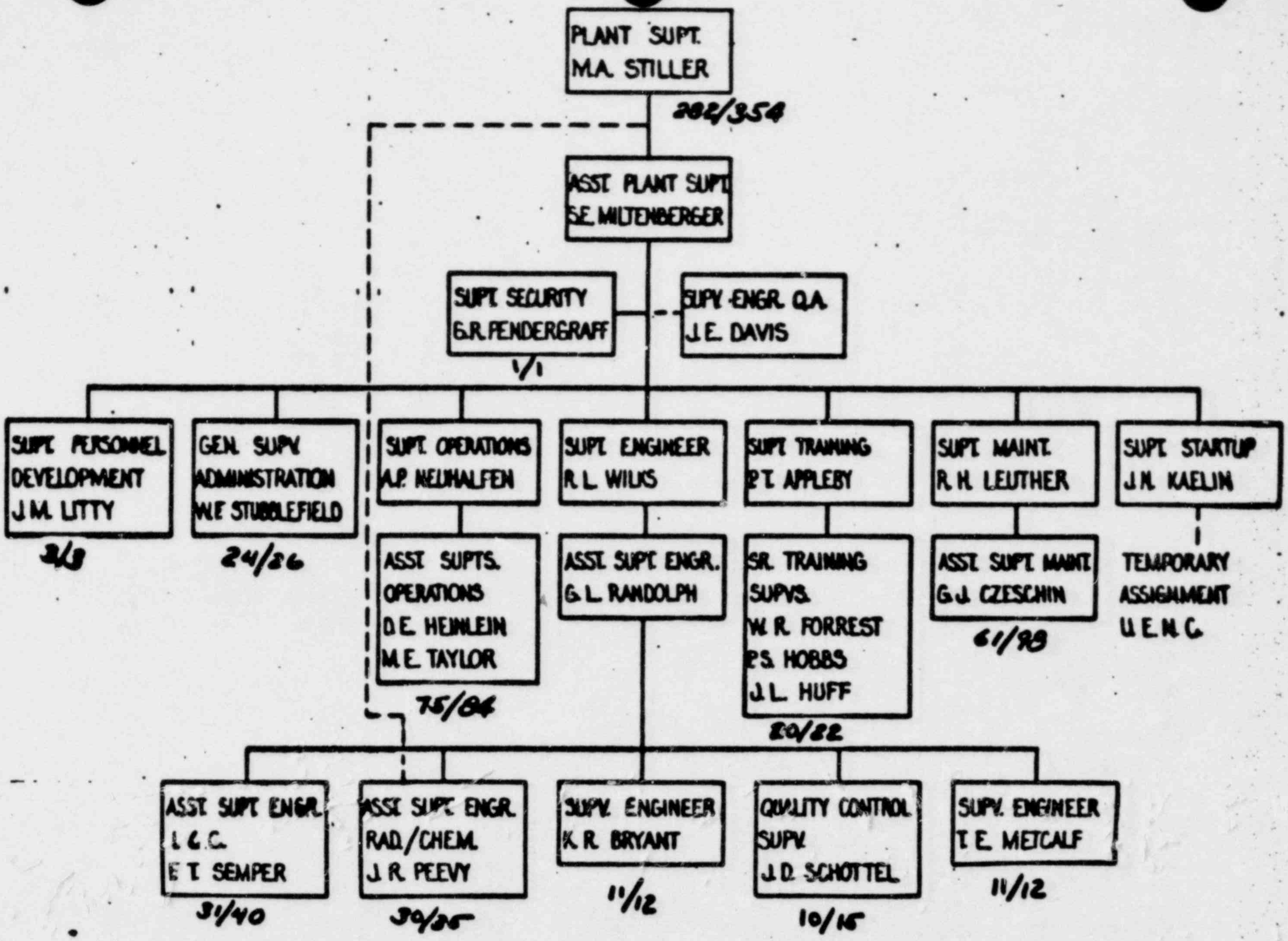

NOTE: ρ = PERSONNEL ASSIGNED 11-1-81 /BUDGET FOR FUEL LOAD

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PLANT OPERATIONS MANAGEMENT

Position	Individual	Degree	Total Experience	Callaway Experience	Other Nuclear Experience
Plant Superintendent	M. A. Stiller	BS, MSME	28	7	*
Asst. Plant Supt.	S. E. Miltenberger	BSEE, MSNE	14	2	*
General Supv. Administration	W. F. Stubblefield	BS Bus. Adm.	25	2	
Supt. Security	G. R. Pendergraff	BS Biology	10	1	4
Supt. Training	P. T. Appleby	BSNE	20	5	10*
Supt. Operations	A. P. Neuhalfen	BSEE	14	2	6*
Asst. Supt. Operations (2)	M. E. Taylor	BSME	10	3	6*
	D. E. Heinlein	BSME	10	3	*
Supt. Maintenance	R. H. Leuther	BSME	26	1	*
Asst. Supt. Maint.	G. J. Czeschin	BSME	10	2	*
Supt. Engineering	R. L. Wilks	BSME, MBA	14	5	*
Asst. Supt. Engrg.	G. L. Randolph	BSEE, MSEE	10	4	6*
Asst. Supt. Eng. Rad. Chem.	J. R. Peevy	BS Nuc. Rad.	9	3	6*
Asst. Supt. Eng. I&C	F. T. Semper	BSEE	10	3	6*
Supt. Startup	J. N. Kaelin	BS Chem E.	21	4	13
Supt. Personnel Development	J. M. Litty	-	22	2	

*Denotes assignment to comparable operating PWR for a complete refueling outage and/or a limited period of operations observation.

PLANT TECHNICAL STAFF

ONSITE STAFF	NO. DEGREED ENGINEERS	AVERAGE TOTAL EXPERIENCE	AVERAGE NUCLEAR EXPERIENCE CALLAWAY	AVERAGE NUCLEAR EXPERIENCE OTHER
Supervisors	3	11	2	9
Nuclear Engineers	8	3	2	1
Electrical Engineers	10	5	2	
Mechanical Engineers	12	6	1	3
Health Physicists/Chemists	4	8	1	4

OPERATORS AND TECHNICIANS

POSITION	NO. PERSONNEL	AVERAGE TOTAL POWER RELATED EXPERIENCE	AVERAGE NUCLEAR-RELATED EXPERIENCE CALLAWAY	AVERAGE NUCLEAR-RELATED EXPERIENCE OTHER
Senior Reactor Operator	16	12	2	4
Reactor Operator	18	8	1	4
Equipment Operator	26	7	2	5
Technical Foremen	5	13	1	9
Rad Chem Technicians	23	9	1	8
I&C Technicians	23	6	1	5

SELECTION CRITERIA

FOR OPERATING PERSONNEL AND TECHNICIANS

REVIEW OF RESUME/EXPERIENCE

BMST (BASIC MATH AND SCIENCE TEST)

IPAT (PSYCHOLOGICAL TEST SERIES)

PHYSICAL EXAMINATION

INTERVIEW

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SOURCES OF OPERATING EXPERIENCES

- INPO SIGNIFICANT OPERATING EXPERIENCE REPORTS (SOER'S)
- INFORMATION TRANSMITTED VIA INPO'S NOTEPAD PROGRAM
- CALLAWAY LER'S
- NRC LER MONTHLY REPORTS
- LICENSING INFORMATION SERVICE ADVISORIES
- NRC INSPECTION REPORTS

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TRAINING

P. T. Appleby, Superintendent of Training

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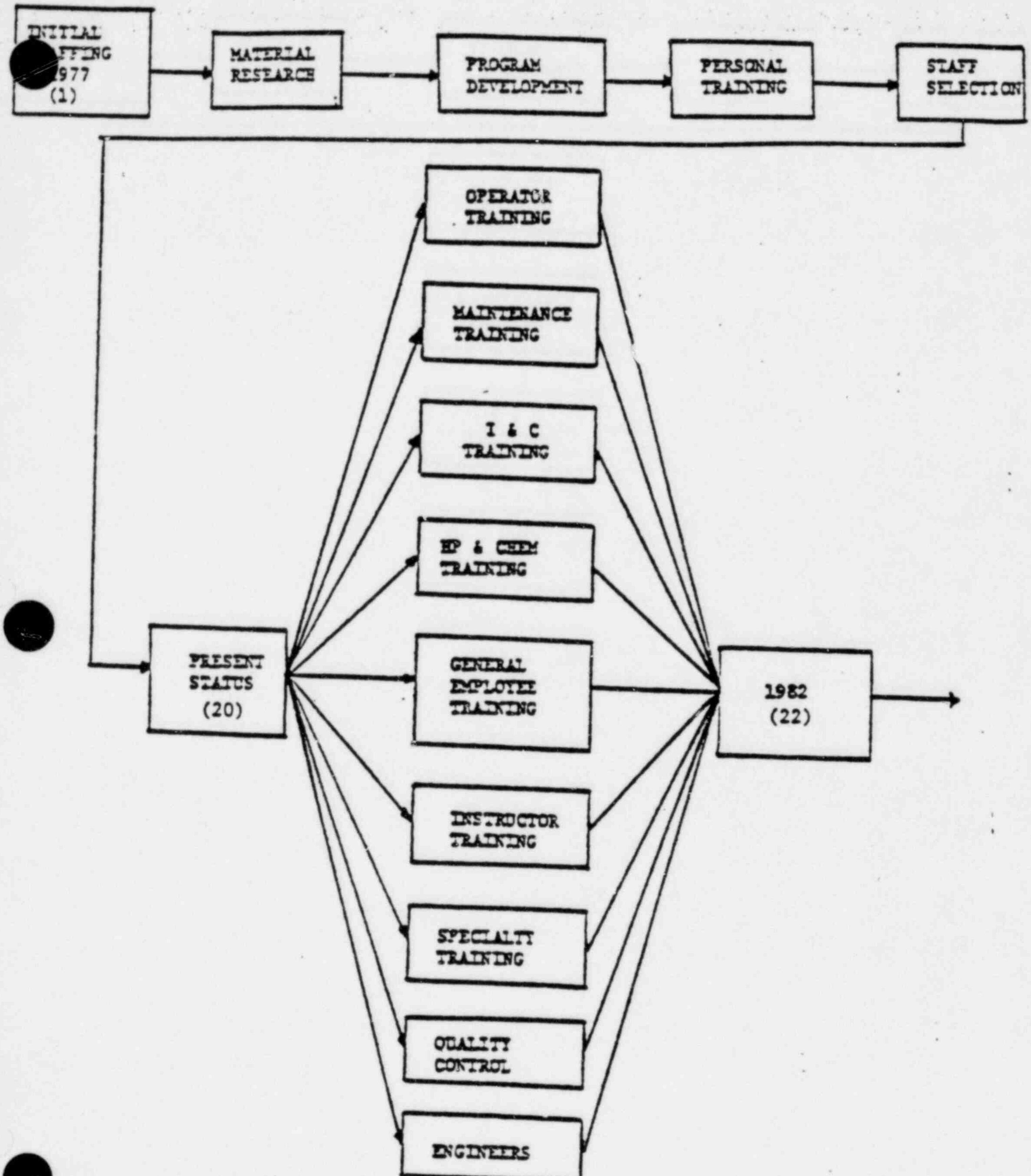
TRAINING DEPARTMENT



VG 1

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SUPERINTENDENT, TRAINING
(Sequoyah) 20 years exp.
SRO Research
SRO Certified Zion
BSNE Degree

SENIOR TRAINING SUPV.
(SRO CERT-ZION)
21 YEARS EXPERIENCE
(AA DEGREE)

TRAINING SUPERVISORS

1. 11 years exp.
   (SRO Cert. SNUPPS)
2. 10 years exp. (AA Degree)
   (SRO Cert. SNUPPS)
3. 7 years exp. (Phase I)
4. 12 years exp.
   (WPPSS)

SENIOR TRAINING SUPV.
(SRO BEAVER VALLEY & MSU)
15.5 YEARS EXPERIENCE
(COMPLETING WORK ON ME DEGREE)

TRAINING SUPERVISORS

1. 14 years exp.
   (SRO Cert. SNUPPS)
2. 20 years exp.
   (Phase II)
3. 14 years exp. (STA)
   (BS Degree)
4. 7 years exp. (MS Degree)
   (Phase I)
5. 9 years exp. (Phase II)
6. 10 years exp. (AS Degree)
   (SRO Cert. SNUPPS)

SENIOR TRAINING SUPV.
(GINNA)
19 YEARS EXPERIENCE

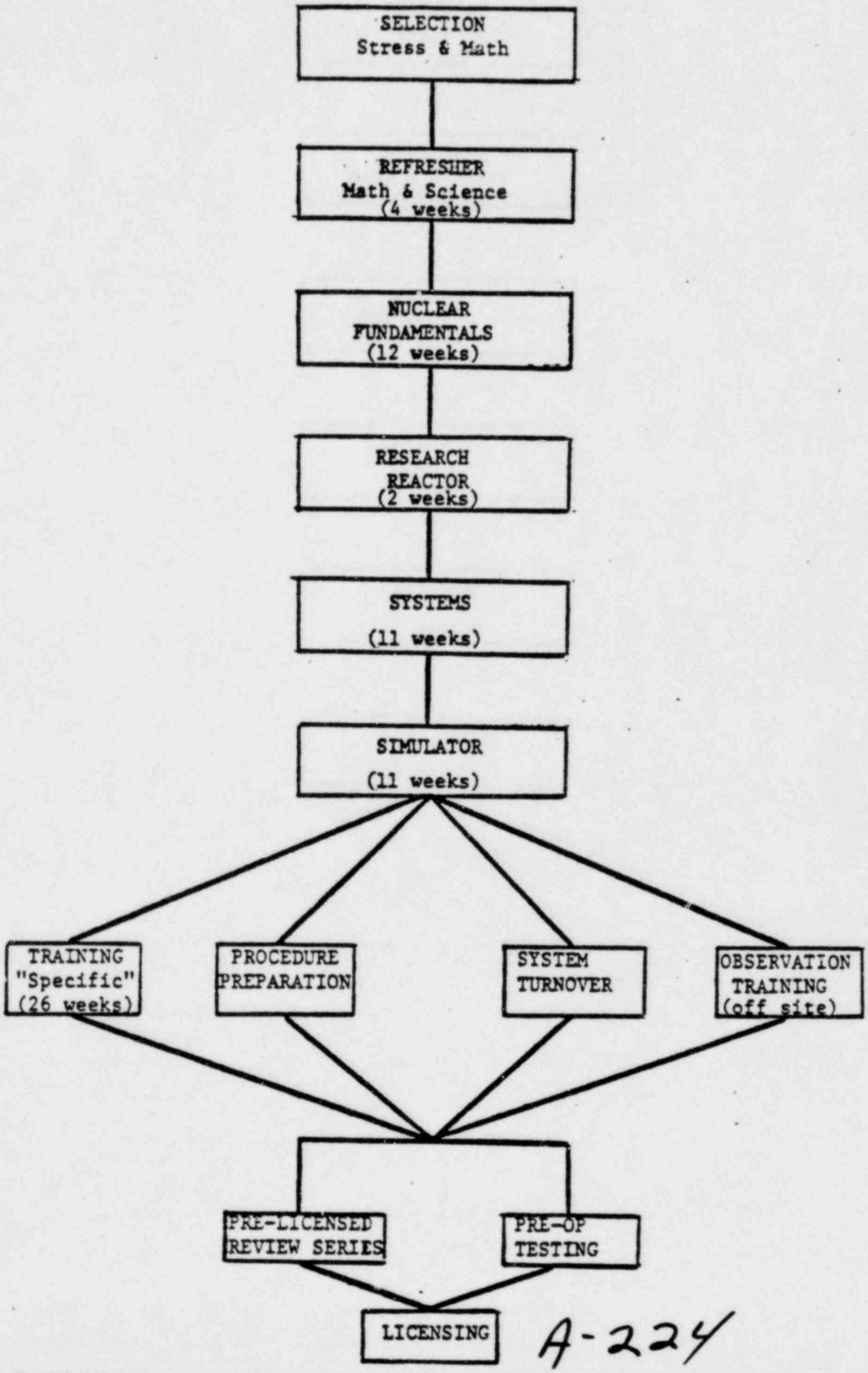
TRAINING SUPERVISORS

1. 22 years exp.
2. 9 years exp.
   (SRO Research &
   RO Oconee 1,2, &3)
3. 11 years exp.
4. 19 years exp.
5. 11 years exp.
6. 14 years exp. (BA Degree)
   (Commonwealth Edison)

TRAINING PROGRAMS

TITLE	DURATION
REFRESHER [OPERATIONS, MAINTENANCE]	
. MATH	2 WEEKS
. SCIENCE	2 WEEKS
PHASE I (THEORY) [OPERATIONS, STAFF, MAINTENANCE]	
. NUCLEAR FUNDAMENTALS	12 WEEKS
. THERMO, FLUIDS, HEAT TRANSFER	
. HEALTH PHYSICS/CHEMISTRY	
. INSTRUMENTATION	
RESEARCH REACTOR [OPERATIONS]	
. SU AND SD OPERATIONS	2 WEEKS
. HP CONTROLS	
. LAB DEMONSTRATIONS	
PHASE II (SYSTEMS) [OPERATIONS, STAFF, MAINTENANCE]	
. PLANT SYSTEMS	7 WEEKS
. OBSERVATION TRAINING AT OPERATING PWR PLANT (ZION)	4 WEEKS
PHASE III (SIMULATOR) [OPERATIONS]	
. NORMAL OPERATIONS	11 WEEKS
. TRANSIENTS/CASUALTIES	
. COLD LICENSE CERTIFICATION EXAMS	
PHASE V (ON-SITE TRAINING) [OPERATIONS]	
. SYSTEMS	≈ 26 WEEKS
. PROCEDURES	
. CHECKOUTS	
. REVIEW	

OPERATOR TRAINING PROGRAM



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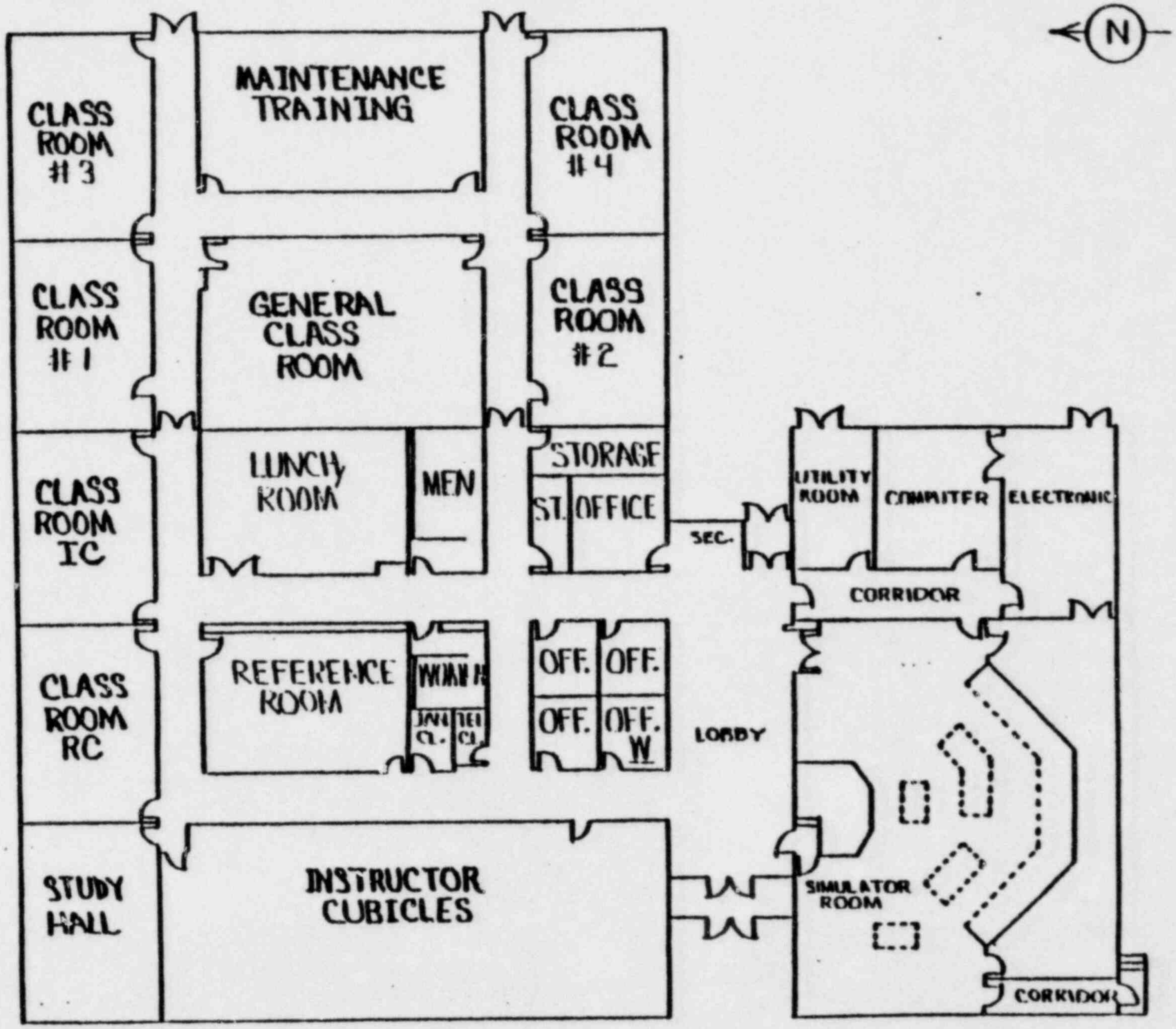

CALLAWAY TRAINING CENTER
(21,000 SQ. FT.)

STA TRAINING

I. STA RESPONSIBILITY

. EVALUATE PLANT CONDITIONS AND PROVIDE ADVICE TO THE SHIFT SUPERVISOR DURING PLANT TRANSIENTS, ACCIDENTS AND MATTERS RELATED TO OPERATIONAL SAFETY.

II. EDUCATION

A) PREREQUISITES BEYOND HIGH SCHOOL
- . MATH
- . CHEMISTRY
- . PHYSICS

B) COLLEGE LEVEL EDUCATION

. NUCLEAR SCIENCES

1) REACTOR THEORY	6 HOURS
2) CHEMICAL ENGINEERING	3 HOURS
3) ENGINEERING MATERIALS	3 HOURS
4) RADIATION DETECTION	3 HOURS
5) RADIATION SAFETY	3 HOURS
6) ENGINEERING ANALYSIS	3 HOURS

. NUCLEAR THERMOSCIENCES

1) NUCLEAR POWER GENERATION (THERMO)	3 HOURS
2) INTRO. TO NUCLEAR ENGINEERING (FLUIDS)	3 HOURS
3) NUCLEAR HEAT TRANSPORT (HEAT TRANSFER)	2 HOURS

. ELECTRICAL SCIENCES

1) PROBLEMS	1 HOUR
2) POWER SYSTEMS	3 HOURS

III. TRAINING

A) MANAGEMENT/SUPERVISORY SKILLS	1 WEEK
B) PLANT SYSTEMS	7 WEEKS
C) ADMINISTRATIVE CONTROLS	1 WEEK
D) PROCEDURES	1 WEEK
E) TRANSIENT/ACCIDENT ANALYSIS	1 WEEK
F) SIMULATOR TRAINING	2 WEEKS

CALLAWAY TRAINING SUMMARY

	COMPLETED	IN PROGRESS	TOTAL
REFRESHER COURSE	90	---	90
PHASE I (6 - 12 WEEKS)	192	28	220
PHASE II (7 - 11 WEEKS)	161	33	194
PHASE III (11 WEEKS)	39		39*
PHASE V - 1	53	---	53
I & C	---	12 @ 75%]	24
] SPECIALITY		12 @ 30%	
RAD/CHEM	---	13 @ 90%]	26
		13 @ 30%	
SS/STA	0	16	0

* CERTIFIED:

