ACRS-2428
PDR 3/25/87

CERTIFIED

TABLE OF CONTENTS
MINUTES OF THE
314TH ACRS MEETING
JUNE 5-7, 1986
WASHINGTON, D.C.

DESIGNATED ORIGINAL

Certified By EMB

TABLE OF CONTENTS
APPENDICES TO MINUTES OF THE 314TH ACRS MEETING JUNE 5-7, 1986

314th
6/5-7/86

1100 Pennsylvania Avenue, NW., Washington, DC 20506.

This meeting will be open to the public on a space available basis. The topics will include guidelines and policy.

If you need accommodations due to a disability, please contact the Office for Special Constituencies, National Endowment for the Arts, 1100 Pennsylvania Avenue, NW., Washington, DC 20506, 202/682-5532, TTY 202/682-5496 at least seven (7) days prior to the meeting.

Further information with reference to this meeting can be obtained from Mr. John H. Clark, Advisory Committee Management Officer, National Endowment for the Arts, Washington, DC 20506, or call 202/682-5433.

John H. Clark,

Director, Office of Council and Panel Operations, National Endowment for the Arts.

May 23, 1986.

[FR Doc. 86-12105 Filed 5-29-86; 8:45 am]

BILLING CODE 7537-01-M

Expansion Arts Advisory Panel; Meeting

Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92-463), notice is hereby given that a meeting of the Expansion Arts Advisory Panel (Overview Meeting) to the National Council on the Arts will be held on June 16-17, 1986, from 9:00 a.m.-5:30 p.m. in room 714 of the Nancy Hanks Center, 1100 Pennsylvania Avenue, NW., Washington, DC 20506.

This meeting will be open to the public on a space available basis. The topics will include guidelines, policy and the Five-Year Plan.

If you need accommodations due to a disability, please contact the Office for Special Constituencies, National Endowment for the Arts, 1100 Pennsylvania Avenue, NW., Washington, DC 20506, 202/682-5532, TTY 202/682-5496 at least seven (7) days prior to the meeting.

Further information with reference to this meeting can be obtained from Mr. John H. Clark, Advisory Committee Management Officer, National Endowment for the Arts, Washington, DC 20506, or call 202/682-5433.

Dated: May 23, 1986.

John H. Clark,

Director, Office of Council and Panel Operations, National Endowment for the Arts.

[FR Doc. 86-12181 Filed 5-29-86; 8:45 am]

BILLING CODE 7537-01-M

Media Arts Advisory Panel; Meeting

Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92-463), as amended, notice is hereby given that a meeting of the Media Arts Advisory Panel (Radio Programming in the Arts Section) to the National Council on the Arts that was to be held on May 29, 1986, from 9:00 a.m.-5:30 p.m. in room 716 of the Nancy Hanks Center, 1100 Pennsylvania Avenue, NW., Washington, DC has been changed. This meeting will not be held on June 12, 1986, from 9:00 a.m.-5:30 p.m. in room 716 of the Nancy Hanks Center, 1100 Pensylvania Avenue, NW., Washington, DC 20506.

This meeting is for the purpose of Panel review, discussion, evaluation, and recommendation on applications for financial assistance under the National Foundation on the Arts and the Humanities Act of 1965, as amended, including discussion of information given in confidence to the Agency by grant applicants. In accordance with the determination of the Chairman published in the **Federal Register** of February 13, 1980, these sessions will be closed to the public pursuant to subsections (c)(4), (6) and (9)(B) of section 552b of Title 5, United States Code.

Further information with reference to this meeting can be obtained from Mr. John H. Clark, Advisory Committee Management Officer, National Endowment for the Arts, Washington, DC 20506, or call (202) 682-5433.

Dated: May 23, 1986.

John H. Clark,

Director, Council and Panel Operations, National Endowment for the Arts.

[FR Doc. 86-12182 Filed 5-29-86; 8:45 am]

BILLING CODE 7537-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Nuclear Regulatory Commission; Meeting Agenda

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards will hold a meeting on June 5-7, 1986, in Room 1046, 1717 H Street, NW., Washington, DC. Notice of this meeting was published in the **Federal Register** on May 19, 1986.

Thursday, June 5, 1986

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

8:45 A.M.-12:30 P.M.: South Texas Project, Units 1 and 2 (Open/Closed)—The members will hear and discuss the reports of its subcommittee, the NRC Staff, and the Applicant regarding the request for an operating license for this facility.

Portions of this session will be closed as required to discuss Proprietary Information applicable to this facility and detailed security arrangements for this project.

1:15 P.M.-1:45 P.M.: Topics for Meeting with NRC Commissioners (Open/Closed)—The members will discuss the contents of its report of January 14, 1986 to the NRC regarding the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design Applicable to Future Plants.

Portions of this session will be closed as necessary to discuss Proprietary Information and detailed arrangements for plant security for this class of nuclear plants.

2:00 P.M.-3:30 P.M.: Meeting with NRC Commissioners (Open/Closed)—The members of the committee will meet with the NRC Commissioners to discuss the Committee's report of January 14, 1986 regarding the GESSAR II Final Design Approval as noted above.

Portions of this session will be closed as necessary to discuss Proprietary Information and detailed arrangements for plant security for this class of nuclear plants.

3:45 P.M.-5:45 P.M.: NRC Safety Research Program (Open)—The members will discuss portions of the proposed ACRS report to the NRC regarding the proposed safety research budget for FY 1988-89.

5:45 P.M.-6:45 P.M.: Future ACRS Activities (Open/Closed)—The members will discuss anticipated ACRS subcommittee activity, and proposed items for consideration by the full Committee.

Portions of this session will be closed as required to discuss National Security Information.

Friday, June 6, 1986

8:30 A.M.-10:30 A.M.: Recent Operating Experiences at Nuclear Facilities (Open/Closed)—the members will hear and discuss the reports of its subcommittee, and representatives of the NRC staff. Representatives of the nuclear industry will participate as appropriate.

Portions of this session will be closed to discuss Proprietary Information and detailed security arrangements for the facilities being discussed.

10:45 A.M.–12:00 Noon: Reactivation of Deferred and Cancelled Nuclear Plants (Open)—The members will hear a briefing regarding major issues in reactivation of nuclear power plant construction projects.

1:00 P.M.–1:30 P.M.: ACRS Subcomitee Activities (Open)—The members will hear and discuss a report by its subcommittee on thermal Hydraulic Phenomena regarding proposed NRC activities in this area.

1:30 P.M.–3:00 P.M.: NRC Safety Research Program (Open)—The members will continue discussion of the Committee's proposed report to NRC regarding the proposed NRC safety research program for FY 1988–89.

3:15 P.M.–5:15 P.M.: Source Term for Nuclear Power Plant Accidents (Open)—The members will hear and discuss proposed revisions to the accident source term used in evaluation of nuclear power plants.

5:15 P.M.–5:45 P.M.: ACRS Subcommittee Activities (Open)—The members will hear and discuss the report of its Management Subcommittee regarding procedural topics considered during its subcommittee meeting on June 4, 1986.

5:45 P.M.–6:30 P.M.: Appointment and Activities of ACRS Members (Open/Closed)—The members will discuss the report of its Nominating Panel regarding candidates nominated for appointment to the ACRS. The members will also discuss the proposed reappointment of a member of the Committee and the non-ACRS activities of ACRS members.

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

Saturday, June 7, 1986

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

Portions of this session will be closed as necessary to discuss Proprietary Information, detailed security arrangements, National Security Information, and information concerning initiation, conduct, or disposition of a formal agency adjudication applicable to the matters being discussed.

1:30 P.M.–2:00 P.M.: ACRS Procedures (Open)—The members will discuss proposed changes to ACRS Bylaws and procedures for the conduct of ACRS activities.

2:00 P.M.–3:00 P.M.: Miscellaneous (Open/Closed)—The members will hear a report by a member of the Committee regarding participation on an ANS Panel to discuss ACRS recommendations on severe accidents. The Committee will also complete discussion of matters considered during this meeting.

Portions of this session will be closed as necessary to discuss controlled and classified information as noted above.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 2, 1985 (50 FR 191). In accordance with these procedures, oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Committee, its consultants, and Staff. Persons desiring to make oral statements should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a prepaid telephone call to the ACRS Executive Director, R.F. Fraley, prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with section 10(d) Pub. L. 92–463 that it is necessary to close portions of this meeting as noted above to discuss Proprietary Information (5 U.S.C. 552b(c)(4)) applicable to the facilities being discussed, detailed information related to the security arrangements at a nuclear power plant (5 U.S.C. 552b(c)(3)), information the release of which would represent a clearly unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)), classified restructed data (5 U.S.C. 552b(c)(3)), and information concerning initiation, conduct, or disposition of a formal agency adjudication (5 U.S.C. 552b(c)(10)).

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

Dated: May 27, 1986.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 86–12154 Filed 5–29–86; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50–369 and 50–370]

Duke Power Co.; Consideration of Issuance of Amendments to Facility Operating Licenses and Proposed No Significant Hazards Consideration and Determination

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF–9 and NPF–17 issued to Duke Power Company for operation of the McGuire Nuclear Station, Units 1 and 2, located in Mecklenburg County, North Carolina.

The amendments would authorize on an emergency basis a one-time release of the existing contents of the Conventional (non-radioactive) Wastewater Basin, containing trace amounts of tritium, into the Catawba River. Technical Specifications (TS) 3.11.1.1 and its referenced Figure 5.1–4, "Site Boundary for Liquid Effluents" define the authorized discharge point for radioactive material released in liquid effluents to unrestricted areas as being only to Lake Norman. The proposed authorization would be accomplished by the addition of a footnote to TS Figure 5.1–4 at the discharge point for the Conventional Wastewater Basin into the Catawba River, stating that this discharge point is authorized for a one-time discharge of water which contains trace amounts of tritium in addition to the normally processed effluents of the Waste Water Collection Basin, effective the date of Commission approval. The change would not affect any existing limits or procedures regarding the processing of conventional (i.e., non-radioactive) contaminants.

These revisions to the technical specifications would be made in response to the licensee's application for amendments dated May 20, 1986.

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

An unexpected release of tritium into the Conventional Wastewater Basin has created the need for prompt action as proposed above for two reasons, both stemming from the fact that the Basin is

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

Revised: June 3, 1986

SCHEDULE AND OUTLINE FOR DISCUSSION
314TH ACRS MEETING
June 5-7, 1986
WASHINGTON, D. C.

Thursday, June 5, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

1)	8:30 - 8:45 A.M.	Report of ACRS Chairman (Open) 1.1) Opening Statement (DAW) 1.2) Items of current interest (DAW/RFF)
2)	8:45 - 12:30 P.M. (BREAK - 10:30-10:45)	South Texas Nuclear Plant, Unit 1 (Open) 2.1) Report of ACRS Subcommittee regarding an OL for this unit (JCM/MME)
	TAB 2 ---------	2.2) Meeting with NRC Staff and Applicant (Note: Portions of this session may be closed to discuss Proprietary Information and security arrangements for this facility.)
	12:30 - 1:15 P.M.	LUNCH
3)	1:15 - 1:45 P.M.	Preparation for Meeting with NRC Commissioners (Open/Closed) (Note: Portions of this session will be closed as necessary to discuss detailed security provisions and Proprietary Information applicable to GESSAR II.)
4)	2:00 - 3:30 P.M. TAB 4 --------	Meeting with NRC Commissioners (Open/Closed) 4.1) Discuss ACRS report on GESSAR II dated January 14, 1986 (Note: Portions of this session will be closed as necessary to discuss detailed security provisions and Proprietary Information applicable to this matter.
	3:30 - 3:45 P.M.	BREAK
17)	3:45 - 4:15 P.M.	Future Activities (Open) 17.1) Briefing by H. R. Denton, NRR, regarding IAEA meeting on the Chernobyl reactor accident
5)	4:15 - 6:00 P.M.	Reactor Safety Research Program (Open) 5.1) Discuss proposed ACRS report to NRC regarding the Safety Research Program for FY 1988-89 (CPS, et al./SD et al.)

6) 6:00 - 6:45 P.M. Future ACRS Activities (Open/Closed)

TAB ---------- 6.1) Anticipated Subcommittee activity (MWL) (Open)

TAB ---------- 6.2) Proposed items for ACRS consideration (DAW/RFF) (Open)

TAB ---------- 6.3) Consideration of N-Reactor review (Open/Closed)(SJSP/RKM)

(Note: Portions of this session will be closed as required to discuss National Security Information.)

Friday, June 6, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

7) 8:30 - 11:15 A.M. (10:00 - 10:15 - BREAK)	NRC Safety Research Program (Open) 7.1) Discuss proposed ACRS report (CPS, et al/SD, et al)
6) 11:15 - 12:00 Noon	Future ACRS Activities (Open/Closed)
TAB ----------	6.1) Anticipated Subcommittee activity (MWL) (Open)
TAB ----------	6.2) Proposed items for ACRS consideration (DAW/RFF) (Open)
TAB ----------	6.3) Consideration of N-Reactor review (Open/Closed)(SJSP/RKM) (Note: Portions of this session will be closed as required to discuss National Security Information.)
12:00 - 1:00 P.M.	LUNCH
10) 1:00 - 3:00 P.M.	Recent Operating Experiences at Nuclear Facilities (Open/Closed) 10.1) ACRS Subcommittee Report (JCE/HA) 10.2) Briefing by representatives of NRC Staff (Note: Note: Portions of this session will be closed as required to discuss Proprietary Information and Safeguards Information applicable to these facilities.)
3:00 - 3:15 P.M.	BREAK
11) 3:15 - 5:15 P.M.	Source Term for Nuclear Power Plant Accidents (Open)
TAB 11--------	11.1) Report of ACRS Subcommittee (WK/MDH) 11.2) Meeting with representatives of NRC Staff and contractors as appropriate
12) 5:15 - 5:45 P.M.	ACRS Activities (Open) 12.1) Report of Management Committee regarding June 4, 1986 meeting items (DAW/RFF)
13) 5:45 - 6:45 P.M.	Appointment/Activities of ACRS Members (Open/Closed)
SEE HANDOUT----	13.1) Report of ACRS panel regarding nomination of candidates for appointment to the ACRS (HWL/ALN) (Closed) 13.2) Reappointment of ACRS member whose term is expiring (WK/ALN) (Closed)

13.3) Non-ACRS activities of ACRS members (Open/Closed)
   13.3-1) H.W.Lewis testimony regarding nuclear future (Open)

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

Saturday, June 7, 1986, Room 1046, 1717 H Street, NW, Washington, D.C.

14) 8:30 - 12:30 P.M. ACRS Reports to NRC (Open/Closed)

14.1) Discuss proposed ACRS reports to NRC regarding:

14.1-1) NRC Safety Research Program (SD,et al)

14.1-2) South Texas, Unit 1 (JCM/MME)

14.1-3) Reassessment of Source Term (WK/MDH)

14.1-4) Recent operating experience at nuclear facilities (JCE/HA) (tentative)

TAB ---------------- 14.1-5) Examples of systems interactions (DO/RPS) (Tentative)

(Note: Portions of this session may be closed to discuss Proprietary Information, detailed security arrangements for the plants being considered, and information that will be involved in an adjudicatory proceeding.)

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

15) 1:30 - 2:00 P.M. ACRS Procedures (Open)

15.1) Proposed change in ACRS Bylaws regarding

TAB ---------- Preparation of Minority Reports (DAW/RFF)

8) 2:00 - 2:45 P.M. ACRS Subcommittee Activity (Open)

8.1) Report of Thermal Hydraulic Phenomena Subcommittee regarding activities in this area (DAW/PAB)

16) 2:45 - 3:00 P.M. Miscellaneous (Open)

16.1) Complete discussion of items considered during this meeting

(Note: Portions of this session may be closed to discuss Proprietary Information, Detailed Security Arrangements for plants being discussed, information the release of which would represent an unwarranted invasion of personal privacy, National Security Information, and information concerning initiation, conduct, or disposition of a formal agency adjudication.)

CERTIFIED

MINUTES OF THE
314TH ACRS MEETING
JUNE 5-7, 1986

The 314th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street, N.W., Washington, D.C., was convened by Chairman D. A. Ward at 8:30 a.m., Thursday, June 5, 1986.

[Note: For a list of attendees, see Appendix I. D. Okrent, F. J. Remick, and C. P. Siess did not attend the meeting on Saturday, June 7.]

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

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

I. Chairman's Report (Open)

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

Chairman Ward indicated that he and T. G. McCreless, ACRS Assistant Executive Director, visited the Wingspread site and found it very satisfactory. He noted that an invitation has been extended for participation by representatives of the Soviet Union. The new NRC Chairman, L. W. Zech, Jr., will formally open the meeting.

Chairman Ward noted the retirement of R. B. Minogue, the Director of the Office of Nuclear Regulatory Research, and the retirement from Government of D. Eisenhut as of June 13, for a position with NUS Corporation.

II. South Texas Nuclear Plant, Unit 1, Operating License Review (Open)

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

J. C. Mark described the site of the South Texas project indicating that it is a Westinghouse 4-loop PWR of 3,800 MWt. He suggested that the three-train system for cooling might be of interest to some members as is the fact that the RHR pumps, which are in the containment, can withstand the containment environment under all conditions. D. Okrent indicated that he has two general questions that he would like to have addressed during the presentations. The first of these involved quality control and quality assurance activities at the South Texas plant and a discussion of the detailed results of a preliminary scoping probabilistic risk assessment study.

J. H. Goldberg, HL&P, presented a brief history of the South Texas Project (see Appendix IV). He described the site and the ownership of the project by four partners, Carolina Power & Light, HL&P, and the cities of Austin and San Antonio, Texas. He indicated that in November 1979 a special NRC inspection team investigated a number of quality concerns that focused on harassment of quality control inspectors and difficulties with concrete and nuclear welding. In December of that year a stop-work order was issued on complex concrete placement. In the spring of 1980, the NRC stopped work on nuclear welding and shortly issued a show-cause order and assessed a civil penalty of $100,000. With the help of numerous consultants, by October 1980 the project was able to demonstrate that key problems that impeded the quality of the job were under control and welding was restarted. The pouring of complex concrete was restarted in January 1981. F. J. Remick asked the reason for the April 1980 show-cause order. J. H. Goldberg indicated that it was to show cause why the construction permit should not be suspended. It was directly related to welding and concrete placement problems and the harassment of quality control inspectors.

J. H. Goldberg indicated that by September 1981, after years of frustratingly slow progress on the project, the project owners reluctantly agreed that the project's interest would be better served with a more experienced architect-engineer. Bechtel Corporation was hired in the fall of 1981 and Brown & Root elected to withdraw totally from the project. In February 1982 Ebasco Services was hired to take over the duties as constructor. The current project structure is one with Houston Light & Power Company functioning as project manager, Bechtel functioning as architect-engineer and construction manager, and Ebasco Services functioning as constructor. J. C. Ebersole noted again that the plant is a Westinghouse design. He asked if the balance of plant is basically a Brown & Root design, an Ebasco design, or a Houston Light & Power design. J. H. Goldberg indicated that the basic structural configuration of the station is a Brown & Root design with a considerable amount of the nuclear analysis done in 1975 by NUS Corporation. Nevertheless, Brown & Root did not do much of the design of the safety-related cable trays and raceways and virtually none of the nuclear piping design. Most of the mechanical and electrical auxiliary building design was done by Bechtel Corporation. The containment design was a collaboration of Bechtel and Westinghouse. G. A. Reed complimented HL&P on the turbine orientation, noting that the layout of the turbine building is such that this plant is one of the first to have an arrangement where turbine missiles are not a factor in the penetration of key areas such as the diesel rooms, control rooms, or the containment. He suggested that Brown

& Root should be given credit for recognizing early on the best turbine building arrangement. M. R. Wisenberg, HL&P, pointed out that the orientation of the turbine building was an issue of concern in the early design stages of the project and modifications were made as a result of NRC staff questions.

J. H. Goldberg discussed the construction organization, its philosophy and status. He briefly summarized HL&P's management philosophy. These include a commitment to build and operate the South Texas station in full compliance with applicable regulations, to require that people who do work take full responsibility for its quality, and to require quality assurance to independently confirm the quality of activities being performed. There is extensive management oversight of the entire program to ensure compliance with applicable program requirements. HL&P also reports in a timely and forthright manner all matters requiring attention by a regulatory authority (see Appendix V). HL&P upper management organization was discussed.

D. Okrent asked how many of the managers have reasonable technical insight into the potential causes of severe accidents with respect to potential scenarios that severe accidents can follow given the different sets of constraints. J. H. Goldberg indicated that HL&P has two basic engineering organizations. An engineering group on the project and an off-project engineering team called Nuclear Engineering which handles most of the analytical work (core physics, thermal hydraulics analysis, and probabilistic risk assessment). He noted that within the off-project engineering team at least a half dozen engineers might well qualify regarding knowledge of severe accidents. P. Dodson, HL&P manager of engineering, indicated that a half dozen individuals on the project engineering team would also quality. The nuclear engineering group on the project had participated in actual running of some of the Westinghouse codes. South Texas operations people have been very close to the Westinghouse Owners' Group Emergency Response Guidelines and emergency operating procedures. These individuals would also be knowledgeable of severe accident scenarios. D. Okrent agreed that very few in operating engineering groups may have knowledge of severe accidents but he noted that no mention was made of upper management. J. H. Goldberg mentioned the capabilities of J. G. Dewease, Vice President of Operations; W. Kinsey, Plant Manager; and K. K. Chitkara, Manager of the Off-Project Nuclear Engineering Group, as well as E. Dodson, Manager of Project Engineering, who bring extensive experience to the South Texas project.

J. C. Ebersole pointed out the fact that HL&P has a number of problems with the Westinghouse turbine. He spoke of the extensive turbine inspections performed by Westinghouse to protect the plant against physical damage potentially caused by turbine explosion.

J. H. Goldberg agreed with J. C. Ebersole's assessment of the situation.

E. Dodson discussed the plant layout and identified some of the project's unique features (see Appendix VI). The 46 acre essential cooling pond was described. He noted that it functions as the ultimate heat sink for the plant to provide cooling water for safety-related systems. W. Kerr asked the maximum temperature of the pond during a worst-case accident. M. Wisenberg indicated that the temperature used in the design calculations that were performed for a thirty-day capable pond was 102° Fahrenheit. E. Dodson pointed out that the site consists of two plants that are slide-along duplicates. Each plant is physically separated from the other and has its own safety-related and nonsafety-related systems. Only a few nonsafety-related support systems are common to both plants. C. Michelson asked if the heating, ventilating and air conditioning system is just for the control room. E. Dodson indicated that there is an entirely separate system for the control room besides the rest of the electrical auxiliary building. J. C. Ebersole pointed out that the distribution of air occurs in common duct work which is fed by redundant chillers in air moving systems. D. W. Moeller noted that all have a common outside air intake. E. Dodson agreed but indicated that it is a concrete duct. D. W. Moeller wondered if isolation of that concrete duct to the control room would turn off the air supply to the electrical auxiliary building. E. Dodson agreed to discuss it later in the session. He mentioned that the diesel generator building is compartmentalized into three compartments for the three identical Cooper diesel generators which service Class 1E sources for on-site AC power. C. Michelson asked if the diesel is self-supporting in terms of its own battery supplies. E. Dodson indicated that it does not have to have external power. C. Michelson noted that water spray deluge systems are used for fire protection. E. Dodson indicated that the water spray deluge is in the diesel room itself actuated by a pre-actions signal. It is seismically qualified. C. Michelson thought it significant that South Texas can operate the diesel generators with the deluge system on. He wondered if all the equipment in the rooms was qualified for deluge including the switchgear, batteries and other electrical equipment. In answer to a question by F. J. Remick, E. Dodson indicated that all three diesel generators are independent from the standpoint of fuel and air supply.

E. Dodson explained that the South Texas project uses a three-train design instead of a two-train design which is physically segregated and electrically independent. There are no shared components for heat removal from the core or containment atmosphere or heat rejection to the ultimate heat sink. He pointed out that the South Texas project has the capability to shut the plant down with one of three trains rather than one of two. Three trains also provide greater margins since, for the majority of the analyzed possible accidents, one of three trains can successfully mitigate the accident. There is single train shutdown capability for fire protection for small break loss of coolant accidents, small breaks in general, and normal shutdown. It was revealed during a brief Committee discussion that the South Texas project can be credited with three 100 percent trains with the exception of the design basis accident, the non-mechanistic double-ended pipe break which requires two trains. E. Dodson added that if the double-ended pipe break is postulated to occur it is assumed that one train would spill and one train would experience active failure. One train would be expected to inject. Therefore, three 100 percent trains are required for this scenario.

The electrical auxiliary building air distribution was discussed extensively including fire protection features, including fire dampers and chilled water systems in places such as the switchgear room where the circulating air cannot remove enough heat in a fire. D. W. Moeller voiced some concern regarding the fact that water deluge systems are placed on the charcoal filters on the recirculating system to comply with Regulatory Guide 1.52 even though the ACRS has complained that water deluge systems should not be required in this application. There have been cases where they have activated and inadvertently shorted electrical cables. C. Michelson asked if this deluge system is seismically qualified as was the case with the diesel generator room. E. Dodson indicated that the deluge systems are seismically qualified not to operate. D. W. Moeller asked if there is a plan to test the emergency ventilation system for the control room regarding measurement of the rate at which temperature increases in the control room. W. Kinsey, HL&P, indicated that HL&P plans to do a preoperational test of the control room ventilation system and will check temperature rise, recirculation and leakage in accordance with technical specifications at 18-month intervals. D. W. Moeller asked if HL&P was familiar with control room habitability studies that the NRC staff has had underway for the past few years. E. Dodson indicated that they were.

E. Dodson explained that the three-train design coupled with the plant layout provides considerable advantages for fire protection including two ways to shut the plant down in the event of a fire in any area. He pointed out that the three-train capability extends to auxiliary shutdown capabilities including control of all three trains and the capability to maintain cold shutdown from the auxiliary panel. He noted that HL&P has compartmentalized the plant to limit the vertical propagation of a fire by creating separate fire areas at each of the elevations. He explained that the auxiliary feedwater system is a four-train system including a steam-driven feed pump with segregation of the trains with one train for each of the four steam generators. Three independent

Class 1A power sources feed the three trains. G. A. Reed pointed out a potential vulnerability of the auxiliary feedwater system on loss of all AC power (station blackout) since there is only one turbine-driven pump. E. Dodson acknowledged the weakness. C. Michelson pointed out that the NRC Staff's SER mentions a safety-related cooling water system with three 50 percent capacity trains. He asked the applicant to explain how HL&P characterizes them as three 100 percent capacity trains. E. Dodson explained that the trains are characterized by HL&P as 100 percent trains depending upon the amount of load shedding that is necessary. HL&P considers them 100 percent trains for the purposes of accidents since all that is needed is one out of those three trains. J. Bailey, HL&P, indicated the situation can properly be characterized as three 50 percent trains regarding the design basis accident but 100 percent trains for lesser events. G. A. Reed asked if the South Texas Project has feed-and-bleed capability. E. Dodson indicated that the plant does have feed-and-bleed capability but not with a loss of AC power because there are no other feed pumps except the one turbine-driven pump that can operate with no AC at all.

E. Dodson explained that the control room of the South Texas Project fully complies with NUREG-0737, Supplement 1. The control room design review integrated the human factors design post-accident monitoring instrumentation, safety parameter displays, emergency operating procedures, safety-grade cold shutdown capability, bypass and inoperable status monitoring for engineered safety features equipment and enunciator alarm prioritization (see Appendix VII). F. J. Remick asked where the SPDS CRTs are read. E. Dodson indicated that they are on the main panel and also on various boards. R. L. Balcom, HL&P, explained that the dedicated SPDS CRT is on the operator's console and there are also CRTs in the Technical Support Center and the Emergency Operations Center. C. Michelson asked about the fire protection system for the cable spreading rooms and the switchgear. S. West, NRC, indicated that the switchgear rooms are considered heavily cabled areas and the Applicant has agreed to put in a fixed deluge suppression system manually actuated.

E. Dodson explained that another major aspect of the control room integration effort was the Qualified Display Processing System (QDPS). The objectives of this system was to optimize the instrumentation design to include evolving regulatory requirements such as Three Mile Island, Appendix R and Safety-Grade Shutdown Criteria, as well as to provide optimized cable routing using the latest digital technology. J. C. Ebersole asked about the degree of redundancy of the QDPS. E. Dodson indicated that it is a fully separated three-train system. E. Dodson explained that the QDPS is a digital monitoring system which offers graphic displays which support the operating procedures, while using fewer panel indicators and simpler control panels. It relieves the operator of the burden of cross checking redundant indicators, performs quality checking of input signals, simplifies instrumentation of signal distribution using the data links while monitoring itself through on-line diagnostics and self-calibration. D. Okrent requested further information on quality checking on input signals. He was concerned regarding the possibilities and under what circumstances the computed averages and deviations of redundant input sensors might lead to wrong information for the operator. T. H. Crawford, HL&P, indicated that the chances are very small and would require multiple channel failure of that signal. E. Dodson described the extensive verification and validation program to ensure the proper functioning of software and hardware. He noted that the actual software and hardware is being tested in the Unit 2 system. R. L. Balcum explained that the data are displayed either as questionable data or bad data based upon the redundant sensor calculation. The operator has backup indications and is not taught to believe the indicators blindly but to use the other plant parameters and the knowledge of events to further analyze the situation.

D. Okrent reintroduced his concern regarding the adequacy of the design and construction from a quality and quality assurance point of view. He raised the specter of finding significant gaps in the overall plant quality after the ACRS issues its operating licensee report. C. J. Wylie and C. P. Siess assured D. Okrent that the Applicant would talk later in the session regarding the unique features of his nuclear assurance program.

E. Dodson discussed station blackout and the fact that the South Texas Project falls into the 4 hour plant station blackout general design criteria. He noted that a plant-specific procedure for station blackout has been developed in conjunction with the Westinghouse Owners' Group Emergency Response Guidelines to ensure that proper operator action would be taken. The maximum seal leakage has been calculated to be approximately 25 gpm per pump or 100 gpm total. Each Class 1E battery can supply station blackout loads for approximately 8 hours. E. R. Dodson noted that seal cooling can be maintained by operator action through a positive displacement pump powered from a balance of plant diesel which is in addition to three standby diesels on-site. In addition to this there are five balance of plant diesels that can be hooked up if necessary and five balance of plant batteries that can be made available. He noted that the reactor coolant pump seal leakage was based upon results of tests that were run in France on a 7-inch seal assembly which showed 16 gpm. D. Okrent pointed out that the French have chosen to provide a direct source of reactor coolant pump seal cooling water as part of a backfit to all of their plants. He noted that the British Sizewell B plant will have a similar modification. He asked if HL&P had specifically considered that technique and discarded it. E. Dodson indicated that HL&P has not considered that actual technique but had looked at the situation primarily from the reliability of the electrical grid. HL&P has not looked at the cost benefit or option of backfitting the steam-driven charging pump as being done in France.

E. Dodson briefly discussed prevention of an explosion involving the diesel generator oil storage tank. He indicated that the room in which the tank is located is continuously ventilated from ceiling to floor to remove potential flames. The fan is spark-proof on the 1E bus and the rooms are provided with a foam-water fire suppression system. Doors to the rooms are water-tight and locked closed and the tank level is monitored in the control room. As a result, an explosion is not deemed to be a credible event. C. Michelson and J. C. Ebersole expressed particular interest in the fire protection aspects regarding overfilling of the diesel fuel tanks. J. C. Ebersole suggested that the problem is the level indication system on the fuel tank and its singular capacity to fail. N. P. Kadambi, NRC, agreed that it is a hazardous situation but indicated that the Staff had not actually looked at the scenario of overfilling the diesel fuel tank. He suggested that this question could be addressed in an SER Supplement. M. W. Carbon asked if there might be some sort of common mode failure of filling all the oil tanks with a supply of bad oil. M. L. Balcum indicated that there are technical specification limits on the fuel oil put into those tanks and HL&P has a rigid sampling program prior to filling the tanks. Only one tank is filled at a time and a source would be sampled before filling. J. C. Ebersole asked if HL&P has sought to avoid crash cold starts on these diesels. M. L. Balcum indicated that HL&P has a surveillance program for a once-a-month start of the diesels from an emergency start signal. The diesels are unique, however, in the fact that they are not cold started since there are support systems that maintain them hot. C. J. Wylie asked if a vibration analysis is done using extended runs on the diesels to ensure that piping disconnections do not take place. He asked if such <u>in situ</u> tests will be run to pick out vulnerable spots for vibration and fatigue of pipe connections. It may take 750 hours of continuous running to do such a vibration analysis. E. Dodson indicated that HL&P is doing vibration analyses throughout the plant but not such a test on the diesels.