OPERATING SUPERVISORS	15 SRO	
REACTOR OPERATORS	6 SRO	7 RO
TRAINING SUPERVISORS	4 SRO	
OTHER	7 SRO	

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SER OPEN ISSUES

R. L. Stright, Licensing Manager-SNUPPS

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EMERGENCY PLANNING

N. G. Slaten, Supervising Engineer-Environmental

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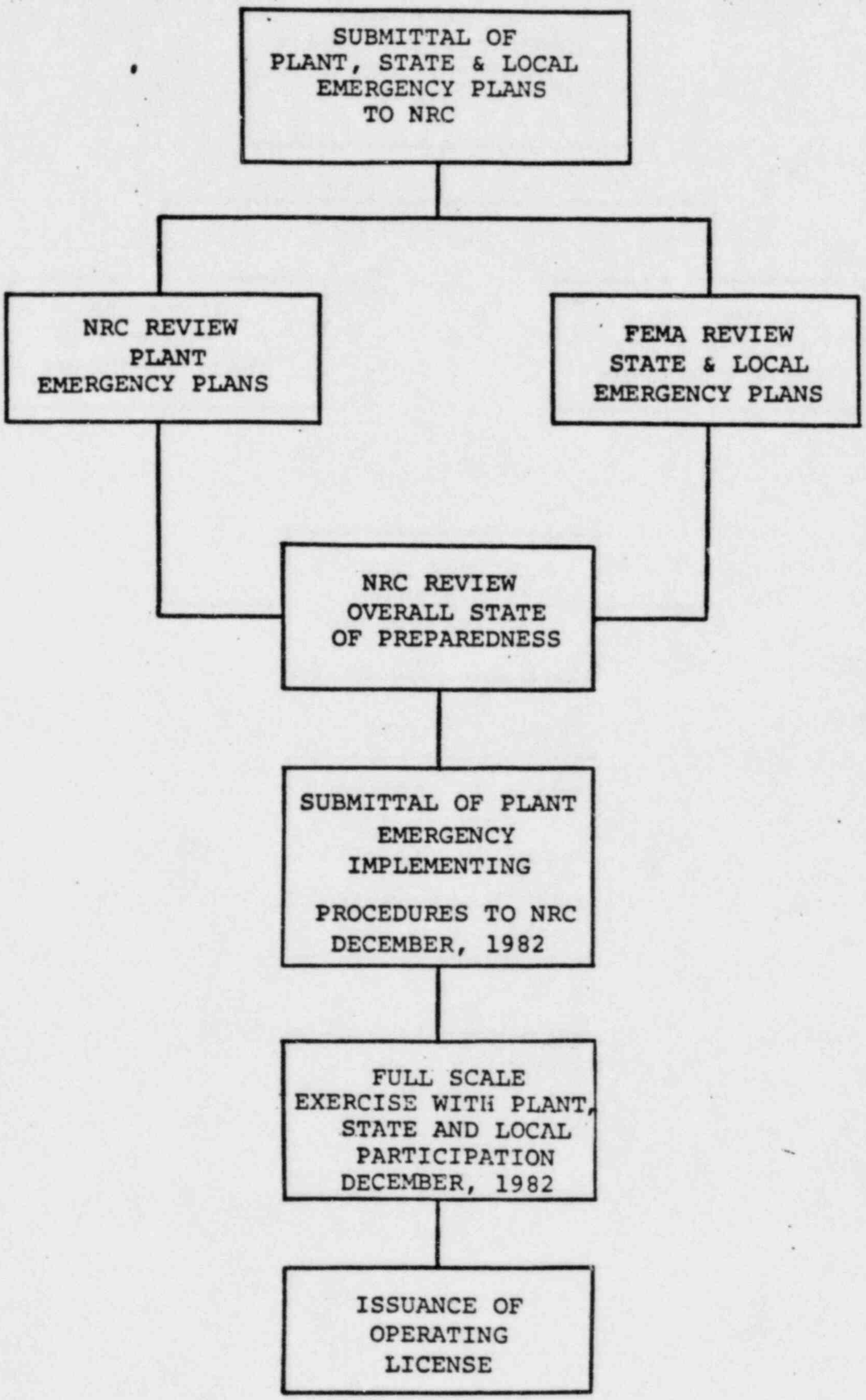
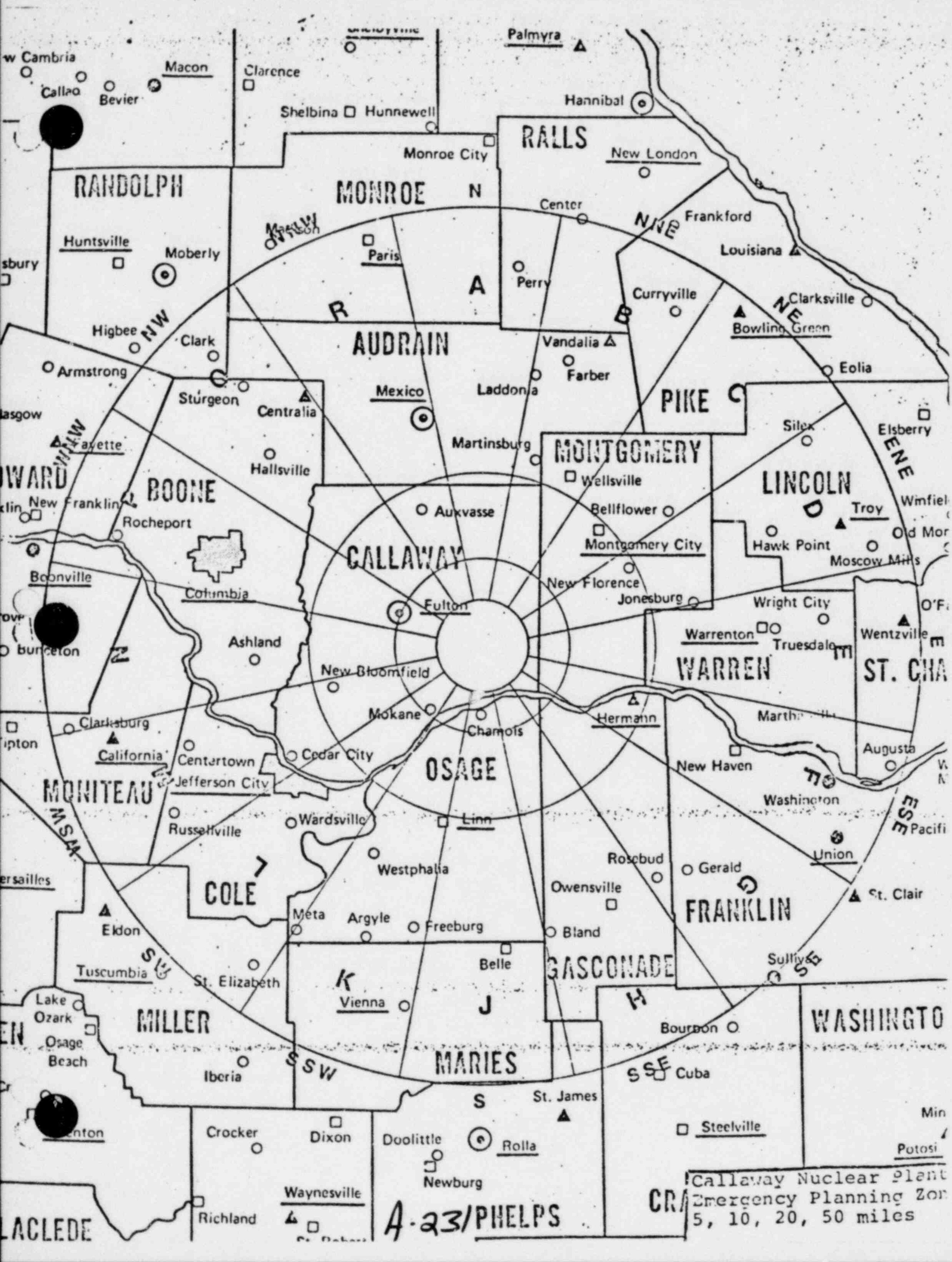

SUBMITTAL OF PLANT, STATE & LOCAL EMERGENCY PLANS TO NRC
NRC REVIEW PLANT EMERGENCY PLANS
FEMA REVIEW STATE & LOCAL EMERGENCY PLANS
NRC REVIEW OVERALL STATE OF PREPAREDNESS
SUBMITTAL OF PLANT EMERGENCY IMPLEMENTING PROCEDURES TO NRC DECEMBER, 1982
FULL SCALE EXERCISE WITH PLANT, STATE AND LOCAL PARTICIPATION DECEMBER, 1982
ISSUANCE OF OPERATING LICENSE



Callaway Nuclear Plant
Emergency Planning Zone
5, 10, 20, 50 miles

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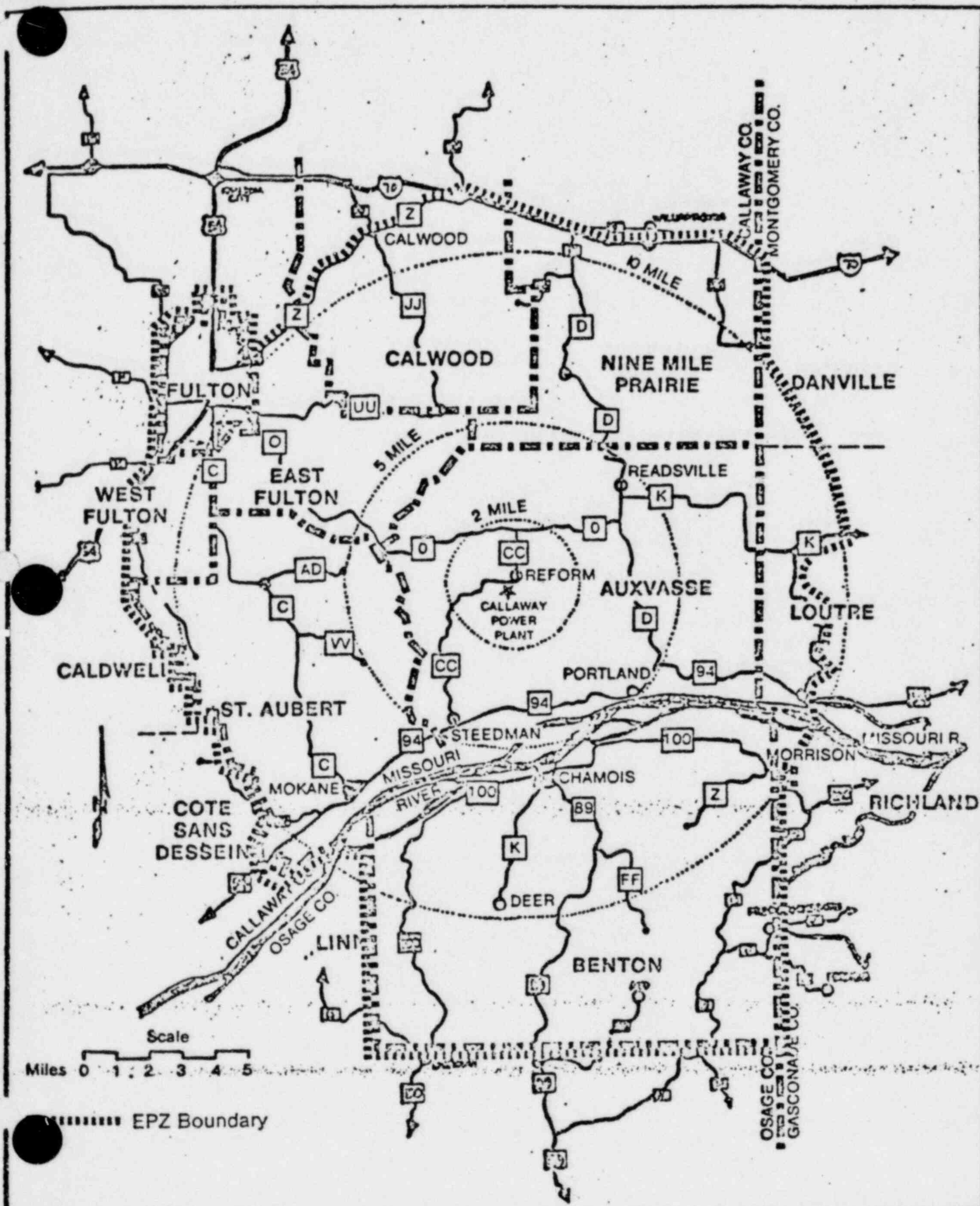

Figure 3. Proposed EPZ Boundary

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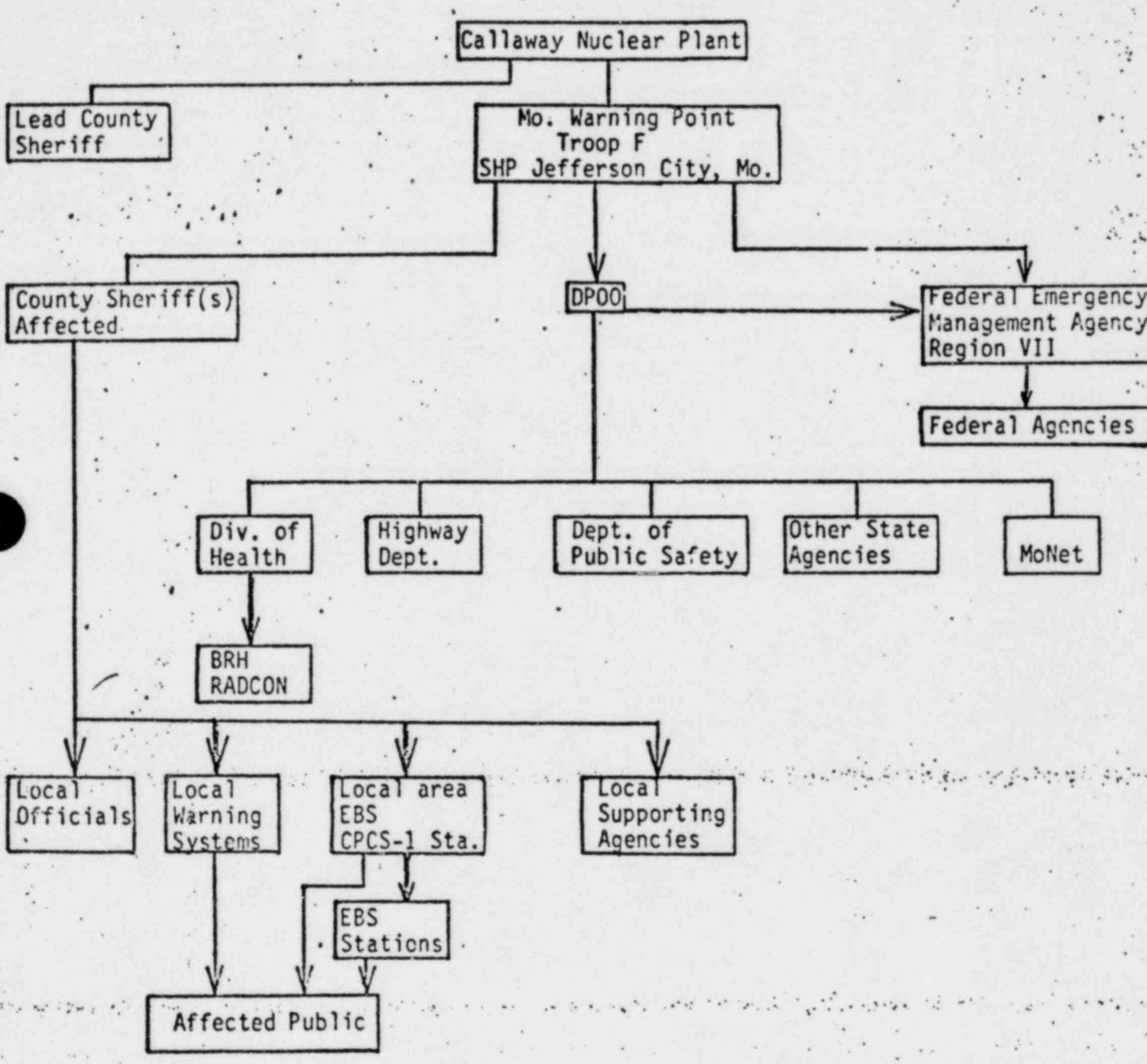
Appendix 3

NOTIFICATION FANOUT FROM CALLAWAY



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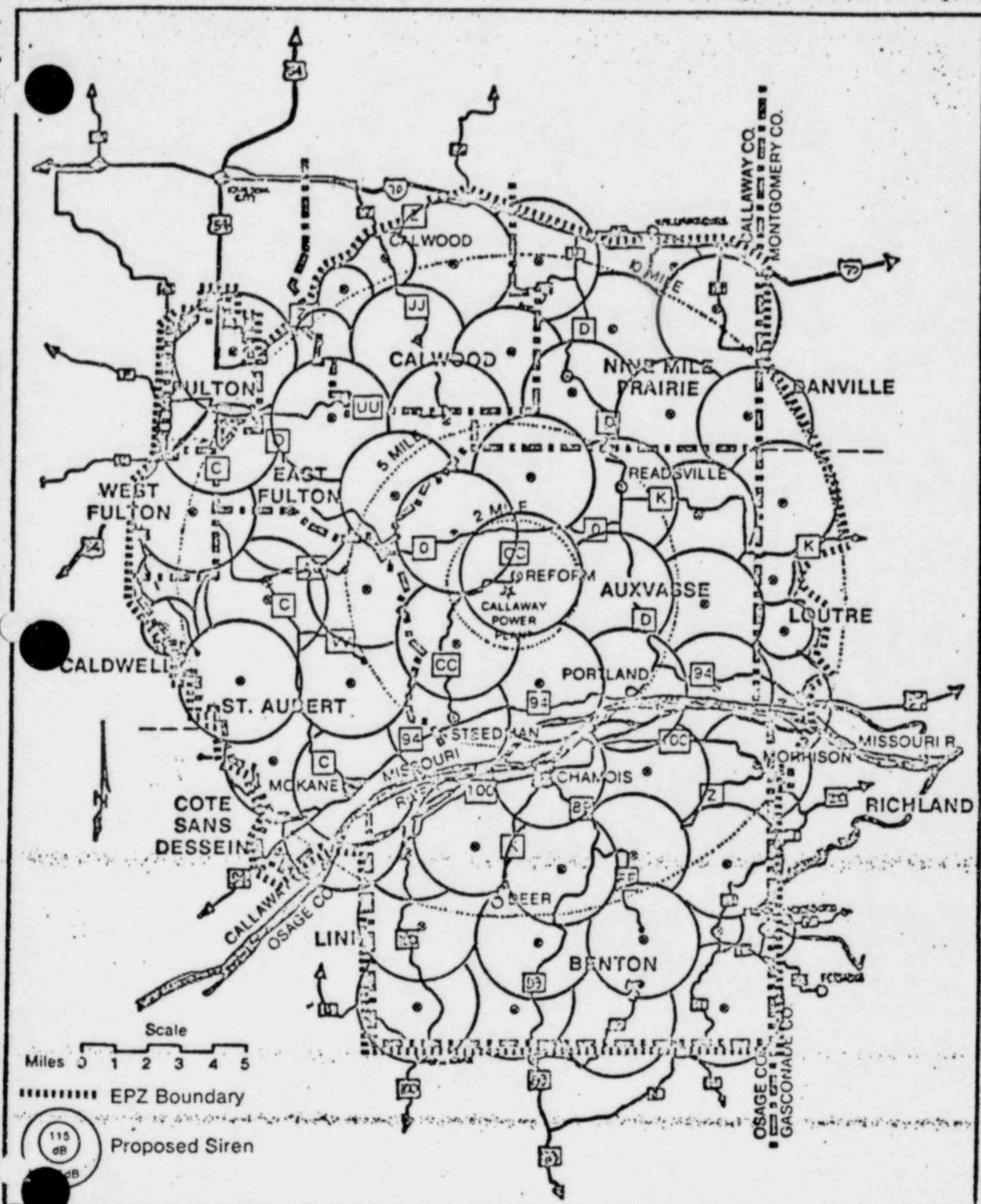


Figure 4. Proposed Siren System

DRILLS AND EXERCISES

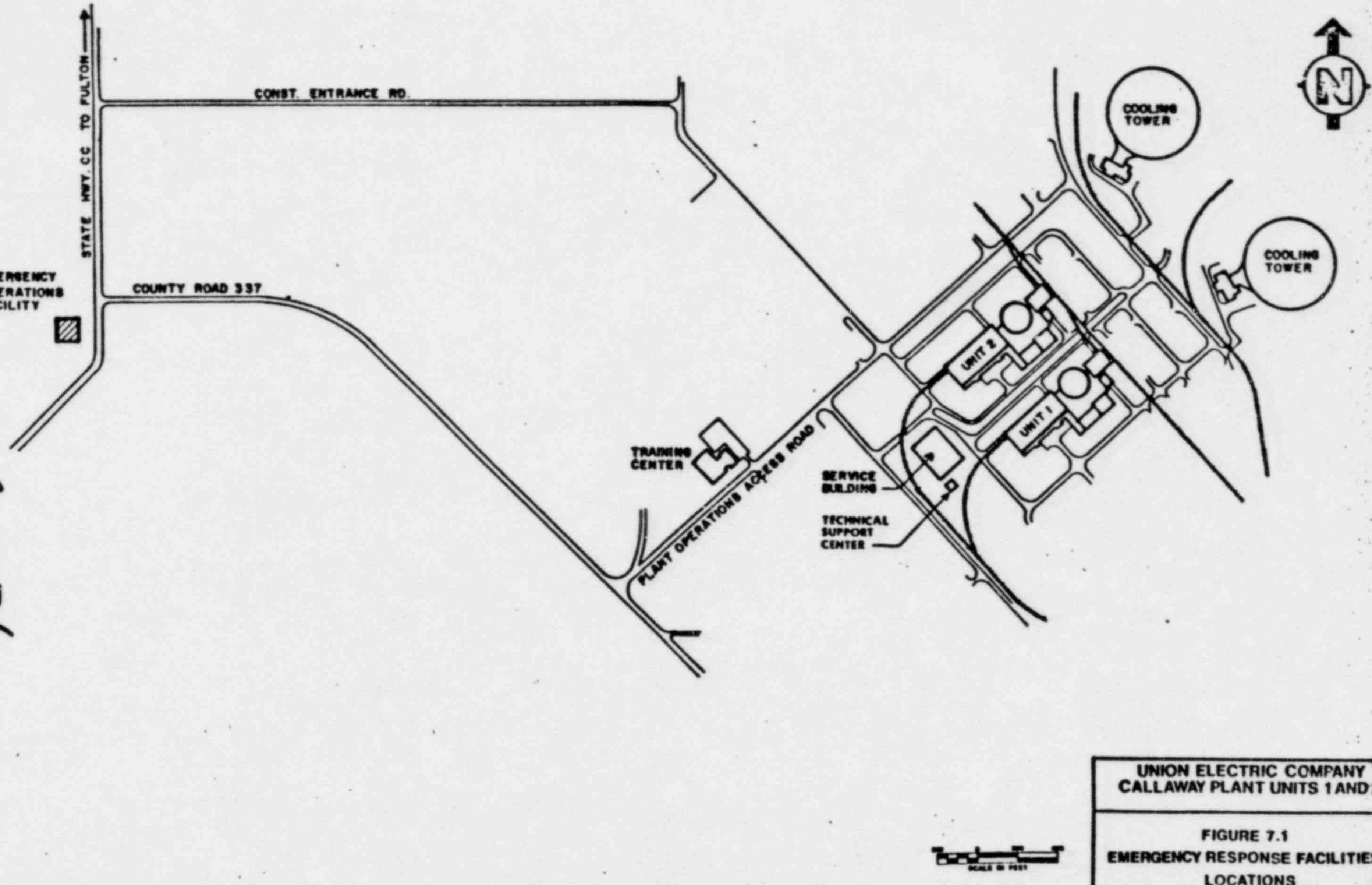
ANNUAL EXERCISE

COMMUNICATIONS DRILLS

MEDICAL EMERGENCY DRILLS

ENVIRONMENTAL AND RADIOLOGICAL MONITORING DRILLS

PUBLIC INFORMATION FORUM



UNION ELECTRIC COMPANY
CALLAWAY PLANT UNITS 1 AND 2

FIGURE 7.1
EMERGENCY RESPONSE FACILITIES
LOCATIONS

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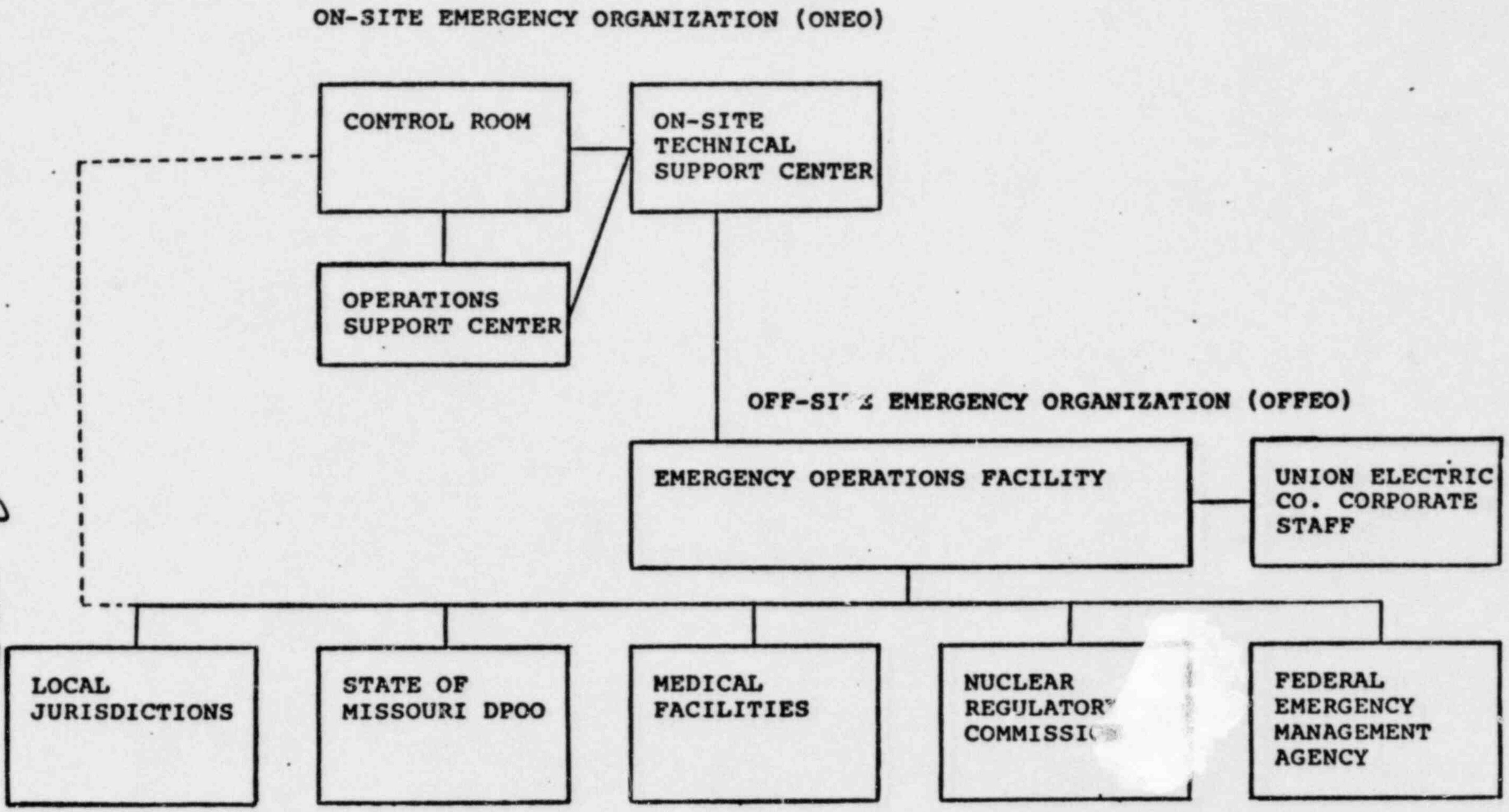

——— Normal Emergency Interface

----- Temporary Interface prior to activation of the Emergency Organization and Facilities

UNION ELECTRIC COMPANY
CALLAWAY PLANT UNITS 1 AND 2

Figure 5.3
Functional Interfaces of Emergency Organizations

CONTROL ROOM DESIGN

M. E. Taylor, Assistant Superintendent of Operations

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OPERATING PROCEDURES

A. P. Neuhalfen, Superintendent of Operations

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PREPARATION OF EMERGENCY OPERATING PROCEDURES

I. CURRENT DEVELOPMENT AND IMPLEMENTATION

II. FORMAT PHILOSOPHY

III. COORDINATED USE

IV. PROCEDURE STRUCTURE

V. EXAMPLE FORMAT

VI. STATUS TREE

VII. INSTRUMENTATION

VIII. SUMMARY

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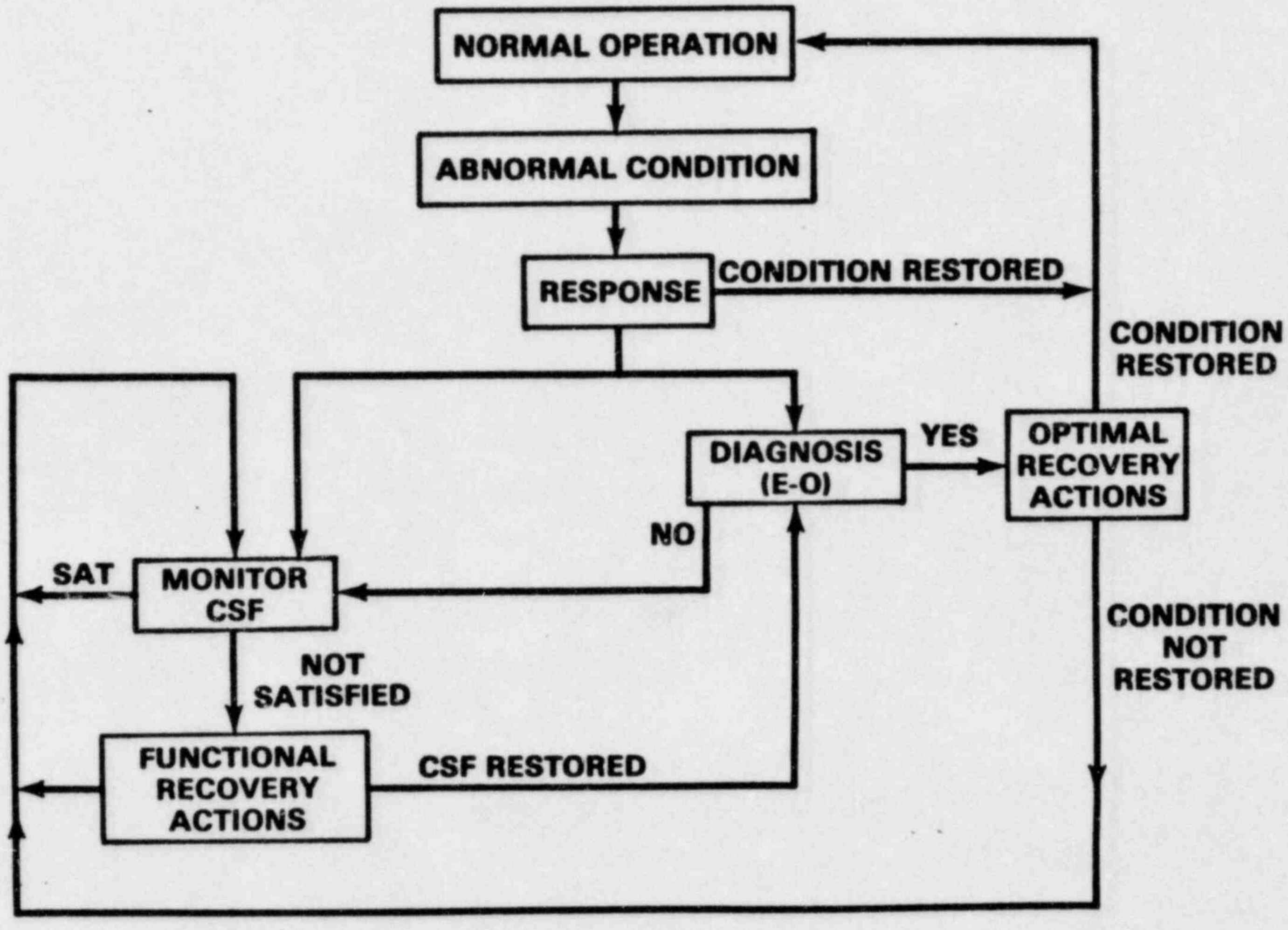
COORDINATED USE OF EMERGENCY RESPONSE GUIDELINES



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PROCEDURE STRUCTURE

I. Optimal Recovery
   1. Emergency Procedures (E)
   2. Emergency Subprocedures (ES)
   3. Emergency Contingency Actions (ECA)

II. Functional Recovery
   1. Critical Safety Function Status Trees
   2. Functional Restoration Guidelines (FRG)

TABLE 1A

OPTIMAL RECOVERY GUIDELINES

E-0 REACTOR TRIP OR SAFETY INJECTION

- ES-0.1 REACTOR TRIP RECOVERY
- ES-0.2 NATURAL CIRCULATION COOLDOWN
- ES-0.3 SI TERMINATION FOLLOWING SPURIOUS SAFETY INJECTION

E-1 LOSS OF REACTOR COOLANT

- ES-1.1 SI TERMINATION FOLLOWING LOSS OF REACTOR COOLANT
- ES-1.2 POST-LOCA COOLDOWN AND DEPRESSURIZATION
- ES-1.3 TRANSFER TO COLD LEG RECIRCULATION FOLLOWING LOSS OF REACTOR COOLANT
- ES-1.4 TRANSFER TO HOT LEG RECIRCULTATION

E-2 LOSS OF SECONDARY COOLANT

- ES-2.1 SI TERMINATION FOLLOWING LOSS OF SECONDARY COOLANT
- ES-2.2 TRANSFER TO COLD LEG RECIRCULTATION FOLLOWING LOSS OF SECONDARY COOLANT

E-3 STEAM GENERATOR TUBE RUPTURE

- ES-3.1 SI TERMINATION FOLLOWING STEAM GENERATOR TUBE RUPTURE
- ES-3.2 ALTERNATE SGTR COOLDOWN
- ES-3.3 SGTR WITH SECONDARY DEPRESSURIZATION

TABLE 1B

EMERGENCY CONTINGENCY ACTIONS

ECA-1 ANTICIPATED TRANSIENTS WITHOUT SCRAM

ECA-2 LOSS OF ALL AC POWER

- ECA-2.1 LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED
- ECA-2.2 LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED

ECA-3 SGTR CONTINGENCIES

Table 2

FUNCTION RESTORATION GUIDELINE SET (FRGs)

FR-S.1 Response to Nuclear Power Generation
FR-S.2 Response to Loss of Core Shutdown

FR-P.1 Response to RCS Overpressurization
FR-P.2 Response to High RCS Pressure

FR-C.1 Response to Inadequate Core Cooling
FR-C.2 Response to Potential Loss of Core Cooling
FR-C.3 Response to Saturated Core Cooling Conditions

FR-I.1 Response to Pressurizer Flooding
FR-I.2 Response to Low System Inventory
FR-I.3 Response to Voids in Reactor Vessel

FR-H.1 Response to Loss of Secondary Heat Sink
FR-H.2 Response to Low Steam Generator Level
FR-H.3 Response to Loss of Noraml Steam Dump Capability

FR-Z.1 Response to Containment Above Design Pressure
FR-Z.2 Response to High Containment Pressure
FR-Z.3 Response to High Containment Radiation Level
FR-Z.4 Response to High Hydrogen Concentration in Containment
FR-Z.5 Response to Containment Flooding

PRELIMINARY

Number:	Symptoms:	Revision:
E-0	REACTOR TRIP OR SAFETY INJECTION (Cont.)	061281

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	Caution *If SI flow cannot be verified, symptoms for (E²01-1) should be monitored.*	
14	**Verify SI Flow:**	
	a. Charging/SI pump flow indicator - CHECK FOR FLOW	a. Manually start pumps and align valves as appropriate.
	b. <u>IF</u> RCS pressure is less than _(1)_ psig, <u>THEN</u> check high-head SI pump flow indicators – CHECK FOR FLOW	b. Manually start pumps and align valves as appropriate.
	c. <u>IF</u> RCS pressure is less than _(2)_ psig, <u>THEN</u> check low-head SI flow indicators – CHECK FOR FLOW	c. Manually start pumps and align valves as appropriate.
	Caution *Do not throttle auxiliary feedwater flow until the water level is above the top of the U-tubes.*	
15	**Verify AFW Flow:**	
	a. AFW flow indicators – CHECK FOR FLOW	a. <u>IF</u> AFW flow <u>NOT</u> verified, <u>THEN</u> go to (E²01-2).
16	**Verify RCS Heat Removal:**	
	a. RCS average temperature – DECREASING TO _(3)_ °F	a. Manually open condenser steam dump valves —OR— b. Manually open steam generator PORVs.

(1) Enter plant specific shutoff pressure of high-head SI pumps.
(2) Enter plant specific shutoff pressure of low-head SI pumps.
(3) Enter temperature (°F) for programmed no-load temperature.

CRITICAL SAFETY FUNCTIONS LISTING

(1) SUBCRITICALITY (S-SERIES)

(2) RCS INTEGRITY (P-SERIES)

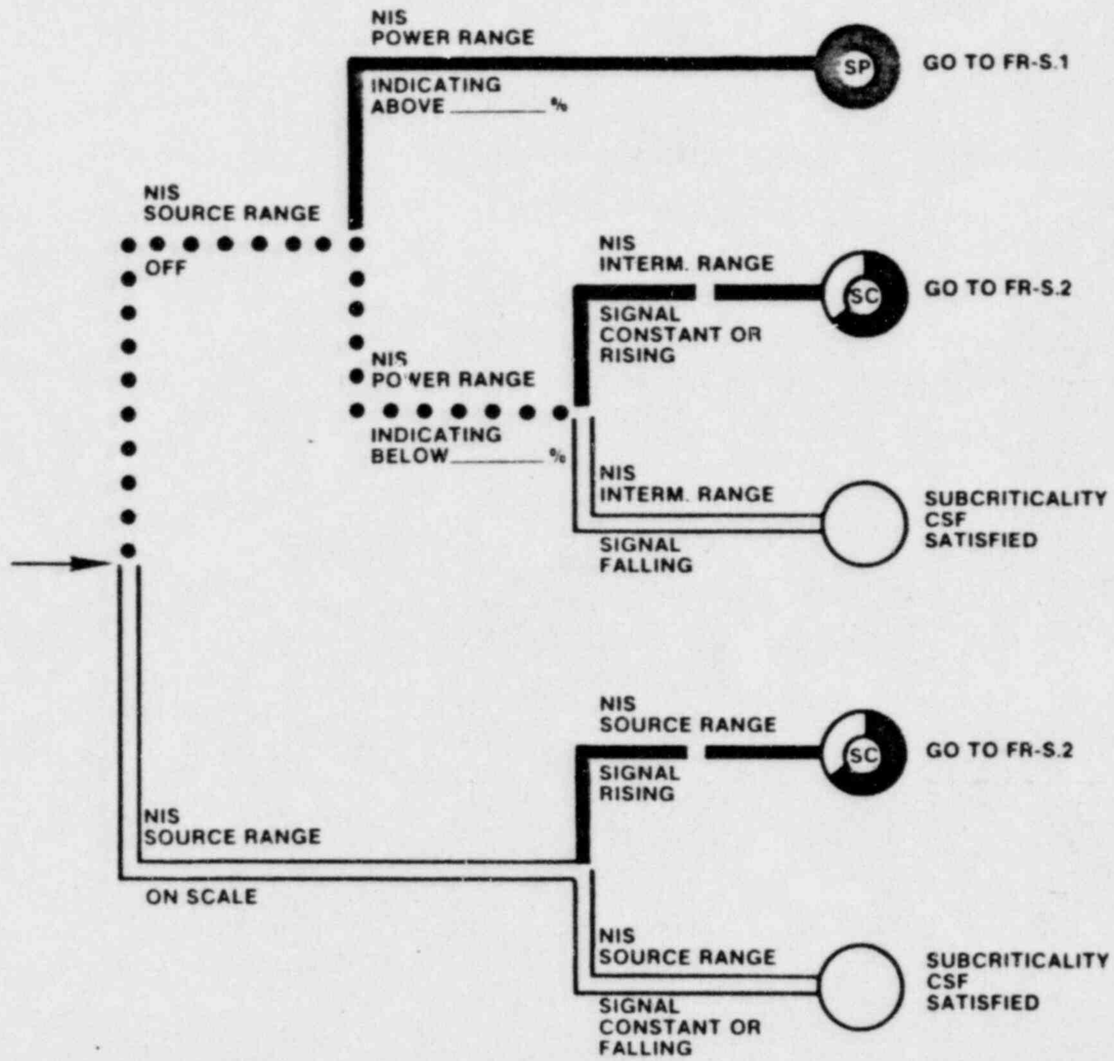
(3) CORE COOLING (C-SERIES)

(4) RCS INVENTORY (I-SERIES)

(5) CORE HEAT SINK (H-SERIES)

(6) CONTAINMENT INTEGRITY (Z-SERIES)

SUBCRITICALITY



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INSTRUMENTATION USED TO MONITOR CRITICAL SAFETY FUNCTIONS

I. SUBCRITICALITY
   A. NIS

II. REACTOR COOLANT SYSTEM INTEGRITY
   A. RCS PRESSURE WR
   B. T COLD WR RTD

III. CORE COOLING
   A. SUBCOOLING MONITOR
      1. RCS PRESSURE WR
      2. CORE EXIT T/Cs
      3. RTDs
   B. REACTOR VESSEL LEVEL INDICATION SYSTEM

IV. REACTOR COOLANT INVENTORY
   A. PRESSURIZER LEVEL
   B. RVLIS

V. HEAT SINK
   A. STEAM LINE PRESSURE
   B. STEAM GENERATOR LEVEL
   C. FEED WATER FLOW

VI. CONTAINMENT
   A. CONTAINMENT PRESSURE
   B. CONTAINMENT RADIATION MONITOR
   C. CONTAINMENT HYDROGEN CONCENTRATION
   D. CONTAINMENT SUMP LEVEL

CORE COOLING MONITOR
(SATURATION MARGIN)

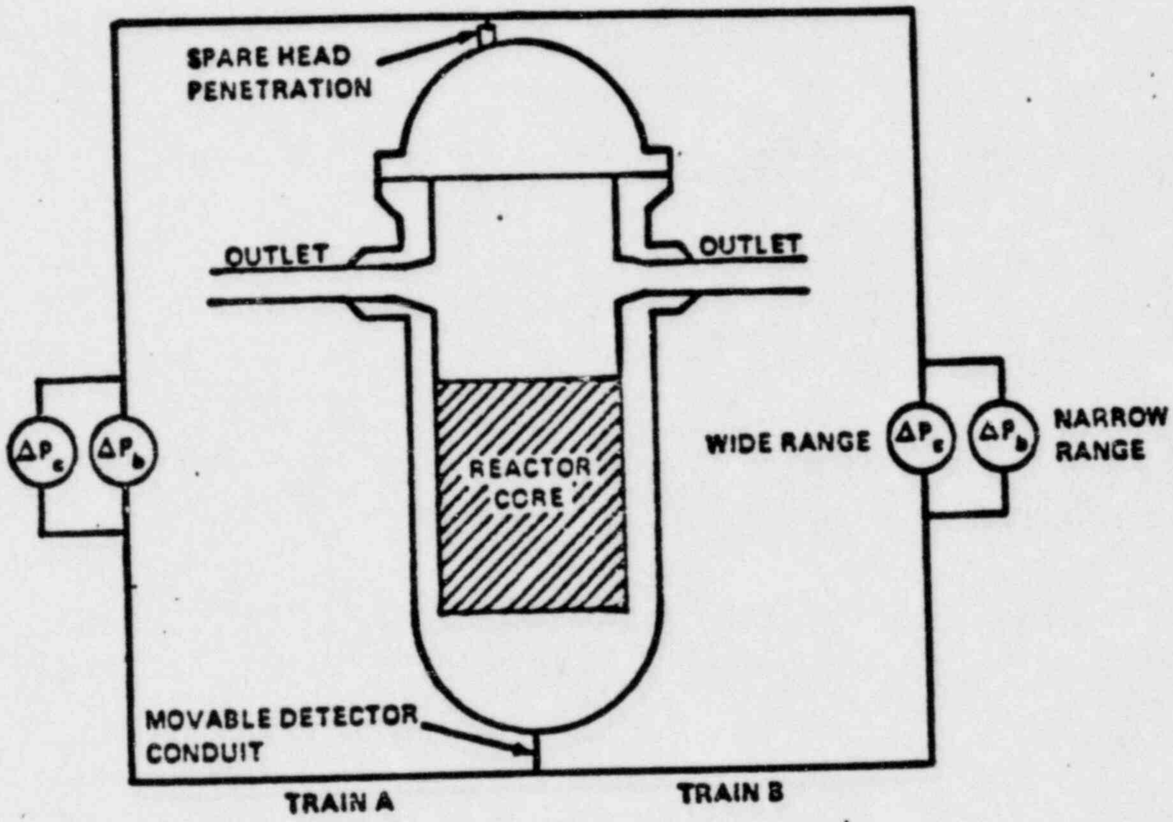
- REDUNDANT CHANNELS AND OUTPUT TRAINS
- SATURATION MARGIN FROM LOWEST OF THREE PRESSURE SIGNALS
  - CORE OUTLET THERMOCOUPLES
  - HOT AND COLD LEG RTD's
- EARLY WARNING SYSTEM
- ELECTRICALLY INDEPENDENT AND CLASS IE FOR ENTIRE SYSTEM
- MEETS NRC REQUIREMENTS

THERMOCOUPLE MONITOR

- REDUNDANT CHANNELS AND OUTPUT TRAINS
- SYSTEMS
  - PRIMARY
    - MEASURES ALL THERMOCOUPLES
    - ELECTRICALLY INDEPENDENT AND CLASS IE POWER SOURCE UP TO ISOLATOR
    - HARDWARE AND DISPLAY BEYOND ISOLATOR IS NON CLASS IE
  - BACKUP
    - TWO CHANNEL
    - EACH CHANNEL MONITORS 25 CORE OUTLET THERMOCOUPLES
    - ELECTRICALLY INDEPENDENT AND CLASS IE FOR ENTIRE SYSTEM
- MEETS NRC REQUIREMENTS

REACTOR VESSEL LEVEL
INSTRUMENTATION SYSTEM

- TWO SETS OF TWO Δ P CELLS (REDUNDANT)
- EACH SET
  - NARROW RANGE
    - BOTTOM OF REACTOR VESSEL TO TOP OF REACTOR VESSEL
    - NO OPERATING REACTOR COOLANT PUMPS (NATRUAL CIRCULATION)
  - WIDE RANGE
    - BOTTOM OF REACTOR VESSEL TO TOP OF REACTOR VESSEL
    - ANY COMBINATION OF OPERATING REACTOR COOLANT PUMPS
- TEMPERATURE MEASUREMENT OF IMPULSE LINES
- USE EXISTING REACTOR COOLANT TEMPERATURE (RTD's) AND WIDE RANGE REACTOR COOLANT SYSTEM PRESSURE
- MEETS NRC REQUIREMENTS



REACTOR VESSEL LEVEL
INSTRUMENTATION SYSTEM

SUMMARY

I. WESTINGHOUSE OWNERS GROUP GUIDELINES

II. PROCEDURES ARE PLANT SPECIFIC, TAILORED TO CALLAWAY CONTROL ROOM, SYSTEMATIC APPROACH DURING HIGH STRESS CONDITIONS

III. THOROUGH UNDERSTANDING THROUGH SIMULATOR DRILLS AND TRAINING

IV. CALLAWAY DESIGN PROVIDES ADDITIONAL INSTRUMENTATION FOR ASSISTANCE IN MANAGEMENT OF POSTULATED ACCIDENTS

DECAY HEAT REMOVAL

F. Schwoerer, Technical Director-SNUPPS

FUNCTIONAL REQUIREMENTS FOR COLD SHUTDOWN

1. HEAT REMOVAL AT HOT STANDBY
2. BORATION OF RCS
3. COOLDOWN OF RCS
4. DEPRESSURIZATION OF RCS
5. RHR OPERATION

SUMMARY OF

OPEN ITEMS FROM

SAFETY EVALUATION

REPORT FOR

CALLAWAY PLANT, UNIT 1

OPEN ITEM SUMMARY

- TOTAL OF 8 ITEMS + 5 TMI-RELATED ITEMS

  9 ITEMS - UE ACTION

  2 ITEMS - NRC ACTION

  2 ITEMS - UE AND NRC ACTION

  13 TOTAL

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OPEN ITEMS FROM SER

	ITEM	NEXT ACTION	EXPECTED CLOSE
1.	ICE LOAD ANALYSIS FOR ESW	UE	BY 12/31/81
2.	HIGH-ENERGY PIPE BREAK HAZARDS ANALYSIS	UE	NEARLY COMPLETE 2/1/82
3.	VIBRATION DAMPING ANALYSIS FOR CABLE TRAY & CONDUIT SUPPORT SYSTEMS	BOTH UE & NRC	POSSIBLY BY 11/12/81
4.	PUMP & VALVE OPERABILITY ASSURANCE PROGRAM	NRC	MID-1982 OR LATER

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ITEM	NEXT ACTION		EXPECTED CLOSE
5. PIPE SUPPORT BASEPLATE FLEXIBILITY: EFFECT ON ANCHOR BOLT LOADS		CLOSED	
6. SEISMIC & DYNAMIC QUALIFICATIONS OF MECHANICAL AND ELECTRICAL EQUIPMENT	UE		MID-82 OR LATER
7. ENVIRONMENTAL QUALIFICATION OF SAFETY - RELATED ELECTRICAL EQUIPMENT	UE		MID-82 OR LATER
8. FUEL ASSEMBLY STRUCTURAL RESPONSE TO SEISMIC & LOCA FORCES		CLOSED	

ITEM	NEXT ACTION	EXPECTED CLOSE
9. STEAM GENERATOR LEVEL MEASUREMENT ERRORS DUE TO ENVIRONMENTAL TEMPERATURE EFFECTS ON INSTRUMENT REFERENCE LEGS	-	CLOSED
10. FIRE PROTECTION PROGRAMS - ALTERNATE SHUTDOWN PANEL	BOTH UE & NRC	LATE 1982
11. SYSTEMS AND COMPONENTS ON THE "Q" LIST	UE	POSSIBLY BY 11/12/81

ITEM	NEXT ACTION	EXPECTED CLOSE
12. TMI ACTION PLAN ITEMS		
I.C.1 GUIDANCE FOR EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS	UE	8/82
I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR TERM OL APPLICANTS	UE	LATE 1982
I.D.1 CONTROL ROOM DESIGN REVIEW	UE	LATE 1982
II.B.2 PLANT SHIELDING FOR ACCESS TO VITAL AREAS AND TO PROTECT SAFETY EQUIPMENT FOR POST - ACCIDENT OPERATION	UE	MID-1982
III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES	NRC	LATE 1982

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

November 12, 1981

MEMORANDUM FOR: M. Bender, Chairman
ACRS Subcommittee on Comanche Peak Units 1 and 2

FROM: H. Alderman H. Alderman
Reactor Engineer

SUBJECT: SUBCOMMITTEE ON COMANCHE PEAK UNITS 1 AND 2 MEETING OF NOVEMBER 11, 1981

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

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
J. Arnold
F. Binford
E. Case, NRR
E. Goodwin, NRR
G. Edison, NRR
J. Youngblood, NRR

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(Insert proper classification)

November 12, 1981

PROPOSED SUMMARY OF THE NOVEMBER 11, 1981 MEETING OF THE
ACRS SUBCOMMITTEE ON COMANCHE PEAK ELECTRIC STATION UNIT 2

Purpose:

The purpose of the meeting was to review the application of Texas Utilities Generating Company for a license to operate the Comanche Peak Steam Electric Station Units 1 and 2.

Principal Attendees:

ACRS	Staff
M. Bender, Chairman	S. Burwell
J. Ray, Member	B. Youngblood
D. Moeller, Member (Part-time)	L. Crocker
J. Arnold, Consultant	W. Hazelton
F. Binford, Consultant	P. Collins
J. C. McKinley, Designated Federal Employee	D. Vassalo
H. Alderman, ACRS Staff	E. Sylvestor

TUGCO

H. Schmidt	C. Turner
J. Marshall	M. Blevins
B. Clements	R. Estes
J. Kuykendall	R. Calder
D. Jones	F. Madden
T. Vega	D. Braswell
R. Seidel	C. Feist
J. Rumsey	S. Kumar

Meeting Highlights, Requests, and Agreements:

1. Mr. Spottswood Burwell, NRC Staff Licensing Manager, noted the milletones of the review. The FSAR was docketed April 25, 1978. The Subcommittee toured the plant June 29, 1981. The SER was issued July 14, 1981. The first supplement was issued on October 16, 1981. The hearing for this project starts on December 2, 1981.

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2. Mr. Burwell noted several unique features in Comanche Peak. They are the lead plant conforming to IEEE 323-1974. They are the lead plant using Hafnium control rods. They are the lead plant to use the protection system upgrade package.

3. Mr. Marshall, TUGCO, noted the reason for the change to hafnium rods was primarily economic. The material currently used in Westinghouse plants contains large amounts of silver. Hafnium has been used in other applications widely in the nuclear industry, particularly in military applications. The Staff is concerned about the potential for dimensional changes and TUGO has agreed to a surveillance program to inspect these rods as they are used throughout the early life of the plant.

4. Mr. Burwell noted that when the SER Supplement was issued there were 8 non-TMI items and 9 TMI items open. One of the non-TMI items has been resolved and progress has been made on the others.

5. Mr. Bender requested that someone from the NRC Staff discuss with the ACRS what guidance is used to determine when preservice and inservice inspection is satisfactory.

6. Mr. Moeller noted that containment pressure was the only indicator for actuation of containment sprays, and inquired why other indicators were not used for backup. Mr. Siva Kumar, Gibbs & Hill, responded that the containment spray activation is mainly to maintain the integrity of the containment and pressure is the correct indication for integrity of the containment.

7. Mr. Bender inquired how inadvertent spraying of sodium hydroxide into the containment is prevented. Mr. Jones, TUGCO, responded that inadvertent sodium hydroxide has occurred in the past by improper valve lineup. Comanche Peak has a checking procedure in which one operator will do the lineup and another operator will come along and check his work.

8. Mr. Moeller noted that in the Final Environmental Station, page 5-50, it states "... for example, the chances are about one in a hundred million per reactor year, that 100,000 or more people might receive doses of 200 rem or greater." Mr. Moeller remarked that he thought this was off by a factor of 10. He requested that someone from the Staff discuss this with the Committee.

9. During the discussion of the Comanche Peak turbine, Mr. Bender requested that the Staff discuss how close the Staff is to the German review on Friday, November 13th.

10. Mr. Schmidt noted that the construction was about 89% complete on Unit 1, and 52% complete on Unit 2. The overall project is about 78% complete. Estimated fuel load date is June 1983. Currently, there are two active intervenors.

11. Mr. Marshall noted that their calculations regarding manual switchover of containment are being reviewed to eliminate some of the conservatism. He noted there is a difference of about 2.5 minutes in the switchover time between what is acceptable and what is not.

12. Mr. Clements, TUGCO, discussed their organization. He emphasized the dedication of their staff. He felt this partially compensated for the lack of commercial nuclear experience.

FUTURE MEETINGS:

The full ACRS is scheduled to review Texas Utilities Generating Company's application for a license to operate the Comanche Peak Steam Electric Station, Units 1 and 2 on Friday, November 13, 1981, from 8:30 a.m. to 12:30 p.m.