E. Dodson discussed additional design features of the South Texas Project, including the qualified residual heat removal system inside the containment, the backup power for the chemical and volume control positive displacement pump, and steam generator sludge supports at the preheater. Similar design aspects such as the three-train systems that are being used in both France and Belgium were mentioned. He noted that the South Texas Project has made several modifications to the Model E Westinghouse steam generators at the secondary side of the plant to protect their investment. J. C. Ebersole remarked about the commonality of all the steam generators regarding the header arrangement from main feedwater and recent check valve failures at other plants. He postulated a burst in the main feedwater header system with improper function of check valves in common which could drain all the steam generators backwards through the feedwater system. He asked if HL&P had considered the dynamic reverse flow problem and the rapid closure of valves. E. Dodson indicated that the block valve is a gate valve and it closes equally well in either direction as does the flow control valve which is upstream of that valve. G. A. Reed pointed out recent removal of tilting disk check valves in that same location and the insulation of swing checks because of problems with valves. E. Dodson indicated that HL&P removed the swing check valves because of the problems that that type of valve has been experiencing.

C. Michelson brought up the issue of the detailed acceptance criteria developed by the NRC staff for the selection of pipe systems under GDC 4 to which revisions to the pipe break hangers will be applied. He noted that in the case of South Texas Unit 1 this provision is to be applied now. He asked the Staff how they intend to apply their criteria. V. Noonan, NRC, indicated that the appropriate Staff members were not in the meeting room to address that question. He indicated that the Staff would make a submittal to the ACRS on this subject. C. Michelson thought that since the Staff intends to apply these criteria to South Texas now he did not believe it unfair to ask the Staff for the acceptance criteria at this time. C. Michelson asked the Applicant what they have requested of the Staff regarding GDC 4. M. Wisenberg indicated that HL&P has asked the Staff for permission to take advantage of the existing rule on GDC 4 for main loop breaks. There is a discussion pending regarding a submittal relative to balance of plant breaks. All of those breaks will be inside containment. C. Michelson expressed concern since the Staff appears to have decided on application of GDC 4 for breaks outside of containment. He noted that the South Texas SER suggests this fact. He did explain that his concern would go away if the piping were only inside containment. D. Okrent expressed his interest in also seeing criteria for pipes outside of containment.

J. E. Geiger, HL&P, indicated that the South Texas Nuclear Assurance Program during the operational phase will consist of Operations QA, an independent safety engineering group (ISEG), a safe team program (employee concerns), and the Fitness for Duty Program (see Appendix VIII). He defined the responsibilities of the Quality Engineering Group and the Quality Control Inspection Group. He mentioned a Technical Services Division which will perform other necessary and important tasks to implement a comprehensive QA program. Technical Services Division assignments are not in the nature of day-to-day support as is the operations QA division previously described. J. C. Ebersole brought up the issue of welding problems that the South Texas Project had had back in the early days with Brown & Root. He asked how the issue of weld inspection (metallurgical review) was handled from a quality control and quality assurance standpoint.

J. Geiger discussed the formation of the Independent Safety Engineering Group whose responsibilities include providing continuing systematic and independent assessments of plant activities including maintenance and modifications. D. Okrent asked what, if anything, this group would be doing about systems interactions. J. Geiger indicated that they will perform reviews and some analyses of selected problems that occur at South Texas and will do, on a selected basis, some root cause analysis. He stressed that he was not singling out systems interactions as an independent activity but said that the group would undertake activities with one of the important features of those activities being systems interactions. He described the Safe Team Program as an administrative program for the purpose of providing a forum for South Texas Project employees to identify concerns in the area of nuclear safety quality. D. W. Moeller wondered about the number of responses from employees regarding concerns and deficiencies. J. Geiger indicated that since September 1984 the company has contacted almost 18,000 individuals and received 580 concerns related to nuclear safety or quality. D. A. Ward asked if any of those 580 concerns have resulted in some significant change to systems or the program. J. Geiger indicated substantiation of roughly 110 concerns which did result in some modifications to the course of business. The Committee discussed verification of the qualifications of welders which arose as a result of an employee allegation.

J. Geiger discussed a fitness for duty program based on the Edison Electric Institute Guide. He indicated that the program at the South Texas Project has ten key elements which include top management support, written policy, programs training, liaison with law enforcement as well as chemical testing. He mentioned a urinalysis test for illegal drugs and a breathalizer test used for drunkenness. He mentioned strong emphasis on behavioral observation of employees by supervisors. A supervisor can request that a subordinate be given a chemical test at random. The Committee discussed the chemical testing program and its implications. G. A. Reed raised the issue of the qualifications of QC personnel with regards to the issue of independence of QC versus technical qualifications. J. Geiger indicated that all inspectors on-site are certified current to relevant ANSI standards.

J. Geiger discussed the construction transition program, the transfer of responsibility from Brown & Root to Bechtel. In answer to D. Okrent's question regarding the possibility of significant quality or quality assurance issues arising during the remainder of the project, he spoke of 230 work packages in the transition program which covered items such as current status of the engineering, including design verification, licensing items that were pertinent, such as I&E bulletins, circulars, and necessary SER changes, recommendations for any significant corrective actions, a summary of work in process, and assumptions of special conditions. All factors were cross referenced. Open non-conformances in this transition from Brown & Root to Bechtel/Ebasco had been made the responsibility of Bechtel/Ebasco. Houston Light & Power personnel performed QA audits and surveillances and the NRC conducted inspections and reviews. J. H. Goldberg added that Bechtel accepted technical responsibility for the work previously performed by Brown & Root as a contractual condition. D. Okrent expressed concern regarding the depth of sophistication and the incentive Bechtel had to find Brown & Root errors. J. H. Goldberg indicated that Bechtel had a strong incentive to do a thorough review of Brown & Root's work in part because their professional reputation was at stake for any mistake that might be committed however unintentional. There was also no financial penalty to conduct extremely detailed reviews. J. Geiger indicated that Bechtel reported several major findings as a result of their reviews in the following areas:

° penetrations through the main cooling reservoir and erosion of the soil around pipe penetrations;

° transformer-size insufficient to handle loads on safety-related buses;

° defective weld joints discovered during inspection of emergency cooling reservoir;

° capacity of safety-related HVAC insufficient to handle safety-related heat loads chiller capacity increased after Bechtel took over as architect-engineer and construction manager.

D. Okrent noted that Bechtel has a penalty clause in its contract. If an error is found some years from now they have to pay for its repair. He asked if Stone & Webster also had such a clause in its contract. R. A. Frazer, HL&P, indicated that there is a participation agreement with Stone & Webster which obligates them to call to HL&P's attention any matter that they deem to be questionable or deficient from a technical point of view. D. Okrent asked how large a penalty could be assessed on Bechtel. J. H. Goldberg indicated that Bechtel could conceivably forfeit its entire fee for the job.

J. Geiger discussed an effectiveness inspection program conducted by HL&P staff as a reinspection of work that had been previously inspected by Bechtel Corporation or Ebasco Services. These inspections were designed to replicate results to reach a determination as to the quality of the inspection effort, not the quality of the hardware. In the 1985-86 time frame, these reviews were identified as a limited readiness review audit program. These independent reviews were performed by independent contractors supervised by HL&P management. Previously troublesome topics which were reviewed were seismic interaction, concrete materials control, environmental qualification, structural steel and settlement monitoring. There were no findings of safety-related problems. F. J. Remick asked if HL&P thought that this readiness review concept was a worthwhile effort. J. H. Goldberg indicated that this review was similar to that undertaken at the Vogtle plant by Georgia Power. It was not particularly useful to the South Texas Project because those issues reviewed were ones with a history of being troublesome and were issues that had been previously solved permanently. Such programs invariably turn up problems that somehow have been missed and the result is a never-ending examination.

J. D. Dewease, HL&P, described a nuclear group organization as consisting of groups assigned to plant operations, licensing, nuclear assurance, engineering and construction, special assignments, nuclear safety review board and corporate services (see Appendix IX). He discussed the Nuclear Training Department's program design to apply the systematic approach to training concepts and the major commitment of the organization to performance-based training. G. A. Reed noted HL&P's use of the Edison Electric Institute POSS and MAST tests for preselection of personnel regarding training and reassignment. J. D. Dewease indicated that HL&P believes in aptitude testing.

J. D. Dewease briefly described the Nuclear Security Department (physical protection and safeguards services) and the nuclear construction organization which is a composite of engineering and construction functions. He indicated that the staffing for the operations phase activities continues essentially on schedule for about 1,400 persons for both Units 1 and 2. G. A. Reed thought that the 1,400 person staffing level was ambitious. He wondered how all of these individuals could be utilized efficiently. C. J. Wylie asked where in the organization plant vulnerabilities and interactions are investigated. J. H. Goldberg indicated that once the plant is operational, the engineering and construction department will continue to have a staff of engineers to conduct basic review of the design from the standpoint of systems interactions.

C. J. Wylie asked who would maintain the PRA reliability analysis. J. H. Goldberg indicated that that would be done in the Nuclear Engineering Group. C. J. Wylie noted that reliability analysis and systems interactions are interrelated. J. H. Goldberg agreed but noted that the engineers who will be conversant with the physical design and design criteria for the systems themselves will be part of the engineering and construction management group. They will be supported in PRA analysis capability by the Nuclear Engineering Group as a coordinated effort.

W. H. Kinsey, HL&P, discussed the nuclear plant operations department which is responsible for the safe operation, maintenance and testing of the station (see Appendix X). He explained that the Reactor Operations Division will operate six shifts, each with a complement of nine personnel per unit. He indicated that there will be a technical support organization responsible for providing engineering support to the other line organizations that report to the plant superintendent. One section of this organization is called the Systems Performance Section which is responsible for monitoring plant performance through testing, observation of operating parameters through plant tours, and review of plant maintenance work requests. The Reactor Performance Section is responsible for routine monitoring of core performance. The engineers in this section will hold an SRO license and serve as shift technical advisors. HL&P believes that the decision to license the shift technical advisors will help to make them an integral part of the shift's group.

W .H. Kinsey discussed the Maintenance Division indicating that the maintenance philosophy of HL&P is a strong preventive maintenance program with close supervision of the work. The preventive maintenance program will account for approximately 60 percent of expended maintenance man-hours. The maintenance organization is responsible for the station's measuring and test equipment program with the exception of chemical laboratory equipment and radiation protection. There is root cause determination for equipment deficiencies. D. A. Ward asked if there is a strategy or plan for the ratio of resource allocation to preventive maintenance versus corrective maintenance. W. H. Kinsey indicated that about 60 percent of the man-hours projected for maintenance performance will go directly into performing preventive maintenance as supporting the preventive maintenance program. C. Michelson asked about color coding of valves and pipes to avoid confusion between Units 1 and 2. W. H. Kinsey indicated that since the two plants are separated by 1,000 yards in distance the confusion between 1 and 2 has been eliminated.

W. H. Kinsey discussed the work of the Radiological Protection Section which is responsible for implementing corporate and station policies regarding radiation protection. He mentioned HL&P's commitment to the concept of ALARA which is reflected throughout the organization including corporate management. He indicated that J. H. Goldberg has set a maximum limit of 5 rem per year for any individual while working at the South Texas Project and has set an administrative limit of 4 rem per year. D. W. Moeller mentioned the Regulatory Guide 1.97 requirement for the calibration of the containment high range monitors which should be capable of reading up to 10^8 r/hr. J. Rosenthal, NRR, explained how the Staff deals with calibration of these instruments while avoiding the horrendous potential radiation exposure to technicians. D. W. Moeller noted that the radiation protection manager is really the Health and Safety Services Manager. J. H. Kinsey indicated that HL&P emphasizes occupational health and safety as well as radiation protection. C. J. Wylie asked which group in the station organization reviews modifications for their safety significance coordination and implementation during the operational phase. J. H. Kinsey indicated that while there is a coordination process between operations engineering and quality assurance departments the primary responsibility lies with the engineering department. The plant manager's staff is obviously very concerned about review of those modifications and the nuclear safety review staff will also be involved, as well as the ISEG. C. Michelson asked how HL&P processes LERs generated by other utilities. The discussion centered on the fact that HL&P as well as other utilities rely on INPO to process, categorize and screen all LERs.

L. Constable, NRC Region IV, discussed the overall inspection program at the South Texas Project as well as the status of allegations. He indicated that the NRC has spent a great deal more time inspecting the South Texas Project than it would normally spend at a nuclear plant, in part because of the interesting past of this facility, and expects to incur over 30,000 hours of inspection effort through 1986 (see Appendix XI). He explained that the preoperational inspection program is just getting started and the bulk of the system testing inspection effort so far has been procedural reviews. Generally, the staff has been impressed with most SALP results for this utility. A CAT team inspection in late 1985 uncovered major problems and the Staff is considering appropriate escalated enforcement actions. An enforcement conference was held with the utility. D. A. Ward asked what the problem areas are. L. Constable indicated that some of the problems involved isolated incidents with individuals regarding QA inspections of welding. G. A. Reed asked if this is influenced by the organizational structure of quality control. L. Constable indicated that it was too early to say if that was the problem. The NRC does seek the benefit of the Applicant's input and has not gotten to the point of issuing violations. The allegations are typically well founded and fairly normal for a project at this stage in construction. The allegations are fairly routine.

N. P. Kadambi indicated that the ACRS Subcommittee members requested additional information on five questions after the Staff's presentation of license conditions, open items and confirmatory items. There were no comments on the Staff SER. He indicated that the Subcommittee raised concern regarding the degree of protection afforded by the separation between trains in the bunkered system which characterizes the South Texas design. Reliability problems associated with fire dampers was an issue. He noted that the Staff has issued two information notices and received a report from Ruskin, one of the manufacturers of these fire dampers. C. Michelson indicated that he was under the impression that there were no cross ties between ventilation trains and no need for fire dampers. H. Dodson, HL&P, indicated that there are dampers from the common supply exhaust intake and outlets into those trains, and there are cases where a common wall exists between the trains. C. Michelson asked regarding the powering of these dampers and their failure mode. H. Dodson indicated that they are DC powered. J. C. Ebersole indicated that it does not matter how they are powered. They are not redundant and of questionable reliability. N. P. Kadambi indicated that the Staff has received 50.55(e) reports from the Applicant describing deficiencies in their damper systems and corrective actions. HL&P has concluded that the problem is resolved. The implementation, of course, is subject to NRC inspection. S. West, NRR, indicated that the fire dampers in question are most likely released by fusible links. Any release by a smoke or heat detector would be powered by the diesel generators so they would stay open until there actually is a fire. J. C. Ebersole wondered whether the fusible link would function early enough in temperature rise to preclude overheating of electrical apparatus. S. West indicated that fusible links are available with different temperature readings anywhere from 165°F on up. J. C. Ebersole wondered whether any electrical apparatus could tolerate a 165° ambient temperature. S. West indicated that you would presume the equipment in the fire area actually lost and, even though there might be 165°F in the duct work at the damper, one is not probably to have the same temperature in areas adjacent to the fire area. J. C. Ebersole thought the Staff was counting on an ill-defined temperature gradient.

N. P. Kadambi discussed the ACRS' concern regarding the issue of fire protection in the diesel fuel oil storage areas, especially the proximity of these storage areas to the control room. He indicated that the Staff has found acceptable the design changes proposed by the Applicant to augment fire protection of these storage areas. He explained that an ACRS question regarding separation of battery rooms from the balance of plant was a case of ambiguous wording in the SER. The staff had in mind not the separation between the battery room and the balance of plant but the balance of the equipment in that train for the particular battery room.

N. P. Kadambi indicated that in the case of fires in cable trays, the Staff has taken into account applicable Sandia National Laboratory tests in conjunction with the combustion tests on IEEE 383 qualified cables. He noted that the Staff does not believe that some recent tests having to do with fire in cabinets are applicable to the fire potential for cable trays. The spread of fires within cabinets is characterized by different mechanisms than for cable trays. J. C. Ebersole indicated that the question revolves around whether you have an autocatalytic and progressive fire on the trays caused by burning of a great mass of cables. S. West explained that the ignition resistance and flame spread properties of the qualified cables is much less than for unqualified cables. J. C. Ebersole suggested that this is just skirting the issue, that the issue is that the cables would burn but not perhaps as briskly as unqualified cables.

N. P. Kadambi indicated that the last of the questions from the Subcommittee meeting involved the relevance of the San Onofre November 21, 1985, event to the South Texas design. The San Onofre event, a water hammer phenomenon, led to check valve failures. He indicated that there is a strong defense against a similar event at South Texas. The defense has to do with the fact that there are separate lines for the feedwater and the auxiliary feedwater systems. In addition, the feedwater line has an ESF actuated isolation valve in addition to the check valve.

N. P. Kadambi indicated that the current SER does not speak about an exemption from GDC 4. In Section 362 of the SER the Staff states specifically that South Texas conforms to GDC 4 and pipe rupture postulation and associated effects. The Staff has received exemption requests from South Texas related to the primary coolant loops and pressurizer surge line. The exemption requests for the primary coolant loops has been rendered moot by the limited scope rule recently approved by the Commission in its final form. The surge line exemption is being reviewed at this time. The Staff has not developed any criteria by which to accept or reject it. C. Michelson noted that the broad scope rule out for public comment pertains to all piping inside and outside containment. He asked if the Staff is considering eliminating required breaks including arbitrary intermediate breaks and terminal point breaks outside containment. N. P. Kadambi indicated that the request from the Applicant only applies the leak before break concept inside containment. C. Michelson referred the Staff to Appendix G of the SER which implies elimination of arbitrary intermediate breaks both inside and outside containment. N. P. Kadambi indicated that this Appendix is not an exemption from GDC 4 but is viewed by the Staff as a deviation from the Standard Review Plan. M. R. Wisenberg explained that relief regarding arbitrary intermediate breaks are implied both inside and outside containment. This issue which was handled by the NRC Staff is intended to be outside containment and part of what will be covered ultimately by the broad scope rule.

C. J. Wylie asked if the Staff intends to require testing of diesel generators to assure their long term capability of operation. Such in situ testing would be of vibration, fatigue, or analysis of vibration fatigue. C. Berlinger, NRR, indicated that the Staff does not have a requirement for licensees qualifying a diesel to run for 10^7 cycles as was done in the case of the TDI requalification program. That program was done primarily because of identified specific problems with regard to design quality assurance, quality control, and manufacturing. C. J. Wylie pointed out that during the TDI testing program, it was found that piping designed by the architect/engineer such as oil piping to the diesels had corroded after many hours. There were a number of reports of broken oil pipes. He suggested that the piping systems connected to the diesels be vibration tested. C. Berlinger indicated that the problems experienced with the piping were primarily concerned with adequate restraints on high pressure fuel oil piping or actual mechanical or material defects discovered in some of the tubing. He asserted that if the pipes are adequately supported, there should not be a vibration problem. He also noted that the level of vibration on the diesel engine support structure and the foundation is monitored with sensors. An acceptable way to do vibration testing is to utilize walk-down examinations during engine operation during preoperational tests. That is general industry practice but is not an NRC requirement. C. J. Wylie contended that a 750 hour test (30 days under full load) was necessary to do a proper test of the piping. V. Noonan, NRC, indicated that the Staff intends to issue two more supplements to the South Texas SER this year. He expected that many of the open items will be finalized and closed out in those supplements.

A security briefing on the South Texas Project was held in closed session. The discussion of this portion of the meeting will be found in a supplement attached to the end of the minutes.

III. Recent Operating Experiences at Nuclear Facilities (Open)

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

A. Possible ATWS Event at La Salle Unit 2

D. Allison, NRC, discussed a possible ATWS event that occurred at the La Salle County Station Unit 2 on June 1, 1986 (See Appendix XIII). The plant was operating at about 93 percent power and had a feedwater transient that brought the reactor water level down very close to the trip set point of 12.5 inches instrument level. At that point the operator took action and the level went back up. C. Michelson asked how far above the core 12.5 inches is. D. Allison indicated the zero point is about 155 inches. During the incident, one of the four reactor protection system channels indicated a low level, tripped, and gave a half scram. The other three channels did not. Apparently at the time the operator thought he had made it through the transient. During a later review at Commonwealth Edison of the records of the transient, it began to look like the level had gone down to about 6 inches where all four switches should have tripped. It was concluded that the reactor protection system may have malfunctioned and an alert was declared and the plant shut down slowly. An IE investigation team at the site has so far concluded that the set points on the pressure switches that give the low level trip signal vary by a few inches. It appears that this is what caused the lack of a trip. The switches have displayed setpoint problems at Oyster Creek, but the drift noted there is not nearly enough to cause a real safety problem in this particular application. W. Kerr asked the significance of the formation of an inspection team. D. Allison indicated that they are there to assist the region and make sure that the problem was only because of the failed switches. E. Jordan, NRC, indicated that the Staff plans to issue an information notice promptly communicating what the Staff knows to this point. He explained that the plant would remain shut down until the Staff has investigated the problem. There are 130 of these switches used in that particular plant. W. Kerr indicated that he got the impression that these were fairly newly installed switches. D. Allison acknowledged that the switches are newly installed at La Salle and Oyster Creek. They are part of an upgraded environmental qualification modification and unlike the previous installation, one cannot tell that they are drifting except by doing a calibration check. The previous installation was a Yarway Level Indicator which could be read every shift or every day. These are blind switches in that all that can be done is the application of a test pressure and a calibration check. C. Michelson asked if Commonwealth Edison is using the Yarway to indicate control room reactor level. He surmised that they had replaced the control function of the Yarway with this separate switch. D. Allison agreed. He noted that the investigation will determine why the switches failed, investigate problems of feedwater pumps that caused the transient and operator reaction and whether the incident was reported to the NRC properly. He mentioned that the company did not realize they had a problem for some hours and did not report it to the NRC for 13 hours. G. A. Reed thought that the Licensee ought not to be criticized. He thought this was good performance on the part of the Licensee in the fact that they reported the incident as soon as it was recognized even though it was many hours after it had occurred. E. Jordan noted that once the utility had notified the NRC that their plant had failed to trip, they voluntarily took the plant down to try to find out what had happened. D. Allison pointed it out to the committee that there was a previous trip of the plant on May 9, 1986, where the level was going down and kept going down. During that incident, the operator noticed the level below 12.5 inches and decided to scram. An automatic scram occurred before the operator could execute a manual scram.

B. Reactor Scram at Palisades

W. Hehl, NRC Region III, indicated that the Palisades Plant experienced a reactor trip from 99 percent power in response to a high pressurizer pressure condition on May 19, 1986. The high pressurizer pressure condition was the result of the loss of control power to the turbine EAC system which allowed the turbine valves to close. Upon reactor trip and during the plant recovery, numerous pieces of equipment failed to perform. As a result of the reactor trip and associated multiple equipment failures, as well as the potential serious challenges to safety systems posed by these failures and the burden that these failures place on the operating staff, the NRC dispatched a fact-finding team to review the event prior to the unit returning to power. The Augmented Inspection Team (AIT) was tasked with performing an independent review of the May 19 trip to assure that the scope of the equipment failures was accurately known to evaluate the equipment failures, to gain a perspective regarding the impact of these failures and any existing out-of-service equipment on the operating staff, and to assess their ability to respond to the plant transients (see Appendix XIV). W. Hehl presented background information on the Palisades plant which included the troublesome SALP report covering the period November 1984 to October 31, 1985. The areas of maintenance surveillance, quality programs, and administrative controls were singled out as problem areas due to a lack of aggressive corrective action by the Licensee and poor management controls. Prior events at the facility which began in late 1985 due in part to inadequate maintenance involved safety-related equipment including five separate events related to leaking safety injection tank check valves. Despite maintenance on these valves during the cycle 5 refueling outage, during cycle 6 two of the 8 valves had to be refilled. Additionally, during the cycle 6 refueling, the Licensee elected not to perform maintenance on the primary coolant pumps despite indicated seal oscillations.

W. Hehl explained that during a March 1986 startup from a refueling/maintenance outage, two of four primary coolant pumps developed seal problems. Valve leakage problems were also identified in the primary coolant system loop, in check valves, and in two safety injection tank pressure control valves and a manual isolation valve associated with the safety injection tanks in the three way valve in a CVCS system. Also mentioned were on April 10, 1986, exceedance of the technical specification limit for unidentified primary coolant system leakage and subsequent shutdown. On April 11, 1986, a derating took place because of a pump packing failure and valve leakage problems in the primary coolant system makeup system during the period April 23-29, 1986. D. Okrent asked if this pattern of repeated failures is to be expected and is similar to the average for nuclear plants. W. Hehl indicated that the repetitiveness of equipment failures is far from what would typically be expected. D. Okrent asked if many of these failures would not have occurred had the Licensee done more maintenance during the cycle 5 shutdown for refueling. W. Hehl indicated that there has been significant concern on the part of Palisades operators with regard to maintenance activities at Palisades and the reliability of equipment. There has been experience with leaking check valves since 1983 and vendor assistance has been used in rebuilding these valves. D. Okrent asked if the Staff knows why this problem has occurred. W. Hehl suggested the inadequacies of these valves for the application as a possibility. J. C. Ebersole wondered what the Staff's course of action would be in differentiating between inadequate design and poor maintenance. D. Allison indicated that a new reactor coolant leak is occurring at Palisades every two to three days and the Staff does not know whether it is a design or a maintenance problem. J. C. Ebersole suggested that it is the Licensee's option to put in better equipment. W. Hehl indicated that there are not many nuclear plants like Palisades and Palisades does have some unique problems regarding the location of their safety injection tanks. Palisades is somewhat unusual in its experience with this type of repetitive and continuous problem with check valve leakage. G. A. Reed pointed out that the Palisades plant has been in operation since about 1970 and these valves were basically sound at that time but possibly in need of careful maintenance in the intervening years. It was his impression that the Palisades maintenance organization is lacking.

W. Hehl discussed the sequence of events during the May 19, 1986 reactor trip at Palisades. He noted that the pressurizer spray valve failed to fully close. G. A. Reed suggested that the Staff identify whether the pressurizer spray valve is a bellow-sealed valve or whether it is just a packed valve because that may have been a factor in its failure to close. G. A. Reed mentioned a loss of packing on a pressurizer spray valve at the Zion Station suggesting a connection with this incident. D. Okrent noted that the reactor tripped on pressurizer high pressure indicating that the reactor coolant system was heating up and that there was a subsequent cool down. He asked if there was any violation of cool down rate during the aftermath of the transient. W. Hehl indicated that the failure that occurred did not result in significant worsening of the plant transient. The performance of the operators in the other major plant systems performed as expected and well within design criteria.

W. Hehl indicated that the AIT inspection, based on their review of the apparent failure modes, the maintenance history, and discussions with the Licensee's maintenance organization finds that significant weaknesses exist in the areas of diagnostics, troubleshooting repair, and post-maintenance testing. These were contributors to most of the failures that occurred. J. C. Ebersole asked if the Staff has identified a managerial problem at Palisades. W. Hehl indicated that within the last two years significant management changes have occurred at Palisades. With the abandonment of the Midland facility, there was a truncation of the management at Palisades incorporating part of the Midland management. Part of the problems observed is rooted in the experience level of both the Palisades plant maintenance operations engineering and the inexperience of the management staff. G. A. Reed asked if Consumers Power uses validated aptitude testing for employment and transfer of their personnel at the Palisades plant. He suggested that the Staff look into this matter since it appears that there have been years of problems with respect to the performance of plant people in operations and maintenance. He noted that while the Big Rock Point Plant is much smaller, it is not noted to be a particular problem. W. Hehl indicated that it is not uncommon to have differing levels of performance within the same utility. G. A. Reed thought this might be attributable to a lack of standardization in the evaluation and processing of new employees. Perhaps they need more regimentation. W. Hehl indicated that this might be part of the problem. D. Okrent asked if there are any objective indicators that support the Staff's opinions concerning the quality of maintenance. W. Hehl cited examples of incidences of poor maintenance in the repeated reworking of valves. G. A. Reed suggested that the forced outage rate is probably very high for this plant. This is one of the best indicators of poor performance. W. Hehl cited financial pressures on the utility in part from their harsh treatment by the public service commission of the state of Michigan. Many of their employees at Palisades have had to take significant pay cuts and the attrition rate in their operating department has been of the order of 25 percent. G. A. Reed suggested that possible identification of a parallel case in TVA. He wondered whether the NRC Staff and the regional people will be able to turn this situation around. The committee discussed the fact that because of public utility commission action, Consumers Power may not be able to hire appropriate staff, as with the case with Davis-Besse. E. Jordan noted that the EDO has reviewed the actions of the region taken with respect to

Palisades and believes that it is appropriate. He also noted that there is a periodic or quarterly look at plants that are most troubled and the Staff wishes to return to the Committee with a status report on the Palisades plant. He noted that it is interesting that the Staff is finding that the problems with Palisades are principally based on balance of plant equipment rather than safety-related equipment. While there is no NRC requirement to cover this situation, there are certainly precursors of serious problems out of the large number of balance of plant failures. D. A. Ward noted that from his review of the SALP ratings for Palisades over the last six years, ratings appeared to improve in the 1980-83 period and have now deteriorated again. He wondered if IE and regional activity has increased in response to the decrease in ratings. E. Jordan indicated that the inspection programs are now adjusted and are based on the poor performers. The better performers get less inspection. The poor performers more. Palisades is receiving a great deal of inspection attention. D. W. Moeller observed that is significant that close to 10 percent of the operating nuclear plants are currently shut down and unable to return to power without careful reviews by the NRC Staff.

C. <u>Repeated Snubber Failures at Trojan</u>

T. Chinn, NRR, discussed failure of steam generator hydraulic snubbers at the Trojan Nuclear Plant, Unit 1. The Staff's concern was over stressing of the reactor coolant system piping (See Appendix XV). In February 1985, the Trojan Plant was issued surveillance inspection technical specifications for large bore snubbers. In April, Trojan was shut down for refueling and 16 steam generator snubbers were inspected at that time. Two of the snubbers were tested and both failed. Based on a management decision by Pacific Gas and Electric, all 16 steam generators snubbers were declared inoperable. These failures were attributed eventually to restrictive acceptance criteria for the control valves. All 16 snubbers were disassembled, inspected and reassembled. They were then retested, found to be acceptable, and placed back in service. Based upon marks that were evident within the snubbers which indicated that they had exhibited motion, it was not concluded at that time that any of the snubbers had actually locked up during the 1984-85 cycle. During an April 1985 outage, a hot leg to the Steam Generator B pipe whip restraint to lateral support member was found pulled from the wall about 5/8 of an inch. Since 1982, Trojan had observed erratic pressurizer surge line movement and over the past three years had been monitoring this motion in order to determine its cause. A consultant called in by the Licensee to evaluate the pressurizer line surge movement and was made aware of the fact that two steam generator snubbers did not pass acceptance tests. The consultant concluded that the surge line movement was in fact attributable to the locked snubbers. In addition, it was determined that overstressing under the worst case condition assuming the steam generator snubbers were locked from the cold position at the beginning of the 85-86 cycle, overstressing of the hot leg elbow to B's steam generator could have occurred.

In April 1986, the 16 steam generator snubbers were reinspected and 11 of the 16 were found not acceptable on functional test acceptance criteria. The determination of failure of the snubbers was attributed to inadequacies in the design of the control valve. Because the regional staff were concerned about possible of overstressing of the RCS piping, the Staff requested several follow-up actions to assure the soundness of the piping. A UT test was performed on the steam generator elbow to pipe weld and no indications were found. The Licensee, NRR, and the Region walked down the portions of the RCS piping and observed some evidence of restrained thermal growth. UT testing was also performed on all four hot leg elbows with no indications found. The snubber control valves were replaced with one of a new design. As a follow-up action prior to restart for the 1986-87 cycle, the Licensee is to monitor the thermal growth of the reactor coolant system during heat up and during operations to assure that the predictive thermal growth and clearances are all acceptable. The Licensee will also verify the assumption of the locked snubbers causing the erratic pressurizer surge line movement and the damage which was observed on the pipe whip restraint. The NRR is also reviewing the Licensee stress and fatigue reports to assure that the integrity of the RCS piping is intact.