COMANCHE PEAK SES

OPERATIONS

ENGINEERING

TECHNICAL SUPPORT

NUCLEAR SERVICES

G&H - A-E

W - NSSS

OWNERS

- TESCO
- TPL
- DPL
- TMPA
- BEPC
- TEX-LA

CPSES DESIGN

2 UNITS - 3411 MW_T

W RESAR-3 DESIGN

R. G. 1.70, REV. 2

IEEE 323-74

COMPARISON - FSAR SECT. 1.3

- McGUIRE
- TROJAN
- NORTH ANNA

CPSES SCHEDULE

CONSTRUCTION PERMIT	-	DEC. 1974
O.L. APPLICATION	-	APR. 1978
SER	-	JUL. 1981
SSER	-	OCT. 1981
T-G COMPLETE	-	JAN. 1982E
HYDRO R.C.S.	-	SEP. 1982E
BEGIN HOT FUNCTIONAL	-	DEC. 1982E
FUEL LOAD DATE	-	JUN 1983E
COMMERCIAL OPN.	-	1Q84E

CPSES

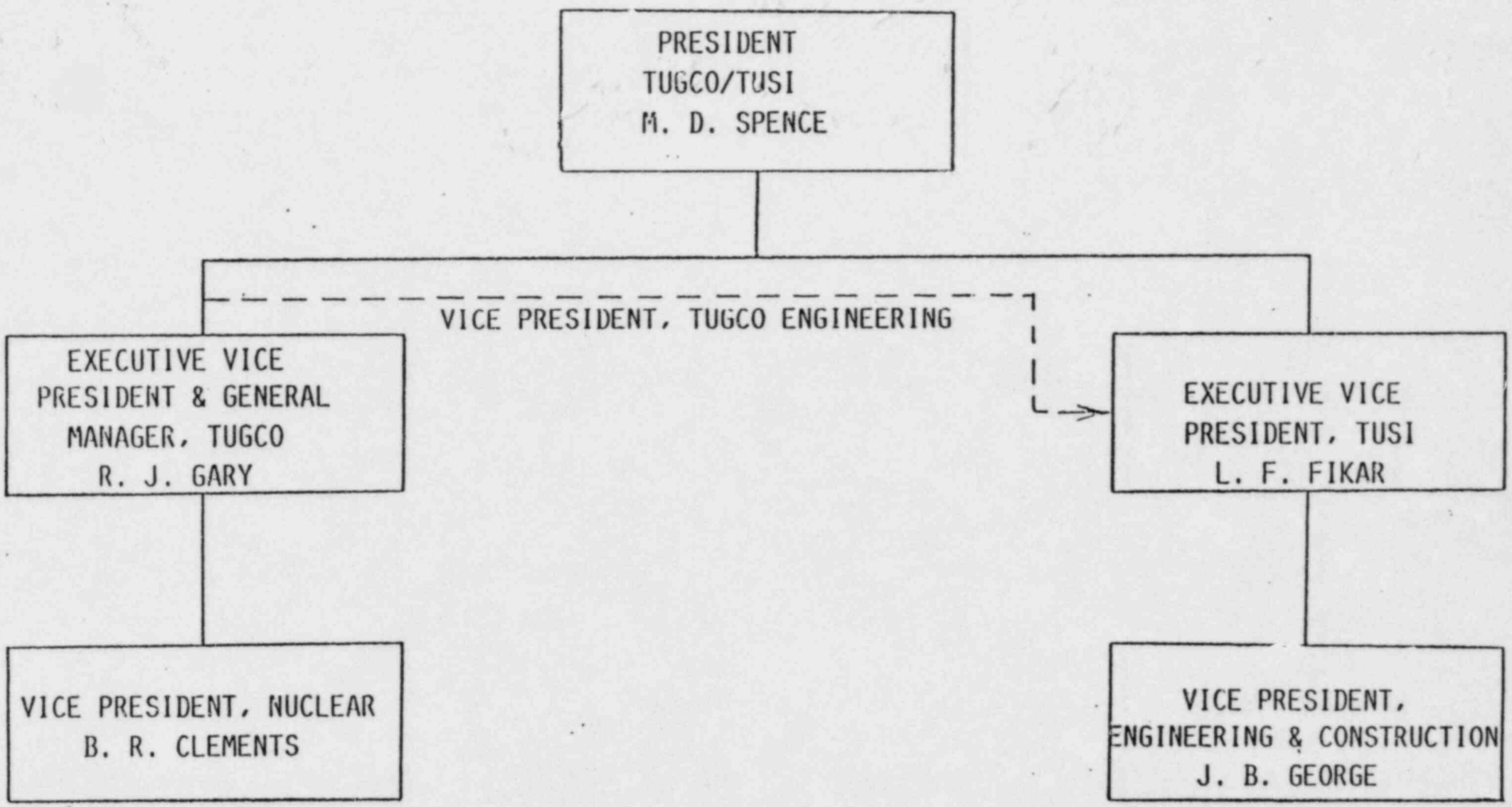
CONSTRUCTION STATUS

FUEL STATUS

LICENSING STATUS

A-271

TUGCO/TUSI ORGANIZATION



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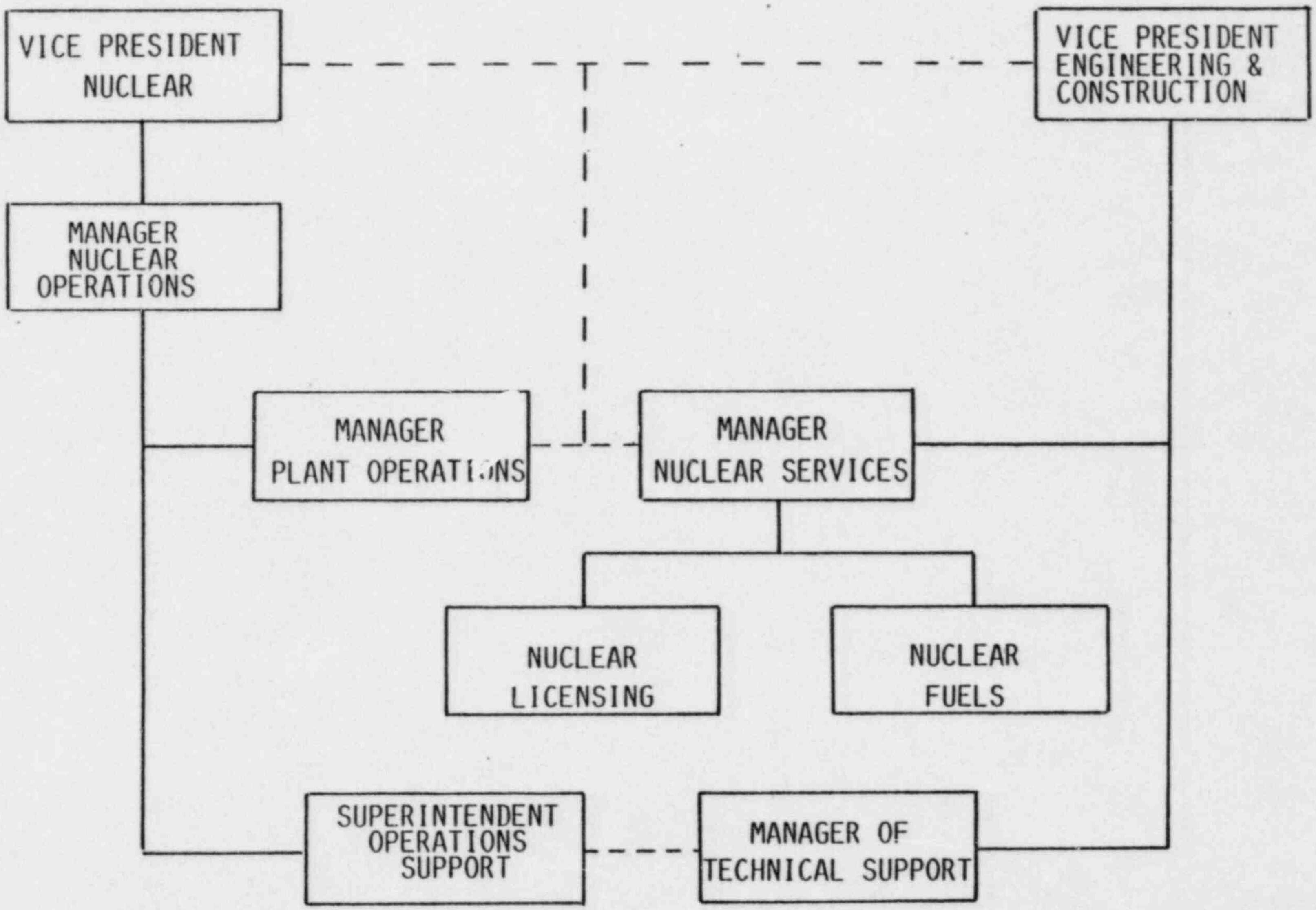
COMANCHE PEAK

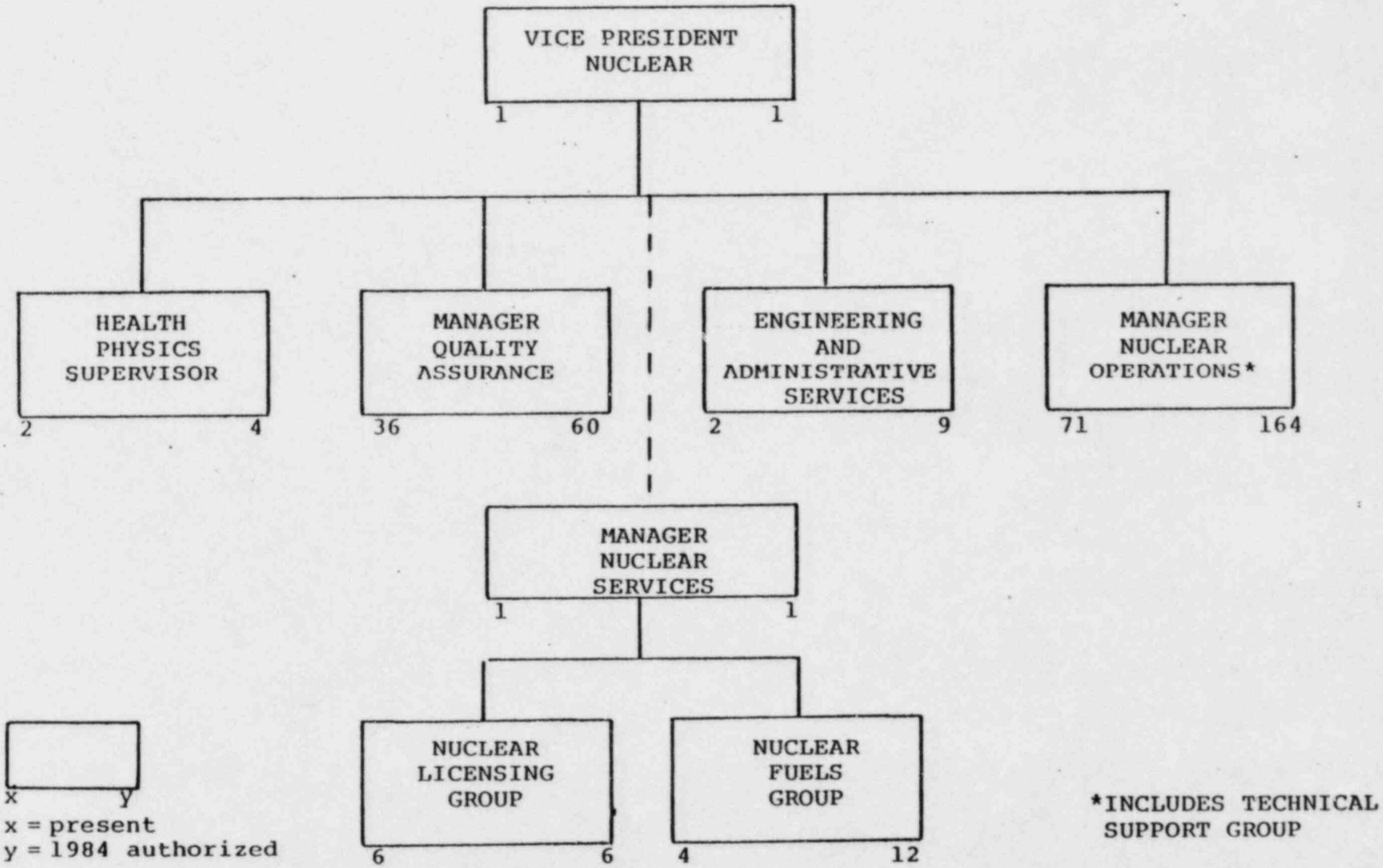
OPERATIONAL ORGANIZATION



A-273

CORPORATE NUCLEAR ORGANIZATION

PERSONNEL



TOTAL (1984) 258

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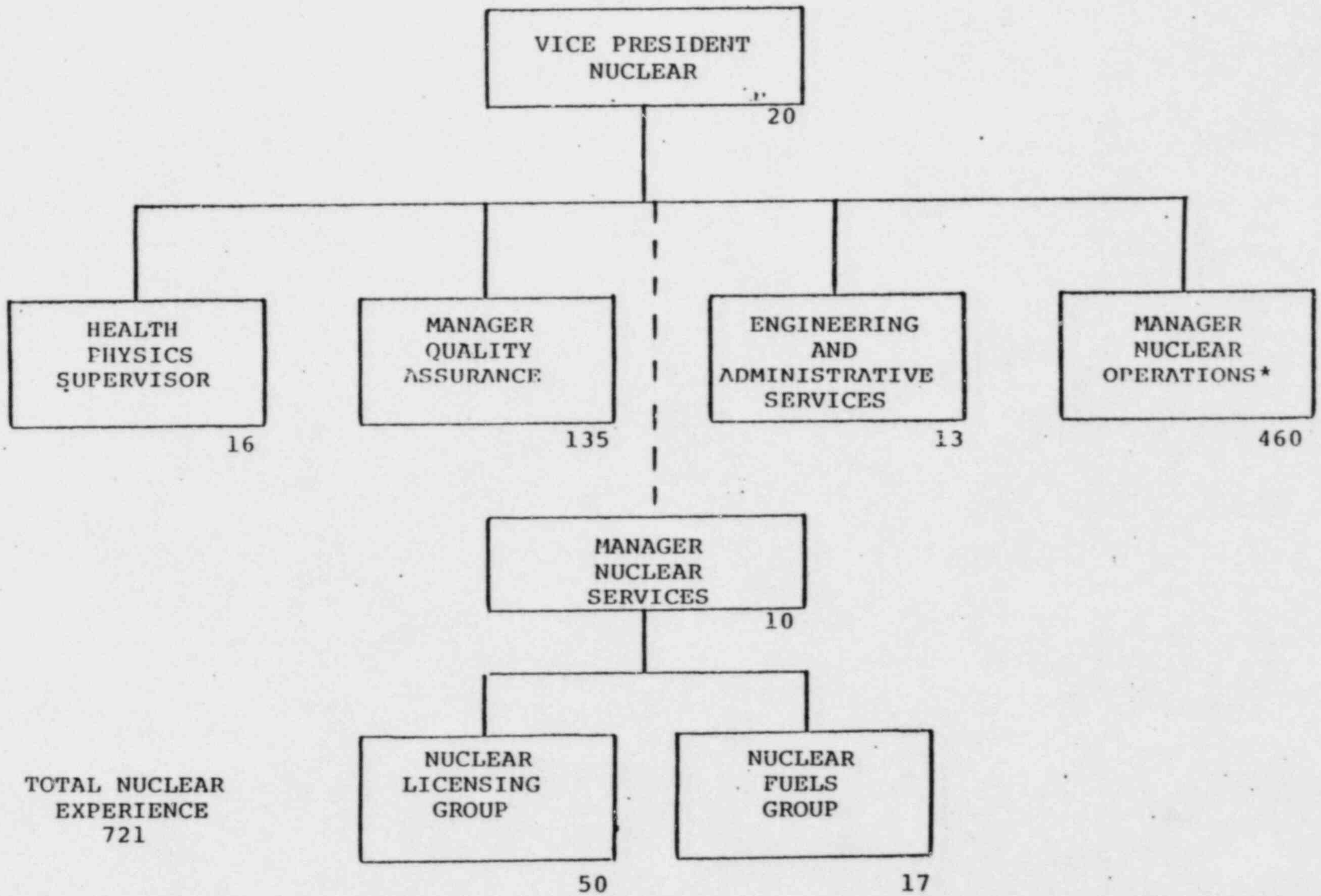
CORPORATE NUCLEAR ORGANIZATION

NUCLEAR EXPERIENCE
(MAN YEARS)



TOTAL NUCLEAR
EXPERIENCE
721

11/6/81

* INCLUDES TECHNICAL
SUPPORT GROUP

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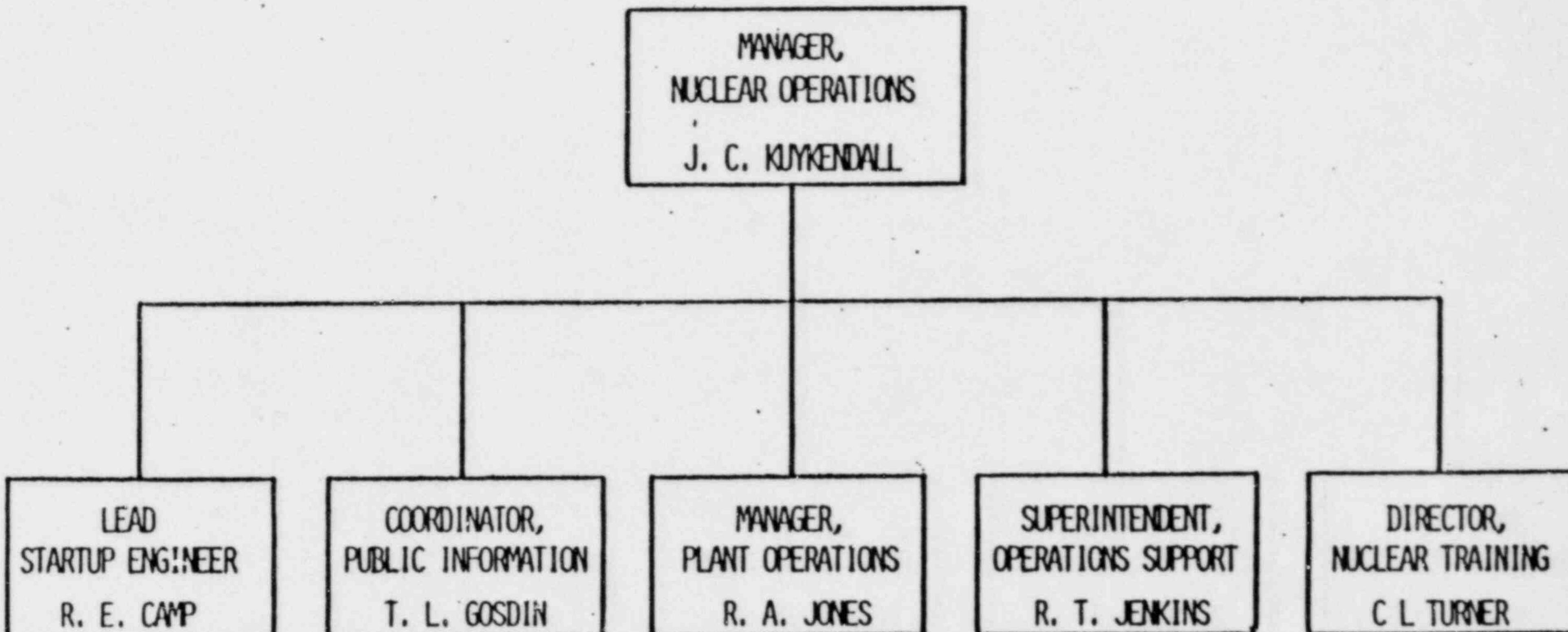
NUCLEAR OPERATIONS

TEXAS UTILITIES GENERATING COMPANY



A-276

Assigned Personnel, effective 9-1-81

AUTHORIZED MANNING LEVELS*

DALLAS OFFICE STAFF	1 UNIT OPERATIONS (1984)	2 UNIT OPERATIONS (1985)
VICE PRESIDENT, NUCLEAR/STAFF	14	14
MANAGER, NUCLEAR SERVICES	1	1
NUCLEAR LICENSING GROUP	6	6
NUCLEAR FUELS GROUP	12	18
MANAGER, QUALITY ASSURANCE/STAFF	60	60
DALLAS TOTAL	93	99

COMANCHE PEAK SITE STAFF		
MANAGER, NUCLEAR OPERATIONS	1	1
SUPERINTENDENT, OPERATIONS SUPPORT/STAFF	36	36
DIRECTOR, NUCLEAR TRAINING/STAFF	28	28
LEAD START UP ENGINEER/START UP STAFF	60	30
MANAGER, PLANT OPERATIONS/PLANT STAFF	313	408
MANAGER, TECHNICAL SUPPORT/STAFF	40	40
SITE TOTAL	478	543
TOTAL STAFF	571	642

*THESE NUMBERS DO NOT INCLUDE CONTRACT SECURITY FORCE.

OPERATIONS STAFF

PLANT NUCLEAR EXPERIENCE

CPSES	310	man years
OTHER COMMERCIAL	51	
OTHER	312	
TOTAL	673	

TRAINING PROGRAMS

GENERAL EMPLOYEE TRAINING

RADIATION WORKER TRAINING

GENERAL INFORMATION TRAINING

MAINTENANCE TRAINING

TECHNICIAN TRAINING

SHIFT TECHNICAL ADVISOR TRAINING

LICENSED OPERATOR TRAINING

SYSTEMS TRAINING

A-280

- REACTOR THEORY
- REACTOR COOLANT SYSTEM
- CORE COMPONENTS
- RCP
- CVCS
- BTRS
- BORON RECYCLE
- RHR
- AUXILIARY FEEDWATER SYSTEM
- ECCS
- CONTAINMENT SPRAY
- CONTAINMENT SYSTEM
- CONTAINMENT
- COMPONENT COOLING WATER
- SERVICE WATER
- SPENT FUEL POOL COOLING
- FUEL HANDLING
- RADIATION PHYSICS
- RADIATION STANDARDS
- BIOLOGICAL EFFECTS
- RAD WASTE PROCESSING
- MAIN STEAM
- CONDENSATE AND FEEDWATER
- EXTRACTION STEAM AND HEATER DRAINS
- COOLING WATER SYSTEM
- MAIN TURBINE AND AUXILIARIES
- MAIN GENERATOR AND AUXILIARIES
- PLANT ELECTRICAL SYSTEMS
- EXCORE NUCLEAR INSTRUMENTATION
- INCORE INSTRUMENTATION
- ROD CONTROL SYSTEM
- ROD POSITION INDICATION
- RCS TEMPERATURE AND N-16 POWER
- PRESSURIZER PRESSURE AND LEVEL CONTROL
- STEAM DUMP CONTROL SYSTEM
- STEAM GENERATOR WATER LEVEL CONTROL
- REACTOR PROTECTION SYSTEM

OPERATOR TRAINING PROGRAMS

- INITIAL LICENSED OPERATOR TRAINING
- REPLACEMENT OPERATOR TRAINING
- REQUALIFICATION TRAINING

INITIAL LICENSED OPERATOR TRAINING

- SELECTION
- PRETRAINING
- WESTINGHOUSE CERTIFICATION PROGRAM
- ON-SITE TRAINING
- PREOPERATIONAL DUTIES
- REVIEWS

REPLACEMENT OPERATOR TRAINING

- CLASSROOM AND IN-PLANT
- 2-1/2 YEARS
- AUXILIARY OPERATOR TRAINING
- LICENSE TRAINING
- STARTUP CERTIFICATION

REQUALIFICATION TRAINING

- THEORY AND SYSTEMS REVIEW
- ON-SHIFT EXERCISES
- SIMULATOR OPERATION
- TWO YEAR CYCLE
- ANNUAL EXAMS

SIMULATOR TRAINING

OPERATOR TRAINING

SHIFT TECHNICAL ADVISOR TRAINING

GENERAL INFORMATION TRAINING

PROCEDURES

STATION ADMINISTRATIVE

DEPARTMENTAL ADMINISTRATIVE

SPECIFIC IMPLEMENTING/TECHNICAL

PROCEDURE PREPARATION

PREPARED BY RESPONSIBLE GROUP

INDEPENDENTLY REVIEWED

SORC APPROVED

PERIODICALLY REVIEWED

PROCEDURE STATUS

84% OF 2390 PREPARED

ON SCHEDULE

STARTUP VERIFICATION

PROBABILISTIC RISK ASSESSMENT

RELIABILITY ANALYSIS

A-289

II.E.1. AUXILIARY FEEDWATER STUDY

SHOWED HIGH RELIABILITY

OPERATIONAL INPUT

SYSTEM INTERACTIONS

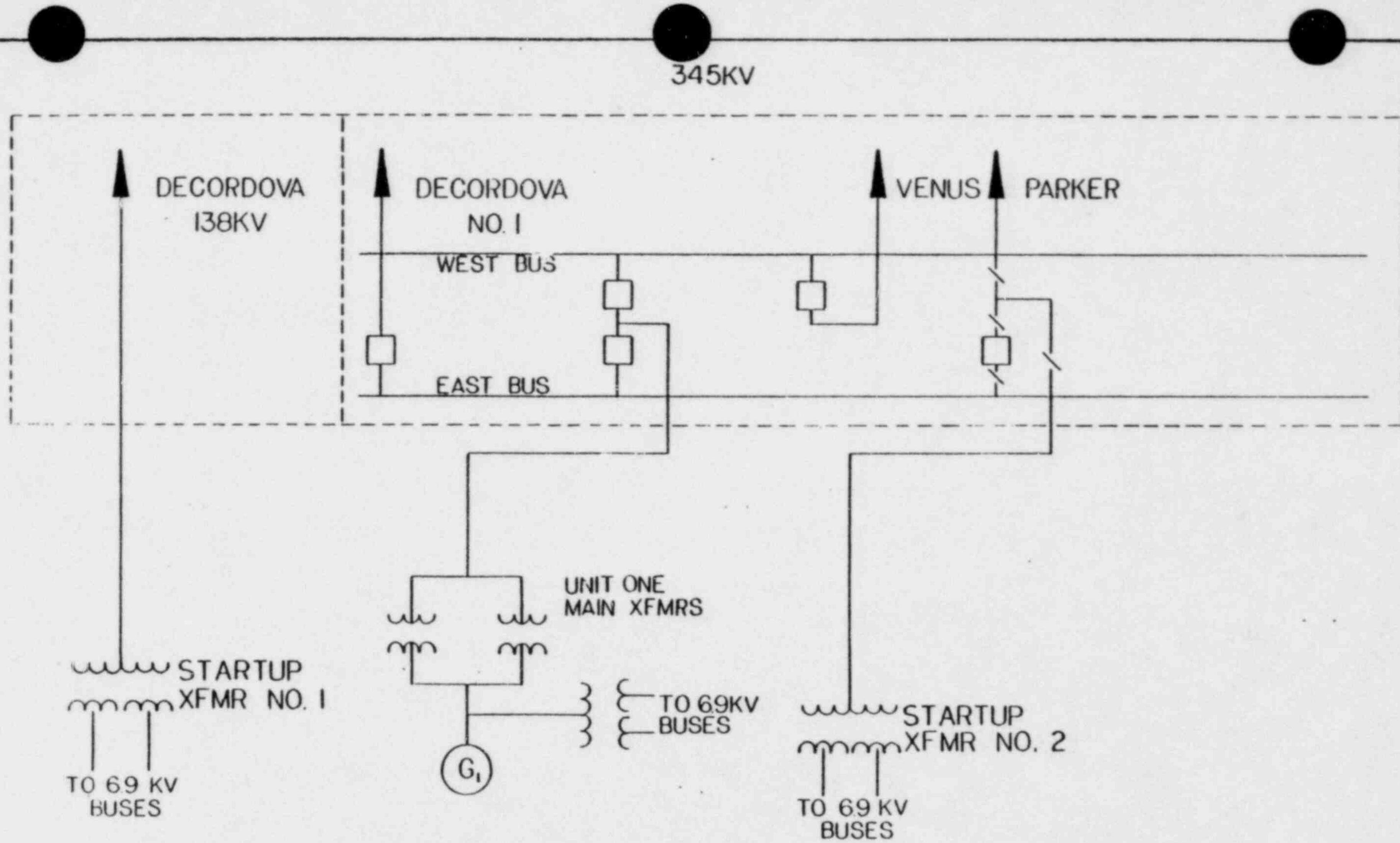
- INTER-DISCIPLINARY REVIEW
- HAZARD ANALYSIS
- CONTROL SYSTEM ANALYSIS
- HEAVY LOADS
- LER REVIEW

Reliability of Station Electric Power and DC Power System

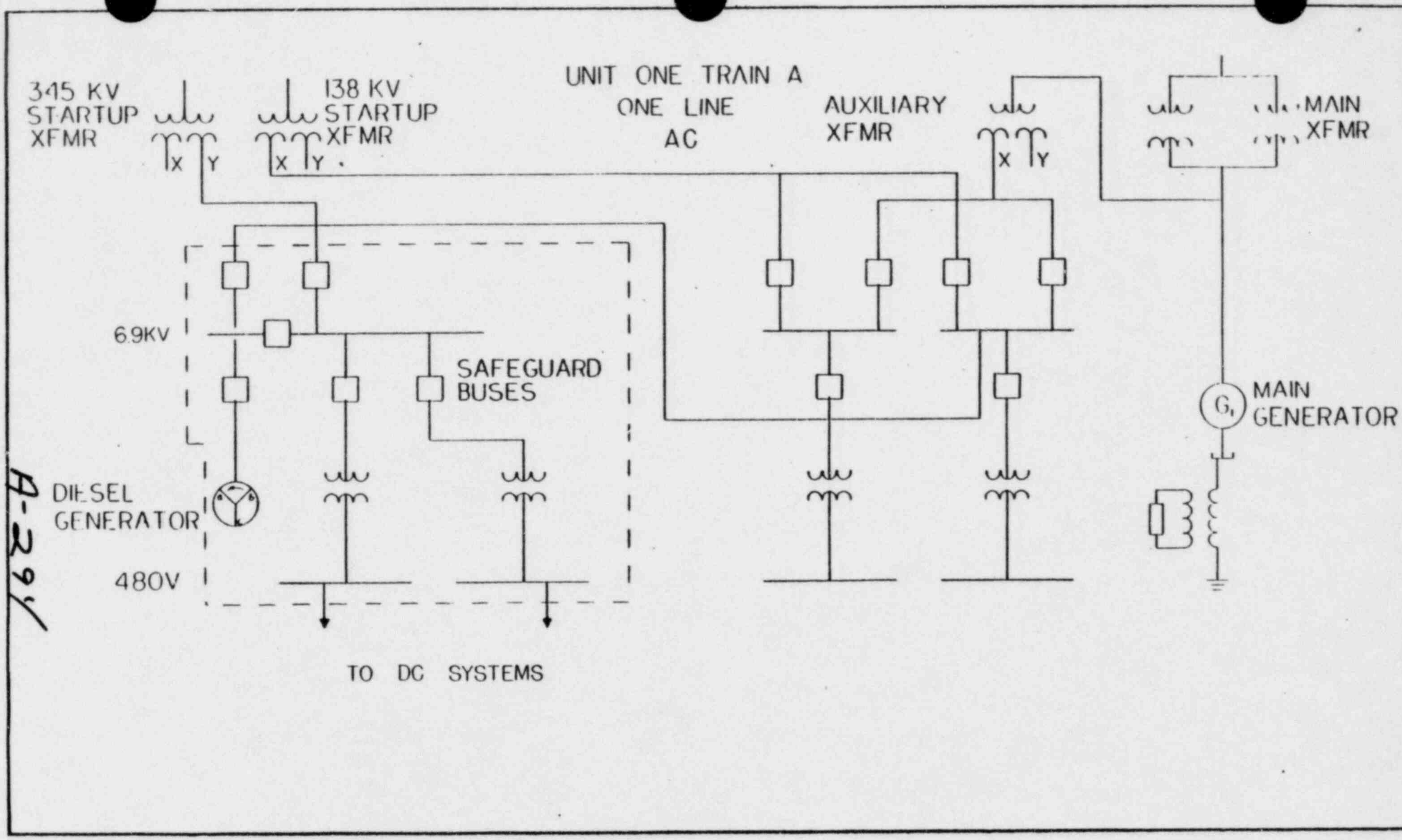
Survival Time for Loss of All AC Power

SLIDE 2

A-293



ONE LINE CPSES (1980) SWITCHYARD

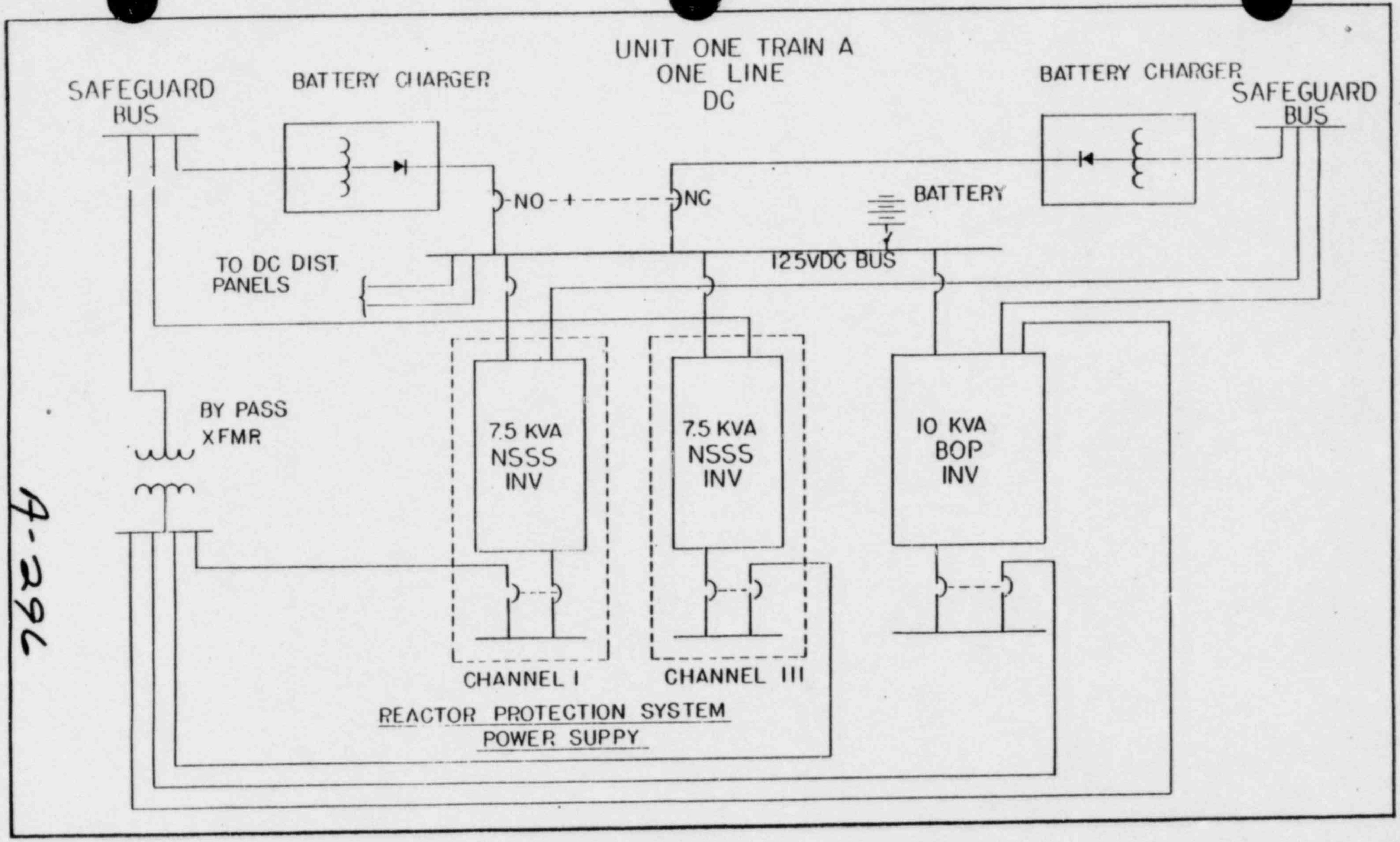

UNIT ONE TRAIN A
ONE LINE
AC
345 KV
STARTUP
XFMR
X
Y
138 KV
STARTUP
XFMR
X
Y
AUXILIARY
XFMR
X
Y
MAIN
XFMR
6.9KV
SAFEGUARD
BUSES
MAIN
GENERATOR
G_1
DIESEL
GENERATOR
480V
TO DC SYSTEMS

A-29Y

SLIDE 5

DIESEL FACTS

1. DELAVAL 7000 KW.
2. 7 DAY FUEL OIL SUPPLY.
3. EACH STARTING AND FUEL SYSTEM HAVE REDUNDANT PARTS.
4. EACH AIR RECEIVER HAS ENOUGH AIR FOR 5 STARTS (2 RECEIVERS PER DIESEL).
5. TESTED MONTHLY AT FULL LOAD.


UNIT ONE TRAIN A
ONE LINE
DC
SAFEGUARD BUS
BATTERY CHARGER
NO
+
NC
BATTERY
BATTERY CHARGER
SAFEGUARD BUS
TO DC DIST PANELS
125VDC BUS
BY PASS XFMR
7.5 KVA NSSS INV
7.5 KVA NSSS INV
10 KVA BOP INV
CHANNEL I
CHANNEL III
REACTOR PROTECTION SYSTEM
POWER SUPPY

A-296

SYMPTOMS OF LOSS OF ALL AC POWER

A. CONTROL ROOM STANDARD LIGHTING OFF; EMERGENCY LIGHTING ON

B. PLANT SAFEGUARDS EQUIPMENT NOT ENERGIZED

C. ZERO VOLTAGE INDICATION FROM PLANT AUXILIARY AND STARTUP TRANSFORMERS

D. ZERO VOLTAGE INDICATION FROM MAIN AND EMERGENCY BUSES.

OPERATOR GOALS

1. MINIMIZE RCS INVENTORY LOSS
2. MAINTAIN AN ULTIMATE HEAT SINK
3. RESTORE POWER
4. RECOVER THE PLANT FOLLOWING RESTORATION OF AC POWER

WESTINGHOUSE ANALYSIS

TIME TO TOP OF CORE UNCOVERY - 100 HOURS

DECAY HEAT REMOVAL

Steam Driven Auxiliary Feedwater Pump

1. Requires no AC
2. Starts automatically on loss of power and loss of main feedwater
3. Flow can be regulated with flow control valves with air accumulators

Auxiliary Feedwater Supply

1. Preferred supply 276,000 gallons reserved out of 500,000 gallon Condensate Storage Tank
2. Additional sources - 400,000 gallons in Demineralized Water Storage Tank

Steam Generators

1. Remain at hot standby with code safeties
2. PORV's have air accumulators

DC POWER SUPPLY

Two vital 125 VDC power supplies - 4 hours

Non-vital DC power supplies

125/250 VDC System - 4 hours

24/48 VDC System - 3 hours

EMERGENCY LIGHTING

Lighting powered by vital batteries - 4 hours

Individual sealed beam battery packs - 8 hours

Turbine building emergency lighting - 4 hours

Communications

1. Sound powered telephone system - not limited
2. Intraplant 2 way radios - not limited
3. Plant-to-offsite 2 way radios - not limited

ENVIRONMENT IN CRITICAL AREAS

1. Control Room - 95°F
2. Battery Rooms - 105°F
3. Inverter Rooms - 122°F
4. Turbine Driven Aux. Feedwater Pump - 116°F
5. Containment - 134°F

SURVIVAL TIMES

	Best Estimate
Auxiliary Feedwater Supply	7 + hours
DC Power	8 + hours
Emergency Lighting	8 + hours
Top of Core Uncovery	100 + hours

ACTIONS TO BE TAKEN

1. GET DIESELS STARTED
2. RESTORE OFF-SITE POWER

Summary

1. In place procedures.
2. Trained operators to mitigate effects.
3. Importance of reliable bulk transmission network to CPSES.
4. Core can be cooled and covered for days.

COMANCHE PEAK
HYDROGEN CONTROL-CURRENT DESIGN BASES

10CFR 50.44 AND REGULATORY GUIDE 1.7

SOURCES OF HYDROGEN -

- 5% METAL-WATER REACTION
- HYDROGEN DISSOLVED IN REACTOR COOLANT SYSTEM
- RADIOLYSIS OF CORE AND SUMP WATER
- CORROSION OF ALUMINUM AND ZINC

IMPLEMENTATION OF TMI LESSONS LEARNED -

OPERATOR TRAINING AND IMPROVED EMERGENCY OPERATING PROCEDURES WILL ENHANCE THE CAPABILITY TO AVERT ACCIDENTS LEADING TO DEGRADED CORE CONDITIONS AND ASSOCIATED HYDROGEN GENERATION.

COMANCHE PEAK
HYDROGEN CONTROL DESIGN FEATURES

HYDROGEN RECOMBINERS

- TWO RECOMBINERS LOCATED IN EACH CONTAINMENT
- CLASS 1E, SEISMIC CATEGORY I

HYDROGEN PURGE SYSTEM

- CAPABLE OF PURGING UP TO 700 SCFM
- RELEASE IS MONITORED AND FILTERED THROUGH HEPA AND CHARCOAL FILTERS
- SEISMIC CATEGORY I

HIGH POINT REACTOR COOLANT SYSTEM VENTS

- REMOTE-OPERATED, CLASS 1E, SEISMIC CATEGORY I VENT VALVES AT TOP OF REACTOR VESSEL AND PRESSURIZER
- VENT GASES TO OPEN AREAS OF CONTAINMENT FOR GOOD MIXING

COMANCHE PEAK

HYDROGEN CONTROL DESIGN FEATURES (CONT.)

HYDROGEN MONITORING SYSTEM

- REDUNDANT, CLASS 1E, SEISMIC CATEGORY I INSTUMENTS LOCATED IN EACH CONTAINMENT
- CAPABLE OF MEASURING HYDROGEN CONCENTRATION AT 4 DIFFERENT LOCATIONS (ELEVATIONS)

POST ASSIDENT SAMPLING SYSTEM

- CAPABILITY OF CAPURING AND ANALYZING DISSOLVED GASES IN REACTOR COOLANT SYSTEM AND HYDROGEN AND OXYGEN CONCENTRATIONS IN CONTAINMENT

HYDROGEN MIXING

- DURING LOCA, HYDROGEN AND STEAM ENTER CONTAINMENT IN THE FORM OF A HIGHLY TURBULENT JET WHICH ENTRAINS THE SURROUNDING ATMOSPHERE AND INDUCES TURBULENT MIXING
- COMPREHENSIVE CONTAINMENT SPRAY SYSTEM INDUCES ADEQUATE MIXING
- SUBCOMPARTMENTS PROVIDED WITH OPENINGS (VENTS) AND DRAINS
- HIGH POINT REACTOR COOLANT SYSTEM VENTS DISCHARGE TO OPEN AREAS OF CONTAINMENT

COMANCHE PEAK

HYDROGEN CONTROL CONSIDERATIONS

THE PROPOSED RULE FOR DRY PWR CONTAINMENTS (SECY-81-245A) REQUIRES ANALYSIS BE PERFORMED, WITHIN 2 YEARS, TO ASSURE THAT CONTAINMENT INTEGRITY, SAFE SHUTDOWN AND ESSENTIAL EQUIPMENT WILL NOT BE JEOPARDIZED BY HYDROGEN RELEASES FROM DEGRADED CORE ACCIDENTS INVOLVING A 75% METAL-WATER REACTION.

LARGE ATOMOSPHERIC CONTAINMENT

- FREE VOLUME $\approx 3.0 \times 10^6$ FT^3
- DESIGN PRESSURE = 50 PSIG

75% METAL-WATER REACTION (INCLUDES CONTROL RODS)

- 1800 LBM. OF H_2 PRODUCED
- 11.1 $^V/_O$ HYDROGEN IN DRY CONTAINMENT

CONSERVATIVE ESTIMATE OF CONTAINMENT STRENGTH

- 83 PSIG AT YIELD

COMANCHE PEAK
PRESSURE RISE DUE TO ADIABATIC HYDROGEN BURN
(75% ZIRCALOY AND HAFNIUM OXIDATION)

TEMP °F	PERCENT WATER VAPOR SATURATED 0%	25%	50%	100%
100	16.4/72.2/11.1	16.6/72.3/10.9	16.9/72.4/10.7	17.3/72.7/10.4
160	18.1/73.5/11.1	19.3/74.2/10.4	20.5/75.1/9.8	22.9/77.0/8.8
180	18.7/73.9/11.1	20.6/75.1/10.0	22.5/76.6/9.2	
200	19.3/74.3/11.1	22.2/76.3/9.6	25.1/78.7/8.5	
240	20.5/75.2/11.1	26.7/80.0/8.5		
260	21.1/75.6/11.1			
280	21.6/76.0/11.1			
300	22.2/76.5/11.1			

BELOW FLAME TEMP CRITERIA

INITIAL PRESSURE (PSIA) / FINAL PRESSURE (PSIA) / H_2 CONCENTRATION (V/O)

COMANCHE PEAK

HYDROGEN CONTROL

CONCLUSION -

COMANCHE PEAK DESIGN FEATURES REPRESENT STATE-OF-THE-ART FOR HYDROGEN CONTROL IN LARGE, DRY CONTAINMENTS.

ADDITIONALLY, INITIAL EVALUATION PROVIDES CONFIDENCE THAT COMANCHE PEAK DESIGN WILL SATISFY PROPOSED RULES FOR 75% METAL-WATER REACTION.

HAFNIUM CONTROL RODS

For all practical purposes, the control rods in the Comanche Peak design are indistinguishable from those employed in other plants using the Westinghouse NSSS. Westinghouse control rods employ a control (poison) material encapsulated in a stainless steel tubing. The Comanche Peak control material is hafnium as opposed to Ag-In-Cd which has been most widely employed. The reactivity worths of the two materials are essentially equivalent. However, hafnium is slightly heavier physically and results in a faster trip reactivity insertion rate.

The material properties of hafnium are well known. The design melting temperature of hafnium is 3913°F which compares with 1454°F for Ag-In-Cd. The lowest eutectic temperature in the Fe-Hf phase diagram is at 2372°F which is greater than the temperature at which the stainless steel cladding would fail by stainless-water reaction or plastic creep deformation (~2200°F). Postulated clad failure at this temperature range would, in the case of Ag-In-Cd, result in molten absorber material leaking out with subsequent loss of control rod worth. However, in the case of Hf, the absorber would remain solid with a much slower loss of rod worth due to oxide formation by reaction with H_2O.

Hafnium has been used extensively in the naval nuclear program and also in some limited commercial applications (Indian Point Unit 1, Yankee Rowe, and Shippingport reactors). Hafnium is also widely used in test reactor facilities. Unlike the Comanche Peak design employing stainless steel encapsulation, this previous experience was with unclad Hf directly exposed to the reactor coolant. The naval experience is normally classified and generally unavailable. Westinghouse has researched all available information on Hf experience including eight unclassified or declassified reports from the naval nuclear program. In all cases, including the limited commercial applications discussed above, the experience indicates that Hf is highly reliable and well suited to reactor application.

RESPONSE TO ACRS QUESTIONS ON N-16 SYSTEM

The following information is provided in response to several ACRS questions on the Comanche Peak N-16 protection system at the sub-committee meeting of November 11, 1981. This information should clarify some of the responses given at the sub-committee meeting as well as correct some erroneous statements that were made.

1) The N-16 power meter is not a spectrometer that measures only high energy gamma rays. Rather, it measures the gross gamma ray production, both high and low energy gamma rays. Thus, the N-16 system will detect the gamma rays of fuel failure.

2) Calculations have been done, at a variety of different conditions, which demonstrate that fuel failures will result in an increase in the N-16 signal. For example, one percent fuel failure would result in 2-3% increase in the N-16 power measurement at full power. The expected fuel failure percentage is much less than this value as demonstrated by current operating experience in Westinghouse plants. Therefore, the effect on N-16 monitors will be insignificant.

3) Since fuel failures result in an increase in the N-16 power signal, this would result in a conservative input to the protection system. That is, the indicated power level

would be higher than the actual power level. Hence, the plant would trip, during a transient, sooner than required.

4) The N-16 power measurement is checked against a secondary calorimetric power measured every 24 hours. The N-16 measurement is recalibrated if it deviates from the secondary side power measurement by more than 2%. Thus, the effect of any fuel failure on the N-16 reading will be checked every 24 hours and the meter will be recalibrated if necessary.

5) The primary purpose of the N-16 power measurement is for use in the overtemperature and overpower protection system. The N-16 power signal is also used in conjunction with the inlet temperature measurement to generate a vessel average temperature which is used in the rod control system.

6) The N-16 power meter is not used to detect fuel failures. Rather, separate measurement systems, the Gross Failed Fuel Monitor System, and primary coolant sampling, are used for this purpose.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JUN 1 4 1966

APPENDIX XXIII
PERIODIC COMPREHENSIVE (10 YEAR) REVIEW
OF OPERATING POWER REACTORS

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: PERIODIC COMPREHENSIVE (TEN YEAR) REVIEW OF OPERATING POWER REACTORS

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards recommends that the Atomic Energy Commission institute a program of periodic comprehensive reviews of operating licensed power reactors by the Regulatory Staff. Such comprehensive reviews would supplement the present annual operating reports, Compliance inspections, and DRL reviews in connection with proposed changes, and would result in reports to the Commission by the Regulatory Staff stating what limitations, if any, are being applied to ensure continued safe operation of each reactor.

The form of the reports to be submitted by reactor operators for these comprehensive reviews might well be outlined in a suitable guide prepared by the Staff. Such reports should contain summaries of operating history with special emphasis on significant problems. Information should be submitted to support the predictions of service lives of components whose failure could lead to a serious accident. These reports would also constitute a valuable source of information for the Regulatory Staff in its continuing effort to apply operating experience to the review of new applications.

The Committee suggests periods of about ten years for the comprehensive reviews. The period should be sufficiently flexible that the review can include recent results of inspections and tests made infrequently, such as containment leakage rate, pressure vessel inspection, and measurements of nil ductility transition temperature.

Attachment 4

RL 15.2

A-317

The Committee believes that each reactor operator should be responsible for the maintenance of appropriate records of the design, fabrication, inspection, installation, testing, and operating history of important plant components. The information should be adequate for the comprehensive reviews by the Regulatory Staff. Careful studies should be started by the Regulatory Staff to ascertain what information can be expected to be significant for future safety assessment. Surveillance and testing programs, where appropriate, should be planned to allow for possible use of improved knowledge developed during the life of the plant.

For reactors already in operation or under construction, the Committee recommends that the Regulatory Staff begin now to plan for the periodic comprehensive reviews.

The Committee believes that a periodic comprehensive review program of the sort proposed would constitute a workable mechanism for providing additional assurance that each reactor can continue to be operated without undue hazard to the health and safety of the public.

Sincerely yours,

ORIGINAL SIGNED BY
DAVID OKRENT

David Okrent
Chairman

1

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

November 17, 1970

APPENDIX XXIV
SAFETY OF OPERATING REACTORS

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: SAFETY OF OPERATING REACTORS

Dear Dr. Seaborg:

With the increasing number of large power reactors coming into operation, increased attention by the Regulatory Staff and by the reactor operators will be valuable in assuring continuation of the current good record of safe operation. Various measures recently developed cooperatively with the nuclear industry, such as Section XI of the ASME Boiler and Pressure Vessel Code for In-Service Inspection of Nuclear Reactor Coolant Systems, should contribute toward this objective.

In a previous report to you, the ACRS recommended the institution of periodic, comprehensive reviews of operating power reactors at intervals of approximately ten years. Such reviews were recommended as a "mechanism for providing additional assurance that each reactor can continue to be operated without undue hazard to the health and safety of the public."

The Committee is pleased to observe that, in addition to developing plans to institute such in-depth reviews, the Regulatory Staff is also considering reviews of specific components and performance aspects of power reactors at shorter intervals as appropriate.

As new knowledge, techniques, and experience are gained, it also becomes appropriate to institute regulatory procedures for assuring that the safety of operating reactors will receive early benefit from such new information on a systematic and timely basis. With the growth in the number of operating power reactors, it will be

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increasingly important that the organization and capabilities of the Regulatory Staff are adequate to meet this need, as well as that for periodic comprehensive reviews, without interfering with the adequacy of the licensing procedures for new reactors.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Vice Chairman

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

October 11, 1979

APPENDIX XXV
SYSTEMATIC EVALUATION PROGRAM

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: SYSTEMATIC EVALUATION PROGRAM

Dear Dr. Hendrie:

During its 213th meeting, January 5-7, 1978, the Advisory Committee on Reactor Safeguards (ACRS) received a presentation from the Nuclear Regulatory Commission (NRC) Staff concerning the Systematic Evaluation Program (SEP) as planned. This program was intended to examine many safety-related aspects of eleven of the older light water reactors (LWR). The purposes of the program were to ascertain the degree to which these reactors complied with current LWR safety criteria and standards, and to enable evaluation in a systematic way of the possible need for backfitting, after the review of each reactor was completed. The program also included the potential for identification of significant deficiencies which might warrant separate, earlier action prior to completion of the review.

The SEP appeared to be generally responsive to the ACRS recommendation for a periodic, comprehensive (10-year) review of older reactors, first made by the Committee in 1966. An important difference was that the ACRS had recommended that the licensee perform the detailed safety analysis of his plant and report his results and conclusions to the NRC Staff for their review and evaluation, while in the SEP the NRC Staff performs the detailed review.

In January 1978, the NRC Staff estimated that the SEP, as they planned it, would take about three years.

During its 233rd meeting, September 6-8, 1979, the ACRS was again briefed on the status of the SEP by the NRC Staff. The Staff reported that progress had been far slower than expected and that the earliest completion date was now three to three and one-half years in the future even if the currently available manpower resources were not diverted to other jobs. The NRC Staff stated that, thus far, they had identified only a few potentially significant deficiencies and stated that no criteria existed for identification of such deficiencies by the Staff.

The ACRS believes that the pace of the SEP has been too slow and that the currently expected completion date is later than desirable, in view of the fact that most of the plants being reviewed in this program were designed prior to the development of the first draft General Design Criteria and otherwise reflect an early era in the evolution of safety criteria.