D. Okrent observed that the reason that these snubbers were tested was because there were snubber technical specifications for the first time. He wondered why this was the first time that this type of failure of hydraulic snubbers was observed in a plant. R. J. Kiessel, NRC, indicated that during the 1970's when the initial technical specifications were issued on hydraulic snubbers, a visual examination of all snubbers was expected, as well as a functional testing of a sample of the snubbers. They also included an exemption for any snubbers that were in difficult or unusual locations. There was also a size limitation such that any snubber over 50 KIPS was not required to be tested. Part of the rationale at that time was that there were not sufficient facilities to perform adequate testing on the large snubbers. In November 1980, NRR issued a generic letter which revised the technical specifications by removing the 50 KIPS limit and modifying the sampling plans. Apparently, the Trojan plant had not resubmitted their technical specifications until late in 1984-85. This was the first time that the snubbers had been tested. He pointed out there have been a number of other instances with large steam generator snubbers that failed to lock up and also continually locked up. Trojan is not the first plant to have encountered the problem. T. Chinn explained that the arrangement of the snubbers and their hydraulic lines at Trojan is such that the design intent is that if one snubber locks up then the flow would not go through the check valves which would activate that particular snubber. The four steam generator snubbers are arranged in a parallel arrangement such that the necessary fluid flow to permit motion could be transferred to the remaining three snubbers. A failure of one snubber would not prevent the necessary motion. J. C. Ebersole asked if there are any generic letters to warn people of this problem. T. Chinn indicated that NRR and IE are working together to evaluate the generic implications of this occurrence. Trojan is one of very few plants which utilizes this particular control valve and the control valves have been changed out to a new design at the Trojan plant as they have at many of the other utilities.

D. Single Failure of Miniflow Logic at Pilgrim

J. C. Ebersole explained that the Pilgrim Nuclear Power Station still retains a system called loop selection whereby a large size break on one side of the reactor in such a two flow system will result in difficulty delivering enough water for low pressure injection. The solution to the problem appears to be a cross tie with a valve which would send the water from the three pumps towards one but not towards the empty loop. Such a modification necessitates an extremely fast transient DP gradient and the signal generation system would never have worked. Subsequent findings are that the low pressure system served more than just a spray system as it also constituted an inventory make-up function as well. It was decided to lock out the loop selection logic at that plant. What was found at Pilgrim was that the loop selection logic was still in existence and its presence could cause a potential for certain valves to close that put all four of the RHR pumps against a closed discharge. On a single failure malfunction, one would be without core cooling pumps. The RHR pumps in that configuration can lock up with zero flow and grind to inoperability permanently. D. Allison indicated that the Staff briefed the subcommittee on the fact that a single failure of the miniflow logic could disable all redundant RHR pumps during the smaller intermediate size break LOCA. J. C. Ebersole suggested that the Staff should pay particular attention to that selection logic design. He asked how many nuclear plants there are like this one. W. Hodges, NRC, indicated that all BWR-3 plants, including Dresden 2 and 3, Millstone 1, Pilgrim, Monticello, Quad Cities 1 and 2, plus two BWR 4 plants are like this one. Duane Arnold and Fermi 2 also have this selection logic design. J. C. Ebersole wondered why the problem did not propagate to these other plants. W. Hodges indicated that those operators were able to demonstrate that they satisfied the regulations with the logic as it exists. They satisfied Appendix K and showed that the loop selection logic selected a proper loop. If it didn't select the proper loop for smaller breaks, they still didn't go above the 2200°F peak cladding temperature. C. Michelson suggested that the logic seems to work fine on paper, but it is in the hydraulics of the actual operation where the difficulty arose. The system has such high hydraulic noise that at any point in time it cannot detect on which side the break is. J. C. Ebersole suggested that the staff consider changing out this flaw for all other susceptible plants. The Committee discussed the possibility of requesting Staff action in an ACRS letter. E. Jordan explained that the Staff has notified all plants that are susceptible requesting that they assess the single failure vulnerability and submit a plan of action to the NRC. Affected licensees have formed an owners group to study the situation and derive a solution.

IV. <u>Source Term for Nuclear Power Plant Accidents</u> (Open)

[Note: M. D. Houston was the Designated Federal Official for this portion of the meeting.]

W. Kerr mentioned the previous Committee letter in December 1985 on the draft report entitled, Reassessment of the Technical Basis for Estimating Source Terms, NUREG-0956. The letter listed a number of comments which the Staff has addressed as part of its review of public comments. Significant changes have been made in the format of the report. He indicated that the Subcommittee met on June 3, 1986, to discuss what was called "Review Copy of NUREG-0956" dated May 23, 1986. It was the consensus of the Subcommittee that significant improvements have been made in the report.

M. Silberberg, NRC, discussed the state of progress on source term technology regarding the implementation of the Severe Accident Policy Statement. He discussed major changes made to NUREG-0956 (See Appendix XVII). The most important change was to add a considerable amount of redundant information directly into the report to deal with the technical basis for the source term. Considerable time has been spent describing the upgraded computer code package to augment the presentation in the draft report. The report mentions only some of the features of the code suite. Also presented are analyses of additional sequences performed with the source term code package for the NUREG-1150 (Risk Perspectives Rebaselining) analysis. The chapter on risk and containment has been removed to an appendix in response to the public comments. A chapter has been added to reflect revision to the severe accident research plan in NUREG-0900. There is also an improved statement of conclusions. He stated that the Staff believes that major advances have been made in this technology since WASH-1400 (the Reactor Safety Study), particularly in the last five years. An important part of this progress is the development of an analytical approach to source term estimation, the source term code package. He briefly discussed several areas of improvement (see Appendix XVII). D. W. Moeller noted that a number of the major advances are also on the list of areas of needed research. M. Silberberg did not believe this unusual since there are still some gaps left in the process of gaining a deeper understanding of severe accident phenomena and understanding the true uncertainties present. Further improvements can be expected from the research that is now in place. Nevertheless, he indicated that the Staff believes that it now has a sound technical basis from which to move forward to use the new source information to reevaluate regulatory practice as mandated by the severe accident policy. The process of using the new source term information is already in progress in the NUREG-1150 risk rebaselining study now being concluded. The second application will deal with the implementation plan for the Severe Accident Policy Statement and the regulatory use of new source term information. NUREG-0956 is an important element of the process of moving forward in examining current regulatory practice with respect to source terms.

J. Mitchell, NRC, discussed specific changes to NUREG-0956 based upon ACRS comments and recommendations in a December 12, 1985 ACRS report to the Commission. She indicated that the ACRS asked whether the Staff finds a significant difference between severe accident research program results to date and the Reactor Safety Study. She indicated that the final NUREG-0956 is less ambiguous than was the draft of the report. It now states that the Staff believes that there are not large systematic reductions. She mentioned that the ACRS questioned whether the selected accident sequences for the five reference plants provided sufficient tests of the capabilities of the computer codes. She indicated that while the Staff has shown that the codes execute and give physically reasonable results, there was now a program for validation of the codes and models. The Staff will compare the results with existing experiments and future research experiments. D. W. Moeller asked if the South Texas Project is one of the five referenced plants. J. Mitchell indicated that the five referenced plants are Surry, Peach Bottom, Sequoyah, Zion and Grand Gulf. These are the plants discussed in NUREG-0956. After its first publication as a draft, material was added to NUREG-1150 on La Salle, a BWR Mark II. L. Soffer, NRR, indicated that one of the earliest applications of the source term code package and the revised methodology was a relook at accident risks in the South Texas Project where there was a discussion using the source term from WASH-1400 and the revised methodology so that the results could be compared side by side. D. W. Moeller indicated that he had found this comparison very helpful in terms of the ACRS review. W. Kerr cautioned care in interpreting the results since it is not a safety document.

J. Mitchell noted the ACRS' desire to know what effect each of the major improvements being made would have on the source term. She indicated that while this appears to be a reasonable request, it is not very easy to satisfy. The Reactor Safety Study methodology was used as a set of small stand alone codes that were woven together to provide results. The Staff believes that it is not really practical to look at the advances one by one compared with the old methodology. For those sequences that are comparable, the Staff has provided a comparison with a bottom line. That includes the effects of all of the advances including the framework. The Staff is evaluating the effect of the iodine chemical form assumption with a forward-looking chemistry package rather than looking backward at what the Reactor Safety Study has said.

In general, J. Mitchell agreed with most of what was said in the ACRS report. J. C. Mark referred to a quotation in chapter 4 of NUREG-0956 which refers to "during the multiple hydrogen burns" in a number of places. He indicated that he could not find any description of the assumptions made for a hydrogen burn. He asked what kind of a burn was assumed and under what conditions did it occur. J. Mitchell indicated that the assumptions in most of the cases were that the hydrogen concentration should be 8 percent. J. C. Mark indicated that that appears reasonable as it comes straight from the TMI 2 experience. J. Mitchell indicated that the steam moisture content should be below about 55 percent. In some cases, there might be steam inerting. J. C. Mark asked if the hydrogen is coming from metal-water reaction or from core-concrete interactions. J. Mitchell indicated that it depends on the time of the accident. In vessel it derives from zirconium oxidation. J. C. Mark indicated that in that case one gets pure hydrogen with steam. In concrete, one gets as much water and carbon dioxide as hydrogen plus carbon monoxide. J. C. Mark and J. Mitchell discussed the course of a hydrogen burn accident including airborne fission products and released fission products as a function of time.

V. <u>Briefing Regarding IAEA Meeting on Chernobyl</u> (Open)

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

H. Denton, Director, NRR, reported on his briefing by Soviet nuclear experts at the May 21, 1986 meeting of the International Atomic Energy Agency's Board of Governors. He indicated that it appears that the Soviets will only participate in an IAEA format and will not engage in bilateral discussions with the U. S. Some time during July, H. Denton said that Soviet Union will report to the IAEA on the causes of the accident in a post-accident review meeting. Another IAEA scheduled meeting is intended to produce binding international early warning and coordination agreements. The objective will be to get all nations to sign such a binding agreement. H. Denton also indicated that the IAEA plans to assemble experts from around the world to propose ways to consider additional safety features or measures to improve the safety of all nuclear plants. Finally, a conference of governments will meet to consider binding agreements from the three previous meetings and consider the expert's recommendations. He spoke of a unified U. S. approach to Chernobyl through the National Science Advisory Board. He suggested that the ACRS should consider developing a factual report of what happened at Chernobyl.

H. Denton indicated that the chief Soviet spokesman at the May 21 meeting was Boris Siminoff who described a postulated event sequence which began with an intensive evaporation of cooling water, an overpressurization, and a hydrogen explosion. B. Siminoff claimed that the accident occurred in the reactor core as a result of a sudden power surge from 7 percent to over 50 percent power. There was a thermally disruptive metal-water reaction, a hydrogen leak and hydrogen release. A fire started. Based on what he learned at that meeting, H. Denton said that the first priority of the Soviet fire fighters at Chernobyl was to prevent the fire from spreading to the adjacent Unit No. 3. The Soviets are very concerned with water contamination and are now concentrating on keeping the apparently molten core from penetrating the suppression pool basemat by pumping concrete into the pool cavity. Their aim is to "entomb the reactor" while providing some internal cooling.

H. Denton explained that the IAEA meeting intended to discuss early notification of trans-boundary releases with the objective of signing a binding agreement will go into emergency response and discuss a strengthening of the incident reporting system. The IAEA may ask the United Nations organizations, WHO and UNSCEAR to review world dose contamination as a result of the Chernobyl event. H. Denton indicated that he expects an increase in NRC resources allocated to the study of the Chernobyl event. He noted that little has been learned of a technical nature about Chernobyl since NRR last briefed the Committee in May.

W. Kerr asked if H. Denton had any additional comments regarding the event's scenario. H. Denton indicated that it appeared to have begun on the morning of April 26 and was not a slowly developing accident. He was not sure if there was a positive void coefficient involved. The event was characterized by overpressurized pressure tubes and a rupture of pressure tubes with the release of steam and disruption in the upper part of the core. A metal-water reaction ensued from water which was released from the pressure tubes with a subsequent release of hydrogen which seeped into the graphite moderator. A fire ensued. D. W. Moeller asked what would have happened had the Chernobyl reactor had a large dry containment. H. Denton indicated that he did not know the nature of the pressures generated during the event and could not make any comment on the impact of containment. It was noted that it is possible that the Soviets were doing some manipulations of an experimental nature at the time of the event. He also indicated that 300 individuals who were reported hospitalized were employees of the facility. No residents were involved, according to the Soviets.

H. Denton mentioned international reaction to the Chernobyl event. The filtered-vented containment is now a principal issue in Sweden. There is a also a sense of urgency to move on the IAEA meeting. The KWU Plant built in Austria may be scrapped in reaction to Chernobyl. H. Denton indicated that the U. S. must organize its efforts by incorporating DOE and FEMA and participating through the IAEA process. He speculated that the states in the U. S. will want emergency planning regulations reexamined. IDCOR was asked by the Staff to speed up its efforts to examine containment performance to mitigate severe core damage without containment failure. The NRC intends to develop the outline of a report on Chernobyl and present it to the ACRS. A delegation of five individuals from the U. S. will participate in the IAEA meeting when the Soviet report is given.

VI. Meeting with NRC Commissioners (Open)

[Note: Commissioners present were: N. J. Palladino, Chairman; T. M. Roberts; J. K. Asselstine; F. M. Bernthal and L. W. Zech, Jr.]

Chairman Palladino indicated that the Commission intends to discuss ACRS views on the GESSAR II BWR/6 nuclear island design for future plants per the Committee's January 14, 1986 report to the Commission. The ACRS report indicates that although the GESSAR II design has improved safety features, there are questions whether the design satisfactorily addresses all concerns in the NRC's severe accident policy. He indicated that the concerns expressed by the ACRS bear on the question regarding the Commission's role with respect to ultimate approval of standard designs such as GESSAR II.

F. J. Remick discussed the ACRS findings and recommendations in its January 14 letter. He indicated that the Committee believes that the GESSAR II design includes features that have the potential to provide a significant improvement in safety over current BWR designs. If this were an application for a construction permit for one or more plants of this design, the Committee would not hesitate to recommend its approval. The Committee, however, was unable to agree with the Staff for various reasons that the design satisfactorily or completely addresses all the concerns described in the Commission's Severe Accident Policy Statement. The Committee did approve the GESSAR II design provided that it was for a limited time such as five years and provided that the Staff's review procedure not be viewed as a precedent for handling of future applications. In particular, the information provided to the ACRS in connection with GESSAR II would not be sufficient to support an application for a one step licensing process. He mentioned receipt by the Committee of NUREG-0979 Supplement 5, the fifth supplement to the GESSAR II FDA, but indicated that the Committee had not yet had time to fully analyze it.

M. W. Carbon spoke about his additional comment appended to the January 14 letter. He indicated that he and C. J. Wylie believe that the GESSAR II design represents a definite improvement in safety over BWR designs that have been approved in the past and that the applicant has met all of the NRC requirements. He noted that many items are still open and considerable review will yet take place. Nevertheless, he indicated that he and C. J. Wylie support the Staff's plan to issue an FDA applicable to one step licensing. He indicated that he was personally not totally happy with GESSAR II as the design for a standard plant, but was encouraged by the kinds of improvement that were made in this application. He thought a long term standard plant design which might result in the construction of many plants ought to be handled in a somewhat different fashion. Chairman Palladino asked for further clarification of that remark. M. W. Carbon indicated that the NRC should cue vendors who might submit such a potential standard plant design of some of the features that would be highly desirable in such a design. He did not think that it would be adequate for the Commission to wait until the vendors bring the design to the NRC. It would be best to enter the process at an early phase so that the vendors are aware of some of the features the Commission thinks would be desirable in future standard plants. In answer to a question by Commissioner Asselstine, M. W. Carbon indicated that his willingness to sign off on the Staff's review was in part because of the fact that very few of these plants will ever be built. Commissioner Asselstine asked for examples of principal areas of improvement over BWR/6s built into the GESSAR design. M. W. Carbon cited the ultimate plant protection system (UPPS) which he thought a definite step in the right direction.

Commissioner Bernthal thought it interesting that the same argument has come up in the past, that the Commission get involved in the early stages of review of what the industry and the vendors are developing in the way of advanced reactors. Chairman Palladino thought that it best to have essentially a complete design when the NRC begins its review. He asked if the UPPS is essentially complete. C. J. Wylie indicated that the UPPS is a design concept which does not have hardware and criteria associated with it as do some other parts of the plant. But the Staff has made a provision that it would be reviewed in the course of a construction permit application.

D. Okrent presented several questions he thought should be explored as the Commission reviews the GESSAR II matter.

- What is an FDA? What commitment is the NRC making when it issues one? What commitments would it be making if it approves GESSAR II? How much detailed information should be provided by an applicant for an FDA?

- What should be the level and depth of the PRA? Should it treat uncertainties as well as the state-of-the-art will permit? How should interface requirements with the balance of plant be specified in view of the fact that the PRA makes assumptions on the performance of the balance of plant?

- What seismic fragility requirements should be established by the GESSAR PRA and by the Staff review?

- What performance requirements for GESSAR II systems are established by the PRA, if any? What level of PRA evaluation and review is required of the NRC Staff for it to accept an FDA? Should the Staff make use of mean or mode values which are evaluated to the state of the art?

- What should be the quantitative safety objectives for a future plant?

- Should there be some kind of containment performance criterion for a future plant?

- How will a future plant design deal with a terrorist threat and sabotage?

- How does one deal with cost benefit analysis for possible design improvements? How does one insure defense-in-depth at the same time?

- How does one ensure that the frequency of challenges to safety systems is acceptable?

D. Okrent explained that his additional remarks at the end of the ACRS report did not imply a disagreement with the letter except that it would be better to call it an interim letter to issue a limited FDA. He noted his original intent was to write an interim letter and discuss the matter with the Commission. He also noted that the Staff and its consultants evaluated the seismic design and seismic PRA for GESSAR II and contended that it was inadequately designed for purposes of a full review. Missing were the seismic contribution to core melt, as well as a discussion of the elimination of relay chatter. The UPPS was really ill-defined. It has no seismic capability as proposed, and if seismic events turned out to be a major contributor, it would not have alleviated this aspect of the plant risk. He mentioned his concerns regarding sabotage protection and the inability of the Staff to require anything of GESSAR II beyond what is already in the existing regulations regarding access control and fences. He noted that the control room appears to be vulnerable to the kinds of terrorism that has caused severe damage in the Middle East. He also questioned the ability of the drywell to maintain its integrity should a core melt occur and much of the core got through the vessel to the concrete basemat. The concrete would heat up the vessel pedestal, and if the sacrificial shield failed, the vessel could tip over pulling the piping, such as steam lines, from their penetrations. He noted that the NRC Staff was not concerned by this risk and that Brookhaven National Laboratory thought that this would be a case of late containment failure with a small radioactive release.

D. Okrent indicated that he was generally concerned regarding the quality of the NRC Staff review. Commissioner Bernthal wondered what this catastrophic core melt scenario described by D. Okrent would involve. The Commission and the ACRS discussed various aspects of such a catastrophic core melt which might involve rupture of the drywell and bypassing of the suppression pool. D. Okrent suggested that the result might be a small radioactive release or the release might be quite significant. What is important is that the Staff should analyze the scenario before an FDA is issued or enough is known about the scenario to rule it out as a viable possibility. He stressed that in his own opinion future plants including GESSAR II should include the features that are included in his added remarks to the January 14 report. These features should include independent decay heat removal systems, as well as features for sabotage protection. Future plants should be surrounded by a containment designed especially for core melt accidents.

J. C. Ebersole suggested that the practicality of the boiling water reactor gave it the potential even as far back as 1968 of being a potential workhorse power plant. Nevertheless, the boiler still has a completely inadequate reactivity control system with hydraulic drives and a multiplicity of valves and complications which have led to a record of less than optimal performance. Perhaps relatively modest enhancements to the GESSAR design, such as a complete description of the UPPS, might be all that is necessary. He agreed that the seismic area, the area of sabotage protection, and protection from fire, in addition to modifications in the reactivity control system, are issues that need to be addressed. Commissioner Asselstine spoke in favor of a high-quality design that is a near perfect plant that will not require constant modifications and the constant addition of complexities. W. Kerr suggested that his experience in the design of large industrial systems is that they are subject to considerable surprises. He indicated that he was more comfortable with an evolutionary approach to changes with a goal in mind rather than pursuing a complete, perfect design. W. Kerr also expressed some misgivings regarding the process of NRC review which was primarily associated with the use of the PRA in arriving at decisions about severe accidents. He did not think this review should represent a complete review but hoped that the process could be improved. Commissioner Bernthal noted that since there does not appear to be a rush to build one of these GESSAR-II plants, he thought it might be a good idea to insist that the Staff review be more thorough. D. A. Ward agreed that the ACRS thinks that the Staff has come up short in defining the GESSAR-II design as an appropriate standard design.

Chairman Palladino thought it unfair to hear only one side of the issue. He thought that the Commission ought to hear from the Staff regarding the process and their conclusions. Then the Commission should decide for itself what its role is in the FDA process. It has never been defined. Commissioner Bernthal speculated on what would happen to the FDA assuming that the Commission approved it. He wondered what GE intends to do with the FDA. Commissioner Asselstine did not believe that GE would go to licensing hearings in the U. S. He speculated that GE might be helped in the export market.

H. Denton indicated that the Commission approved the policy statement that provided certain provisions for the CESSAR and GESSAR plants which had been under review for some time. The Staff attempted to review those applications against the standards established and found that they met the regulations. He noted that he could not speak to the use GE plans to make of the FDA. He thought it best that GE address that question. He noted that GE has also under development an advanced boiling water reactor which he presumed the Commission would want the Staff to review. Chairman Palladino took note that the Staff reviewed the GESSAR-II application under Commission direction and indicated that it is incumbent upon the NRC to process proposals such as this one from industry. It is the job of the Commission to assure that it is a good, safe design that meets the regulations and the objectives set forth by the Commission. D. Okrent pointed out that there is a difference of opinion between the NRC Staff and the majority on the

ACRS as to whether in fact the Commission severe accident policy was properly implemented in this review. C. P. Siess noted that the GESSAR-II design is a future plant in the sense that if one is built it will be built in the future but it is not a future design. He suggested that the health and safety of the public would not be endangered if one or two of these plants were built. The ACRS main concern was with the review process.

D. A. Ward took note of the retirement of N. J. Palladino as Chairman of the NRC and congratulated him on his record and accomplishments in the last five years. He pointed out that his experience and judgment would be missed by his fellow Commissioners and the Staff.

VII. Executive Sessions (Open)

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

A. Subcommittee Assignments

1. ACRS Review of the Hanford N-Reactor/Chernobyl Reactor

A Subcommittee consisting of D. A. Ward, D. W. Moeller, W. Kerr, and F. J. Remick as Chairman was directed to become familiar with the design and the course of the recent accident at the Chernobyl Nuclear Plant. This Subcommittee will act in an information gathering capacity to follow the course of ongoing studies including the DOE Special Committee and the National Academy of Sciences review of this matter. The Safety Philosophy, Technology, and Criteria Subcommittee (D. Okrent, Chairman) was asked to consider the implications of the Chernobyl accident to reactor safety in the United States.

2. Membership on the ACRS Management Committee

A proposal by H. W. Lewis regarding the composition o the Management Committee was deferred until the August meeting when the Procedures and Administration Subcommittee will consider the question of a two-year term for the Chairman as well as the membership of the Management Committee. Procedures for selection of the ACRS Chairman should also be discussed at this time.

3. Subcommittee Assignments

Distribution of advanced water reactor reviews between the Advanced Water and Advanced Non-water Reactor

Subcommittees needs further discussion by the ACRS Chairman and the Subcommittee Chairmen (M. W. Carbon and C. J. Wylie).

It was agreed that the review of Improved Technical Specifications will be handled by the Subcommittee on Plant Operating Procedures (C. Michelson) rather than the Subcommittee on Operating Reactors (J. C. Ebersole).

4. <u>Report of the ACRS Management Committee</u>

The Chairman reported on the June 4, 1986 meeting of the ACRS Management Committee (Note: Several of the specific recommendations/decisions are reported elsewhere in this list. The remainder are as follows:

° Members were asked to provide comments to the ACRS Executive Director regarding the Status Report on Implementation of the Recommendations of the Panel on ACRS Effectiveness (distributed along with the June 4, 1986 meeting summary). Members were asked to devote particular attention to the status/action proposed for implementation of the recommendations

° The Chairman noted that several specific requests for reassignment of specific tasks and additional resources by ACRS chairmen and subcommittees were considered and have been addressed in memoranda to the proposers (see specific list in the summary of the June 4, 1986 Management Committee meeting). These items were dealt with as follows:

(1) <u>Steam generator overfill</u> should be assigned to the Safety Philosophy, Technology, and Criteria Subcommittee (DO). Reassignment was proposed by the Management Committee provided a real risk from this event can be shown to exist. C. P. Siess and P. G. Shewmon should be added to the subcommittee for this review

(2) <u>Thermal hydraulic bases for EOPs</u> to the Plant Operating Procedures Subcommittee. It was agreed that this should be reassigned

(3) <u>Functioning of isolation valves under accident loadings</u> to Subcommittee on Reliability Assurance. It was agreed that this should be reassigned provided a problem can be shown to exist

(4) Lack of adequate EOPs for severe accidents (e.g., should cooling water be turned on or not after the core is molten?) to the Class 9 Subcommittee. The Management Committee agreed to this reassignment

Proposed addition of a subcommittee per discussion during the 313th ACRS meeting - For example, D. Okrent's suggestion for a Containment Subcommittee - D. Okrent will be queried by M. W. Libarkin regarding the specific tasks for such a group since a Subcommittee on Containment Requirements already exists

Assignment of additional resources to generic subcommittees as requested by Subcommittee Chairman:

(1) Waste Management (DWM) request for 59 person-days vs. 40. It was agreed that this reassignment should not be made at this time

(2) Severe Accidents (WK) request for 4 subcommittee meetings vs. 2 allocated. This request will be revisited in 6 months

(3) G. A. Reed regarding a 1-day meeting for IE programs and 3-4 subcommittee meetings for WAPR review. One subcommittee day will be allocated for review of IE programs. This should include discussion of IE's proposed use of PRA to identify important areas for attention. The WAPR review now has 1 subcommittee meeting assigned. Remaining meetings should be deferred until firm schedules from the NRC Staff are available.

(4) C. Michelson request for Auxiliary Systems review (2 Subcommittee meetings) of the Fire Protection Provisions in nuclear plants. T. G. McCreless was directed to have an ACRS Fellow examine the situation regarding fire protection including the work of Dr. Apostolakus.

° The numerical ranking of ACRS reports considered during 1986 by 10 ACRS members produced the following results:

(1) The highest ranking was 2.8, the lowest was 0.6, and the average is 1.7

(2) Generally, ACRS reports with high ratings (above 1.5) have been sent although this is not true in all cases as noted below

(3) Highly rated reports (State of Nuclear Safety and Safety Considerations in Future Reactors) have not been sent, primarily because of difficulty in preparing them. A means should be considered to reactivate these reports

(4) Reports on waste management, although sent by the Committee in most cases, generally had lower rankings (1.5-1.8 range)

(5) Reports with rankings below 1.0 were not sent. Only one report with a ranking below 1.4 was sent

(6) The move to Bethesda got very low rankings [1.0 (sent) and 0.7 (not sent)]

Monideep De provided a more statistically meaningful evaluation regarding the means and standard deviations of member ratings.

The members did not agree that difficult reports such as the reports on the state of Nuclear Safety and Safety considerations in Future Reactors should be reactivated as a result of this poll.

B. <u>Reports, Letters, and Memoranda</u>

1. <u>ACRS Comments on the NRC Safety Research Program and Budget for Fiscal Year 1988</u>

The Committee prepared a report to the Commissioners on its review of the proposed program and associated budget for the NRC Safety Research Program for Fiscal Year 1988.

2. <u>ACRS Report on South Texas Project, Units 1 and 2</u>

The Committee prepared a report to the Commissioners of its review of the application of Houston Lighting and Power Company (HL&P, the Applicant), acting on behalf of itself and as agent for the City Public Service Board of San Antonio, Central Power and Light Company, and the City of Austin, for a license to operate the South Texas Project, Units 1 and 2.

3. <u>ACRS Comments on NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms -- Review Copy/ Final</u>

The Committee prepared a report to the Commissioners of its review of the final version of NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms."

4. ACRS Recommendations on Hope Creek

The Committee prepared a report to the Commissioners noting its concern regarding proposed resolution of recommendations made in its December 18, 1984 Hope Creek Operating License report concerning a structured turbine over-speed test program and habitability requirements for the Hope Creek control room.

5. Proposed NRC Policy Statement on Standardized Nuclear Power Plants

The Committee authorized a memorandum to N. J. Palladino, NRC Chairman, (Attention: N. Haller, Executive Assistant) from the ACRS Executive Director regarding the pending Commission action on publication of a proposed NRC Policy Statement on Standardized Nuclear Plants. (Ref. V. Stello, EDO memorandum of May 14, 1986 to the NRC Commissioners regarding Standardized Nuclear Power Plants.)

6. ACRS Status

H. W. Lewis proposed a letter to the Commissioners regarding changes by the Commissioners in a cable inviting representatives of the Soviet Union to the Wingspread international meeting on reactor safety. The ACRS decided not to send the letter.

C. Future Agenda

1. Future Agenda

The Committee agreed on tentative agenda items for the 315th ACRS meeting, July 10-12, 1986 (see Appendix II).

2. Future Subcommittee Meetings

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

D. B&W Program on Trip Reduction

F. J. Remick explained that Duke Power Company plans to set up a review group with four outside members as a part of the B&W Owners Group Stop Trip Program (curtail unnecessary trips) and has asked for his participation on the review group. Several members pointed out the conflict-of-interest possibilities involved when any B&W plant is reviewed by the ACRS. The consensus of the Committee was against participation by F. J. Remick and he agreed to decline the invitation.

E. Memorandum of Understanding with the EDO

It was noted by the Committee that on occasions when the NRC Staff comes before the ACRS with a preliminary position on a proposed NRC rule or policy statement and the ACRS declines to comment, the NRC Staff has bypassed the Committee during later review stages. It was decided that subcommittee chairmen should set out a course of action for anticipated ACRS review. The ACRS Executive Director, R. F. Fraley, was directed to send a memorandum to the EDO regarding this proposed course of action indicating ACRS interest in later review of the preliminary Staff position. Copies of the memorandum should be sent to the cognizant Staff members at the branch level and project level.

F. Proposed Amendment to ACRS Bylaws

The Committee approved a change in the ACRS Bylaws which provides a mechanism for individual members to express their personal views in meetings with individual Commissioners. The ACRS Executive Director was directed to inform the Commissioners of the change in such a way as to allow an opportunity for comment as to whether this change is responsive to their needs.

The Committee approved a proposed Bylaw change regarding the authority of the ACRS Chairman to hold up a completed ACRS report if he discovers that it contains a serious error or misstatement which was not evident during its preparation.

The Committee approved guidelines (as a change to the Bylaws) regarding preparation of added (minority) comments to ACRS reports. Changes proposed by H. W. Lewis which were part of the draft made available during the 314th meeting are to be incorporated. Any comments regarding these changes should be directed to T. G. McCreless as soon as possible.

G. Conduct of Members

The Committee discussed the recent comments by P. G. Shewmon to Dr. M. B. McNeill, NRC/RES regarding the NRC research program at Battelle, Columbus, and suggested that it would be most appropriate to preface such comments by individual members with a standard disclaimer. This disclaimer should stress that the comments are the member's own personal views and do not necessarily represent a position of the Committee. If the intention is to interpret a Committee position, a member should first seek guidance from the full Committee.

H. Testimony by H. W. Lewis

H. W. Lewis noted his intent to testify during hearings of the House Subcommittee on Energy and the Environment regarding the viability of the nuclear power program in the United States. His comments will represent his own personal opinions regarding subjects to be addressed rather than the opinion of the Committee. The Committee agreed to support this effort logistically.

I. Implications of Chernobyl

A letter from the State Department was circulated during the meeting which stated that certain specific information regarding the Chernobyl accident is to be considered classified. R. F. Fraley was asked to determine the authority under which such matters should be declared classified.

J. Report of the Nominating Panel

The ACRS panel regarding nomination of candidates for appointment to the ACRS produced a list of six potential candidates for final consideration by the Committee. An additional name was added to the list and two candidates are to be invited to the 315th ACRS meeting (July 1986).

K. Reappointment of ACRS Member

The Ad Hoc Subcommittee set up to consider the reappointment of F. J. Remick whose term ends in September 1986 recommended the reappointment of F. J. Remick to the Committee. The ACRS endorsed the Ad Hoc Subcommittee's recommendation and the ACRS Executive Director was directed to inform the Commissioners of the Committee's recommendation.