The ACRS still believes that the SEP should be carried out in a manner similar to the safety reviews at the OL stage; that is, the licensee should prepare a Safety Analysis Report for those portions of the plant being reviewed, this analysis should be reviewed and evaluated by the NRC Staff, and appropriate actions should be required to remedy any significant deficiencies. The Committee believes also that criteria appropriate to the nature and intent of the SEP be developed on which to base the judgment of potentially significant safety deficiencies.

The Committee recognizes that the SEP is in an intermediate stage wherein a reformulation of the responsibility for the safety reevaluation is not straightforward. However, in view of the potential importance of the safety reevaluation of the reactors under review, and in view of the importance of developing a suitable process for other reactors, the ACRS recommends that the NRC undertake an early reevaluation of the current structure of the SEP.

Sincerely,

Max W. Carbon

Max W. Carbon
Chairman

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SEP PHASE II BRIEFING
OUTLINE

- BACKGROUND
- SAFETY FINDINGS TO DATE
- PROGRESS
  - TOPIC REVIEW STATUS
  - LEAD PLANT (PALISADES) STATUS
- INTEGRATED PLANT SAFETY ASSESSMENT
- SCHEDULE

CONTACT: WILLIAM T. RUSSELL
49-29794

BACKGROUND

PURPOSE

- REVIEW AND DOCUMENT COMPARISONS WITH CURRENT CRITERIA
- PROVIDE BASIS FOR INTEGRATED AND BALANCED BACKFIT DECISIONS

SCOPE

- ORIGINALLY 11 PLANTS (NOW 10)
- 137 TOPICS TAKEN FROM COLLECTION OF LISTS

OTHER CONSIDERATIONS

- PROVIDES BASIS FOR POL - FTL CONVERSIONS
- PROMPT ACTION ON MAJOR DEFICIENCIES

RESOURCES

- STARTED FEBRUARY 1978
- EXPENDED TO DATE: 93 STAFF YEARS; $5.8 M
- FY 82 BUDGETED 23 SY; $1.25 M

SAFETY FINDINGS TO DATE

- SAFETY ISSUES REQUIRING PROMPT ACTION
  - LACROSSE soil LIQUIFACTION (ORDER ISSUED)
  - SEISMIC REANALYSIS ON 5 PLANTS (50.54(F) LETTERS)
  - ELECTRICAL EQUIPMENT ANCHORAGES (50.54(F) LETTERS)
  - SAN ONOFRE TURBINE BUILDING (50.54(F) LETTERS)
  - YANKEE ROWE REACTOR BUILDING (50.54(F) LETTERS)
  - YANKEE ROWE FLOODING

- OTHER ITEMS BACKFIT BY LICENSEES
  - BATTERY MONITORING IN CONTROL ROOM AT PALISADES
  - NEW BATTERIES (2 HR VS 30 MIN CAPACITY) AT PALISADES
  - I & C MODIFICATIONS TO SI RESET AT GINNA
  - SAFE SHUTDOWN MODIFICATIONS AT YANKEE ROWE
  - SIGNIFICANT SEISMIC REANALYSIS AT ALL PLANTS
  - IMPROPER SWING DIESEL SELECTOR SWITCH LINE UP AT DRESDEN
  - REDUNDANT LEVEL INDICATION ON RWST AT HADDAM NECK

OVERALL PROGRESS

• 137 TOPICS AT EACH OF 10 PLANTS	1370
• TOPICS NOT APPLICABLE OR DUPLICATING TMI ACTION PLAN OR USI	466
• TOTAL TOPICS TO BE REVIEWED IN SEP	904
• COMPLETE AS OF 9/30/81	461

GENERAL OBSERVATIONS

- EXTERNAL EVENTS MOST DIFFICULT AND RESOURCE INTENSIVE
  - SEISMIC REVIEWS OF NON-SEISMIC PLANTS
  - WIND, TORNADO, FLOODING HAZARD
- SITING TOPICS GENERALLY HAVE NOT IDENTIFIED SIGNIFICANT ISSUES
  - EXCLUSION AREA AUTHORITY AND CONTROL
  - POPULATION DISTRIBUTION
- CONSIDERABLE NUMBER OF DIFFERENCES IN DESIGN VERSUS NRC'S CURRENT LICENSING CRITERIA

SEP TOPIC STATUS
by REVIEW DIVISION
SUMMARY TABLE

PLANT (IAPM) 1/		PALISADES Michaels	GINNA Wang	DRESDEN 2 Cwalina	OYSTER CREEK Fell	MILLSTONE 1 Persinko	SAN ONOFRE McKenna	BIG ROCK POINT Scholl	HADDAM NECK Brown	YANKEE ROWE Wang	LACROSSE Michaels	TOTAL
NO. OF TOPICS		137	137	137	137	137	137	137	137	137	137	1370
DELETED 2/		44	42	47	51	48	45	50	44	44	51	466
APPLICABLE		93	95	90	86	89	92	87	93	93	86	904
TECHNICAL REVIEW ASSIGNMENT 3/	DL	53 / 57	35 / 39	31 / 36	31 / 34	27 / 37	27 / 40	26 / 38	23 / 38	25 / 38	21 / 35	299 / 392
	DOE	16 / 19	16 / 22	7 / 23	4 / 21	6 / 22	9 / 19	3 / 18	4 / 22	6 / 22	3 / 22	74 / 210
	DSI	9 / 17	27 / 34	6 / 31	10 / 31	13 / 30	9 / 33	8 / 31	3 / 33	1 / 33	2 / 29	88 / 302
COMPLETE		78 / 93	78 / 95	44 / 90	45 / 86	46 / 89	45 / 92	37 / 87	30 / 93	32 / 93	26 / 86	461 / 904
TARGET DATE - ALL TOPICS COMPLETE		SEP 81	SEP 81	OCT 81	NOV 81	DEC 81	FEB 82	MAR 82	APR 82	JUN 82	JUL 82	

1/ Division of Licensing Integrated Assessment Project Manager

2/ Deleted as not applicable, or duplicated by USI, TMI Action Plan, or another topic.

3/ All topic review completion dates are before end of FY 1982. These schedules and assignments are consistent with NRR budget requests.

= No. Completed / No. Assigned

Taken From NUREG-0485 Published Monthly

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STATUS OF PALISADES REVIEW

PRELIMINARY CONCLUSIONS ON TOPICS

• NOT APPLICABLE; BEYOND SEP SCOPE	44
• MEET CURRENT CRITERIA OR EQUIVALENT	52
• ACCEPTABLE WITH LICENSEE PROPOSED MODIFICATION	3
• DO NOT MEET CURRENT CRITERIA OR EQUIVALENT--BEING EVALUATED FOR BACKFIT DURING INTEGRATED ASSESSMENT	23
• NOT YET COMPLETE	15
TOTAL	137

TOPICS ACCEPTABLE WITH LICENSEE

PROPOSED MODIFICATIONS

VIII-2 ONSITE EMERGENCY POWER SYSTEMS - DIESEL GENERATOR

- MODIFICATIONS HAVE BEEN IMPLEMENTED TO UPGRADE DIESEL GENERATOR ALARMS AND INDICATION IN THE CONTROL ROOM

VIII-3.B D.C. POWER SYSTEM BUS VOLTAGE MONITORING AND ANNUNCIATION

- MODIFICATIONS HAVE BEEN IMPLEMENTED TO UPGRADE DC SYSTEM INDICATION AND MONITORING IN THE CONTROL ROOM

III-6 SEISMIC DESIGN CONSIDERATIONS

- MODIFICATIONS HAVE BEEN IMPLEMENTED TO UPGRADE EQUIPMENT AND COMPONENT ANCHORAGE AND SUPPORT
- DIESEL GENERATOR FUEL OIL DAY TANK SUPPORT WELDS WILL BE UPGRADED FOLLOWING COMPLETION OF ANALYSIS OF WELDS
- CONTROL ROOM ELECTRICAL PANEL ANCHORAGES WILL BE UPGRADED AT THE NEXT REFUELING OUTAGE

VII-3 (PARTIAL) SYSTEMS REQUIRED FOR SAFE SHUTDOWN

- NEW BATTERIES WITH 2 HOUR MINIMUM CAPACITY HAVE BEEN INSTALLED

TOPICS WHICH DO NOT MEET CURRENT CRITERIA OR EQUIVALENT

II-1.A	EXCLUSION AREA AUTHORITY AND CONTROL
II-3.B	FLOODING POTENTIAL AND PROTECTION REQUIREMENTS
II-3.B.1	CAPABILITY OF OPERATING PLANT TO COPE WITH DESIGN BASIS FLOODING CONDITIONS
II-3.C	SAFETY-RELATED WATER SUPPLY [ULTIMATE HEAT SINK (UHS)]
III-2	WIND AND TORNADO LOADINGS
III-4.A	TORNADO MISSILES
III-7.A	INSERVICE INSPECTION, INCLUDING PRESTRESSED CONCRETE CONTAINMENT WITH EITHER GROUTED OR UNGROUTED TENDONS
III-7.C	DELAMINATION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES
V-5	REACTOR COOLANT PRESSURE BOUNDARY (RCPB) LEAKAGE DETECTION
V-10.B	RHR RELIABILITY
V-11.A	REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS
V-11.B	RHR INTERLOCK REQUIREMENTS
VI-2.D	MASS AND ENERGY RELEASE FOR POSSIBLE PIPE BREAK INSIDE CONTAINMENT
VI-3	CONTAINMENT PRESSURE AND HEAT REMOVAL CAPABILITY
VI-4	CONTAINMENT ISOLATION SYSTEMS
VI-7.A.3	ECCS ACTUATION SYSTEM
VI-10.A	TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES INCLUDING RESPONSE TIME TESTING
VII-1.A	ISOLATION OF REACTOR PROTECTION SYSTEM FROM NON - SAFETY SYSTEMS, INCLUDING QUALIFICATIONS OF ISOLATION DEVICES
VII-3	SYSTEMS REQUIRED FOR SAFE SHUTDOWN
VIII-3.A	STATION BATTERY CAPACITY TEST REQUIREMENTS
IX-3	STATION SERVICE AND COOLING WATER SYSTEMS
IX-5	VENTILATION SYSTEMS
XV-2	SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

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PALISADES

DL PALISADES

TOPIC NO.	TITLE	SAR 1/ RECEIVED	SER 2/ COMPLETE
III-1 (SYS)	Classification of Structures Components and Systems (Seismic and Quality)	NONE	
III-5.A	Effects of Pipe Break on Structures, Systems and Components Inside Containment	08/13/81R	
III-5.B	Pipe Break Outside Containment	08/25/81R	
III-7.A	Inservice Inspection, Including Prestressed Concrete Containments with Either Grouted or Ungrouted Tendons	NONE	08/25/81
III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	NONE	

DE

TOPIC NO.	TITLE	SAR 1/ RECEIVED	SER 2/ COMPLETE
III-3.C	Inservice Inspection of Water Control Structures	09/30/81	
III-4.B	Turbine Missiles(MP B-46)	NONE	08/28/81
III-7.D	Containment Structural Integrity Tests	NONE	08/17/81
V-1	Compliance with Codes and Standards (10 CFR 50.55a) (MP A-01, A-14)	NONE	
IX-6	Fire Protection(MP B-02)	NONE	

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DSI

TOPIC NO.	TITLE	SAR 1/ RECEIVED	SER 2/ COMPLETE
III-8.A	Loose Parts Monitoring and Core Barrel Vibration Monitoring (NRR B-60, C-12)	NONE	
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures	07/31/81R	10/16/81
V-7	Reactor Coolant Pump Overspeed (NRR B-68)	NONE	
VI-6	Containment Leak Testing (MP A-04)	NONE	
VIII-1.A	Potential Equipment Failures Associated with a Degraded Grid Voltage (MP B-23)	NONE	
IX-1	Fuel Storage	09/03/81	
IX-4	Boron Addition System	10/09/81R	
XI-1	Appendix I(MP A-02)	NONE	
XI-2	Radiological (Effluent and Process) Monitoring Systems(NRR B-67)	NONE	

1/ SAR = Licensee Safety Analysis Report-Scheduled or Actual (Actual(status 4b) when date is followed by R)

2/ SER = Staff Safety Evaluation Report (Status 4a,5,6,7)

CATEGORIZATION OF 23 TOPICS

WHICH DO NOT MEET CURRENT CRITERIA OR EQUIVALENT

- RELATE TO INITIATION OF ACCIDENTS
  - RHR RELIABILITY OR INTERLOCKS (2)

- RELATE TO SAFE SHUTDOWN
  - EXTERNAL FLOOD (3)
  - WIND OR TORNADO (2)
  - BATTERY RELIABILITY (1)
  - ACCESS TO OFF-SITE POWER, INDEPENDENCE OF BORON ADDITION FROM LOSS OF OFFSITE POWER (1)

- RELATE TO ACCIDENT MITIGATION
  - CONTAINMENT INSPECTIONS (2)
  - CONTAINMENT INTEGRITY (3)
  - HIGH/LOW PRESSURE ECCS INTERFACE (1)
  - RPS AND ESF ACTUATION, TESTING & ISOLATION (3)
  - ACCIDENT HEAT LOADS -- SERVICE WATER (1)
  - VENTILATION SYSTEMS (1)

- OTHER
  - EXCLUSION AREA AND LAND TITLES (1)
  - ACCIDENT ANALYSIS ASSUMPTIONS (1)
  - RCS LEAKAGE DETECTION SENSITIVITY (1)

INTEGRATED PLANT SAFETY ASSESSMENT

PURPOSE

- BASIS FOR BALANCED AND INTEGRATED BACKFIT DECISIONS

FACTORS TO BE CONSIDERED

- SAFETY SIGNIFICANCE
- TYPE OF IMPROVEMENT (OPERATION, PREVENTION, MITIGATION)
- COST TO IMPLEMENT (BOTH NRC AND LICENSEE)
- PERSONNEL RADIATION EXPOSURE TO IMPLEMENT

APPROACH

- USE POINT SYSTEM
- USE PRA, IF AVAILABLE
- DOCUMENT BASIS FOR EACH RECOMMENDATION

COORDINATION WITH OTHER NRC REQUIREMENTS

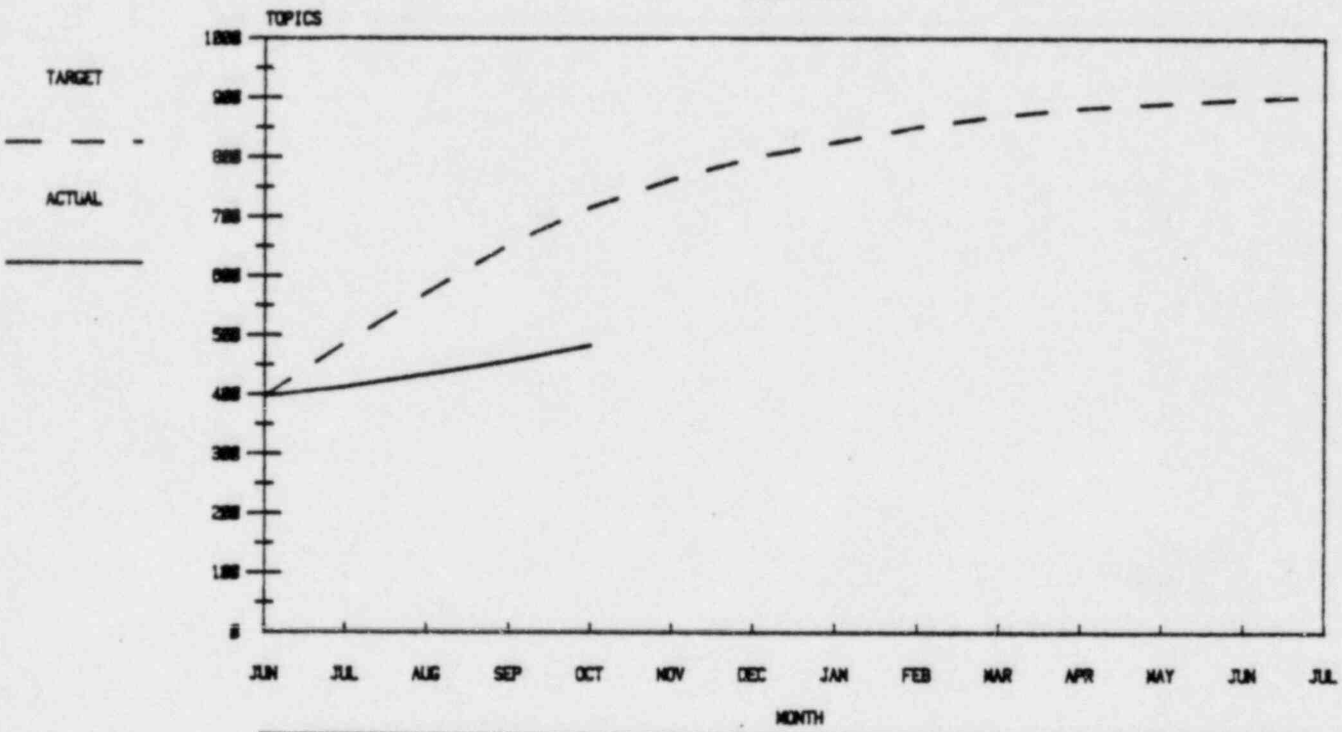
- TMI
- USI
- OTHER GENERIC
- LICENSEE INPUT TO "OPTIMIZE" AND IDENTIFY "COMMON FIXES"

SCHEDULE

PALISADES	MARCH 82
GINNA	APRIL 82
DRESDEN 2	MAY 82
OYSTER CREEK	JUNE 82
MILLSTONE 1	JULY 82
SAN ONOFRE	SEPT. 82
BIG ROCK POINT	OCT. 82
HADDAM NECK	NOV. 82
YANKEE ROWE	DEC. 82
LACROSSE	JAN. 83

PROGRESS OF TOPIC REVIEWS

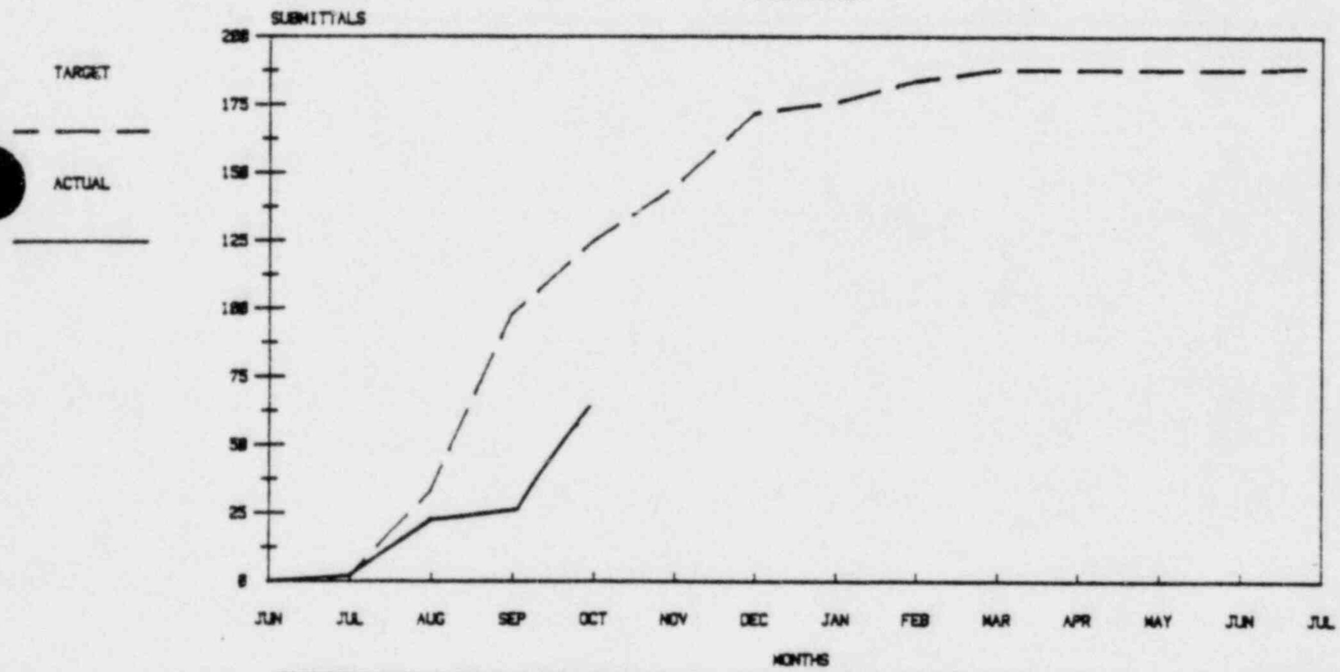
TOTAL PROJECT



PER MONTH	Target	90	88	77	64	48	37	26	26	19	13	8	8	4
	Actual	16	13	16	20									
CUMULATIVE	Target	486	574	651	715	763	800	826	852	871	884	892	900	904
	Actual	412	425	441	461									

LICENSEE TOPIC SUBMITTALS

TOTAL PROJECT



PER MONTH	Target	2	31	65	26	20	27	4	8	4	0	0	1	0
	Actual	2	21	5	37									
CUMULATIVE	Target	2	33	98	124	144	171	175	183	187	187	187	188	188
	Actual	2	23	28	65									

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

APPENDIX XXVII
SUBCMTE ON HUMAN FACTORS MEETING OF
NOV. 2, 1981

MEMORANDUM FOR: D. Ward, Chairman
ACRS Subcommittee on Human Factors

FROM: D. Fischer, Reactor Engineer

SUBJECT: SUBCOMMITTEE ON HUMAN FACTORS MEETING OF NOVEMBER 2, 1981

A summary of the subject meeting is attached for your review. Copies are being distributed to the other ACRS members and Subcommittee consultants for their information and comment. Corrections and additions will be included in the minutes of the meeting.

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
J. Buck
A. Debons
M. Keyserling
R. Pearson
E. Case, NRR
E. Goodwin, NRR
S. Hanauer, NRR/DHFS
J. Kramer, NRR/DFFS
J. Szwolinski, NRR/DHFS
V. Moore, NRR/DHFS/HFEB
D. Vassallo, NRR/DHFS/LQB
D. Beckham, NRR/DHFS/PTRB
L. Beltracchi, NRR/DHFS/HFEB
L. Crocker, NRR/DHFS/LQB
R. Froelich, NRR/DHFS/HFEB
M. Greenberg, NRR/DHFS/HFEB
R. DiSalvo, RES/DFO/HFBR
J. Norberg, RES/DFO/HFBR

11/09/81

SUMMARY
OF THE
NOVEMBER 2, 1981
MEETING OF THE ACRS SUBCOMMITTEE ON HUMAN FACTORS

PURPOSE:

The purpose of the meeting was to brief the newly formed ACRS Subcommittee on Human Factors on the development and programs that have been initiated within the Division of Human Factors Safety, Office of Nuclear Reactor Regulation since that division's inception over a year ago.

ATTENDEES:

ACRS

D. Ward, Chairman
J. Ebersole, Member
W. Mathis, Member
J. Buck, Consultant
A. Debons, Consultant
W. Keyserling, Consultant
R. Pearson, Consultant
R. Major, Designated Federal Employee
D. Fischer, Staff

NRC STAFF

S. Hanauer, NRR/D/DHFS
J. Kramer, NRR/DHFS
J. Szwolinski, NRR/DHFS
V. Moore, NRR/DHFS/HFEB
D. Vassallo, NRR/DHFS/LQB
D. Beckham, NRR/DHFS/PTRB
L. Beltracchi, NRR/DHFS/HFEB
L. Crocker, NRR/DHFS/LQB
R. DiSalvo, RES/DFO/HFBR
R. Froelich, NRR/DHFS/HFEB
M. Greenberg, NRR/DHFS/HFEB
J. Norberg, RES/DFD/HFBR

OTHER

Ms. L. Lund, Lund Consultants Inc.
Mr. W. Coley, Chairman, AIF Subcommittee on Control Room and Emergency Response Facility

MEETING HIGHLIGHTS, AGREEMENTS, AND REQUESTS:

1. Mr. Ward opened the meeting with a brief statement on the purpose and goal of the meeting. He informed the attendees that there had been two requests from members of the public to make oral statements.

2. Dr. Hanauer, Director of the Division of Human Factors Safety (DHFS) of NRR, provided an overview of the Division's role and responsibility. He

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mentioned that the Division was organized as a result of the accident at TMI-2. He summarized his understanding of the human operator's role in reactor safety. He outlined the safety functions that operators must be capable of performing and categorized the behavioral principles associated with the satisfactory performance of these functions (skill-based, rule-based, and knowledge-based behavior). The Division's human factors programs are aimed at ensuring correct operator action given the various sensory inputs to which plant operators are exposed. The human factors programs may be categorized into four general areas:

- Qualification and Training Programs
- Procedures Programs
- Control Room Programs and
- Organization and Management Programs

Dr. Hanauer summarized the Division's efforts in each of these areas.

Dr. Hanauer emphasized the need for the various human factors programs to be coordinated. The various NUREG documents associated with the program areas will be provided to the Commissioners for their approval in the December 1981 to January 1982 time frame. If approved, industry will be required to develop Program Plans for implementing the NUREGs. After a detailed review of those plans (review scheduled to be completed in late 1982), the new procedures, control room improvements, and SPDS will be validated simultaneously.

Dr. Hanauer requested a timely ACRS review of the NUREG documents related to the DHFS programs. He asked that the Committee provide him with any comments on the technical content of those documents as well as any comments on the scope and phasing of the DHFS programs.

3. Mr. R. Froelich (NRR/DHFS/HFEB) discussed NUREG-0700, "Guidelines for Control Room Design Review." This NUREG was published in its final form in September 1981. The purpose of NUREG-0700 is to describe methods for conducting control room design reviews. The control room design modifications which result from this review should improve the operator's capability to prevent and cope with accidents.

   The Staff estimated that cost to licensees for implementing NUREG-0700 to be about 3-5 man-years of effort plus about $10-40 thousand per modification. However, the cost could vary considerably depending on the scope and quality of previous control room review(s) conducted at the utility. Dr. Hanauer stated that NUREG-0700 will not require major modification to all plant control rooms. He said that some control rooms might only require paint, label, and tape type fixes.

4. Mr. M. Greenberg (NRR/DHFS/HFEB) made a presentation on NUREG-0801, "Evaluaion Criteria for Detailed Control Room Design Reviews." He related the NRC's actions required by NUREG-0801 to the actions required of the licensee/applicant as described in NUREG-0700.

A-2V/?

5. Mr. Beltracchi (NRR/DHFS/HFEB) provided the Subcommittee with a discussion of NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System." He said that the Safety Parameter Display System (SPDS) should integrate a minimum set of plant parameters into a display. Using this display, plant operators should be better able to assess the plant safety status during both normal and abnormal conditions. The SPDS must respond to the design criteria specified in NUREG-0696. While NUREG-0835 discusses acceptance criteria for CRT type displays only, other types of displays may be used.

   Mr. Beltracchi briefly described the validation and verification process that the Staff will use to evaluate licensee/applicant's compliance with NUREG-0835.

6. Ms. L. Lund of Lund Consulting, Inc. expressed her opinion that an earlier draft version of NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Review," should be adopted. She stated that a late August version of NUREG-0801 was reviewed by a task group at the Myrtle Beach Conference and provided "for a productive human factors review of nuclear power plants." She feels that the later draft detracts from the clear thinking, flexibility, and approach of the earlier draft.
   Dr. Hanauer agreed to take another look at Ms. Lund's comments and asked that she provide him with additional supporting information. Ms. Lund agreed to Dr. Hanauer's request.

7. Mr. W. Coley, Chairman, Atomic Industrial Forum (AIF) Subcommittee on Control Rooms and Emergency Response Facilities, provided the ACRS Subcommittee on Human Factors with a list of comments on Regulatory Guide 1.97, NUREG-0696, NUREG-0814, and NUREG-0801. The majority of his comments centered around the unclear relationship between the various NRC documents related to human factors.

8. Mr. D. Beckham (NRR/DHFS/PTRB) discussed NUREG-0799, "Criteria for Preparation of Emergency Operating Procedures." He said that if symptom-based procedures alone are used an incredible number of possible event sequences result. By identifying critical safety functions and developing procedures based on these, the number of event sequences becomes manageable. NUREG-0799 provides safety function maintenance and optimum recovery guidelines for accident situations.

9. Mr. L. Crocker (NRR/DHFS/LQB) made a presentation on utility management organizations and infrastructure. He identified TMI-2 items which relate to management organization and technical resources that have already been implemented and summarized those TMI Action Plan items still under development. This discussion was followed by a description of NUREG-0731, "Utility Management Guidelines and Technical Resources." Mr. Croker provided a history and status of NUREG-0731. He discussed the Staff's ongoing efforts to revise NUREG-0731 so that less subjective evaluations of utility management result.

The Subcommittee was given a brief description of how a utility management review is conducted by the Staff and the qualifications of the NRC reviewers. Mr. Crocker addressed two ACRS letters provided to the Staff dealing with utility management organization and infrastructure (ACRS letter to Mr. W. Dircks dated May 12, 1981 and Mr. M. Bender letter to Dr. Mark dated September 23, 1981). He briefly addressed each point in the letters. Mr. Ward asked if there was a Staff position on utilities having in-house versus contractor-run training organizations. Mr. Crocker stated that the Staff required the utility be able to adequately monitor its training program.

10. Mr. J. Zwolinski (NRR/DHFS) provided the Subcommittee with an overview of the Safety Technology Program being conducted by DHFS. This program will support the licensing decision-making process by providing a sound technical basis for the resolution of numerous TMI Action Plan items.

11. Mr. J. Kramer (NRR/DHFS) made a brief presentation on DHFS's interaction with the NRC's Office of Nuclear Regulatory Research (RES). RES "user needs" were identified for FY 1981, FY 1982, and FY 1983-1987.

12. Mr. J. Norberg (NRC/RES/DFO/HFBR) led off the Subcommittee's discussion on the NRC Safety Research Program for FY 1983 in the area of human factors. He provided the Subcommittee with some historical perspectives on human factors research and then described the organizational structure

of the Human Factors Branch (HFBR) of the Division of Facility Operations within the Office of Nuclear Regulatory Research. HFBR is structured to mimic the DHFS organization.

13. Mr. R. DiSalvo (NRC/RES/DFO/HFBR) presented the Office of Research's program for human factors research in FY 1983. He informed the Subcommittee that a significant increase in human factors research is underway. The issues being addressed by HFBR (e.g., Task Analysis, Validation of Control Room Modifications) and how HFBR plans to meet its user needs were discussed in detail. Mr. DiSalvo said that HFBR is trying to improve its understanding of the impact that humans have on nuclear safety and the factors which affect human performance. The HFBR research program should provide the technical data necessary to develop defensible regulatory positions related to human factors. Ultimately, the goal of their research program is to reduce the human contribution to risk to an acceptably low level. Mr. DiSalvo identified the various sources used in developing the human factors research program. He gave examples of the research work which the branch is doing in the areas of human factors engineering, licensee qualification, plant procedures, human reliability, and quality assurance. The proposed funding for FY 1983 in each of these areas was discussed in closed session. Mr. DiSalvo said that the research plan is generally responsive to the Staff's expressed needs. He also believes that the program is consistent with Commission and ACRS guidance. HFBR has coordinated its research efforts with INPO and EPRI.

In response to a question from Mr. Pearson, Mr. DiSalvo said that HFBR research results are selectively provided to the scientific community via NUREG reports.

FUTURE MEETINGS

The Subcommittee Chairman asked the Subcommittee consultants to provide him with written comments on the overall DHFS program within one week. The Subcommittee will report the results of this meeting during the November full Committee meeting. The Subcommittee Chairman will brief the Committee on human factors considerations in the design and operation of nuclear power plants including the qualifications and organizations of management, operators, and supporting infrastructure.

major

APPENDIX XXVIII
REPORT ON HUMAN FACTORS SUBCMTE MEETING
HELD ON NOV. 2, 1981

North Carolina State University

School of Engineering

November 6, 1981

Department of Industrial Engineering
Box 5511 Zip 27650

Mr. Richard Major, Staff Engineer
United States Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, D. C. 20555

Dear Mr. Major:

This is my report on the ACRS Sub-committee on Human Factors meeting in Washington on November 2. Generally, my reaction was quite positive. I thought the staff did a good job of briefing the sub-committee on current status, future plans, priorities, and budgeting. In terms of the over-all plan and schedule of work, I thought this involved a most logical integration of things that need to be accomplished. Somewhere down the line, we need to reach a point where control room design, procedures, CRT display integration, and simulator training all fit together in rational fashion and are aimed at a common objective--efficient performance of the control room personnel.

Apart from the above general observations, I have some specific comments as follows:

1. The projected budget allocations seem most reasonable. I was pleased to hear Dr. Hanauer say that "Procedures" would receive highest priority. Indeed, emergency procedures must ultimately be related to such other system ingredients as SPDS, DASS, and simulator training. I would expect that many (if not all) utilities are in a holding pattern on this topic until all the pieces fall into place--or they are "temporizing" with old or revised approaches. Beyond this, I do hope some effort will be devoted to collecting actual performance data on various procedures involving their controlled comparison. What is best here--fault tree diagrams, charts with JPA's, CRT displays, etc.? In short, what is the best format and mode of "display?"

2. I was pleased to see that 0700 took a stand on color coding, and hope it will hold up. We all recognize the "red-green" issue as a controversial one, and I expect many utilities will object. I understand the "pros" and "cons" from both sides. If the proposal stands, of course, simulator training and re-training will have to follow suit. Again, there should be some performance data collected to determine that operators do not make errors in response to the "new" color codes under conditions of task overload, e.g. emergency procedures.

3. With regard to auditory alerts (0700) I believe some clarification might be desirable in the case of some "master" emergency warning (if used) as opposed to the annunciator alarms. If a single signal is used to indicate

an abnormal transient or emergency condition, then it would be necessary for this to be distinguished from other auditory signals in terms of frequency, amplitude, rise and decay times, and/or intermittency characteristics.

4. With increasing use of CRT's and graphics displays, I believe greater attention needs to be given to the visual requirements of the job and the visual screening of operator personnel. The visual demands for CRT operators are unique and increase with age over 40. The Snellen chart currently used is not an appropriate test in this regard. I would strongly recommend a yearly examination using a device like the orthorater or Titmus optical vision tester that examines near and far visual acuity (both eyes) as well as color vision, the phorias, and depth perception. There are visual profiles published by the manufacturers of these devices, as well as in certain textbooks, and these could be adapted for initial screening or referral purposes.

5. From what I have reviewed, I gather there is a trend toward upgrading R/O qualifications. If I may play the devil's advocate, I wonder whether ultimately--if the job principally becomes one of monitoring a group of CRT displays--the job might be performed by individuals with lesser qualifications? How much engineering will the R/O of the future need? A good deal of my experience is with air traffic controllers who usually are neither pilots nor aeronautical engineers. They do not even need to be college graduates. But they are <u>well trained</u> in ATC procedures, and do perform their job well under pressure. Could similar monitors be so trained for NPP control rooms?

6. As you may recall, I was disturbed by the content of Messrs. Crocker's (on management organization) and Zwolinski's (on safety audit groups, PORC's, and ISEG's) presentations. If the importance of human factors and human resources is so highly recognized (as indeed it has been by NRC) then it would seem to follow most logically that, with regard to this critical man-machine system (i.e. NPP control rooms), greater representation should be accorded these areas within the management structure and audit groups (and in a revised NUREG 0731). Indeed, some of the other NUREG's emphasize this in the control room evaluation and backfit processes. I do, of course, recognize the unlikelihood of having a <u>qualified</u> human factors engineer within every utility, but someone who is <u>dedicated</u> to a concern for <u>human</u> performance should be involved within the organizational structures advocated. If not an HFE, this could be a personnel specialist, industrial/organizational psychologist, or qualified training specialist (not an R/O retreat), but desirably, even these people should have had some exposure to human factors engineering/ergonomics. This leads to my next point.

7. Since there is such a shortage of human factors engineers and, more specifically, of those who have had exposure to plant operations of any kind, why not push for NRC funding of a traineeship program in human factors engineering that would include some training in NPP design and operations? Obviously, NIOSH and other federal agencies recognize shortage areas and support educational programs through fellowships or traineeships.

8. Next, I would address the importance of NRC keeping members of this new sub-committee informed of relevant developments as contained in contractor reports, NUREG's, etc. While you provided us with those documents most relevant to the briefing, there were many others mentioned which I do not have in my possession, or to which I do not have access. These would include (but not be limited to) INPO, EPRI, ORNL, EG&G, SANDIA, and consulting firm (e.g. Essex, Biotechnology) reports. I do not think we should have to purchase these or otherwise request them from NRC; as relevant, such reports should be automatically sent to us.

9. In the same vein, I would appreciate consideration of some effort to provide us, as consultants, with additional familiarization with programs which are active in support of NRC/human factors interests such as ORNL, SPDS, etc. I suggested that our sub-committee meetings be held in conjunction with other professional gatherings (e.g. IEEE working groups, IEEE--Myrtle Beach, ANS, etc.) or at sites which staff might visit (e.g. INPO, ORNL, EPRI, etc.) I recognize some inconvenience to staff, but I would point out that other federal agencies often do this, i.e combine advisory group meetings with other functions, in order to reduce over-all travel costs across all involved. For those of us short on travel time and desiring to learn more about "what's new" in the business, it would seem most expeditious to schedule future meetings in this fashion. It really doesn't make much sense for some to spend the better part of two days travelling to Atlanta, Knoxville, or San Francisco, return "home," and then shortly thereafter devote additional travel time to a one-day ACRS sub-committee meeting in Washington, D.C. What do you think?

I trust this information will be helpful for your meeting next week.

Sincerely,

Richard G. Pearson

Richard G. Pearson, Ph.D.

RGP:ijv

APPENDIX XXIX
COMMENTS ON HUMAN FACTORS SUBCMTE
MEETING OF NOV. 2, 1981

W. MONROE KEYSERLING, PH.D.
26 ST. PAUL STREET NO. 5
BROOKLINE, MASSACHUSETTS 02146
(617) 277-1987

7 November 1981

MEMORANDUM

TO: Richard Major, Staff Engineer

FROM: Monroe Keyserling, ACRS Consultant

RE: Meeting of ACRS Subcommittee on Human Factors

Please forward the attached materials to Mr. David Ward. They will be used in preparing his report of the 2 November 1981 Subcommittee meeting.

W. MONROE KEYSERLING, PH.D.
26 ST. PAUL STREET NO. 5
BROOKLINE, MASSACHUSETTS 02146
(617) 277-1987

7 November 1981

MEMORANDUM

TO: David Ward, Chairman, ACRS Subcommittee on Human Factors

FROM: Monroe Keyserling, ACRS Consultant

RE: Comments on Subcommittee meeting, 2 November 1981

Attached are written comments regarding the presentations and discussions during the recent meeting of the ACRS subcommittee on Human Factors. I hope that you will find them useful in preparing your report to the full committee.

Due to the short preparation time, the attached comments are rather terse. If any points are either unclear or need further elaboration, please feel free to call me at (617) 732-1167.

A-350

A. HUMAN FACTORS ENGINEERING BRANCH (HFEB) PRESENTATIONS

1. NUREG-0700 "Guidelines for Control Room Design Reviews"

Comments:

1. The HFEB should be commended for its efforts in developing this document. The table of reference materials included in the Appendix is a particularly nice feature.

2. Similar guidelines should be developed for evaluating Human Factors deficiencies for facilities, equipment, and procedures external to the control room.

2. NUREG-0801 "Evaluation Criteria for Detailed Control Room Reviews"

Concerns:

1. As an educator in the Human Factors area, I am concerned that an insufficient number of qualified Human Factors specialists (see Section 2.1.1.1, page 8) exist at the present time to meet the demand which will be created by this document.

2. With increased Federal defense spending and the development of new weapons systems, the defense industry will experience an increased demand for Human Factors specialists in the near future. This will aggravate the shortage faced by the nuclear power industry.

3. The NRC must be prepared to deal with this shortage of qualified personnel. Careful audit procedures should be implemented to assure the qualifications of the Human Factors "expert" on each review team.

3. NUREG-0835 "Acceptance Criteria for the Safety Parameter Display System (SPDS)"

Comment:

1. The SPDS is clearly needed in every nuclear power installation. NUREG-0835 is a good initial step in accomplishing this goal.

Concerns:

1. At the present time, the HFEB favors a CRT-type display over other alternatives because of its relatively low cost. A potential problem with the CRT concept is that a single tube is not large enough to display all SPDS parameters at a single time.

2. Because of this limitation, operators will have to select the parameters which are displayed at any instant. This could result in two basic types of problems:

    a. Important information will not be seen by operators because it is not currently displayed.

    b. Unless formats are uniquely distinguishable for each display mode, operators could associate displayed information with the wrong parameter. The effects of this error could be disasterous in an emergency situation.

3. The HFEB should give strong consideration to using conventional displays (e.g., dials, tiles) for the SPDS. If CRTs are used, safeguards must be developed to assure that the problems described above do not arise.

B. PUBLIC PRESENTATIONS

1. Ms. Lund

Comments: None.

2. Mr. Coley

Comments: None.

C. PROCEDURES and TEST REVIEW BRANCH PRESENTATIONS

1. NUREG-0799 "Draft Criteria for Preparation of Emergency Operating Procedures"

Comments:

1. The document recognizes the reality that human and equipment failures are going to occur in nuclear power operations, and that highly reliable job aids (i.e., operating procedures) must be developed to assure that operators restore the plant to a safe condition.

2. The present draft of this NUREG is vague, and considerable enhancements are needed. It is recommended that the NRC should sponsor additional research in this area because of the ultimate contribution of good emergency procedures to the public safety. (This is reiterated in the budget section below.)
Inputs from INPO and other operator groups should be encouraged in developing the final document.

D. LICENSEE QUALIFICATIONS BRANCH PRESENTATIONS

1. Management Organization and Infrastructure

Comments:

1. During Mr. Crocker's presentation, he listed nine areas of corporate technical expertise. (See page 6 of his handout.) It is strongly suggested that Human Factors expertise be added as a tenth area.

E. DHFS SAFETY TECHNOLOGY PROGRAM PRESENTATION

Comments:

1. A major concern of this program is in the area of operator licensing. Specifically, improved procedures are recommended for operator selection, training, and examination.

2. It is suggested that new licensing procedures should not be implemented until they have been validated. This is easier said than done. It is difficult to validate testing procedures under the best of circumstances (i.e., when it is easy to evaluate operator performance). This is not the case in nuclear power operations because of tbe great difficulty in developing relevant criteria for evaluating an operator's on-the-job performance.

3. It is recommended that the Office of Research become more heavily involved in this area. Specifically, methods for evaluating operator performance must be developed through new research. Once developed, these procedures can be used to improve the validity of operator licensing procedures.

F. OFFICE OF RESEARCH PRESENTATIONS

Comments:

1. In general, the Office of Research has developed a progressive and well-balanced program of human factors studies.

2. Because of the slowdown in the design of new facilities, current and future research activities should concentrate on improving the reliability and safety of existing designs. Only minimal research funds should be spent on developing knowledge for designing the "ideal" control room, unless a stronger national commitment is made for nuclear power.

Recommendations:

1. The research budget in the areas of human reliability and plant procedures should be reevaluated. While it is desirable to model human error rates and predict the reliability of a system, such an analysis becomes moot after an accident (either due to human error or equipment failure) has occurred. After an accident occurs, the risk to public safety is minimized by observing well-planned emergency procedures. I would favor reassigning $1 million from the human reliability area to the plant procedures area.

2. Mr. DiSalvo described several studies concerned with the design of CRT displays. Problems with CRTs are not unique to the nuclear power industry. It might be considerably more cost effective to perform an intensive literature search in this problem area rather than trying to "re-invent the wheel".

3. One or more studies should be funded to determine the best design format for the Emergency Parameter Display System. The current commitment to the CRT format may be premature. (See discussion above.)

4. Funding should be provided to conduct a formal "Critical Incidents" survey among experienced RO's and SRO's. The results of such a study could provide important insights to generic human design deficiencies in nuclear power facilities.

REPORT BY Jim Buck
SUBJECT: ACRS HUMAN FACTORS, NOV. 2 SUBCOMMITTEE MEETING

In regards to NUREG-0799, entitled, "Draft Criteria for Preparation of Emergency Operating Procedures:" This document pertains to features which those operating procedures must have to be acceptable. The term "emergency" was not directly defined here but reference to NUREG-0737 purports to define transients and accidents as emergencies. I can only hope that this very important definition is defined in this earlier document since it is not in Section 2 of 0799. The philosophy of this procedure used to create and assure adequate emergency action procedures for CR operators is that a symptom initiate an action sequence, that the action sequences be singular for a specific set of reactor/sensor system responses, and that the specified sequence be justified by explicit supporting analysis, assumption, and fact, and that these procedures are available to the operators. These are clear necssary conditions. Assuming that the procedures are correct and appropriate for the specific plant, sufficient condition extensions require symptom detection by the operators or an automatic form of detection, proper procedure identification, and a proper execution of the identified procedure. I'm a bit uneasy about these final aspects of sufficiency based on NUREG 0799 due to the limited amount of study time possible to date. Part of this uneasiness stems from the single sequence because the procedure will be written for the lowest level of operator qualification when a more qualified operator might find and correct the fault faster when allowed some flexibility of sequence. I'm also worried about memory limitations of operators. While I strongly support many features of this document (e.g. consistency in describing the desired procedure and justificational support of them), there are some subtle philosophical questions and uncertaintities remaining in my mind. Also I note that almost all of the references refer to type legibility, preference, and understanding; with few references to the amount of information depth, diagram-guided instruction, audio-supplemented instruction, or instructional flexibility. In the author's defense, I don't know of many studies on these alternative approaches.

1. I first state that I consider the efforts of the Human Factors Staff on the Nuclear Regulatory Commission to have done an extremely good job, in the short time frame from last year's meeting. Dr. Steve Hanhuer and his Staff are to be congratulated. In my opinion, the progress is __________ sufficient now that the Human Factors group of the NRC should take a longer more integrated and philophical view point of the regulatory as well as the advisory role which they are ~~di~~ to serve.

2. Some general comments over the meeting follow below:

   a. I ~~said~~ sense an attitude of corrective regulations eminating from the Human factors group as opposed to preventative point of view. This attitude tends to prevail in the sense that which is designed cannot be changed even tho it may or not be in a physical state. Quite clearly engineering drawings are one heck of a lot easier to take than ~~amx~~ are mechanical things but mechanical things can be changed when in fact they are much better. Accordingly, I would urge that the group would approach more of the regulatory aspects from a design in ______________ viewpoint rather than restricting so much ~~effort~~ attitudes towards the paint, patch and tape or ~~and~~ making minor corrections as needed. The point is a situation where a control rooom may have the standard ______________ forms of display. In this case there may very well be a ____________________ retrofit to a TRT control without destroying the existing display panels. Such a change of retrofit may be quite cost effective particularly if __________ the viewpoint because you still got the backup and log display behind them. But irregardless of even such retrofit I think the greatest thing to be gained by the prevention concept as opposed to the corrective concept would be psychological in the minds of the Staff. ~~It was~~ There would be more focus on what should be done than how can we correct some of the problems that currently exist.

b. I believe that the beginning change of attitudes by the Human Factors group toward more task than those that soley pertain to the control room is a very healthy sign. I was particularly pleased to see ~~there~~ their focus shifting to some extent on the ____________ activity as I believe that to be one of the biggest support to the correction of problems that we have in the nuclear electrical power generation.

c. I would urge the Human Factors group to contact other federal agencies whom are involved in regulatory actions such as the FAA, DOT, and find some of the attitudes and cases prevailing there. It's quite clear from the FAA's point of view, for example, that they can regulate the number of people in the commercial airplane cockpit. Similarly to the problems that may prevail in the determination of better crew sizes in the control room. Since the FAA also regulates aircraft maintenance personnel qualifications perhaps this ought to be point of ~~concer~~ concern in the maintenance activities coming up as well. Therefore, I reiterate it would be well for the group to contact the other agencies involved with regulations.to see how they perform their jobs.

d. It seems that there ~~is~~ has been practically zero focus by the Human Factors group on the concept of benefit cost ______ _________ show or cost effectiveness although during the talks we heard several references made to heavy cost of small changes in the control panels of the control room of the nuclear reactors. Of particular concern to me is the marginal benefits to marginal cost ratios . It is a well known economic ~~principle~~ principle that activities which a group must perform ought to be optimally performed in the manner such that the~~y~~ marginal ~~costxbenefits~~ benefits cost ratios are the same for all activities. Accordingly, without a concept of balance in the various regulatory actions of the NRC there is apt to be a heavy emphasis on one part of it and a ______________ to another part of it. It is for this reason I would recommend that some consideration be started, if not be pushed further and the marginal benefit cost ratios and, of course, in the ~~eventxof~~ benefit of cost ratios themselves. I can well understand the timidity of the group towards this concept because of public safety being such a prime aspect. But not public safety can be entirely without risk. and the idea of that would be to put the resources that are finite but available into the most ____________ arrangement for controlling this risk ra$ther than ~~buring~~ burying ~~his~~ one's head in the sand relative to the existence of risk. While the added benefits of some of these regulatory programs may be very difficult to assess I believe that philosophically anyway that the added benefits should not exceed the added cost. This is one of the reasons why I believe in a ~~greatxer~~ greater consideration towards the benefit costs concept.

e. I would recommend that the Human Factors branch of the NRC apply a little bit more human factors in the development of their document of their documentation. for the utilities and the public. I would like to see some type of diagram, for example, which shows the scope of the regulatory activities of human factors and where each document in -its self plays a role.. With the beginning of each document you ~~havexto~~ had to read through alot of prose that sometimes was unclear as to just where this document fit with other documents that were still in active form. Other comments could be made about other parts of the document, for example, a few of the documents had tables of contents which/were so tightly packed together there were extremely hard to read. These are just a few illustrations of utilizing more human factors for the Human Factors Branch as a output of the Human Factors Branch.

g. Most of the reported on activities appeared to me to be heavy in the static form while it is quite necessary to solve the static problems first it is certainly not sufficient that the solution of static problems will solve the dynamic element. It was in TMI-2, as I recall, where some of the dynamics of the situation in the control room were primarily to blame for some of the outcomes. I would, therefore, urge an increase consideration by the Human Factors group, the dynamics that occur along the process as well as the static.

h. Throughout the various documents that were reviewed in that meeting there was a constant reference to the identification of functions that needed to be performed ~~followedxsh~~ followed by the allocation of those functions to either people or machines which needed to perform the functions. While I strongly favor this concept, I philosophically have mixed viewpoints on the elebration of it without the qualifications that the same function can be allocated to a different person or to a person one time and a machine another time, for more effectiveness. There is a tendency with the so-called assignment of functions concept for people to assume that once a function is identified to ~~perxxpersonalxoperatorx~~ a human operator or a computer or whatnot, it should always be so assigned. Nothing can be farther from the truth. My deliemma is that if this implication sort of prevails without any preference in stopping it.

i. There were a number of references during the meeting to simulation. Inevitably, the references, at least from my point of view, appeared to be in what as known as manloop or physical simulation whereby human operators are used in conjunction with equipment that simulates to some degree of fidelity the situation that is performed in a control room or elsewhere. I believe the human factors group is quite correct in their assessment that the physical or maniloop ______ simulation ought to have a strategy for deployment and that the question should be: what the least cost simulation which achieves the desired effect. It would seem to me that many different situations would have many effects that you would really want to achieve. ~~The~~ Some very precise ones in some cases where high fidelity simulation may be very very important. In other cases where you are just trying to train procedures over and over, low fidelity simulation can probably do an inadequate job so therefore I would like and urge the Human factors branch to continue their development toward a simulation strategy which would utilize various types of fidelities of simulators and would also consider the very cost effective system of computer simulation as augmenting concept. In fact, the computer simulation, in my opinion, is a better approach to some of the human

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reliability estimation through the human errors. The Swing Gutman system for human reliability appears to depend almost entirely upon faulty tree ? analysis either angates or orgates to form the logic of the situation. There is no dependency situations in the logic that are easily obtained in this fashion whereas simulation can achieve those subtlcies. Simulation can also serve to show work load effects on the part of the operators, the control rooms or in the maintenance activities. Another feature of computer simulation is that it can give you ~~dynamic~~ its dynamic which cannot be captured in many other ways. Finally, computer simulation will allow inexpensive experimentation to supplement maniloop simulation or physical simulation because they are so cost effective. This brings the use of physical simulators or maniloop simulators back to more of a case of verifying some of the effects that you would have found utilizing the computer simulation. For these several reasons, I believe that computer simulation ~~has~~ is _______overlook methodology which the human factors branch ought to embed in their bundle of tools.

j. I support the concept and urge the NRC to provide more in the form of support to graduate students education primarily for the purpose of bringing human ~~factors~~ factors together with nuclear engineering and industrial engineering in the concept of utility management as well as to supply people with sufficient educational background to perform many of the activities needed through the NRC regulations. Some of the students in this program through their graduate studies could perform some of the small research projects that are currently being contemplated in the NRC research office.

k. In regard to guard lines and utility management structure ~~at~~ and technical resources NUREG-0731, this document appears to be quite generic not of the refined use with exceptions of providing some necessary communication linkages and advising the utilities on some of the aspect and background of shift work. While I would have preferred to see more in this document, the state of the art is not such that a great deal can be said along these lines as to ______________________ as so forth. Perhaps further research will amend this deficiency but as it exists I ~~faun~~ find the document reasonable. Perhaps long for what it says.

l. In regard to NUREG-0835 entitled "Human Factors Acceptance Criteria" for the safety peraimeter display system." This document primarily pertains to the display side and then only ~~S~~ CRT oriented display sides of the control room display system during dinoses and fault correction of the system. When I first read the title I expected a broader form of document than the one it pertains here. There is nothing said about input aspects of ~~thi~~ display information such as a keyboard so that the operator can call up different display formating or could select amongst a variety of display forms that might be of concern to the operator during the diagnosis ______. There is some said but not completely said in regard to response time characteristics. There is some said but not a great deal said on various character formats and so forth. There is also very little said on the multi tubes forms of displaying as opposed to single tubes although this is not ruled out my reading of the document. In general, this document does reference/back to ~~the fact~~ NUREZ 007 in 32 of the 88 different criteria indicated in the document. I was a little surprised in this document not to read more of some of the diagnostic display concepts that have been utilized.