L. Agenda for the Wingspread International Meeting

A proposed agenda and outline for this meeting was distributed. Arrangements for the meeting were discussed. Members who plan to bring their wives were asked to advise the ACRS Office (T. G. McCreless).

M. Member Complaint Regarding Lack of Documentation in Safety Evaluation Reports

G. A. Reed expressed his unhappiness with the lack of plant layout and system schematic drawings in the Staff's SER for the South Texas Project, Units 1 and 2 which made the OL review of that facility more difficult. Since the South Texas Project appears to be the last OL review to come before the Committee for some time, it was suggested that, for future plant reviews, ACRS Staff engineers prepare a package of drawings from the applicant's Safety Analysis Report prior to Committee consideration of a plant. G. A. Reed was asked to provide a list of the drawings in which he is interested.

N. Topics for the 315th ACRS Meeting

The members agreed to devote considerable time (3-6 hrs) during the 315th (July) ACRS meeting to the consideration of the proposed NRC Policy Statement on Standardized Nuclear Plants. D. A. Ward and C. J. Wylie were directed to work out a detailed agenda for this meeting. D. Okrent suggested that topics which have been set aside (e.g., General Design Criteria update) should be considered as well as those issues included in the proposed policy statement.

The 314th ACRS meeting was adjourned at 1:20 P.M., Saturday, June 7, 1986.

South Texas Project Safeguards Information Supplement

DELETION 3

Due to Safeguards Information

APPENDIXES
TO
MINUTES OF THE 314TH ACRS MEETING
JUNE 5-7, 1986

ACRS-2428

NRC ATTE

314TH ACRS

Thursday, June 5, 1986

OFFICE OF NUCLEAR REACTOR REGULATION

S. Long
B. Mann
R. Hernan
J. E. Rosenthal
K.S. West
J. L. Milhoan
R. P. Goel
D. Choppa
R. Caruso
D. Scaletti
R. Martin
N. Kadambi

OFFICE OF NUCLEAR REGULATORY RESEARCH

A. Datta

REGION IV

G. Costable

INVITED ATTENDEES

314TH ACRS MEETING

Thursday, June 5, 1986

WESTINGHOUSE ELECTRIC CORP.

B. S. Monty
G.E. Lang
B. D. Losen
W. J. Johnson
W.R.Spezialetti
A.G. Dai
K. P. Slaby
F. J. Twogood
M. J. Hitchler
M.Beaumont

HOUSTON POWER & LIGHT COMPANY

M. E. Powell
J. W. Bailey
J. M. Dew
R. L. Balean
E. Dotson
M. R.Wisenburg
J.E.Geiger
I.Crawford, III
E.A.Alexander
R. A. Frazar
J. R. Pendland
J. Newn
J. H. Goldberg
M. A. McBurnett
D. Cody
W. Kinsey
J.G.Dewease
A. O. Hill
R.C. Munter
T. Roberson

BECHTEL POWER CORP.

D. H. Ashton
S. N. Letouirneau
J. K. Atwell
J. Litehiser
A. Zaccaria

NEWMAN & HOLTZINGER

S. Goldberg
A. [illegible] Gutterman

PUBLIC ATTENDEES

314TH ACRS MEETING

Thursday, June 5, 1986

A. E. Nolan, EG&G, Idaho
P. Higgins
P. Guyer, NUS Corp.
J. Nurmi, Qatel

NRC ATTENDEES

314TH ACRS MEETING

Friday, June 6, 1986

W. H. Beach, RES
R. Hernan, NRR
L. Jackiw, Region III
E. Weiss, IE
W. Hodges, NRR
D. Allison, IE

PUBLIC ATTENDEES

314TH ACRS MTG.

Friday, June 6, 1986

J. Trotter, NUS Corp.
E. Fotopoulos, SERCH Licensing, Bechtel
L.Conner, DDA
J. Nurmi, Qatel
J. Kuemin, Consumers Power Co.
G. A.Zimmerman, Portland General Electric Co.
C.R. Klee,Bechtel Power
J. A.Gieseke, Battelle
R.S.Denning, Battelle
D. Runkle, Morgan Assoc.

APPENDIX A
FUTURE AGENDA

JULY MEETING

Davis-Besse Nuclear Plant -- Subcommittee report concerning request from Commissioner Asselstine regarding the regulatory processes associated with review, approval, and operation of this plant and an auxiliary feedwater system that does not meet current safety standards	½ hr
TVA Reorganization -- Discuss TVA reorganization to resolve QA, design, construction, etc., problems at TVA plants	3 hrs
B&W Water Reactors -- Status report regarding review of long range safety of B&W Reactors with briefings by representatives of the NRC Staff and the B&W Owners Group	2½ hrs
Davis-Besse Nuclear Plant -- Proposed restart of this plant following loss of feedwater incident on June 9, 1985	3 hrs
NRC Regulator Guides -- ACRS comments regarding Regulatory Guide revisions for Regulatory Guide 1.35, Rev. 3, ISI of Ungrouted Tendons in Prestressed Concrete Containments and Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	½ hr
EPRI Requirements for Standardized LWRs -- Briefing by EPRI and NRC Staff representatives regarding status of activities to develop requirements for standardized LWRs	1½ hrs
Design, Maintenance and Testing of Safety-Related Check Valves - Briefing by IE Staff	1 hr
Auxiliary Systems -- Report of ACRS Subcommittee regarding provisions for fire protection in nuclear power plants	1 hr
Proposed NRC Policy Statement on Standardized Nuclear Power Plants -- ACRS comments on proposed NRC policy statement and proposed NUREG to identify topics that are important to the implementation of this policy	6 hrs
Meeting with Director, NMSS -- Discuss matters of mutual interest	1 hr

Control Room Habitability Improvement Effort -- Discuss related NRC program and ACRS comments as appropriate	1 hr
Staff Reviews of Chilled Water Systems -- Discuss related NRC program and ACRS comments as appropriate	1 hr
Reactivation of Deferred and Cancelled Nuclear Plants -- Briefing by representatives of the NRC Staff regarding factors to be considered	1½ hrs
Proposed NRC Policy Statement on Technical Specifications -- ACRS comments are requested	1½ hrs
New Members -- Discussion regarding nominations of candidates to fill the vacancy on the ACRS	3/4 hr
NRC Long Range Plan -- Discuss proposed outline for preparation of a long range plan	deferred to August
Restart of San Onofre Nuclear Plant, Unit 1 -- Review of corrective action at San Onofre Unit 1 to correct check valve problems and resulting water hammer	deferred to August
Containment Performance Design Objective -- ACRS comments	deferred to August/ September
Fitness for Duty Requirments --Status report by NRC Staff	deferred to August
Safety System Functional Inspections -- Status report by NRC Staff	deferred

ACRS SUBCOMMITTI

Ad Hoc Subcommittee on TVA, June 12, 1986 and 13, 1986, Chattanooga, TN (Savio), 8:30 A.M. The June 12 discussion will be held in TVA's Chattanooga Office Complex, Tennesee River Room, 1101 Market Street, Chattanooga, TN, and the June 13 discussion will be held at the Sequoyah Nuclear Plant, Managers Conference Room, Daisy, TN. The Subcommittee will discuss TVA reorganization and related technical and management issues. Attendance by the following is anticipated, and reservations have been at the Holiday Inn Downtown, 401 West Martin Luther King Blvd., Chattanooga, TN for the nights of June 11 and 12:

Mr. Wylie	Mr. Ward
Mr. Ebersole	Mr. Hagedorn
Mr. Michelson	Mr. Barton
Mr. Reed	

Long Range Plan for NRC (CLOSED), June 17, 1986, 1717 H Street, NW, Washington, DC (Major), 8:30 A.M., Room 1046. The Subcommittee will review the proposed NRC Five Year Plan and prepare to address Committee comments to the Commission. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 16:

Dr. Carbon	STATE PLAZA	Mr. Wylie (tent.)	NONE
Dr. Remick	NONE		

Babcock and Wilcox (B&W) Reactor Plants, June 25, 1986, 1717 H Street, NW, Washington, DC (Major), 8:30 A.M, Room 1046. The Subcommittee will consider the B&W Owners Group plans to reassess the long-term safety of B&W reactors, including the implications of operating experience on the adequacy of B&W plant designs. The Subcommittee will also be briefed on the NRC Staff's Incident Investigation Team's (IIT) findings related to the 12/26/85 loss of integrated control system power and overcooling transient at the Rancho Seco nuclear power plant. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 24:

Mr. Wylie	NONE	Mr. Michelson	DAYS INN
Mr. Ebersole	CARLYLE	Mr. Reed	DAYS INN
Dr. Kerr	LOMBARDY	Mr. Ward	NONE

Metal Components, June 25, 1986, Pittsburgh, PA (Igne). The Subcommittee will review the status of NDE of cast stainless steel, and changes in steel-making practice. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Shewmon	Dr. Bush
Mr. Etherington	Dr. B. Thompson
Dr. Mark (tent.)	

Auxiliary Systems, June 26, 1986, 1717 H Street, NW, Washington, DC (Duraiswamy), 8:00 A.M. - 12:45 P.M., Room 1046. The Subcommittee will discuss: (1) the status of the Appendix R compliance, (2) differing technical views among the Staff, (3) proposed research and associated budget for FY 1988 and 1989 in the fire protection area, (4) updates on the progress being made in the Sandia experimental program on fire protection, (5) inspection activities to determine compliance with the Fire Protection Requirements, and (6) recent experiences associated with inadvertent actuation of fire protection systems. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 25:

Mr. Michelson	DAYS INN	Dr. Shewmon	MILLERS
Mr. Ebersole	CARLYLE	Mr. Wylie	NONE
Mr. Reed	NONE		

Regulatory Policies and Practices, June 26, 1986, 1717 H Street, NW, Washington, DC, (Quittschreiber) 8:30 A.M., Room 1167. The Subcommittee will review the regulatory process as it relates to the June 9, 1985 Davis-Besse event using the Davis-Besse Ad Hoc Report as te basis for the meeting. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 25:

Dr. Lewis	HYATT	Mr. Michelson (P.M.)	DAYS INN
Dr. Remick	NONE	Mr. Wylie (P.M.)	NONE
Dr. Siess (A.M.)	ANTHONY		

Gas Cooled Reactor Plants, June 26, 1986, 1717 H Street, NW, Washington, DC (McKinley), 1:30 P.M., Room 1046. The Subcommittee will review the applicability of NRC requirements for equipment qualification and cable testing to Fort St. Vrain, an HTGR. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 25:

Dr. Siess	ANTHONY	Dr. Shewmon	MILLERS
Mr. Ebersole	CARLYLE	Mr. Ward	NONE
Mr. Reed	NONE		

Davis-Besse, June 27, 1986, 1717 H Street, NW, Washington, DC (Alderman), 8:30 A.M., Room 1046. The Subcommittee will review start-up activities for Davis-Besse. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 26:

Dr. Remick	NONE	Dr. Siess	ANTHONY
Mr. Reed (p/t,A.M.)	NONE		

Joint Occupational and Environmental Protection Systems and Auxiliary Systems, June 27, 1986, 1717 H Street, NW, Washington, DC (Schiffgens/Duraiswamy), 8:30 A.M., Room 1167. The Subcommittees will: (1) review a draft AEOD report on the effects of ambient temperature on I&C Systems, (2) be briefed on the status of various control room HVAC Systems problems and the Staff's control room habitability improvement effort, (3) discuss with the Staff the 1 mrem/yr "cutoff" dose rate for the calculation of collective population doses, and (4) be briefed on the Staff's evaluation of the Shearon Harris Chilled Water Systems. Attendance by the following is anticipated, and reservations have been made at the hotels indicated for the night of June 26:

Dr. Moeller	CARLYLE	Dr. Shewmon	MILLERS
Mr. Michelson	DAYS INN	Mr. Wylie	NONE
Mr. Ebersole	CARLYLE	Dr. First	NONE
Dr. Mark	LOMBARDY	Mr. Kathren	NONE
Mr. Reed(p/t, P.M.)	NONE	Mr. Till	NONE

Plant Operating Procedures, July 1, 1986, 1717 H Street, NW, Washington, DC (Schiffgens), 8:30 A.M., Room 1046. Ths Subcommittee will review "Proposed Commission Policy Statement on Technical Specifications." Lodging will be announced later. Attendance by the following is anticipated:

Mr. Michelson
Mr. Ebersole
Mr. Reed
Dr. Remick
Mr. Wylie

Metal Components, July 1 and 2, 1986, Columbus, OH, (Igne), 8:30 A.M. The Subcommittee will review the degraded piping program of RES and NRR and visit its primary testing facility at Columbus, OH. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Shewmon
Mr. Etherington
Dr. Lewis
Dr. Okrent
Mr. Ward
Mr. Bender
Mr. Rodabaugh
Dr. Hutchinson

Improved LWR Designs, July 9, 1986, 1717 H Street, NW, Washington, DC (Alderman), 8:30 A.M. (A.M. Only), Room 1046. The Subcommittee will be briefed and discuss the following topics: (1) the Standardization Policy Statement, (2) proposed changes to 10 CFR 50, and (3) the EPRI Advanced Light Water Requirements documents. Loding will be announced later. Attendance by the following is anticipated:

Mr. Wylie
Dr. Carbon
Mr. Ebersole
Dr. Kerr
Mr. Michelson
Mr. Reed
Dr. Siess

315th ACRS Meeting, July 10-12, 1986, Washington, DC, Room 1046.

Human Factors, July 15, 1986, 1717 H Street, NW, Washington, DC (Schiffgens), 8:30 A.M., Room 1046. The Subcommittee will review: (1) SECY-86-153, industry and staff comments on proposed fitness for duty Policy Statement, (2) SECY-86-70, proposed rulemaking on degree requirements for SROs at nuclear power plants and (3) SECY-86-119, the annual status report on implementation of the Commission Policy Statement on training and qualification. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Remick	Mr. Reed
Mr. Ebersole	Mr. Ward
Mr. Michelson	Mr. Wylie

Waste Management, July 21 - 23, 1986, 1717 H Street, NW, Washington, DC (Merrill), 8:30 A.M., Room 1046. The Subcommittee will review: (1) NUREG-0518, Final Environmental Statement pertaining to the salvaging of contaminated smelted alloys, (2) the broader generic question concerning residual radiation limits and the disposition of land, buildings, metals and equipment resulting from the decontamination and decommissioning of nuclear power plants and fuel facilities, and (3) various nuclear waste topics. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Moeller	Dr. Shewmon
Dr. Carbon	Dr. Carter
Dr. Mark	Dr. Orth
Dr. Remick	Dr. Steindler

Naval Reactors, (operation of a nuclear-powered submarine), July 28, 1986 (Boehnert). The Subcommittee will observe the activities of a nuclear submarine crew. Attendance by the following is anticipated:

Dr. Kerr
(majority of ACRS members)

Westinghouse Reactor Plants, July 30, 1986, 1717 H Street, NW, Washington, DC (Houston), 1:00 P.M. - 5:00 P.M., Room 1046. The Subcommittee will continue discussion and comment on NRC Staff actions taken with respect to the SONGS-1 water hammer/loss of AC power event. This will be a follow-up Subcommittee meeting to the February 12, 1986 meeting on the same subject. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Reed	Mr. Michelson
Mr. Ebersole	Mr. Wylie
Dr. Kerr	Dr. Catton

Waste Management Subcommittee Visit to WIPP and NTS Facilities, July 30 - August 1, 1986 (Merrill). The Subcommittee will be briefed and take surface and underground tours of the DOE Waste Isolation Pilot Plant (WIPP) near Carlsbad, NM and the DOE Nevada Test Site (NTS) Facilities near Las Vegas, NV -- G-Tunnel, Climax, Jackass Flats (E-MAD), and Yucca Mountain. The purpose of these visits is for the members to gain a better understanding of the problems associated with the design, construction and operation of underground facilities similar to the geologic repository for High-Level Radioactive Wastes. Attendance by the following is anticipated, and reservations have been made for them at the AMFAC Hotel, 2910 Yale Blvd., SE, Albuquerque, NM for the nights of July 29 and 30. Reservations have also been made for them at the Tropicana Hotel in Las Vegas, NV for July 31 and August 1 and 2 :

Dr. Moeller
Dr. Carbon
Dr. Remick
Dr. Shewmon
Dr. Donoghue
Dr. Krauskopf

Scram Systems Reliability, July 31, 1986, 1717 H Street, NW, Washington, DC, (Boehnert), 8:30 A.M.. The Subcommittee will discuss the status of the ATWS Rule implementation effort. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Kerr
Mr. Ebersole
Dr. Lewis
Mr. Reed
Mr. Wylie
Dr. Davis
Dr. Lipinski

Metal Components, August 4 and 5, 1986, Hanford, WA (Igne). The Subcommittee will visit and review steam generator integrity program and visit its facilities. In addition, the integrated FM/NDE program will be discussed. Attendance by the following is anticipated:

Dr. Shewmon
Mr. Etherington
Dr. Lewis
Mr. Ward (tent.)
Dr. Bush
Mr. B. Thomspon

Reliability Assurance, August 5, 1986, 1717 H Street, NW, Washington, DC (Major), 8:30 A.M. (A.M. Only), Room 1046. The Subcommittee will review the final resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants." Lodging will be announced later. Attendance by the following is anticipated:

Mr. Wylie
Mr. Ebersole
Mr. Michelson
Dr. Siess

Safeguards and Security, August 5, 1986, 1717 H Street, NW, Washington, DC (Schiffgens), 8:30 A.M., Room 1046. The Subcommittee will review Technical Assistance Program on the "Evaluation of Methods of Reduction of Vulnerability to Sabotage (Generic Issue A-29)" with the NRC Staff. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Mark	Dr. Okrent
Dr. Carbon	Mr. Reed
Dr. Kerr	Mr. Ward (tent.)
Dr. Moeller	Mr. Wylie

Extreme External Phenomena, August 6, 1986 (tentative), 1717 H Street, NW, Washington, DC (Savio), 8:30 A.M., Room 1046. The Subcommittee will conduct a workshop to review the importance of seismic risk to nuclear power plants. Seismic hazard will be the principal topic to be discussed. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Okrent	Dr. Lewis
Dr. Carbon	Dr. Siess

(all other ACRS Members, as available)

316th ACRS Meeting, August 7-9, 1986, Washington, DC, Room 1046.

Maintenance Practices and Procedures, August 13, 1986, 1717 H Street, NW, Washington, DC (Alderman), 8:30 A.M. (A.M. Only), Room 1046. The Subcommittee will review the report on Phase I of Maintenance Program Plan. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Reed	Dr. Moeller
Mr. Michelson	Mr. Wylie

Decay Heat Removal Systems, September 24, 1986, 1717 H Street, NW, Washington, DC (Boehnert), 8:30 A.M., Room 1046. The Subcommittee will continue its review of NRR's proposed resolution position for USI A-45, "Shutdown Decay Heat Removal Systems." Lodging will be announced later. Attendance by the following is anticipated:

Mr. Ward	Mr. Reed
Mr. Ebersole	Dr. Catton
Mr. Michelson	Mr. Davis

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

Instrumentation and Control Systems, Date to be determined (July), Washington, DC, (El-Zeftawy). The Subcommittee will review the Westinghouse RVLIS level instrumentation. Attendance by the following is anticipated:

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

Nuclear Plant Chemistry, Date to be determined (July/August), Washington, DC (Alderman). The Subcommittee will discuss fission product source terms, aerosol behavior, emergency planning, etc. Attendance by the following is anticipated:

Dr. Moeller
Mr. Ebersole
Mr. Etherington
Mr. Reed
Dr. Shewmon

Westinghouse Water Reactors, Date to be determined (July/August), Washington, DC (El-Zeftawy). The Subcommittee will begin the PDA review of the Westinghouse Advanced Pressurized Water Reactor (RESAR SP/90). Attendance by the following is anticipated:

Mr. Reed
Dr. Kerr
Mr. Michelson
Dr. Shewmon
Mr. Wylie
Mr. Davis

Spent Fuel Storage, Date to be determined (July/August), Washington, DC (Alderman). The Subcommittee will continue its review of 10 CFR Part 72 and Monitored Retrievable Storage (MRS). Attendance by the following is anticipated:

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

Decay Heat Removal Systems, Date to be determined (mid-August), 1717 H Street, NW, Washington, DC (Boehnert). The Subcommittee will review NRR's Action Plan to address concerns with the reliability of certain plants' AFW systems. Attendance by the following is anticipated:

Mr. Ward
Mr. Ebersole
Mr. Michelson
Mr. Reed
Dr. Catton
Mr. Davis

Thermal Hydraulic Phenomena, Date to be determined (mid-August), 1717 H Street, NW, Washington, DC (Boehnert). The Subcommittee will continue its review of the RES-proposed revision to the ECCS Rule (10CFR50.46 and Appendix K). Attendance by the following is anticipated:

Mr. Michelson
Mr. Ebersole
Mr. Reed
Mr. Ward
Dr. Catton
Mr. Schrock
Dr. Sullivan
Dr. Tien

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

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

Regional Operations, Date to be determined (August-September), Chicago, IL (Boehnert). The Subcommittee will begin its reivew of the activities of the NRC Regional Offices. This meeting will focus on the activities of the Region III Office. Attendance by the following is anticipated:

Dr. Remick	Mr. Reed
Dr. Carbon	Mr. Wylie
Mr. Michelson	

Seabrook Units 1 and 2, Date to be determined (late summer/early fall), Washington, DC (Major). The Subcommittee will review the application for a full power operating license for Seabrook 1 and 2. Attendance by the following is anticipated:

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

Structural Engineering, Date to be determined (late 1986), Albuquerque, NM (Igne). The Subcommittee will visit and review containment integrity and Category I structures, facilities, and programs. Attendance by the following is anticipated:

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

Probabilistic Risk Assessment, Date to be determined (September/October), Washington, DC (Savio). The Subcommittee will review the probabilistic risk assessment for Millstone 3. Attendance by the following is anticipated:

Dr. Okrent	Mr. Michelson
Dr. Kerr	Dr. Siess
Mr. Ebersole	Mr. Ward
Dr. Lewis	Mr. Wylie
Dr. Mark	

FLASH !!!

The meetings listed below have been changed or added to the list of Subcommittee meetings previously issued at the full Committee meeting.

CHANGED!!!

The Long Range Plan meeting previously scheduled for June 17, 1986 has been POSTPONED to JULY 9, 1986.

Long Range Plan for NRC (CLOSED), July 9, 1986, 1717 H Street, NW, Washington, DC, (Major), 1:00 P.M., Room 1046. The Subcommittee will consider the NRC Staff's Five Year Plan and discuss the guidelines for the review of a long range plan. Lodging will be announced later. Attendance by the following is anticipated:

Dr. Carbon
Dr. Moeller
Mr. Remick
Mr. Wylie

ADDED!!!

Management Committee (CLOSED), July 9, 1986, 1717 H Street, NW, Washington, DC (Fraley), 1:30 P.M., Chairman's Office. Specific topics have not been selected as yet. Lodging will be announced later. Attendance by the following is anticipated:

Mr. Ebersole
Dr. Lewis
Mr. Ward

Procedures and Administration, August 6, 1986, 1717 H Street, NW, Washington, DC (Fraley), 1:00 P.M. The Subcommittee will consider the Effectiveness Panel's recommendations regarding ACRS Officers' terms. Lodging will be announced later. Attendance by the following is anticipated:

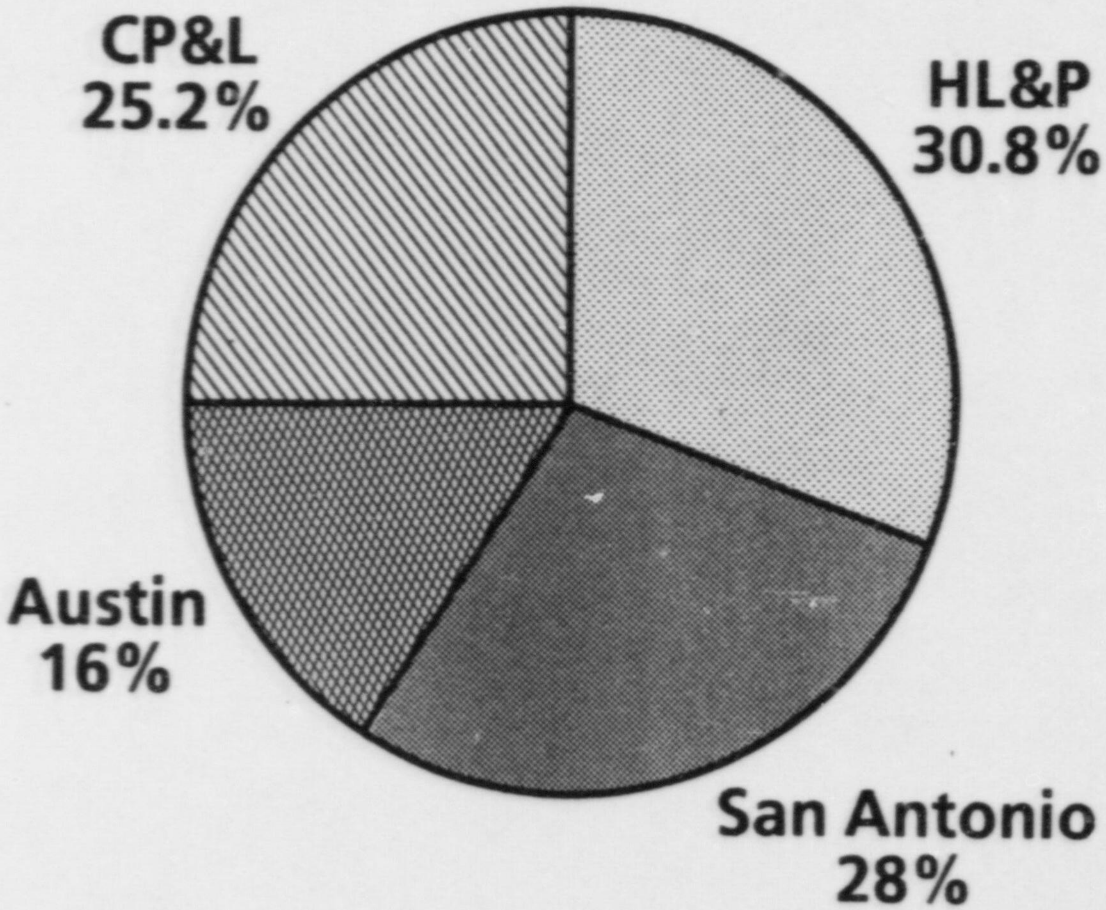
Mr. Ward
Mr. Ebersole
Dr. Lewis
Dr. Moeller
Dr. Remick
Dr. Siess

APPENDIX IV
APPLICANT'S (HL&P) INTRODUCTION

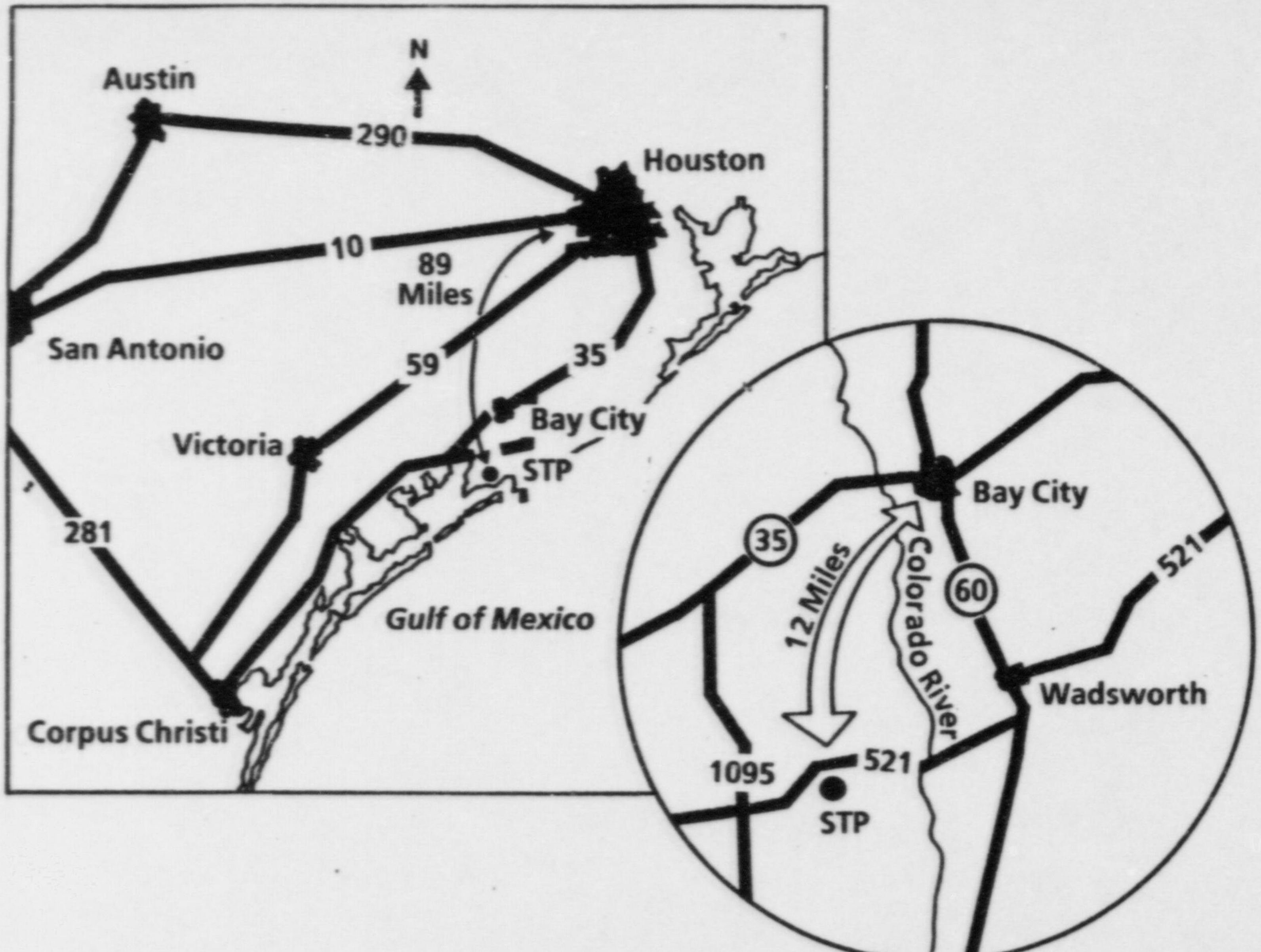
SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

INTRODUCTION

SOUTH TEXAS PROJECT PARTNERS



STP



SITE LOCATION

BRIEF HISTORY

• June 1973	- Plans to build STP announced - Brown & Root named as A/E constructor
• August 1975	- LWA granted
• December 1975	- CP issued
• August 1979	- Unit 1 NSSS Components set
• November 1979	- Special NRC inspection commences
• December 1979	- Stop work on complex concrete placement
• April 1980	- Stop work issued on welding
• April 1980	- Order to show cause issued
• October 1980	- Welding restarted
• January 1981	- Restart on complex concrete
• May 1981	- ASLB hearings on OL commence

BRIEF HISTORY (Continued)

- September 1981 - Brown & Root terminated, Bechtel hired
- February 1982 - Ebasco named as new constructor
- June 1982 - Non safety-related construction resumed
- August 1982 - Safety-related construction resumed
- March 1984 - ASLB Partial Initial Decision
- March 1985 - First systems turned over to Start-up
- May 1985 - Energization
- July 1985 - ASLB Phase II Hearings
- December 1985 - Commenced NSSS Flush
- March 1986 - ASLB Phase III Pre-hearing
- April 1986 - SER issued

STP STATUS - MAY 1986

Percent Complete			Key Dates	
	Scheduled	Actual	Fuel Load	Commercial Operation
Construction				
Unit 1 Power Block and BOP	90.3	89.9	6/87	12/87
Unit 2 Power Block	59.8	60.4	12/88	6/89
Total	77.6	77.5		
Engineering	93.3	92.9		

SOUTH TEXAS PROJECT MILESTONES

• Secondary hydro	July 1986
• Primary hydro	August 1986
• Hot functional test	January 1987
• ILRT / SIT	March 1987
• Fuel load	June 1987

PROJECT STAFFING - MAY 1986

Bechtel Construction Management	695
EBASCO manual	5,410
EBASCO non-manual	1,451
Bechtel Engineering and Home Office	540
HL&P*	1,669
Other	362
Total	10,127

*Includes all testing personnel

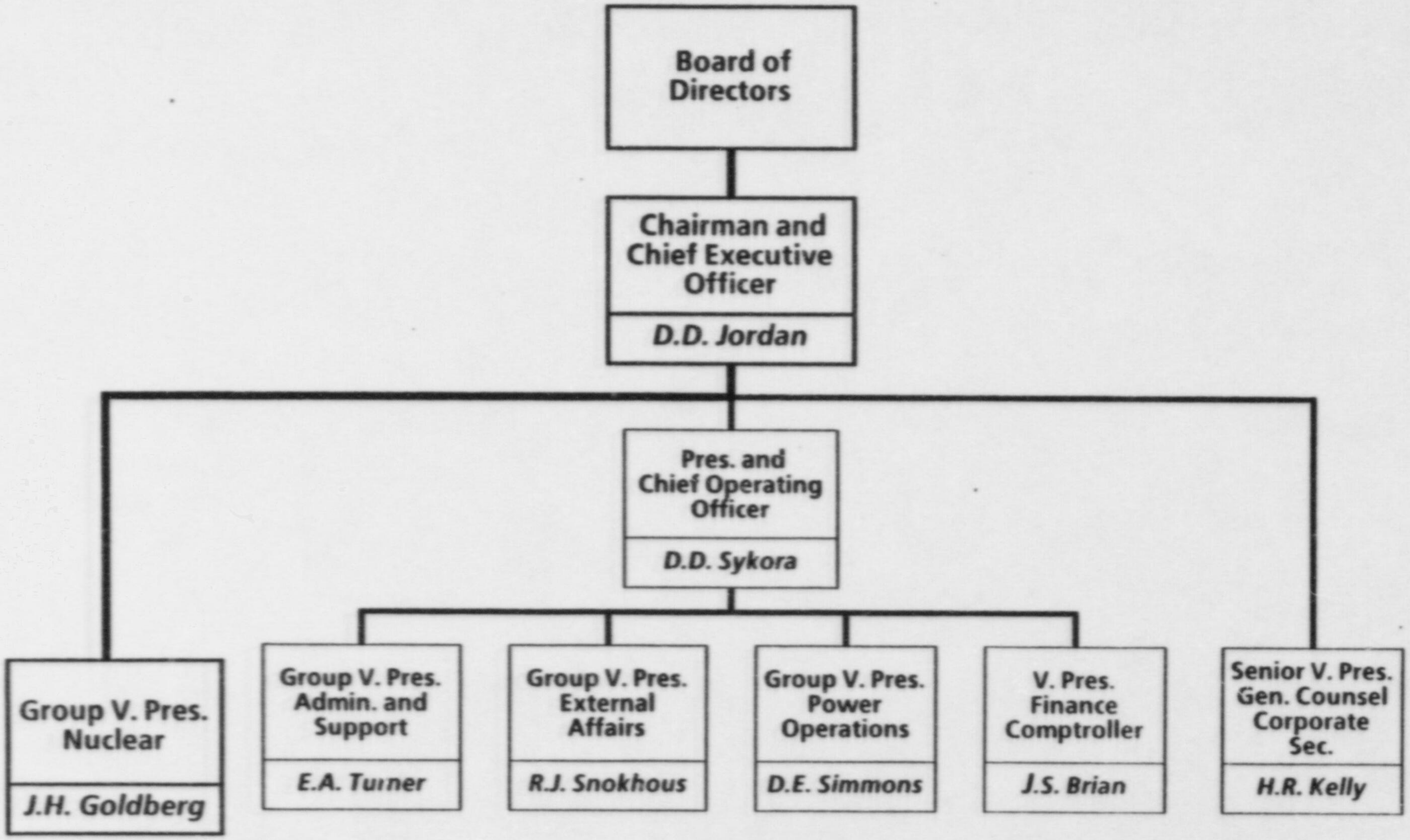
STP

CONSTRUCTION ORGANIZATION, PHILOSOPHY AND STATUS

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

APPENDIX V
CONSTRUCTION ORGANIZATION, PHILOSOPHY
AND STATUS

HOUSTON LIGHTING & POWER COMPANY
Senior Executive Organization



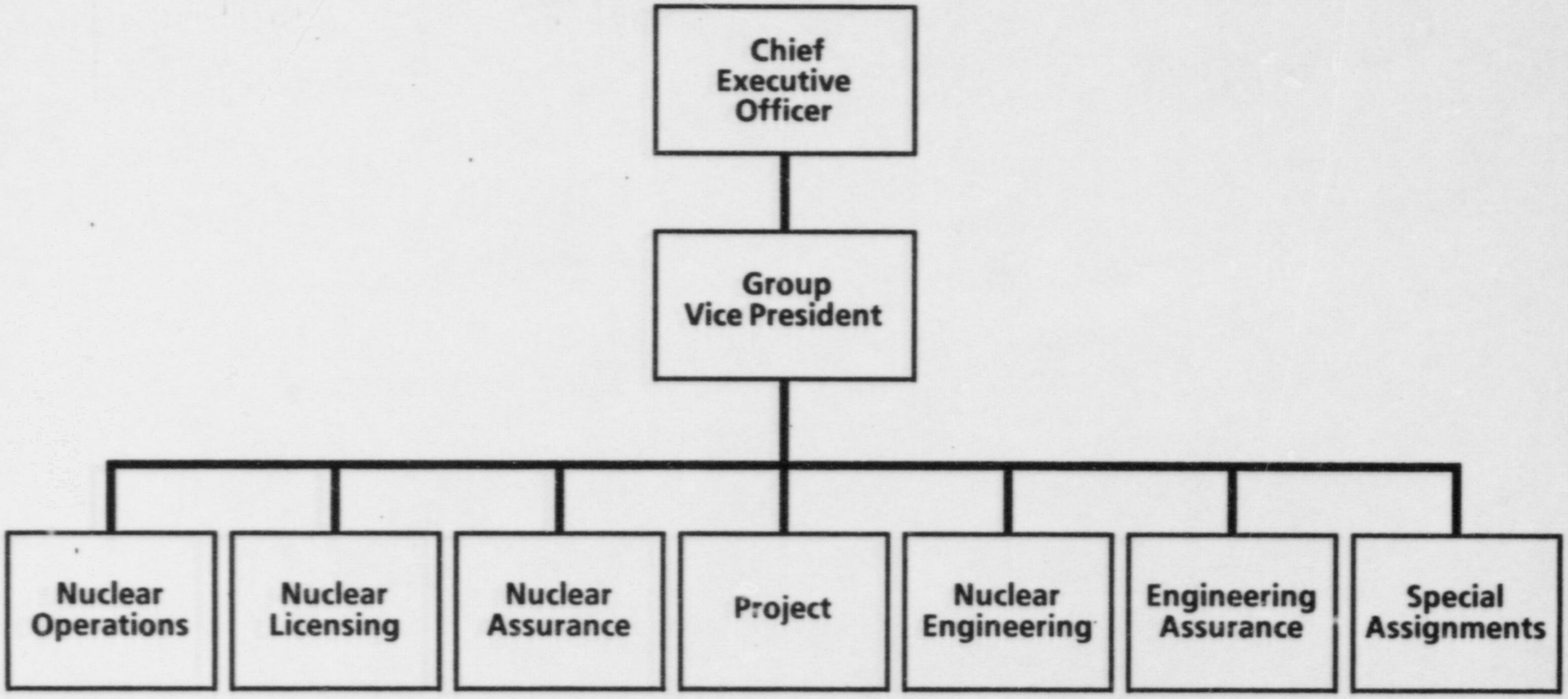
MANAGEMENT PHILOSOPHY

- To build and operate the plant in full compliance with regulatory requirements
- To require the person performing a task to accept total responsibility for performing that task in a quality manner
- To require Quality Assurance to independently confirm the quality of activities being performed
- To perform management oversight of project activities to ensure compliance with applicable program requirements
- To report in a timely forthright manner, all matters required by regulatory authorities

MANAGEMENT PHILOSOPHY (Continued)

- To protect the environment and the health of the public and our employees
- To utilize proven equipment and techniques in design, construction and operation to assure reliable operation of the plant
- To keep abreast of industry occurrences and apply worthwhile experiences to improve programs
- To ensure that the concerns of employees are heard and properly acted upon
- To learn from our mistakes by determining underlying causes and to take necessary actions to preclude recurrence
- To plan for, develop, and retain qualified and trained personnel

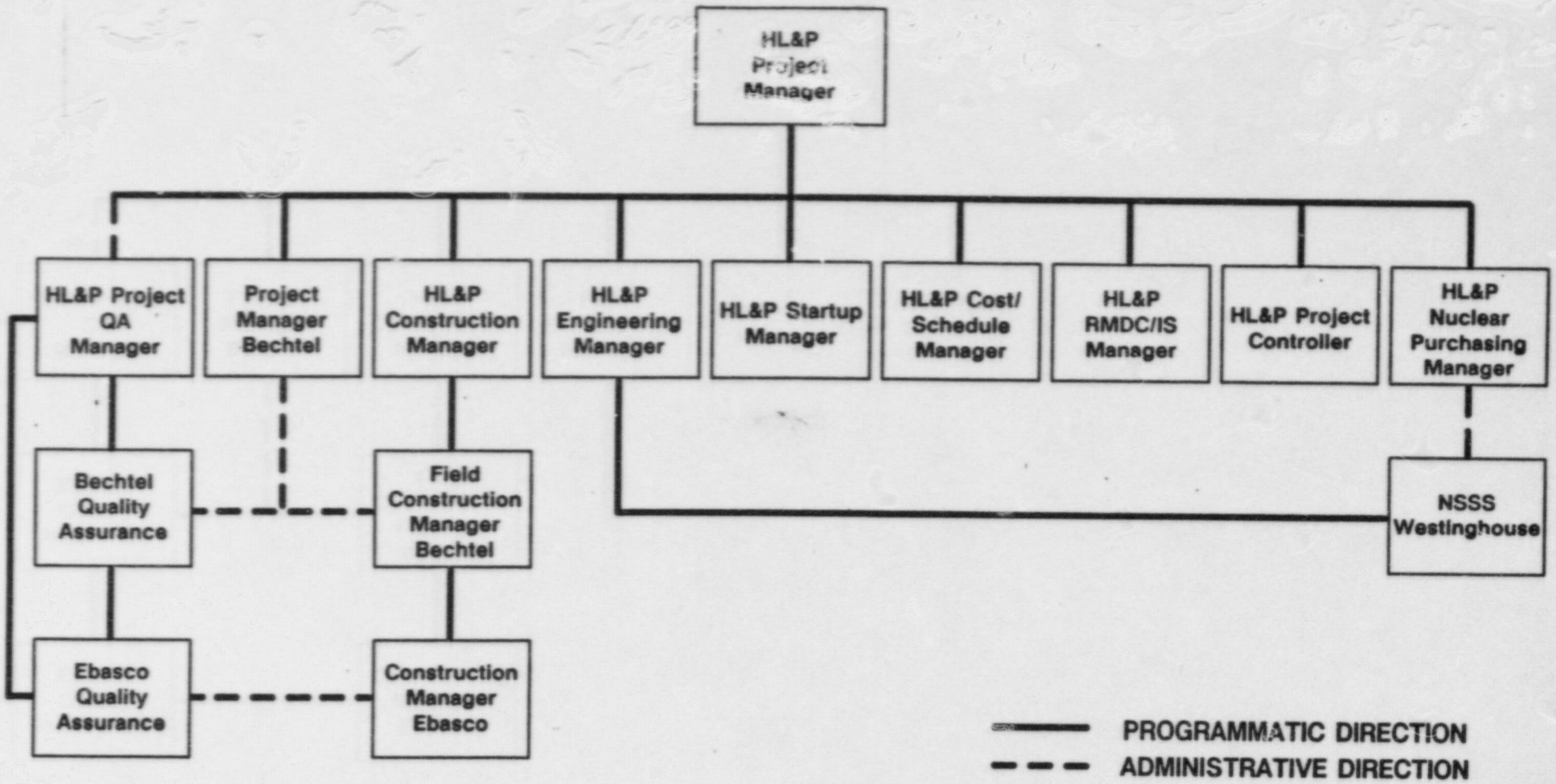
HOUSTON LIGHTING AND POWER ORGANIZATION



STP ORGANIZATIONS

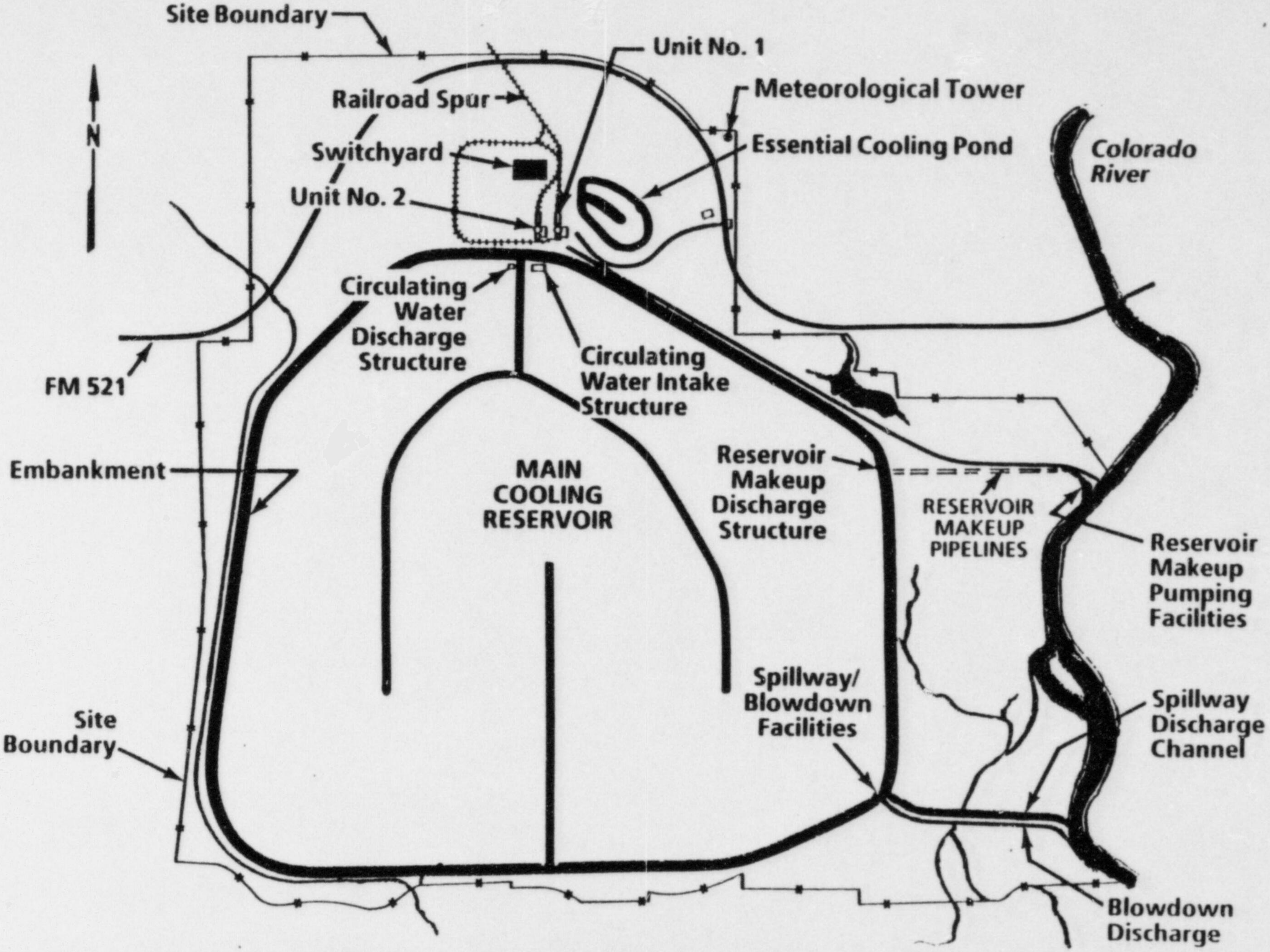
• HL&P	-	Project Manager
• Bechtel	-	Architect/Engineer & Construction Manager
• Ebasco	-	Constructor
• Westinghouse	-	NSSS

SOUTH TEXAS PROJECT



SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

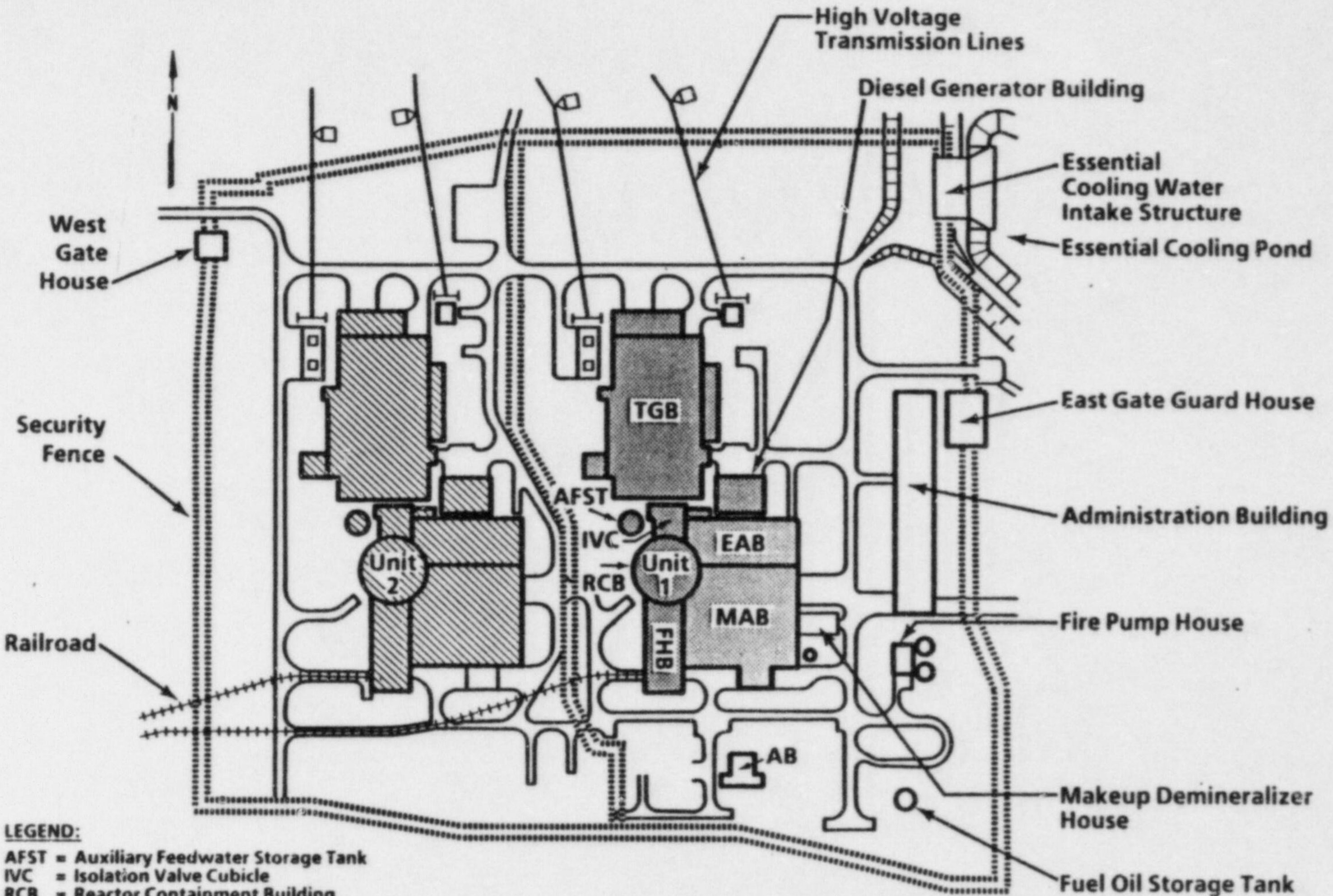
PLANT LAYOUT



Site Boundary
Unit No. 1
Meteorological Tower
Railroad Spur
N
Switchyard
Essential Cooling Pond
Colorado River
Unit No. 2
Circulating Water Discharge Structure
Circulating Water Intake Structure
FM 521
MAIN COOLING RESERVOIR
Embankment
Reservoir Makeup Discharge Structure
RESERVOIR MAKEUP PIPELINES
Reservoir Makeup Pumping Facilities
Spillway/ Blowdown Facilities
Spillway Discharge Channel
Site Boundary
Blowdown Discharge

SITE MAP

STP



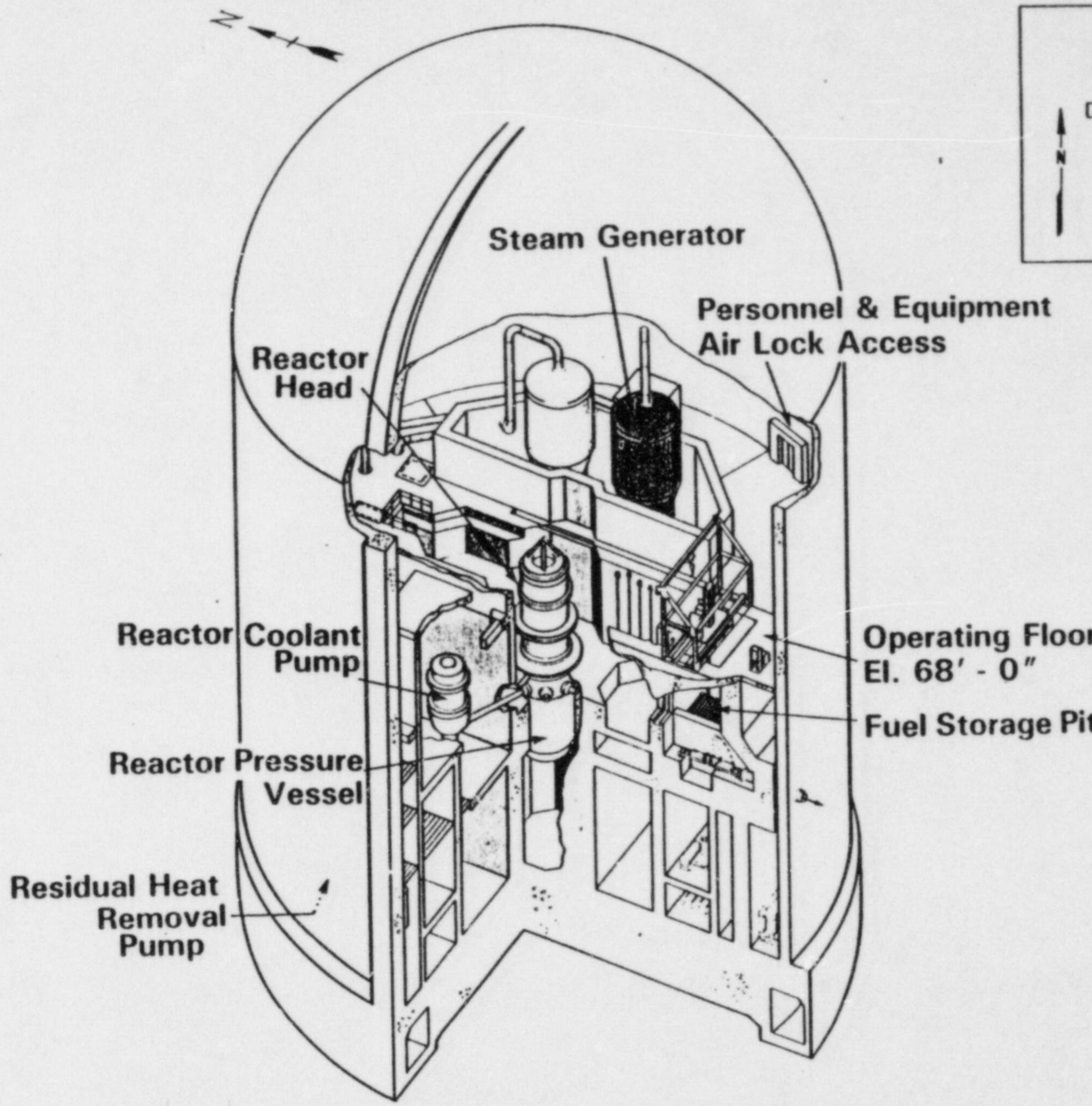
LEGEND:

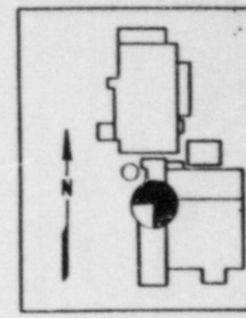
AFST = Auxiliary Feedwater Storage Tank
IVC = Isolation Valve Cubicle
RCB = Reactor Containment Building
FHB = Fuel Handling Building
AB = Auxiliary Boiler
TGB = Turbine Generator Building
EAB = Electrical Auxiliary Building
MAB = Mechanical Auxiliary Building