For example, network display consideration whereby the fault problem be can be isolated to a given degree of resolution. There was also very little ~~aaaaaiaaaxafx~~ implications of __________ human factors imbeded in the document except that which would be inferred from the responding criteria. While there are a number of points to this criteria specification, as far as the time of response being too short or way too long it's not very clear as to ~~where~~ what the acceptability limits are. Perhaps this is asking abit much at this stage of the arts. This document also is an example of a table of contents that needs to be better human factors for display purposes.

m. On regard to NUREG 0801 entitled "Evaluation Criteria for Detailed Control Room Design Review". This document pertains to the control room review procedure for the detection and catagorization of human error deficiences. There are a few comments to be made in regard to this document. One is that is not clear how to catagorize the eminary deficiencies from the documental procedure nor is it clear that such a full catagorization of classified HED's is necessary. Another ~~sixiti~~ critisium of this document is that it ~~ae~~ appears to have insufficient references ~~BAIzzzzz~~ ________20700 or more of an integrated picture. This is a particularly useful if various examples could be shown utilizing 0700 so that the catagorization concept could be understood. A further critism of this document is that nothing is said or was overlooked on my part of the organization of the review team. Particularly on who was organizing, who was managing and who decides when added talent is needed or who decides among disagreements within the team. Exhibit 4-1 in this document seems to me to be acceptive.

n. In regard to NUREG 0700 I have commented on this in the past.

Some of the corrections which I have suggested have been incorporated into this document. I still believe technical deficiencies remain and, of course, everyone recognizes that further research is going to bring forth additional points to be made in the supplements of this document. In fact, I think, NUREG 0700 ought to be viewed as probably a very short term but short period of corrective device needs to be made.as a ongoing activity. This same comment may also be appropriate for other documents in this series but 0700 is the prime one in my opinion.

Final paragraph:

This ~~complete my~~ completes my report. I hope it is sufficient for purposes of your next ACRS meeting. However, if you need further specifics, please call me at Iowa City,.319-353-6083 or at home 351-1919.

P. S. Please share this information with Dr. Hanheau's group.

4-365

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

November 12, 1981

APPENDIX XXXI
REPORT OF REG. ACTIVITIES SUBCMTE ON
REGULATORY GUIDE 1.23, Rev. 1

MEMORANDUM TO: ACRS Members

FROM: C. P. Siess, Chairman
Regulatory Activities Subcommittee

CPS

SUBJECT: REPORT OF REGULATORY ACTIVITIES SUBCOMMITTEE ON REGULATORY GUIDE 1.23, REVISION 1

The Regulatory Activities Subcommittee met on November 11, 1981 to complete its review of proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Measurement Programs for Nuclear Power Plants".

Members present were: D. A. Ward
D. W. Moeller
W. Kerr
M. W. Carbon
J. J. Ray
C. P. Siess

Mr. Bender could not attend; comments provided by him are attached. They were considered by the Subcommittee in its deliberations.

Regulatory Guide 1.23, Revision 1 requires additions to the equipment and program for meteorological measurements, chiefly to meet the need for emergency preparedness and emergency management. Practically none of the additional requirements are needed to for the uses made of meteorological data to satisfy the requirements of Part 100 for Siting, of Appendix I for ALARA, or for environmental impact assessments.

The benefits of the additional programs, in terms of health and safety of the public, seem to be real, in the case of emergency management, but are difficult to quantify.

The costs vary greatly. For many plants the additional cost of a back-up system will be only $20,000 to $30,000. For some older plants, with no on-site meteorological measurement program, the cost may exceed $500,000. For plants at coastal or valley sites (about half of those now in use), the costs of surveys and/or additional instruments may be on the order of a few hundred thousand dollars.

The proposed Guide, together with other NRC Staff positions, such as in NUREG-0737, would require implementation of these augmented programs at all operating plants by the end of 1982 and at all other plants before full-power operation.

The Subcommittee recommends that the ACRS concur in the positions of this Regulatory Guide, but that its implementation for those plants requiring extensive and expensive upgrading be considered by the recently-established Generic Requirements Review Committee to determine whether the cost or other burden on the licensee is justified.

A draft letter to the EDO containing these recommedations is attached.

Attachments:
1. Letter from Mr. Bender, dated November 6, 1981, to Dr. Siess, "Comments on Regulatory Guide 1.23".
2. Draft letter to EDO.

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

November 6, 1981

Dr. Chester P. Siess
Professor Emeritus of Civil Engineering
3110 Civil Engineering Building
University of Illinois
Urbana, IL 61801

Dear Chet:

Comments on Regulatory Guide 1.23

As you know I won't be able to attend the December Regulatory Guide meeting when Regulatory Guide 1.23 will be discussed. If it is to be released as a basis for regulatory requirements concerning emergency response, we ought to first discuss it with the Commissioners. I am convinced that the NRC treatment of site meteorology has a badly distorted emphasis for all purposes, but it is at its worst for emergency planning and response applications. Even though the monetary cost associated with the requirements is relatively small compared to total power plant costs, it is extremely high when related to the actual worth of the information. Of more concern, however, is the waste of time and valuable manpower resources by diverting them from more important safety matters really needing attention.

In a fit of sheer exasperation I prepared the attached comments on Regulatory Guide 1.23, but there wasn't time to review my discussion in detail, and most of it came from mental recollection. Still, I don't think there are any serious errors in it. We should recommend to the Commissioners that Regulatory Guide 1.23 be abandoned because its purpose is no longer in keeping with current regulatory safety approaches.

When the guide was introduced, we were using the traditional arbitrary accident source term derived from small fuel melting experiments to establish containment leak tightness. The nuclide releases were based on a uniform mixture of iodine, noble gases and particulates within the contained atmosphere. We know then and now that this was a poor assumption but it was convenient for safety analysis purposes and it is very conservative. The meteorology was mainly used to show the dispersal of radionuclides from the containment at the prescribed leak rate. It was completely artificial. The use of such complicated meteorological analysis to show site boundary limits was not much better than using astrological principles to predict core melt. It was always a lot of mystical hocos pocus.

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Since TMI-2 we have known that the approach was nonsense. We need to consider realistic source terms, but even if we keep the arbitrary source term basis we shouldn't try to use sophisticated meteorology based on very localized measurements to determine site boundary radiation exposure. Nobody with even a microgram of common sense would think we could use such data to show radiation exposure of the bulk population following a major accident.

I suspect that much of what is proposed in Regulatory Guide 1.23 is the product of ACRS inquiries, improperly interpreted, and of licensee pencil sharpening to obtain approval of the less attractive sites, where stagnant air conditions and inversions caused concern for the dilution rates of nuclides dispersed to the environment. We now understand that we have not defined what is to be diluted. All we know is that leakage rates will be higher than assumed for containments but internal containment decontamination will be a great deal higher than was credited. These are offsetting effects that have never been quantitatively related. I, personally, believe that the decontamination effects are much more of a factor than leak tightness, so the situation will be better than expected, barring, of course, pressurized rupture of containment by some very low probability circumstance. Such gross ruptures were never treated in the dose analysis anyhow.

I don't want to discredit the technological approach used in the Regulatory Guide. If one is going to use a complicated plume analysis to determine mass transport of nuclides and if the leakage is also small, making its behavior dependent on meteorological conditions, then a knowledge of air layer movement and diffusion will be needed. However, if the leakage is small enough for such phenomena to be controlling, then the internal decontamination effects will reduce the nuclide outleakage to levels where their radiation effects can be ignored.

What bothers me most about this analysis is that if public safety were really dependent on such sophisticated understanding of meteorology, we would have great difficulty defending the regulatory posture. We are not depending on such analysis but rather using it as procedural rote intended as a reminder of important phenomenological considerations that affect airborne nuclide transport. We could handle it much simpler if we recognized the decontamination effects of low leakage events, permitting us to ignore the meteorology.

We should remember that when nuclear power was initially being developed it was common practice to vent gaseous nuclides to the atmosphere. ALARA had not been invented for effluent control. We used the meteorology to assure favorable conditions for venting. For that application covered by 10 CFR 20 requirements the meteorological interest made more sense. Our current ALARA limits make this venting issue a moot.

I can't help but note the similarity between this situation and the air-controller strike circumstances. In both cases the "overkill" regulatory practice wasn't recognized until the real need was tested. It is obvious in both instances that safety practice was carried far beyond real need by the momentum of the system. Since we are pushing risk evaluation principles in regulatory practice, this type of calculational mysticism should lead the list of practices to be discarded.

Sincerely,

Mike Bender

M. Bender

MB/cw

Attachment

cc: S. Duraiswamy
R. F. Fraley

COMMENTS ON REGULATORY GUIDE 1.23

Background

Regulatory Guide 1.23 has been used for a number of years as the basis for evaluating site acceptability for nuclear power plants. Its roots lie in the basic accident assumptions used in site analysis. In the early development of nuclear power plant safety principles it was presumed that certain forms of radionuclides would be released and the consequences of such releases would be minimized if sites were selected with favorable meteorology in terms of infrequent air inversions, quiescent air conditions or other factors that would reduce the dilution effects on the nuclides that might be released to the surrounding environment. This was a logical position in that it avoided selection of less desirable plant sites when more suitable ones were available.

When 10 CFR 100 became the basis for site acceptability, the regulatory evaluation process, through its use of prescribed analytical methods using arbitrary accident assumptions, introduced an arbitrary but rigorous method of analyzing site boundary accident effects. The calculational methods became highly refined because some sites could only be shown to meet very conservative 10 CFR 100 limits by refining the interpretation of meteorological conditions. The regulatory staff, in order to apply sophisticated meteorological analysis required substantiating data. This in turn led to the requirement for meteorological towers instrumented to obtain data over an extensive time span. Such data were intended to permit integration of radionuclide release effects over a long period of time. The data were never applied to actual accident release conditions.

The data requirement became ingrained in the Construction Permit processing even though for good nuclear sites it was of little value. The data was only needed because of the arbitrary calculational procedures used for evaluating conformance with 10 CFR 100 limits. The data was not of use for the TMI-2 accident and it would not be of use for accidents of more serious nature since other factors dominate the public risk considerations.

Factors Affecting Airborne Radioactive Dispersions

In core damage accidents the physical characteristics of airborne radioactivity are understood only in a gross qualitative manner. The forms of the radionuclides when released from the fuel vary with time because of nuclide radio-decay. The chemical form of the nuclides depends upon reactions with other materials, the most interesting reaction being the cesium iodide combination which ties two important radioactive constituents together in a chemical compound. The physical movement of nuclides from the fuel to the external containment environment depends upon the pathway of release (calculated by the Corral, March, Contain Codes or analagous techniques), the containment constituents of which moisture and dust particulates are probably most important, and the temperature and pressure in containment when the release is occurring. These factors represent two to three decades of uncertainty in the nuclide concentrations that could leave the contained atmosphere. By comparison, the most refined meteorological analysis can improve over the crudest practice by no more than a factor of three. While we may be able to improve calculational procedures within containment, we can never establish the in-containment accident environment well enough to make the external meteorological conditions a determining factor in evaluating human radiation exposures prior to, during, or subsequent to nuclear accidents. The meteorological data are of use mainly in timing controlled release of radionuclides over long time spans in the range of months and years.

The noble radioactive gases, mainly Krypton 85, will mix with the contained atmosphere and could be released in accidents. However, the quantity is very small even for a full core of spent fuel and when dispersed in the external air it will dilute regardless of meteorology to a concentration which will not cause measurable radiation injury at ground level. Hence, for accident purposes it can be ignored with respect to human exposure during accidents where questions of evacuation, escape routes, and local radiation exposure might be involved.

Dispersal of Airborne Activity Released from Containment

Air plume models are used to analyze the dispersion of particulate radionuclides assumed to be released from containment. The models always assume a point source and are usually analyzed by assuming several release locations. Ground level (a few meters above the surface) and locations at the high point of containment or the containment stack usually bracket the effects.

The shape and energy content of the plume are the important considerations in bulk releases such as a containment rupture. Only gross variations in environmental conditions could be treated in the analysis and many of those would be of marginal importance. The existence or nonexistence of rain, mist, or fog is a major factor. An air inversion might have an important effect for determining dispersal rates, but analysis probably can only discriminate between a stagnant air velocity, one of a few MPH or a few tens of MPH. This will indicate how fast the plume will move laterally, but higher velocity winds will also mean rapid plume mixing and dispersal. The combined effect cannot be calculated with meaningful accuracy.

Variations in air temperature and air layering are of importance only for determining exposure if one were measuring integrated effects for long periods of time when the release is continuous and <u>the physical nuclide form is known</u>. The meteorological tower measurements have meaning only in the localized setting where the measurements are made. They cannot be extrapolated for miles. Integrated effects will be determined primarily by air and surface activity samples taken subsequent to the release. The meteorology would hardly enter the evaluation process.

The CRAC code is the analytical technique currently in use for computing airborne accident effects. It assumes an accident and then ana s the release of the activity from a point source in containment. When last reviewed by the ACRS, it still did not account for gross meteorological variations and it did not have the capability to treat air layer diffusional properties. Whether it has or could be refined for this purpose is unclear, but considering other uncertainties the value of such refinement is doubtful. Nevertheless, if the data required in Regulatory Guide 1.23 is of any use it would have to be shown to be needed in a CRAC-code type of analysis.

<u>Meteorological Factors of Importance When Nuclear Accidents Occur</u>

There are some meteorological considerations of importance when an accident of serious nature involving core melt occurs. These are:

1. Wind directions, if they can influence the time and route of evacuation.
2. Variation in wind conditions as indicated from weather service sources when a planned release such as venting is intended.
3. Air stagnancy (inversion) circumstances that might influence dispersal and cause undesirable localized conditions during a planned release.
4. Flood or tornado conditions that might jeopardize recovery operations. This type of information should come from local weather and emergency warning services.

Value of Regulatory Guide 1.23

With our present and prospective radionuclide dispersal knowledge, Regulatory Guide 1.23 is not of value for emergency evaluation purposes. The expressed need based on requirements stemming from NUREG 0654, 0696, and especially NUREG 0737 only indicates that the reference documents are wrong and need correction

This interest in re[illegible]ed meteorological analysis stems totally from the arbitrary analytical procedures used in determining site suitability to 10 CFR 100 conditions. More than likely, NRC could conserve its own and licensee resources by simplifying the computational procedures and putting more emphasis on such accident assumptions as containment leakage, nuclide dispersal within containment, and lapsed time under which accident conditions prevail.

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ON VYDEC
CPS/SD/da
11/12/81
Draft 1

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ACRS ACTION ON REGULATORY GUIDE 1.23, REVISION 1

Dear Mr. Dircks:

The ACRS has reviewed the proposed Revision 1 to Regulatory Guide 1.23,
1 "Meteorological Measurement Programs for Nuclear Power Plants," dated
October 27, 1981, and concurs in the positions therein.

Although we believe that the upgraded systems required by this revision
will contribute in some degree to the management of an emergency, it is
not clear that the resulting decrease in risk to the public will justify
in all cases the costs, which may be substantial at some plants. For
this reason we suggest that implementation of the requirements of this
2 Guide be reviewed case-by-case on a cost-benefit basis for those plants
requiring large expenditures to comply. Such a review would seem to be
an appropriate task for the recently-established Generic Requirements Review Committee.

Sincerely,

J. Carson Mark
Chairman

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The following pages has been deleted as:

Pages: A-375 to ______ DELETION 1

SUMMARY REPORT OF PROCEDURES SUBCOMMITTEE MEETING
NOVEMBER 11, 1981

APPENDIX XXXII
SUMMARY REPORT OF PROCEDURES SUBCMTE.
MTG. OF NOV. 11, 1981

Purpose: To discuss the following items:

I) Scope of ACRS activities

II) Nature/conduct of ACRS activities

III) Management/prioritization of ACRS activities regarding generic matters

Participants

J. C. Mark, Chairman
C. P. Siess, Member
W. Kerr, Member
R. F. Fraley, Executive Director, ACRS

Discussion

I) Scope of ACRS activities

. NSOC Recommendations

NSOC has recommended that nonmandatory review by the ACRS should be implemented so that the Committee can concentrate on more indepth reviews of selected safety issues and can audit NRC Staff performance.

It appears that nonmandatory review is not likely to pass this Congress. To proceed toward nonmandatory review, however, the Committee should spend less time (e.g., limit to 2-3 hrs per meeting) on a project which has previously been reviewed and approved by the ACRS) with concentration on matters identified by the Subcommittee as those needing full Committee attention (e.g., emergency plans, competence of operating organization, etc.).

In order for the Subcommittee review to be more responsive to the interests/concerns of individual members, additional time should be spent during full Committee discussion of anticipated Subcommittee activity so that members can identify areas of interest/concern for examination during the Subcommittee review and would preclude the need for extended discussion of such items during full Committee meetings unless the Subcommittee considered it necessary to pursue the matter at the Committee level. Members would be encouraged to rely on the report of the Subcommittee in these areas rather than conducting their own detailed inquiry.

The need for the Committee to perform a QA function regarding staff activities was discussed briefly. This is already done in many respects on an "ex-officio" basis. In some cases, however, leadership at the head end of the process or guidance during the process may be more appropriate.

Since approximately 1.5 major projects per month can be expected over the next 3 years with possible peaks of 2-3, better management of generic activities as well as more effective project reviews will be needed. Additional comments/suggestions regarding the conduct and management of Committee activities are included under Items II and III.

. Need for additional NRC Advisory Committees

Additional NRC Advisory Committees to deal with Radiological Effects and/or Radwaste Waste Management/Disposal may be appropriate since few ACRS members are expert in this area and heavy reliance must be placed on consultants. It was agreed that further thought is needed before making such a recommedation. It was also agreed, however, that the Committee should not become involved in any additional areas of review (e.g., fuel fabrication) without giving the matter serious thought.

II) Nature/conduct of ACRS activities

. More effective use must be made of Subcommittee activity as noted in I, above.

. More time should be devoted to discussing items with the NRC Staff technical reviewers rather than the applicant or the NRC project manager so that significant technical issues can be addressed.

. Too much time and attention are being devoted to the list of unresolved issues identified by the Staff and, more recently the concerns of intervenors, than items of high risk potential. The Subcommittee should identify and concentrate on the significant issues needing attention.

Use of Subcommittees

. A better definition is needed of Subcommittee authority with respect to such items as:

- approval of proposed staff plans to take interim action (e.g., approve publication of items for public comment. Promulgation of a proposed final rule would be reviewed by the full Committee).

- provide comments to the NRC Staff regarding proposed actions which do not require full Committee report (e.g., it was proposed that in some cases the Committee would be asked only to endorse the transmittal of Subcommittee comments without the need for full Committee review).

It was noted that some Subcommittees already exercise such authority on an "ex-officio" basis in some areas and it works quite well.

Use of consultants

. The manner by which the Committee makes use of consultants was discussed, including the possibility that senior consultants might be "appointed" as members of an ACRS task force on items where the Committee has limited knowledge or interest.

Meetings with Commissioners

. It was noted that Chairman Palladino has asked that ACRS meetings once again be scheduled with the Commission rather than the Chairman. This will be implemented.

. Use of ACRS meetings with the Commissioners as an opportunity to present a convincing case regarding the basis for ACRS views, the degree of concern, etc. was discussed.

Subcommittee members expressed their belief that it was not appropriate for a collegial body to designate a member (or selected members) to speak on behalf of the Committee except on rare occasions (e.g., Congressional testimony). A collegial body must express its views in written form even though it may be cumbersome. Improved ACRS reports, replies may be appropriate, if considered necessary to provide additional information to the Commissioners regarding Committee recommendations. Written replies to Commission questions can and has also been used.

An additional concern was expressed regarding the adversary relationship such briefings might create with the NRC Staff.

. Items identified for discussion with the Commissioners by the ACRS should be limited to items of substance. In the past some items discussed have lacked a substantive level of enthusiasm by the Committee members.

Preparation of ACRS reports

- Subcommittee members endorsed the attached Guide for content/scope of ACRS Project Reports.

Distribution of documents to ACRS members

- Subcommittee members endorsed a proposal to terminate distribution of project related Category B documents received after the Committee's CP review has been completed and before the OL review has begun (e.g., reports regarding construction deficiencies, etc.). These documents represent approximately 40% of the 1200 Category B reports distributed selectively by the ACRS Office each month.

- A more selective distribution of Category B documents received after the OL review was endorsed. The ACRS staff will review the selective distribution list consistent with these suggestions (e.g., documents related to technical specifications, amendments, environmental qualification of equipment, etc. have little interest while LER's, PNO's, SECY's, etc. are of considerable interest).

III) Management/Prioritization of ACRS Generic Activity

- Several alternates were discussed for improved management of the Committee's generic activities including:

"Established" Generic Items (e.g., those on the NRC/ACRS list of generic items)

(a) The ACRS Generic Items Subcommittee will assign generic items to topical subcommittees bearing in mind the importance of the item, workload of the topical subcommittee, etc.

(b) ACRS topical Subcommittees will select from the list of Established Generic Items (EGI's) those items they consider of high priority.

(c) The Generic Items Subcommittee will provide oversight of generic matters handling those items it is competent to deal with and assigning others to topical Subcommittees as appropriate in much the same manner as the ACRS Subcommittee on Regulatory Activities provides oversight regarding proposed rules and regulatory guides.

(d) The ACRS Executive Driector will assign priorities for generic work.

The Subcommittee members endorsed Alternate C. In this connection the Generic Items Subcommittee should work up a list of priorities for those items that are already established generic issues, e.g., USI's, Task Action Plan Items of priority A, B, and C., etc.

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Newly/proposed potential generic items (e.g., proposed by individual ACRS members, etc.)

(a) Topical Subcommittees would be expected to conduct a preliminary evaluation of potential generic items raised by ACRS members, etc. and report to the ACRS regarding their disposition (e.g., pass on to the NRC Staff for action, take no further action because of low risk-reduction potential, etc.).

(b) The ACRS Generic Items Subcommittee would conduct a preliminary evaluation of proposed/generic items and, with the Committee's concurrence, proceed with further action (e.g., refer to the NRC Staff for action, assign to a topical ACRS Subcommittee for action, etc.).

The Subcommittee members endorsed Alternate (b).

Attachment:
Guide for Preparation of ACRS Reports - Content/Format/Scope of ACRS Project Reports dtd 11/12/81

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11/12/81

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

GUIDE FOR PREPARATION OF ACRS REPORTS

CONTENT/FORMAT/SCOPE OF ACRS PROJECT REPORTS

ACRS reports on specific projects should deal with the project being considered and recommendations regarding areas needing resolution should address the priority, specific items of concern, and degree of ACRS participation in resolution as noted below. Matters of a generic nature should be addressed as they apply to the project being reviewed but project reports should not be the vehicle for making generic recommendations to the Commission. Generic matters normally should be handled in a separate report which addresses not only the nature of the concern but also its degree of applicability to specific types or classes of reactors and, if possible, a target date for its resolution and implementation on a generic basis.

Distinction Between "Recommendations" and "Suggestions"

In preparing its reports the Committee should take into account the difference between a recommendation and a suggestion as follows:

Recommendations - the Committee desires/expects that they will be carried out on the time scale and to the degree defined in the body of the report.

Suggestions - may or may not be carried out by the staff/applicant based on their own good judgment. A suggestion would be an exhortation to good practice but not the equivalent of a recommendation. The Committee could, for example, suggest that an applicant continue to work on an issue or continue to contribute to the resolution of an issue (e.g., by support of EPRI or an Owner's Group activities) and incorporate such improvements as may be forthcoming when and if they develop. This would not be considered a licensing condition, however.

Concluding Paragraph

The concluding paragraph in the Committee's report will address the need to resolve recommendations in the body of the report as follows:

"The Committee believes that if recommendations noted above are resolved as indicated [1] and subject to satisfactory completion of construction [2] and preoperational testing, there is reasonable assurance that the Dreyfus Nuclear Plant can be operated at power levels up to 3500 MW(t) without undue risk to the health and safety of the public."

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Notes:

[1] The recommendations in the body of the report should address three aspects as follows:

A) Timing/Priority for resolution

- Before a license is issued - a major issue which will determine if the basic plant can be built/operated safely.

- Before fuel is loaded - an issue with consequences from an inadvertent criticality that are not acceptable.

- Before power operation is permitted - an issue with consequences from an accident with limited F.P. inventory which are not acceptable.

- Before the first refueling (first year of power operation) - an item of some concen that must be resolved within a reasonable period of time.

- When NRC TMI-2 Action Plan (NUREG-0660 and NUREG-0737), Item XX-XXX has been resolved/implemented

- When proposed NRC Rule (or Regulatory Guide) XX-XXX has been implemented.

- When NRC Task Action Plan Item XX-XXX has been completed.

B) Specific action being recommended (e.g., is a plant change needed or is further study needed)

- ACRS recommends that system (or plant) design (or operational/inspection/testing) changes should be incorporated to ... - a problem does exist and ACRS believes it should be fixed by a change in plant design, operation, etc., with the priority noted in A), above.

- ACRS recommends (further) consideration of - a problem may exist and, if it does, needs resolution with the priority noted in A), above. Further study/evaluation is needed to determine if a problem does or does not exist and an appropriate fix. The Committee should indicate if the study is to be done by the NRC Staff and/or the applicant.

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C) Degree of ACRS Involvement

. ACRS plans/desires to review the proposal - should be referred to the ACRS for review with the priority noted in A), above.

. ACRS desires to be kept informed - ACRS should be informed of resolution in writing. This will provide an opportunity for the Committee to take whatever action appears appropriate.

. should be resolved so it is satisfactory to the NRC Staff - ACRS will rely on staff for resolution. ACRS will be informed of resolution by Supplemental SER and related Category B documents.

[2] The list of items in the concluding paragraph that need to be completed before operation have traditionally been limited to "completion of construction and preoperational testing" since these are areas in which the Committee relies on I&E to make a determination after the ACRS review has been completed. It was agreed during the 252nd ACRS meeting that this list can/should be expanded to include any other substantive items which will be completed following the Committee's review (e.g., staffing of a nuclear plant) depending on the specific situation at the particular plant under consideration.

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, E. F. Goodwin to R. F. Fraley, Revised Proposed NRR Agenda Items for the December, January and February 1981-1982 ACRS Meetings, November 6, 1981

2. Letter, J. C. Mark to W. J. Dircks, NRC EDO, Nuclear Safety Information Center, November 17, 1981