PLOT PLAN

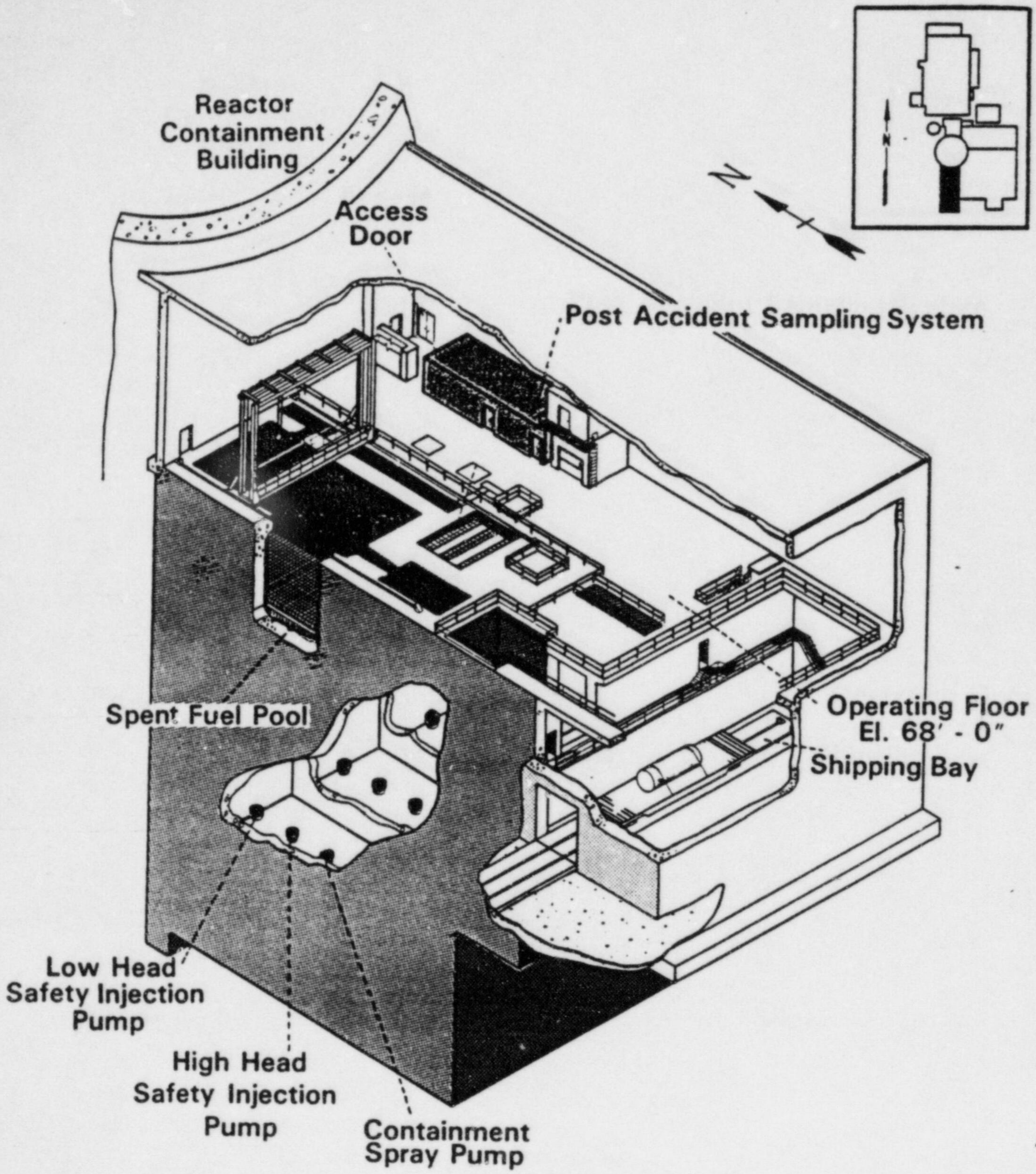
STP

Reactor Containment Building



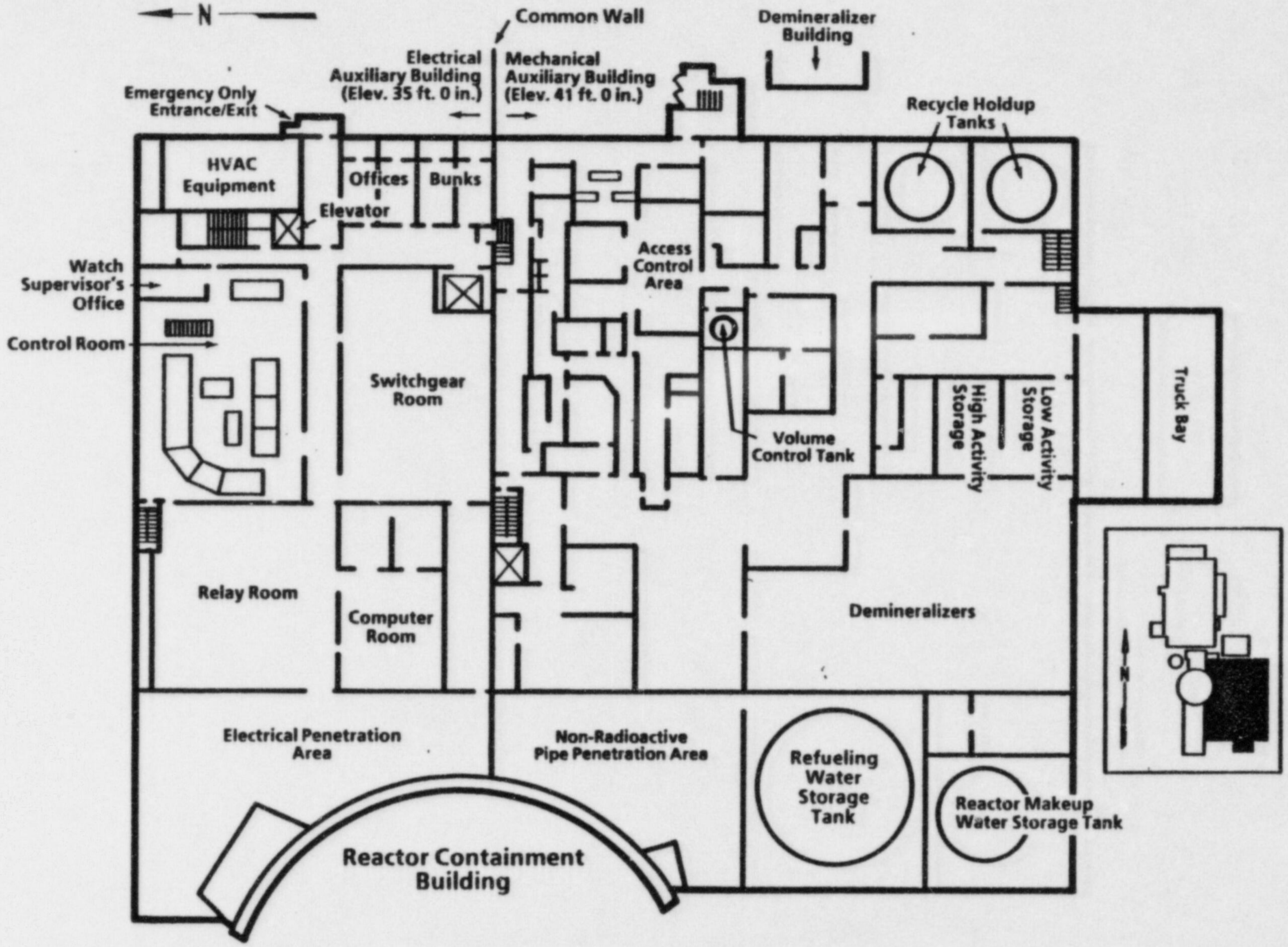


STP



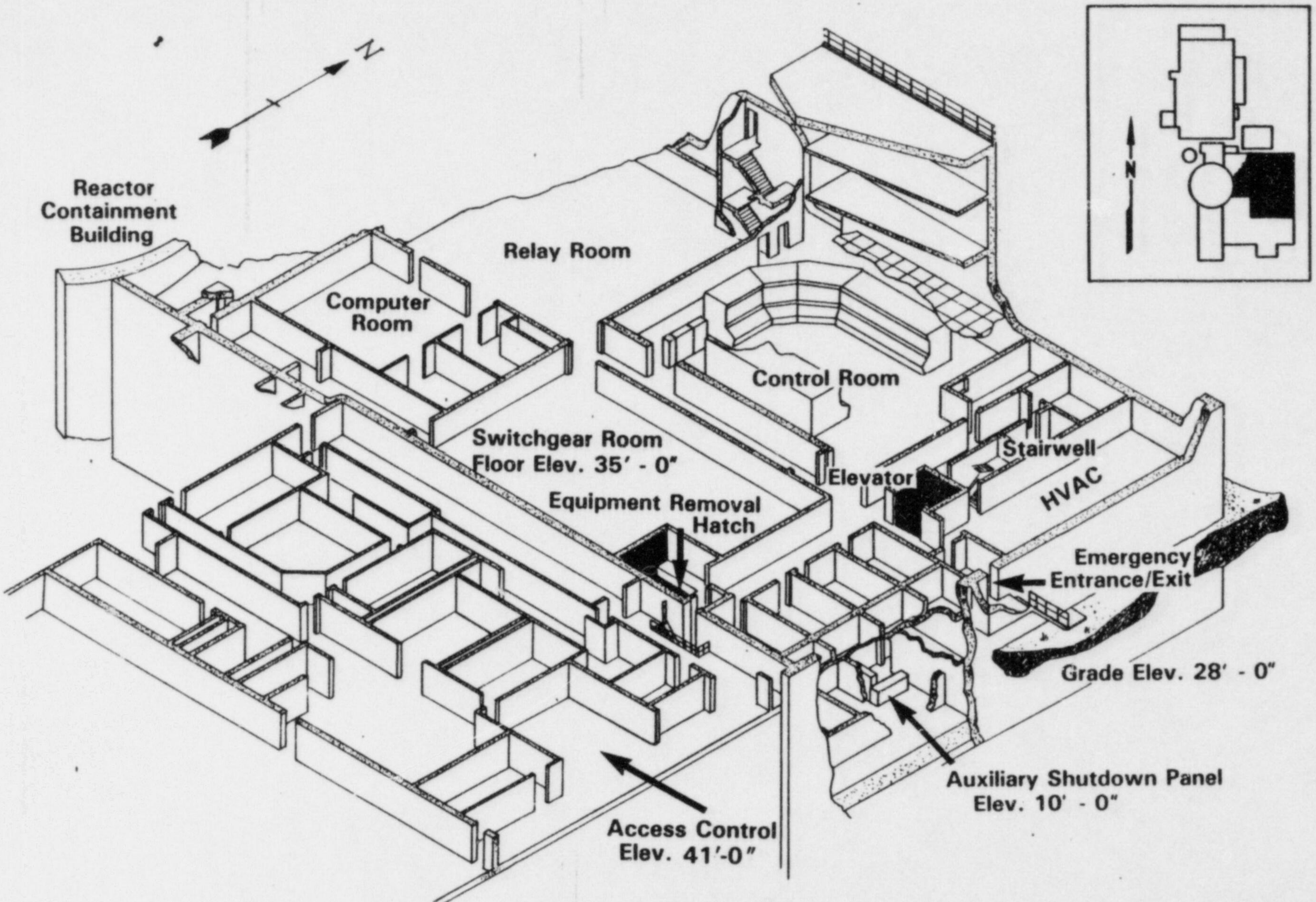
Fuel Handling Building

STP

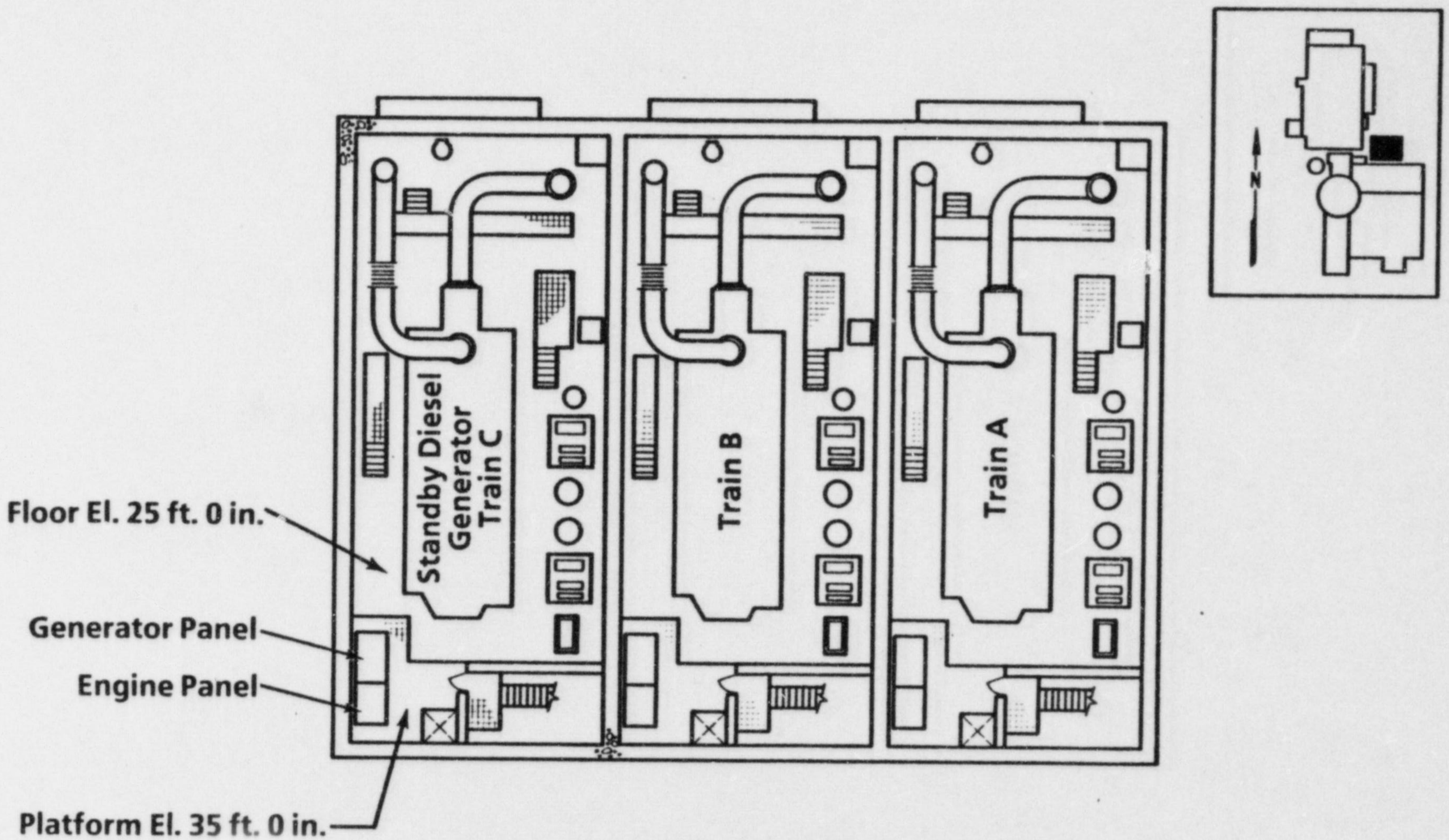


MECHANICAL AND ELECTRICAL AUXILIARY BUILDING

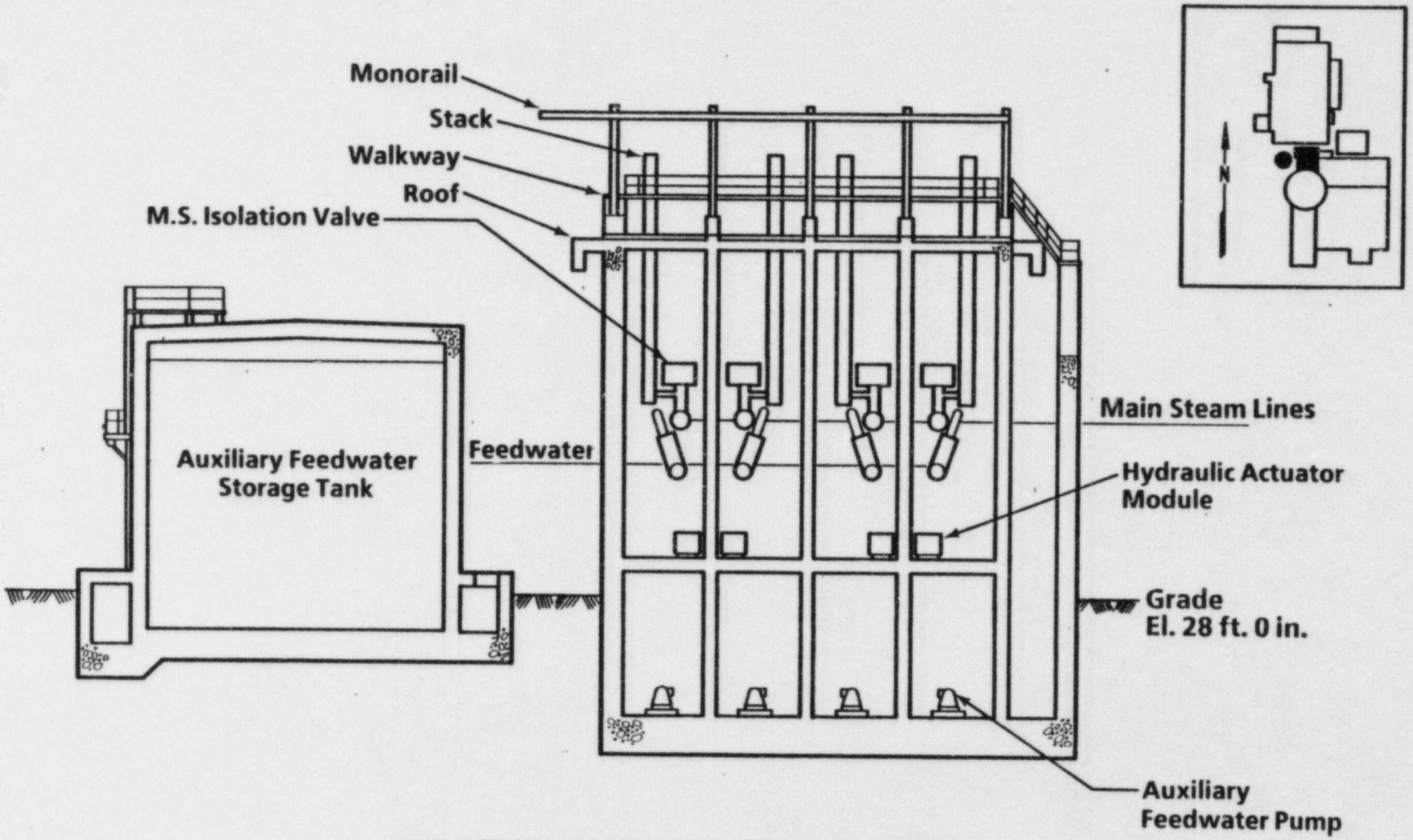
STP



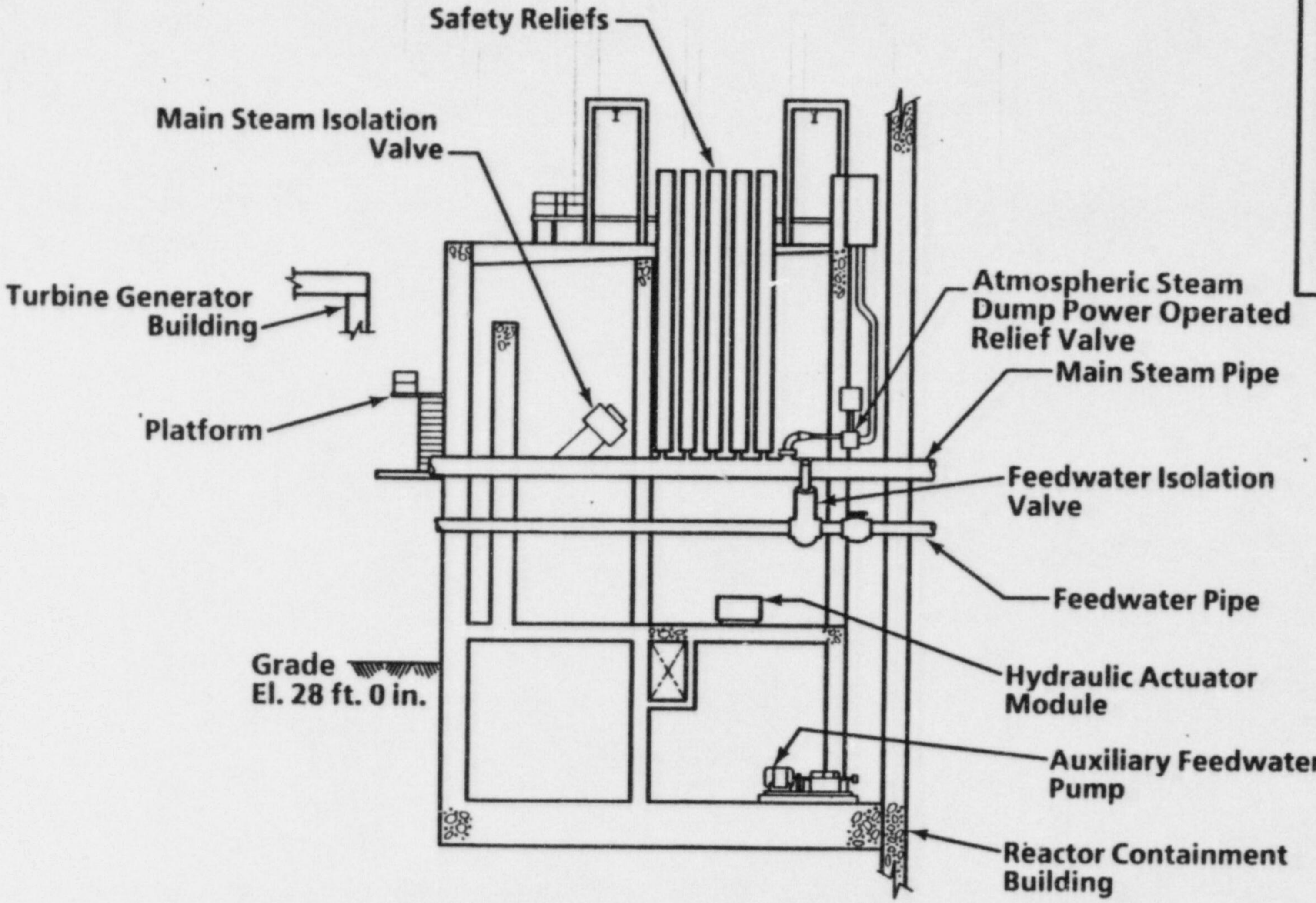
Mechanical and Electrical Auxiliary Building

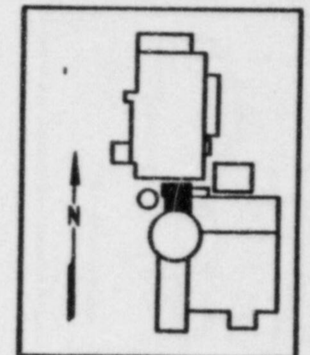


DIESEL GENERATOR BUILDING



ISOLATION VALVE CUBICLE AND AFWST





CROSS SECTION - ISOLATION VALVE CUBICLE

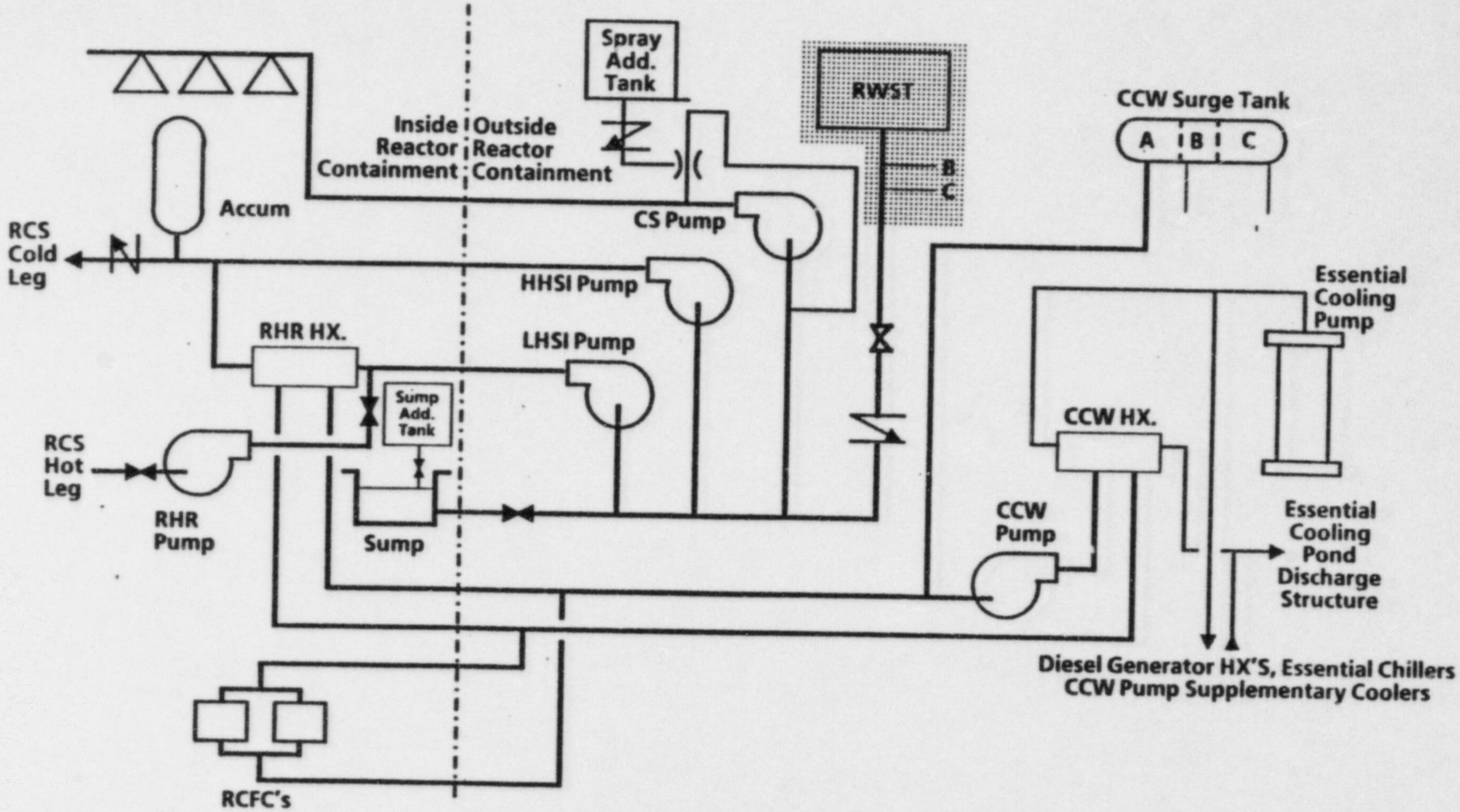
THREE TRAIN DESIGN

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

APPENDIX VII
THREE TRAIN DESIGN

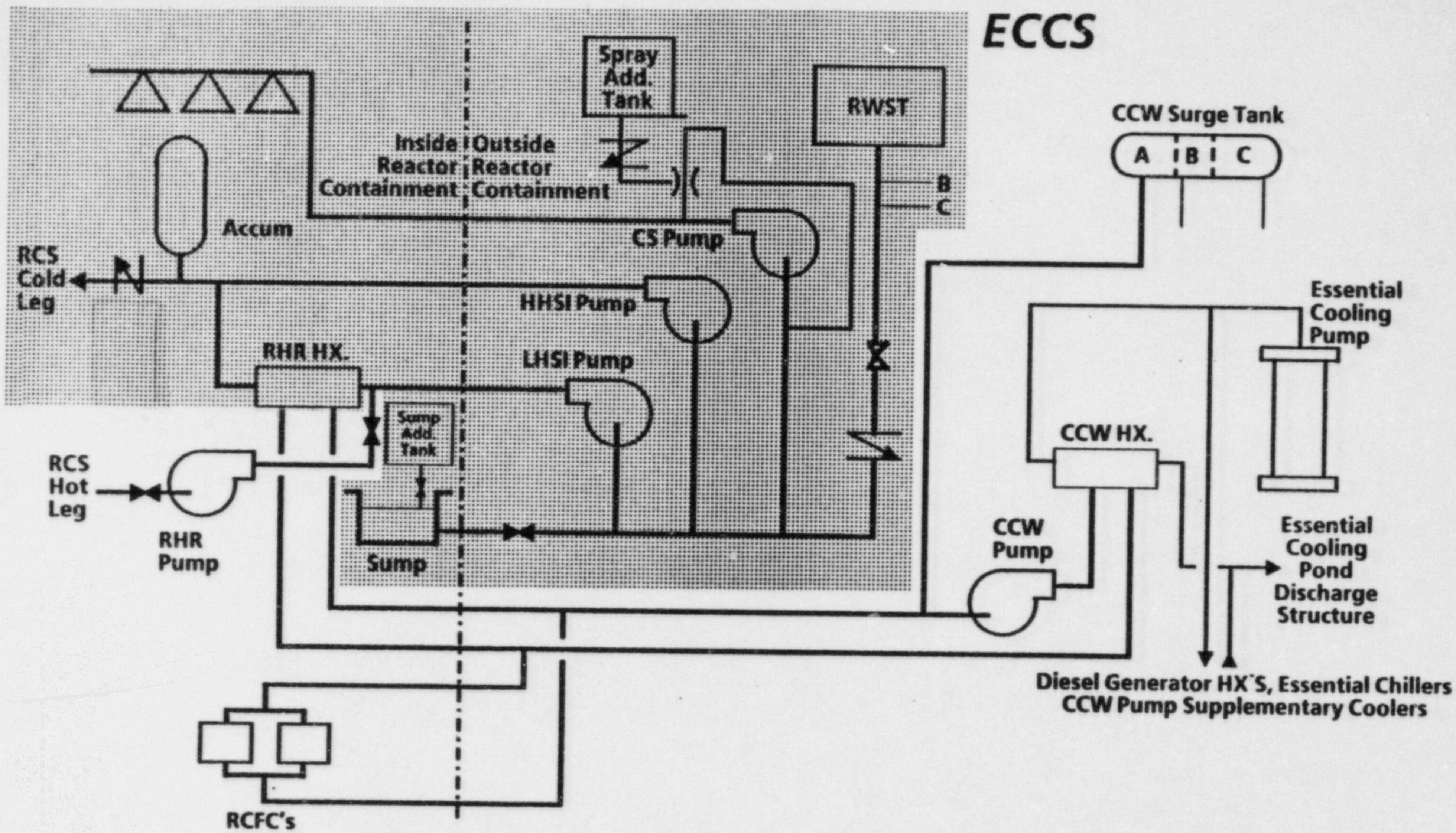
SOUTH TEXAS PROJECT

- Three-train ESF systems
  - Physically segregated
  - Electrically independent



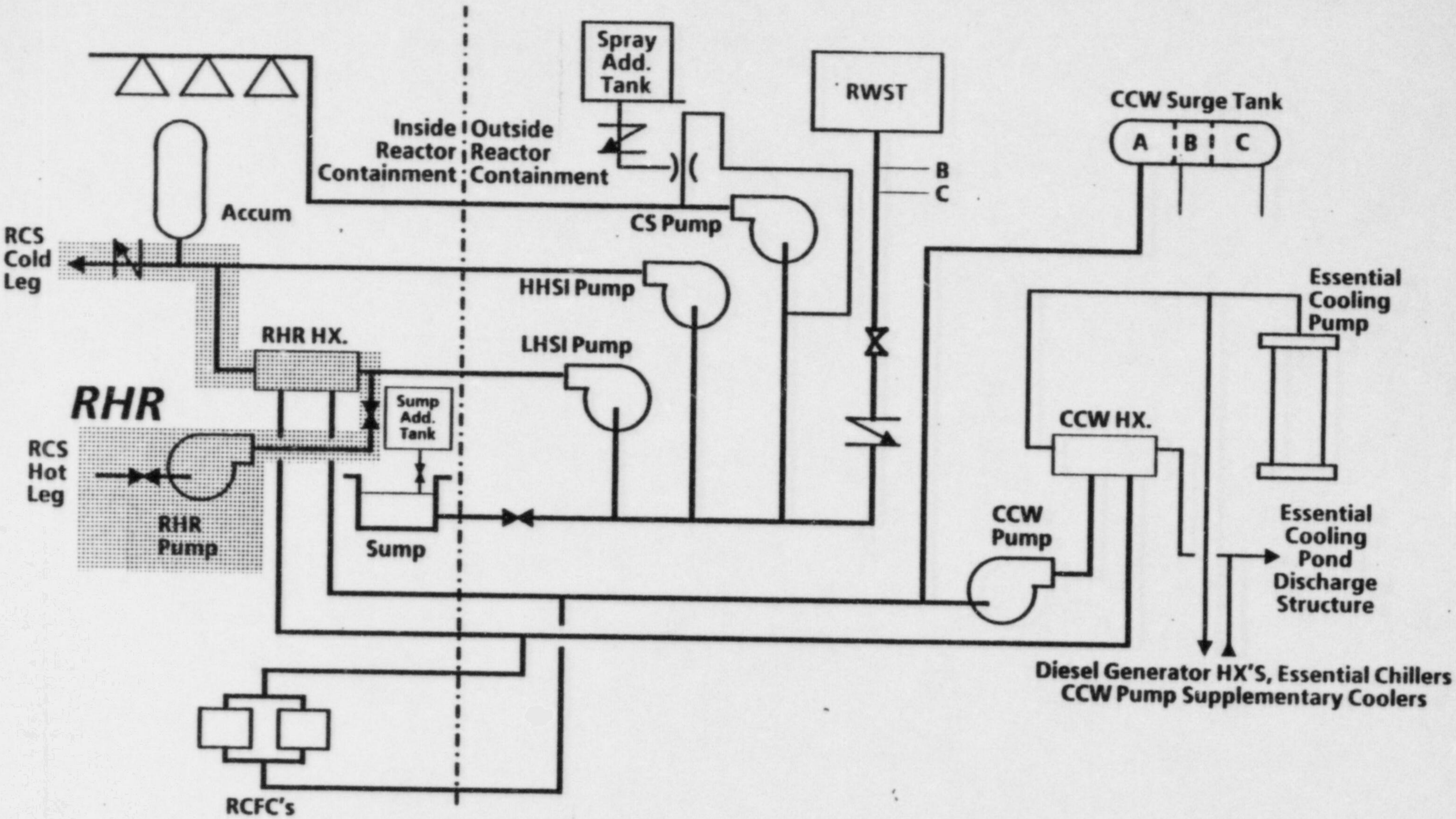
INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT (Typical of Three)

STP



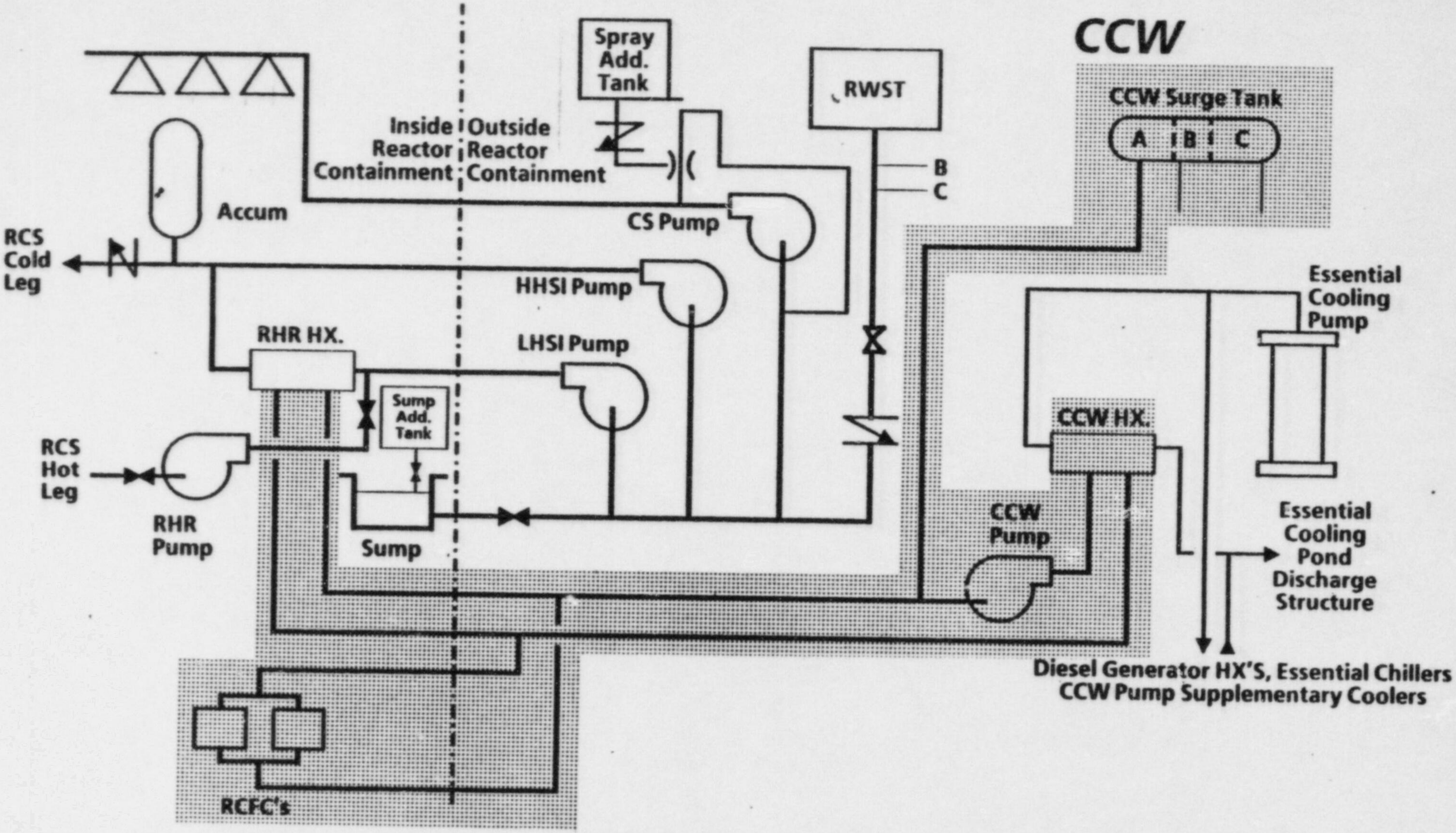
INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT (Typical of Three)

STP



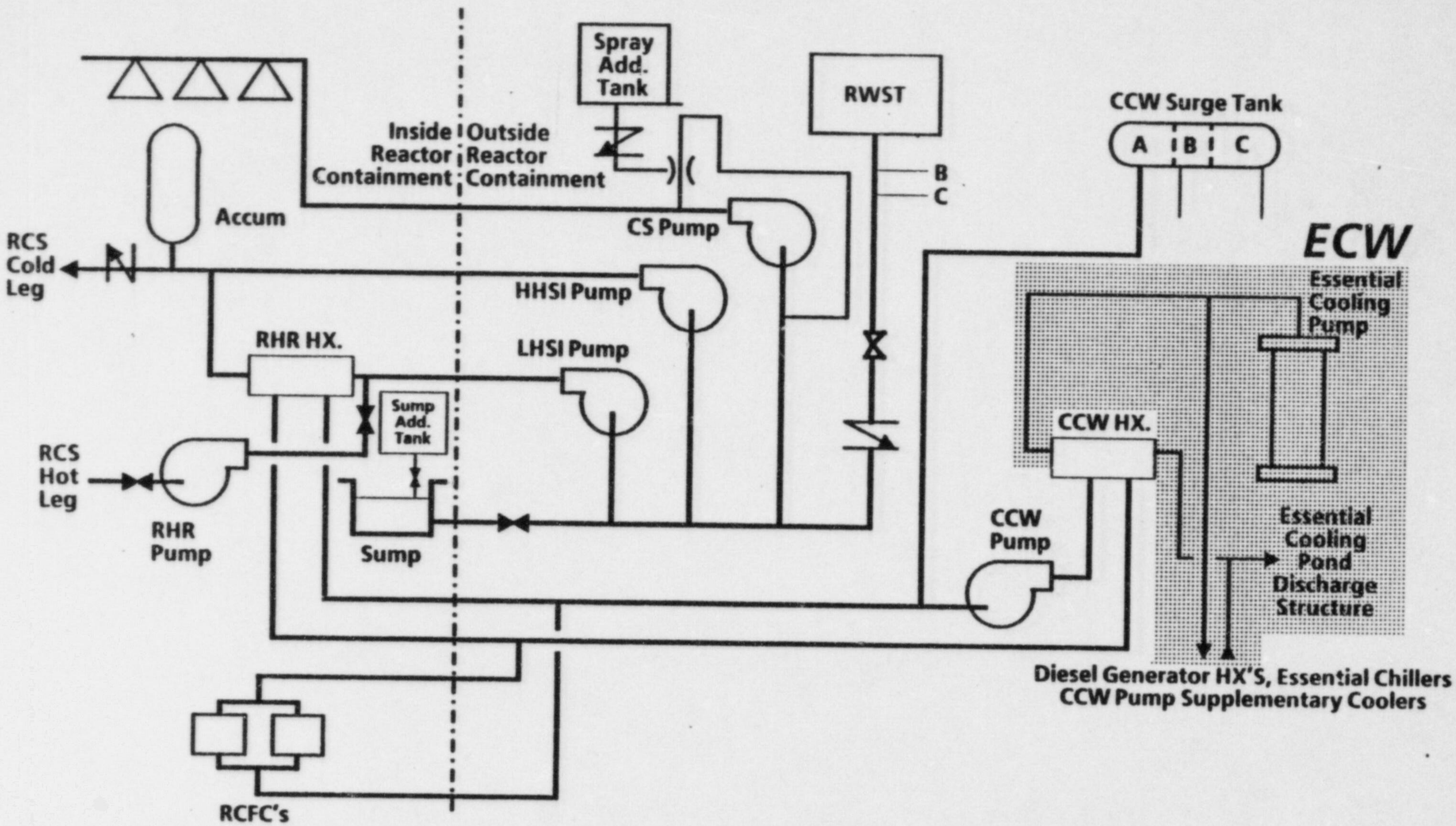
INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT (Typical of Three)

STP



INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT (Typical of Three)

STP



INTERRELATIONSHIP OF ESF SYSTEMS IN THE THREE-TRAIN CONCEPT (Typical of Three)

STP

SOUTH TEXAS PROJECT

Benefits:

- Proven equipment size
- Greater margin provided
- Greater protection for plant investment
  - Single train for small break LOCA
  - Single train for normal shutdown
  - Redundant paths for fire protection
- Simpler piping

SOUTH TEXAS PROJECT THREE TRAIN SYSTEMS

System	Trains Installed	Minimum number of trains required		
		Normal Operation / Shutdown	Large Breaks	Other Accidents
Diesel Generators	3	0/0	2	1
Essential Cooling Water	3	1/1	2	1
Component Cooling Water	3	1/1	2	1
Reactor Containment Fan Coolers	3 (2 RCFC Units per train)	2/2*	2 (3 RCFC Units)	1
Safety Injection	3	0/0	2**	1
Containment Spray	3	0/0	2	0
Residual Heat Removal Heat Exchangers	3	0/1	2	1
Auxiliary Feedwater	4	0/0	2	1

* Normally supplied by RCB chilled water
** Share RHR exhanger - RHR pumps not required

STP HVAC THREE TRAIN SYSTEMS

System	Trains Installed	Minimum number of trains required		
		Normal Operation / Shutdown	Large Breaks	Other Accidents with Load Shedding
Essential Chilled Water*	3	2/2	2	1
EAB HVAC	3	2/2	2	1
Control Room HVAC	3	2/2	2	1
FHB Exhaust	3	2/2	2	1
ECW Ventilation*	3	1/1	2	1
DGB Ventilation ESF*	3	0/0	2	1
IVC ESF Ventilation*	3	0/0	2	1
ECCS ESF Cubicle Cooling*	3	0/0	2	1
CCW ESF Cubicle Cooling*	3	1/1	2	1
Charging Pump Coolers*	3	1/1	0	0
Chiller Cubicle Cooler*	3	2/2	2	1

STP

* Cubicles are provided with individual 100% cooling and ventilation system

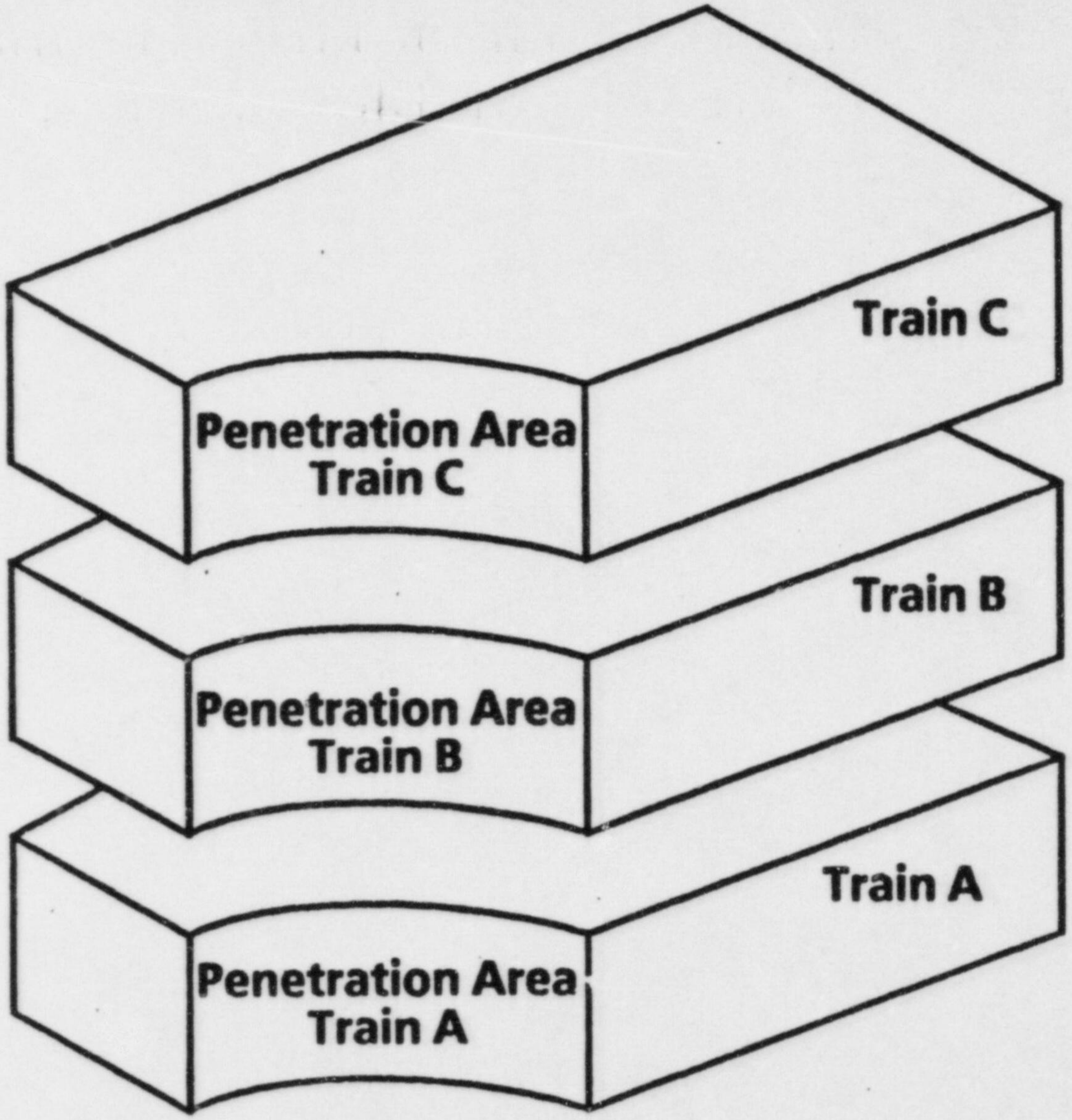
SOUTH TEXAS PROJECT
FIRE PROTECTION FEATURES

- Provides two shutdown pathways assuming a fire in any fire area

  <u>VERSUS</u>

  Only one safe shutdown pathway as required by Appendix R.
- Complies with the requirements of Appendix R and provides equal or better alternatives to the criteria of Appendix A to branch technical position APCSB 9.5-1
- Safe shutdown capability provided outside the control room for all three trains with capability to maintain cold shutdown

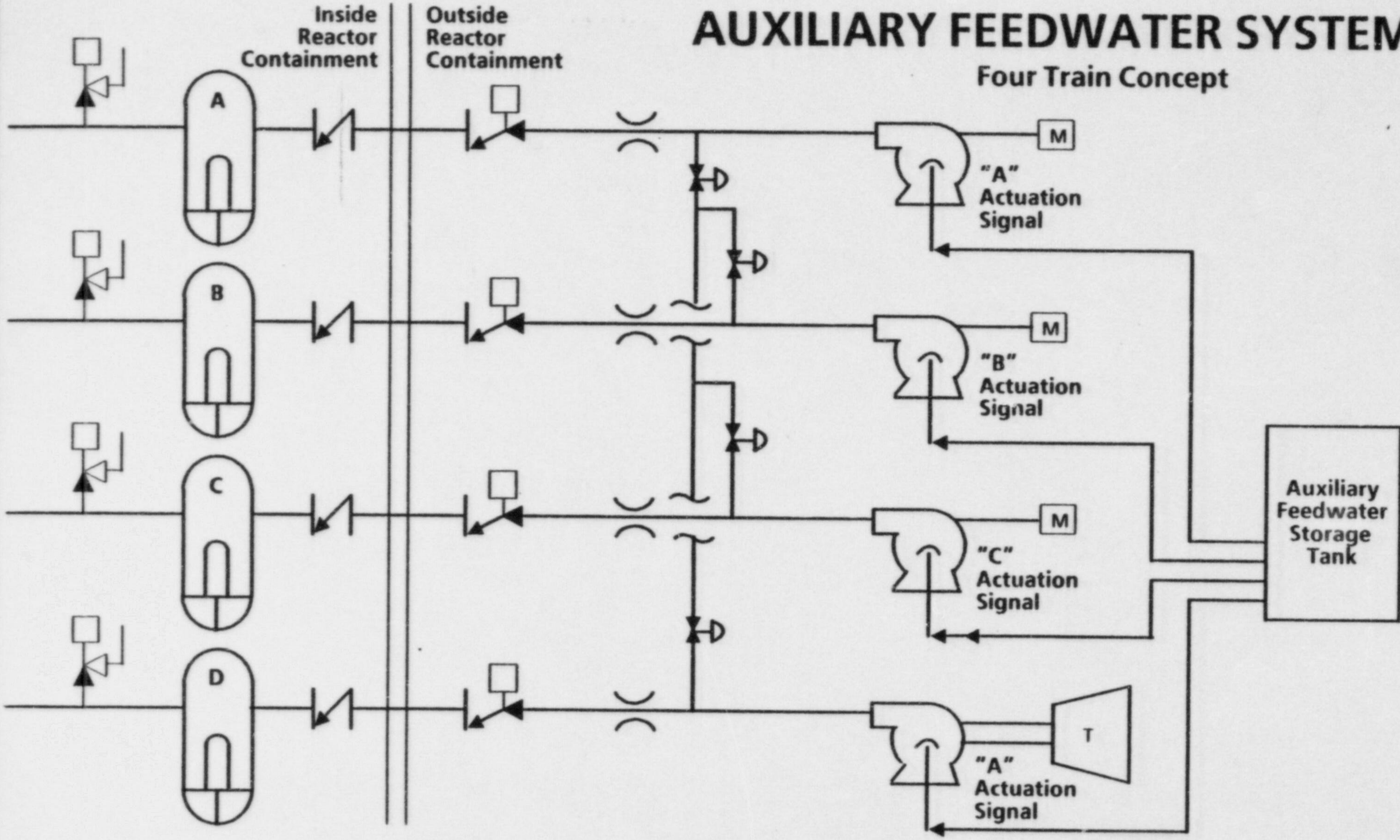
SIMPLIFIED ELEC AUXILIARY BUILDING - VERTICAL SEPARATION



STP

AUXILIARY FEEDWATER SYSTEM

Four Train Concept



STP

CONTROL ROOM INTEGRATION

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

CONTROL ROOM DESIGN REVIEW CONTRIBUTING FACTORS - 1982

- Safety related control panels <20% complete
- Final evolution of TMI criteria
- W OG ERG's available
- Active participation of HL&P Operations

CONTROL ROOM DESIGN REVIEW
INTEGRATED DESIGN CONCEPT

Develop integrated criteria to address:

- Human factors
- Post-accident monitoring (including R.G. 1.97)
- Safety parameter display
- Emergency operating procedures
- Safety grade cold shutdown
- Bypass/inoperable status monitoring
- Annunciator and alarm prioritization

STP

CONTROL ROOM DESIGN REVIEW INTEGRATED DESIGN CONCEPT (Continued)

Constructed full-scale mock-up / utilized simulator

- Performed CRDR
- Performed re-layout
  - Total re-layout of 6 panels including ESF panels
  - Upgraded layout of remaining 4 panels

STP

MILESTONES - CRDR

CRDR started	8/82
Issue program plan/plan report	10/82
Issue implementation plan report	3/83
NRC in-process audit	5/83
Panel re-design released to fabrication	9/83
NRC audit report	10/83
Issue executive summary report	4/84
Panel delivery/installation	6/84
NRC site visit	10/84
Issue executive summary report addendum	4/85

STP

REMAINING WORK ITEMS - CRDR

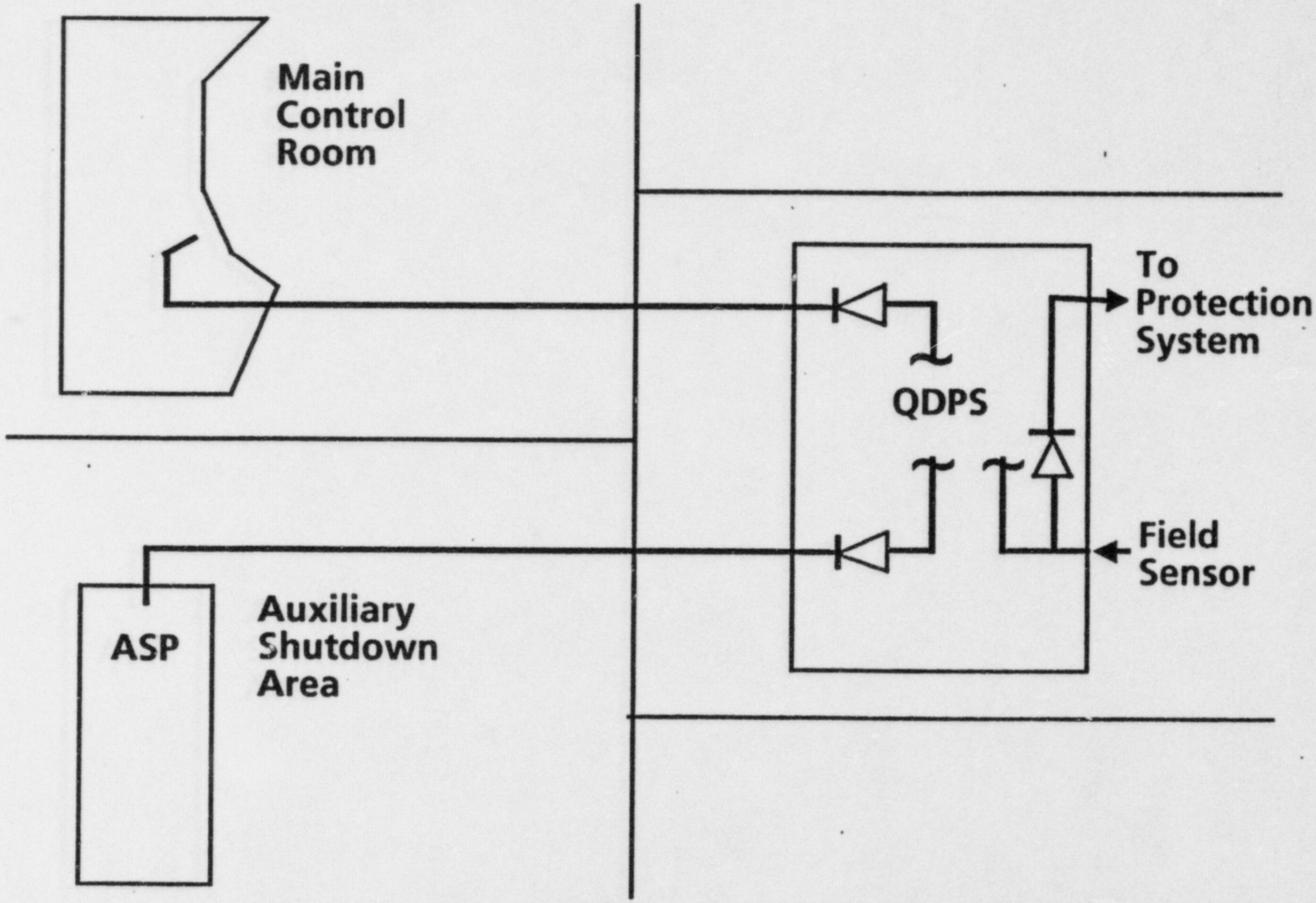
- Checkout of labels and scales
- Complete surveys dependent on control room completion
  - Lighting/indicator visibility
  - Sound/annunciator horns
  - Computer displays
  - Workspace
  - Communications
- EOP validation

ALTERNATIVE SHUTDOWN INTEGRATED DESIGN CONCEPT

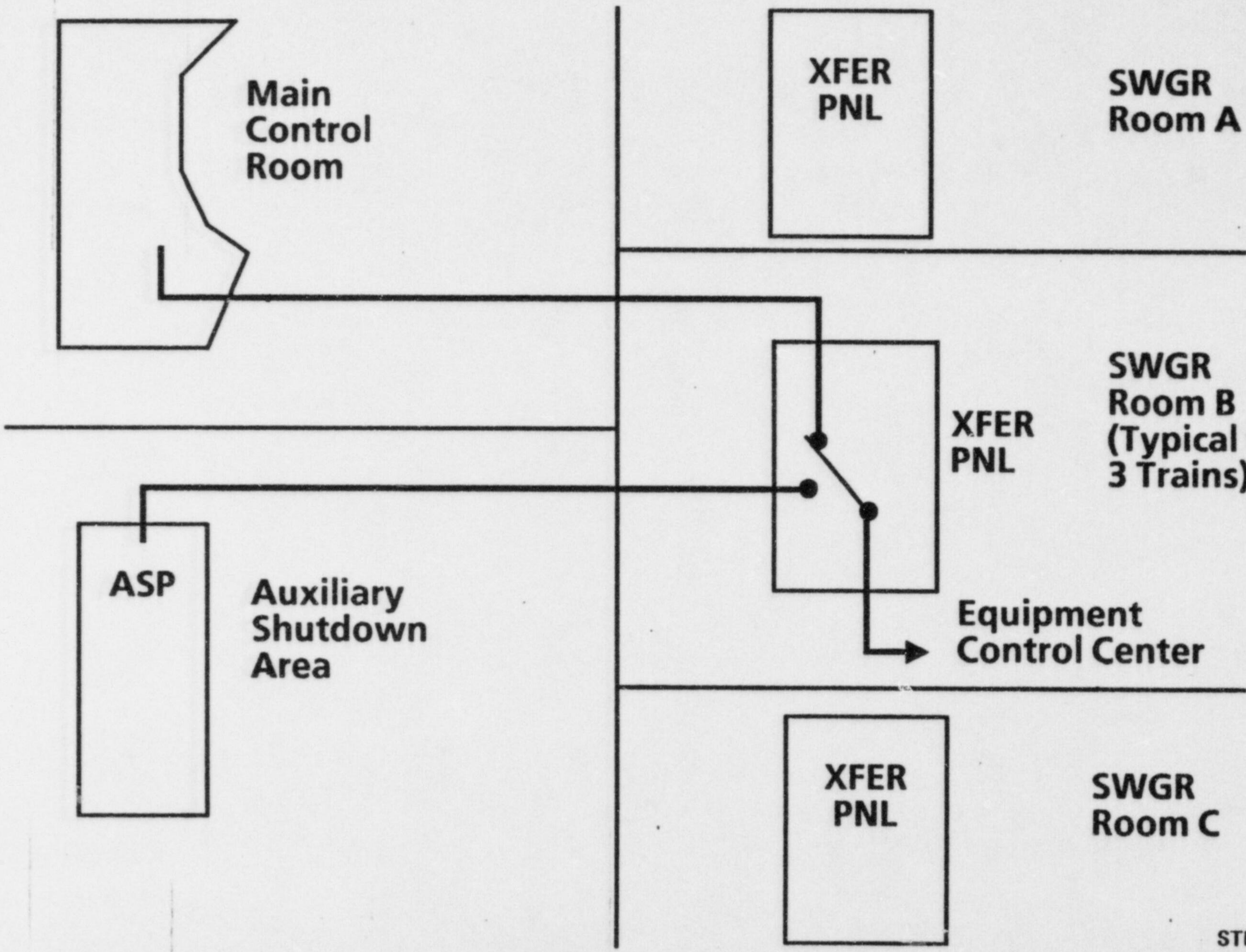
- Appendix R criteria
- Safety grade cold shutdown criteria
- CRDR/human factors criteria for this auxiliary shutdown panel - no fabrication started
- Cable routing design essentially not started

STP

ALTERNATIVE SHUTDOWN - MONITORING



ALTERNATIVE SHUTDOWN - CONTROL



QUALIFIED DISPLAY PROCESSING SYSTEM (QDPS)

INTEGRATED DESIGN CONCEPT
QDPS

- Optimize instrumentation design to address overlapping regulatory guidance
  - TMI
  - Appendix R
  - Safety grade cold shutdown
- Optimize cable routing
- Minimize modifications to existing equipment

INPUTS TO QDPS

- Post-accident monitoring parameters
  - Inadequate core cooling instrumentation
  - RG 1.97 category 1 variables
- Safe shutdown monitoring and control parameters
- Complementary post-accident monitoring, control, and protection system parameters (to enhance display implementation)
- Advanced design modification parameters
  - SGWLCS
  - TAS

QDPS

STP

OUTPUTS FROM QDPS

QDPS

- Qualified PAMS display in Control Room
- Critical Safety Function parameter displays in Control Room
- Safe shutdown system displays in Control Room and ASP
- Isolated data links to SPDS
- Control (modulating) for safe shutdown valves
- Protection system (SGWLCS and TAS)

DIGITAL SYSTEM ADVANTAGES - QDPS

- Graphic displays support operating procedures
- Reduces control panel clutter - fewer panel indicators
- Relieves operator of cross channel checking burden
- Performs quality checking of inputs signal
- Simplifies implementation of signal distribution via data links
- On-line diagnostics and self-calibration

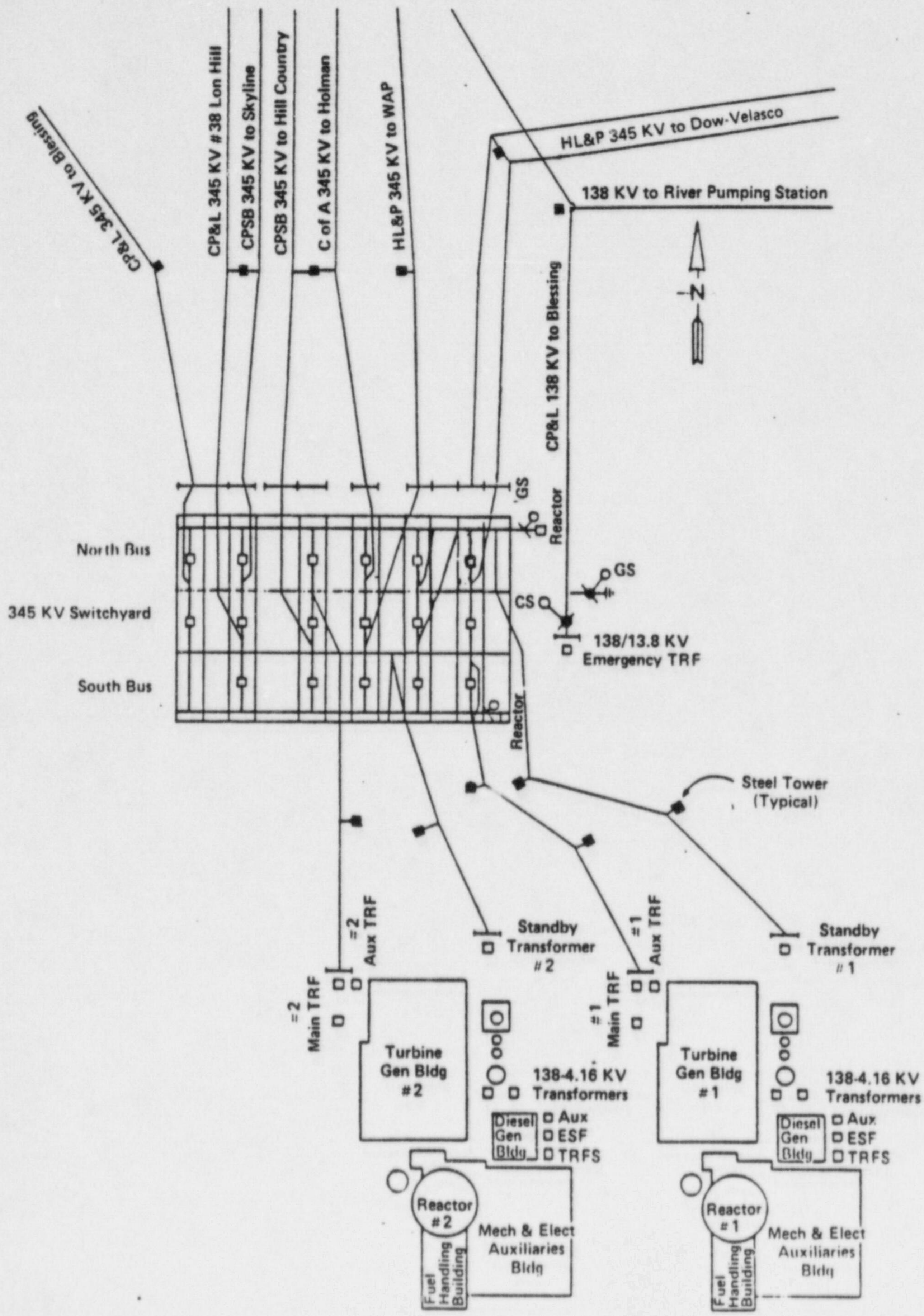
STP

VERIFICATION AND VALIDATION PROGRAM QDPS

- STP V&V plan based on established technology (RESAR 414, ANSI/IEEE-ANS-7-4.3.2)
- Performed by an independent team of verifiers
- Consists of exhaustive functional and/or structural testing of software
- Program will be completed prior to fuel load
- A minimum of four audits will be performed by the NRC staff

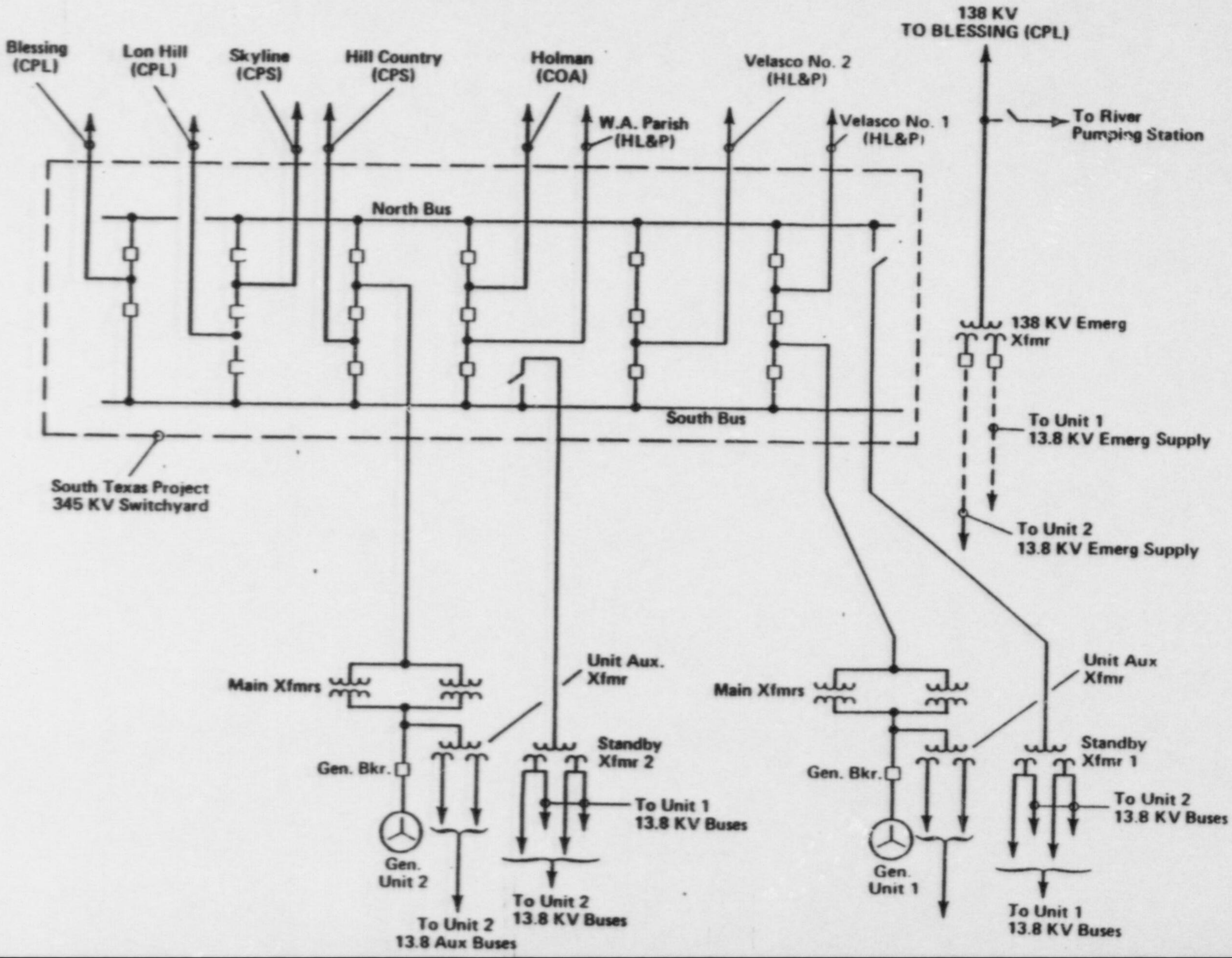
OFFSITE/ONSITE ELECTRIC POWER SYSTEM

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION



CP&L 345 KV to Blessing
CP&L 345 KV # 38 Lon Hill
CPSB 345 KV to Skyline
CPSB 345 KV to Hill Country
C of A 345 KV to Holman
HL&P 345 KV to WAP
HL&P 345 KV to Dow-Velasco
138 KV to River Pumping Station
CP&L 138 KV to Blessing
N
GS
Reactor
North Bus
345 KV Switchyard
South Bus
GS
CS
138/13.8 KV
Emergency TRF
Reactor
Steel Tower
(Typical)
#2
Aux TRF
Standby
Transformer
#2
#1
Aux TRF
Standby
Transformer
#1
#2
Main TRF
Turbine
Gen Bldg
#2
138-4.16 KV
Transformers
Diesel
Gen
Bldg
Aux
ESF
TRFS
#1
Main TRF
Turbine
Gen Bldg
#1
138-4.16 KV
Transformers
Diesel
Gen
Bldg
Aux
ESF
TRFS
Reactor
#2
Mech & Elect
Auxiliaries
Bldg
Fuel
Handling
Building
Reactor
#1
Mech & Elect
Auxiliaries
Bldg
Fuel
Handling
Building

STP 345 KV SWITCHYARD
AFTER TWO UNIT OPERATION



SITE DISTRIBUTION

Summary:

- Nine transmission circuits / four grids
- Generator breaker
- Transformer capacity
- Independent ESF feed

All power system alignments from control room

STATION BLACKOUT

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

DRAFT REGULATORY GUIDE

- STP would fall into the 4 hour category to withstand a station blackout

- STP can withstand at least a 4 hour station blackout

STATION BLACKOUT

Study results:

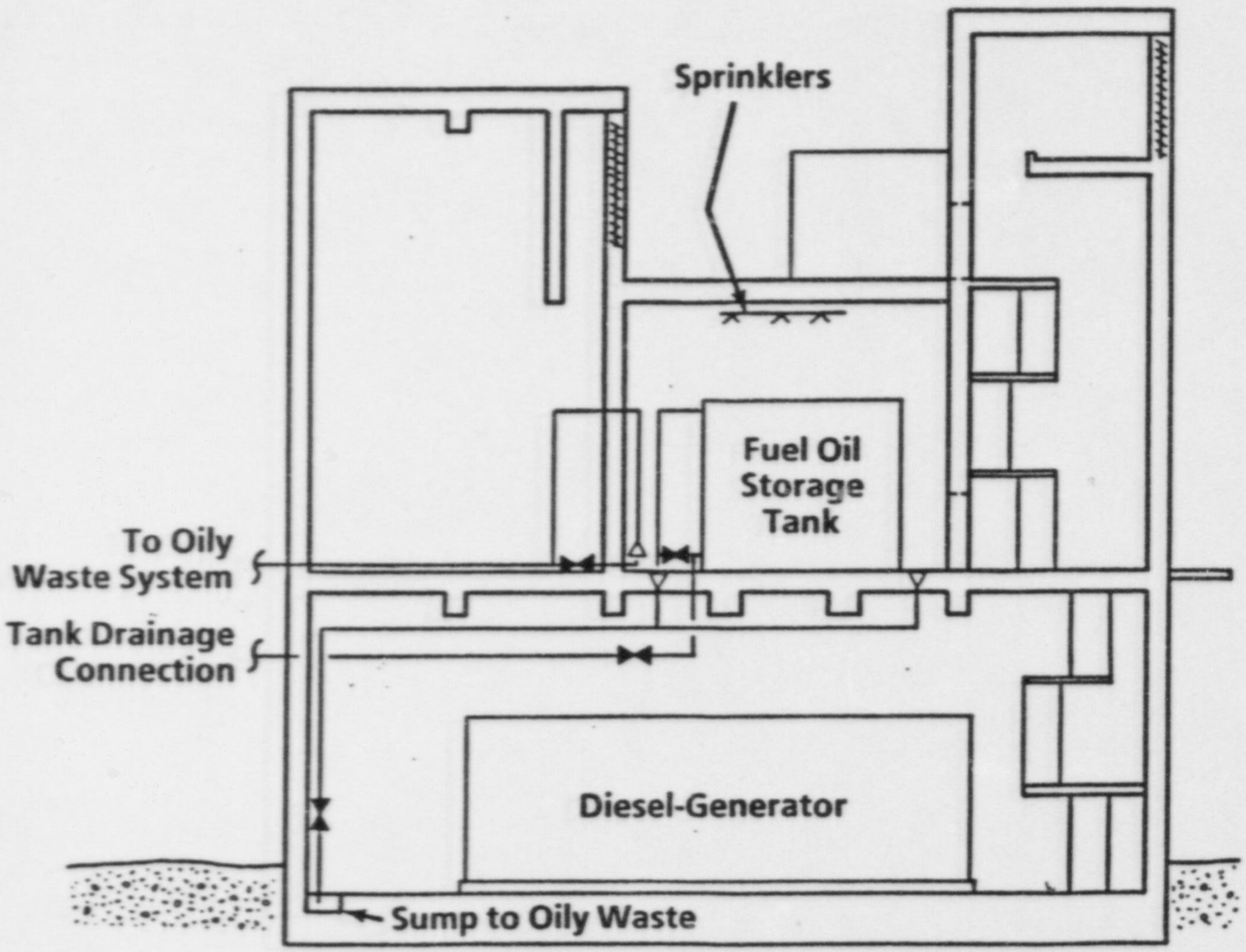
- 22 hours auxiliary feedwater available
- 8 hours class 1E battery available
- RCP seal leakage 25 gpm maximum
- RCP seal cooling available from BOP diesel
- Additional power possible
  - 5 BOP batteries
  - 5 BOP diesels onsite

STP

DIESEL GENERATOR
FUEL OIL STORAGE TANK

SOUTH TEXAS
PROJECT
ELECTRIC GENERATING STATION

DIESEL FUEL OIL STORAGE TANK



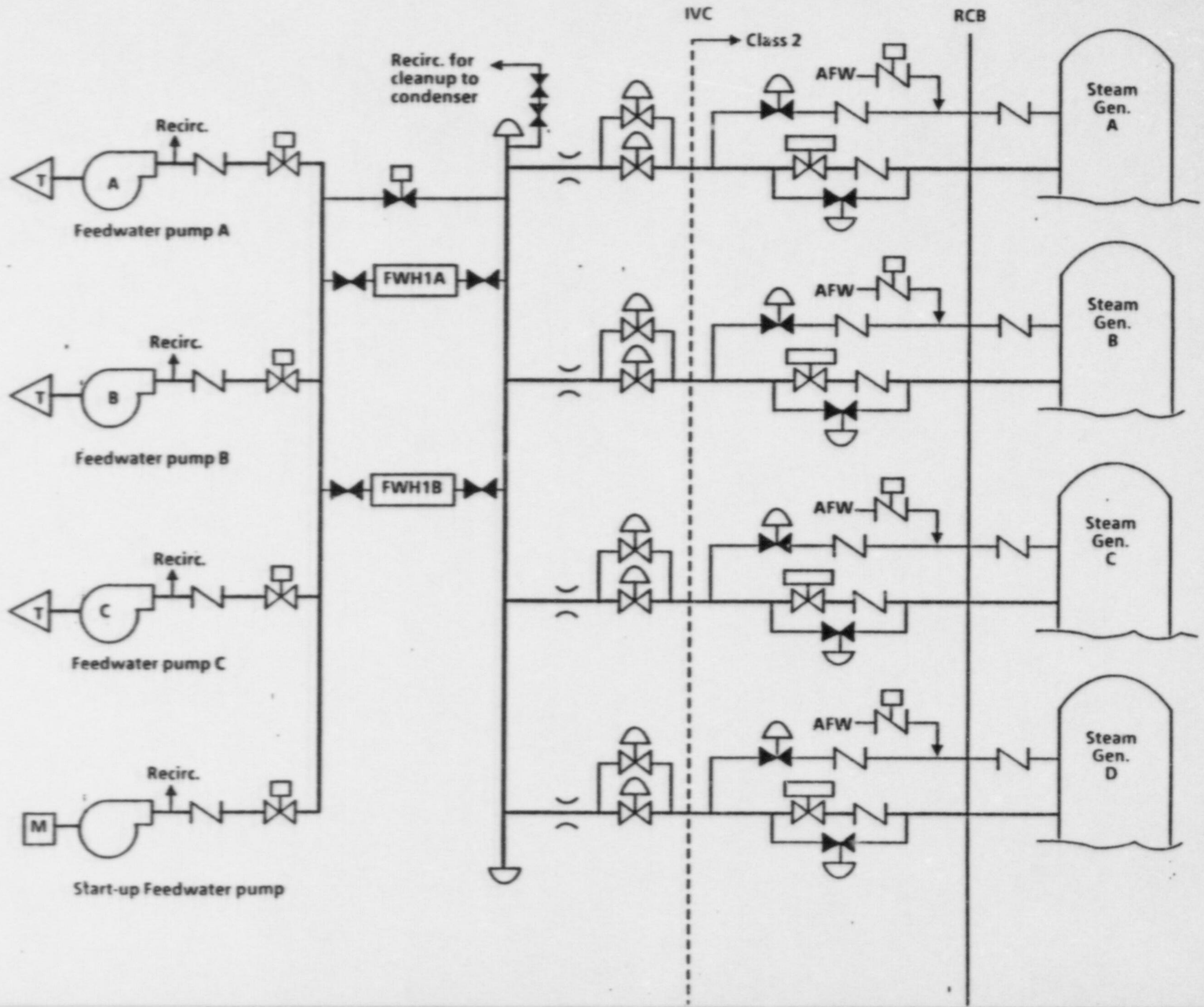
- Each train of diesel fuel oil storage tanks for the three trains are located in separate rooms enclosed by 3 hour rated fire barriers
- Fire protection is provided by automatic foam-water sprinklers. Continuous ventilation will be provided under normal conditions with fans powered by ESF electrical buses, with damper closure under fire conditions
- Room drains will effect removal of foam-water and fuel leakage to the oily waste system. Tank drainage is provided for by a separate connection

TANK ROOM

- Continuously ventilated from ceiling to floor to remove potential fumes
- Vent fans on 1E bus
- Foam-water suppression
- Water tight, locked doors
- Tank level monitored

MAIN FEEDWATER SYSTEM

MAIN FEEDWATER SYSTEM



EUROPEAN EXPERIENCE

EUROPEAN EXPERIENCE

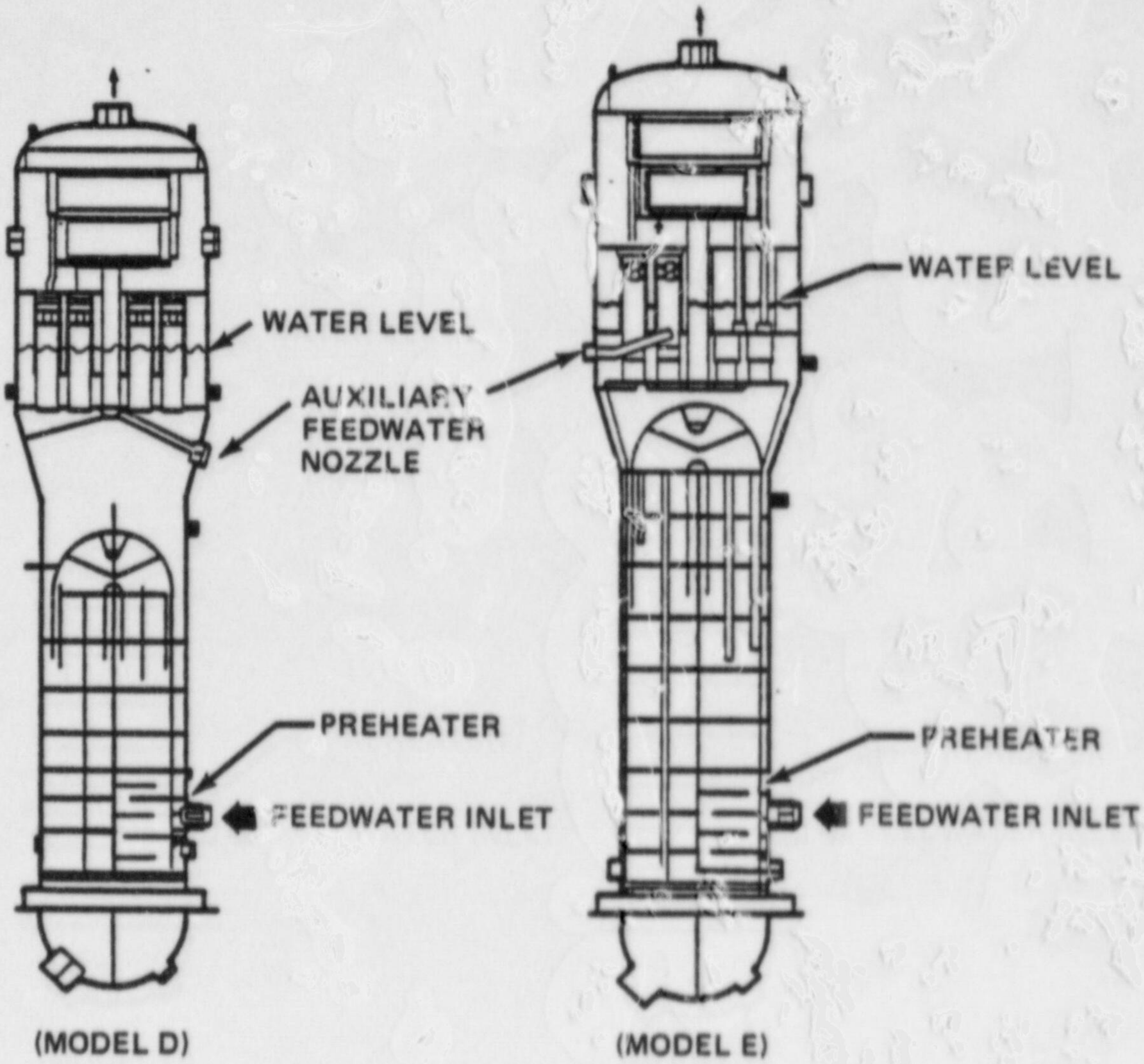
Equipment	European Plants
• 14-foot Fuel Design	D, T, P, S
• Model 100 RC Pump	D, T, K
• Model E Steam Generator	D, T
• Rapid Refueling Head Package	D, T
Design Aspect	
• Three Train Systems	D, T
• Qualified RHRS Inside Containment	D, T, P, S
• Backup Power for CVCS PD Pump	P, S
• Steam Generator Sludge Lance Ports at Preheater	D, T

Legend: D - Doel 4/Belgium
T - Tihange 3/Belgium
P - Paluel/France 1, 2, 3, & 4
S - St. Anton/France 1 & 2
K - Krsko/Yugoslavia

GOOD OPERATIONAL EXPERIENCE DURING DOEL 4 STARTUP

- Rapid refueling performed
- Moisture carryover testing shows low carryover
- Natural recirculation test completed - no problems
- Physics tests - results as predicted

STEAM GENERATOR MODIFICATIONS



WESTINGHOUSE STEAM GENERATOR COMPARISON

STP STEAM GENERATOR MODIFICATIONS

- Tube expansion to 160 tubes in preheater
  - Expansion complete, NDE complete
  - Acceptable startup and power run at DOEL 4
  - In tube accelerometers at DOEL 4 show low vibration levels that are well within guidelines
  - No long-term wear problems predicted
- Moisture Separator Modifications Completed
  - DOEL 4 carryover tests good
- Sludge lancing ports added to preheater area
  - Similar design as DOEL 4
  - Similar locations as DOEL 4
- Tube sheet expansion area has been stress relieved using Rotopeening Process
  - This is same process used at DOEL 4
  - Primary stress corrosion will have reduced potential

ADDITIONAL DESIGN FEATURES

ADDITIONAL STP DESIGN FEATURES

- Full flow deaerator
- Separate startup FW pump
- Full flow polishers/precation and mixed bed
- Feedwater and auxiliary feedwater anti-waterhammer design features
- Elimination of copper material in feedwater/condensate train including Titanium condenser tubes
- Rapid refueling
- 14-foot fuel design

NUCLEAR ASSURANCE

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

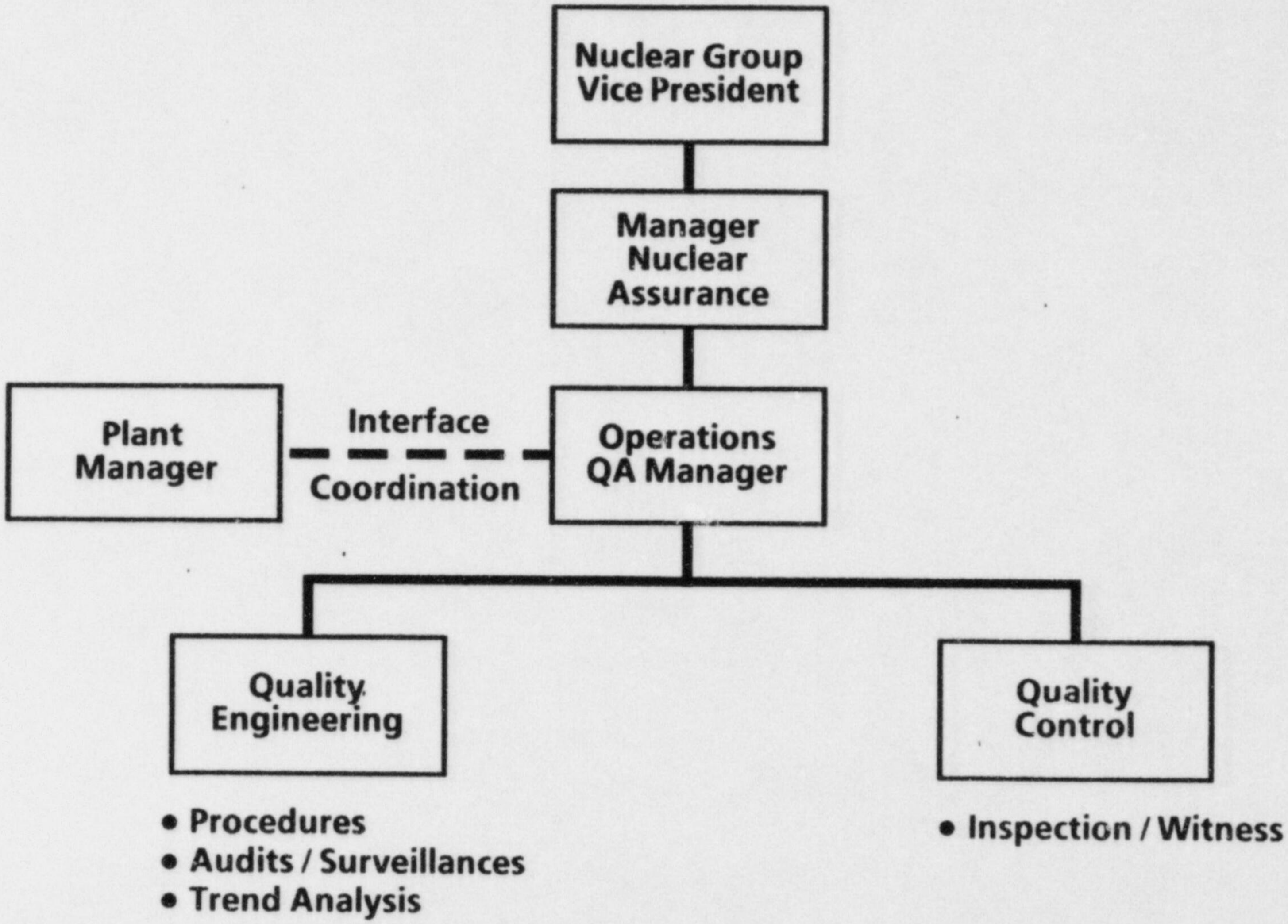
APPENDIX VIII
NUCLEAR ASSURANCE

NUCLEAR ASSURANCE

- Operations QA
- Independent Safety Engineering Group (ISEG)
- Safeteam
- Fitness for duty

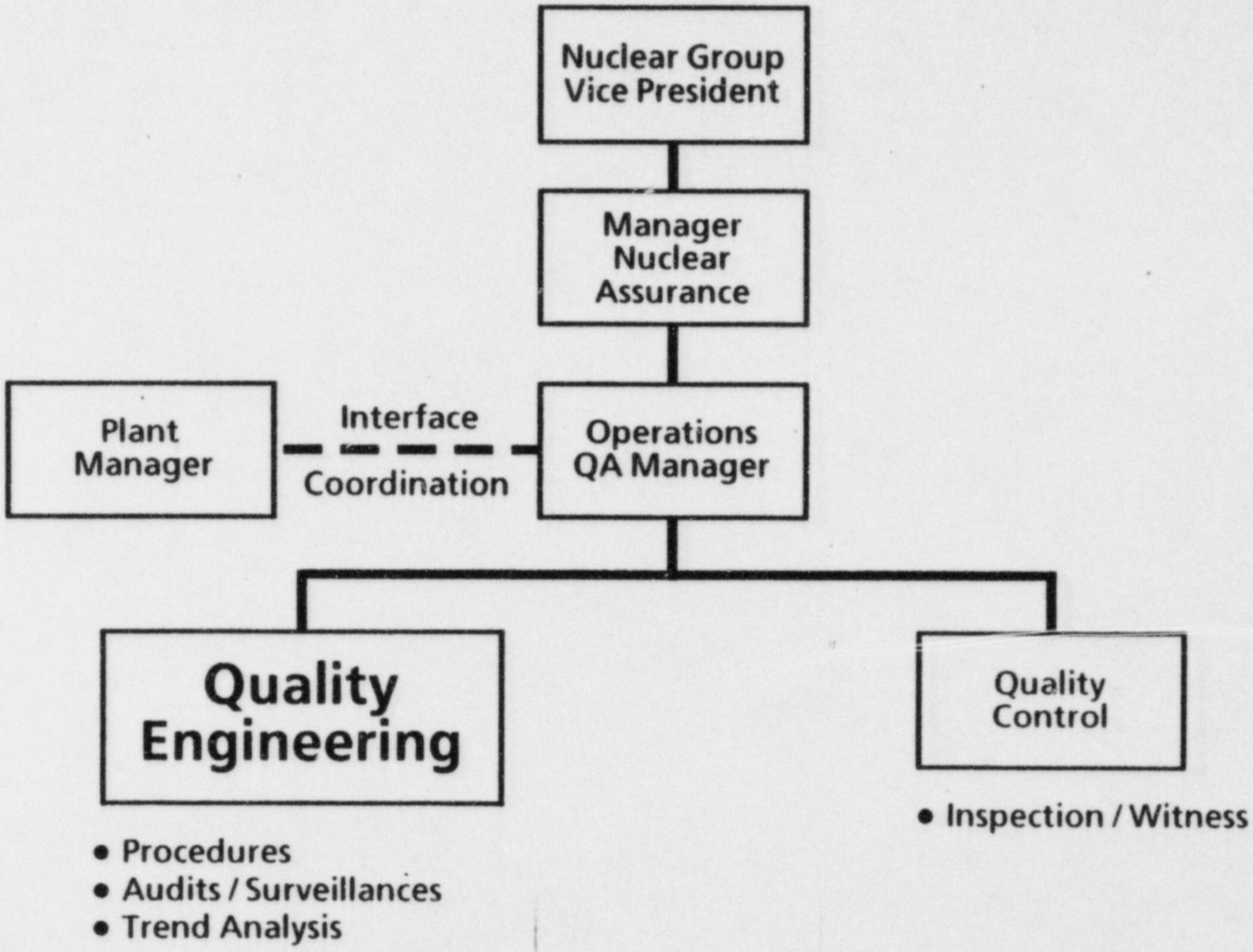
STP

OPERATIONS QUALITY ASSURANCE



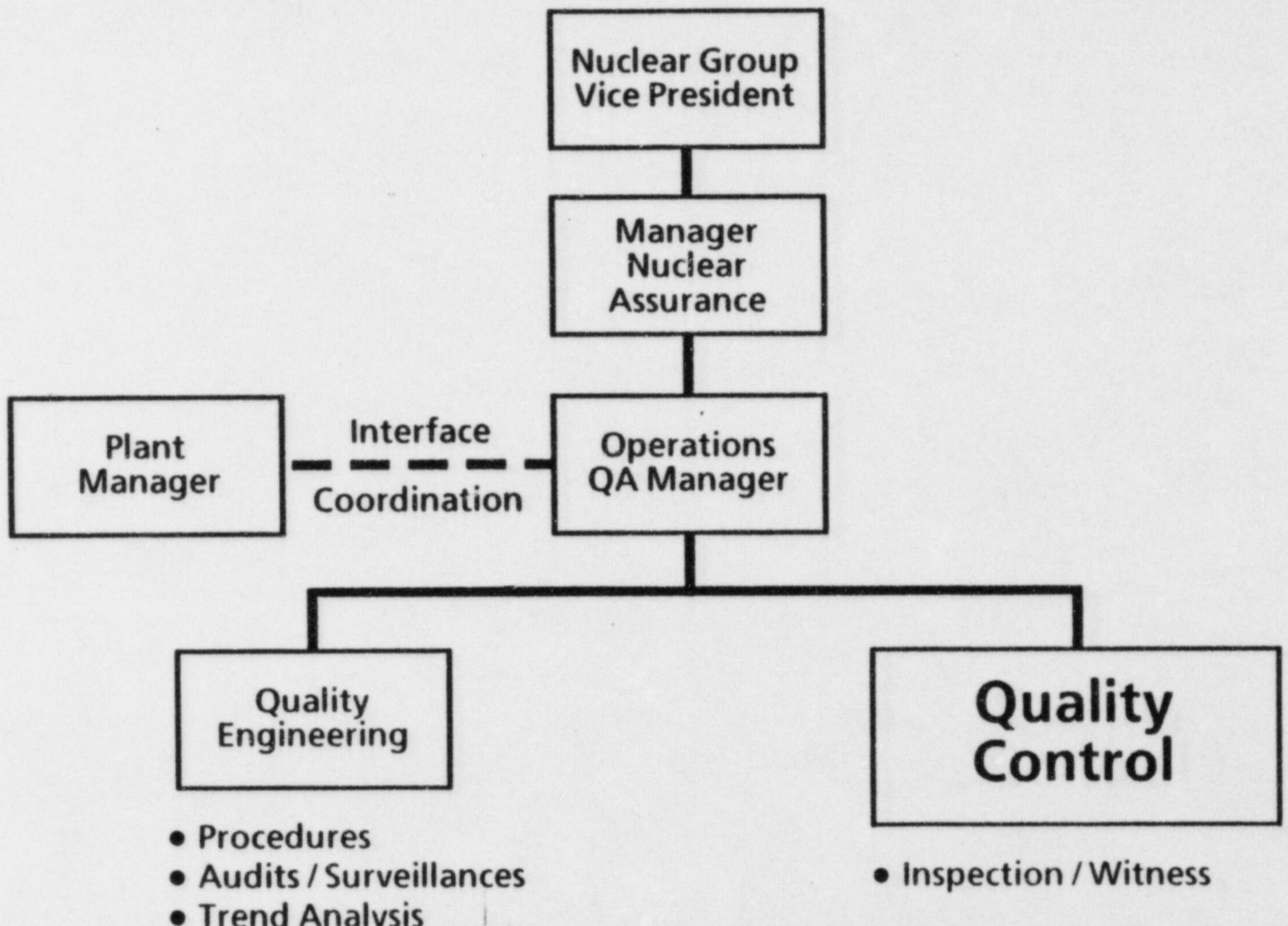
STP

OPERATIONS QUALITY ASSURANCE



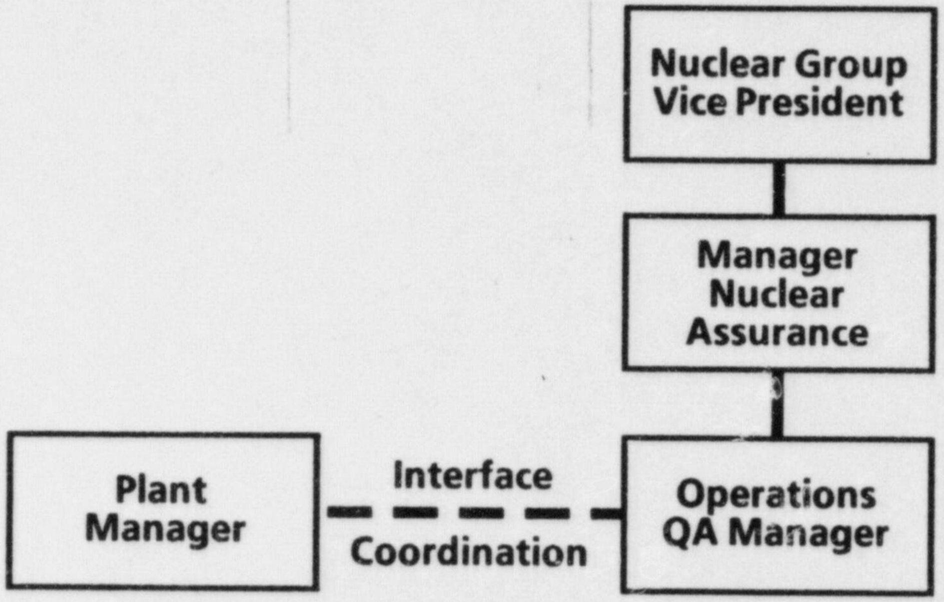
STP

OPERATIONS QUALITY ASSURANCE



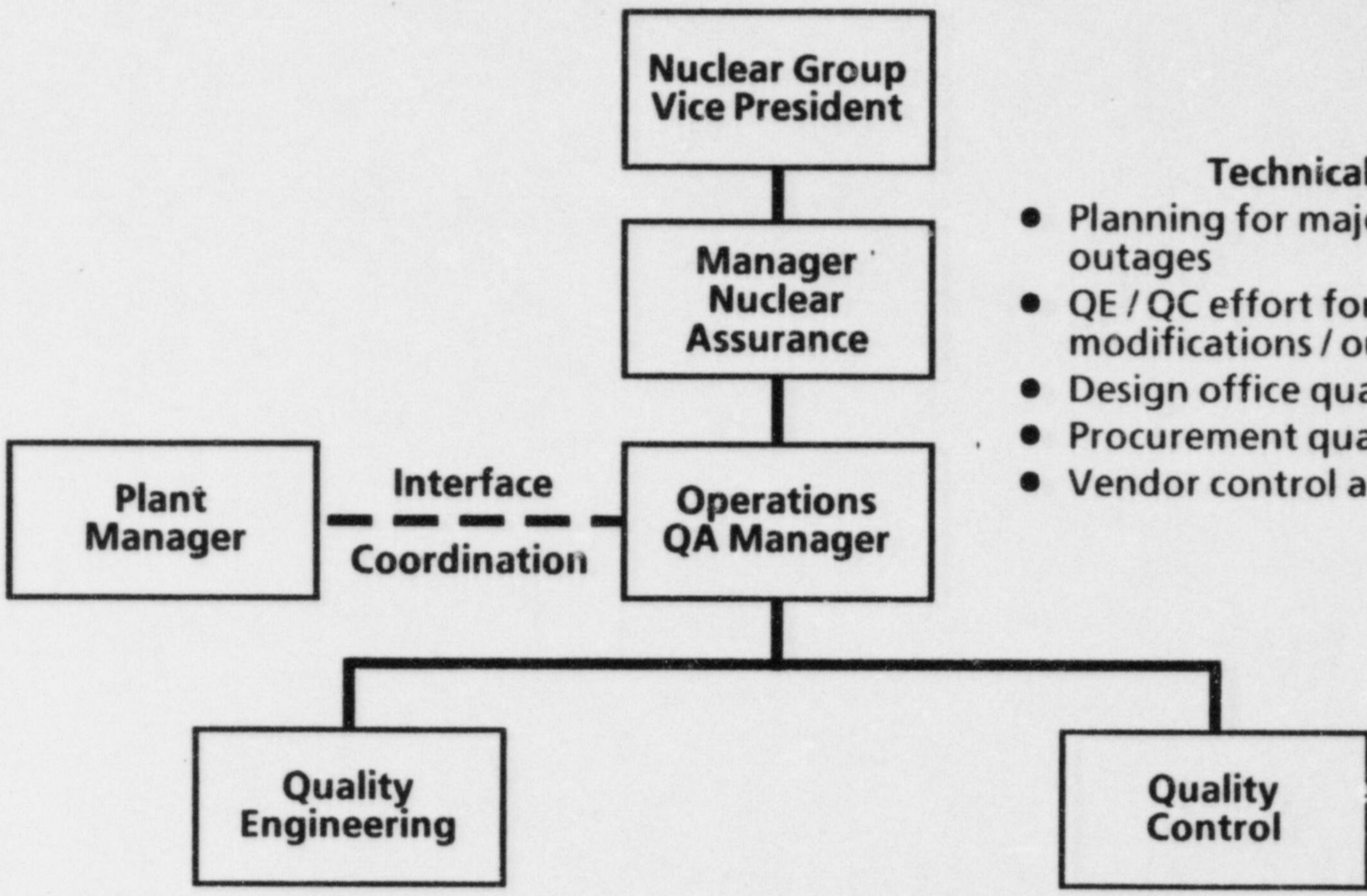
STP

OPERATIONS QUALITY ASSURANCE



Interface / Coordination: Communication channel to assure consistent interpretation and implementation of HL&P quality philosophy

OPERATIONS QUALITY ASSURANCE



Technical Services

- Planning for major modifications / outages
- QE / QC effort for major modifications / outages
- Design office quality assurance
- Procurement quality assurance
- Vendor control activities

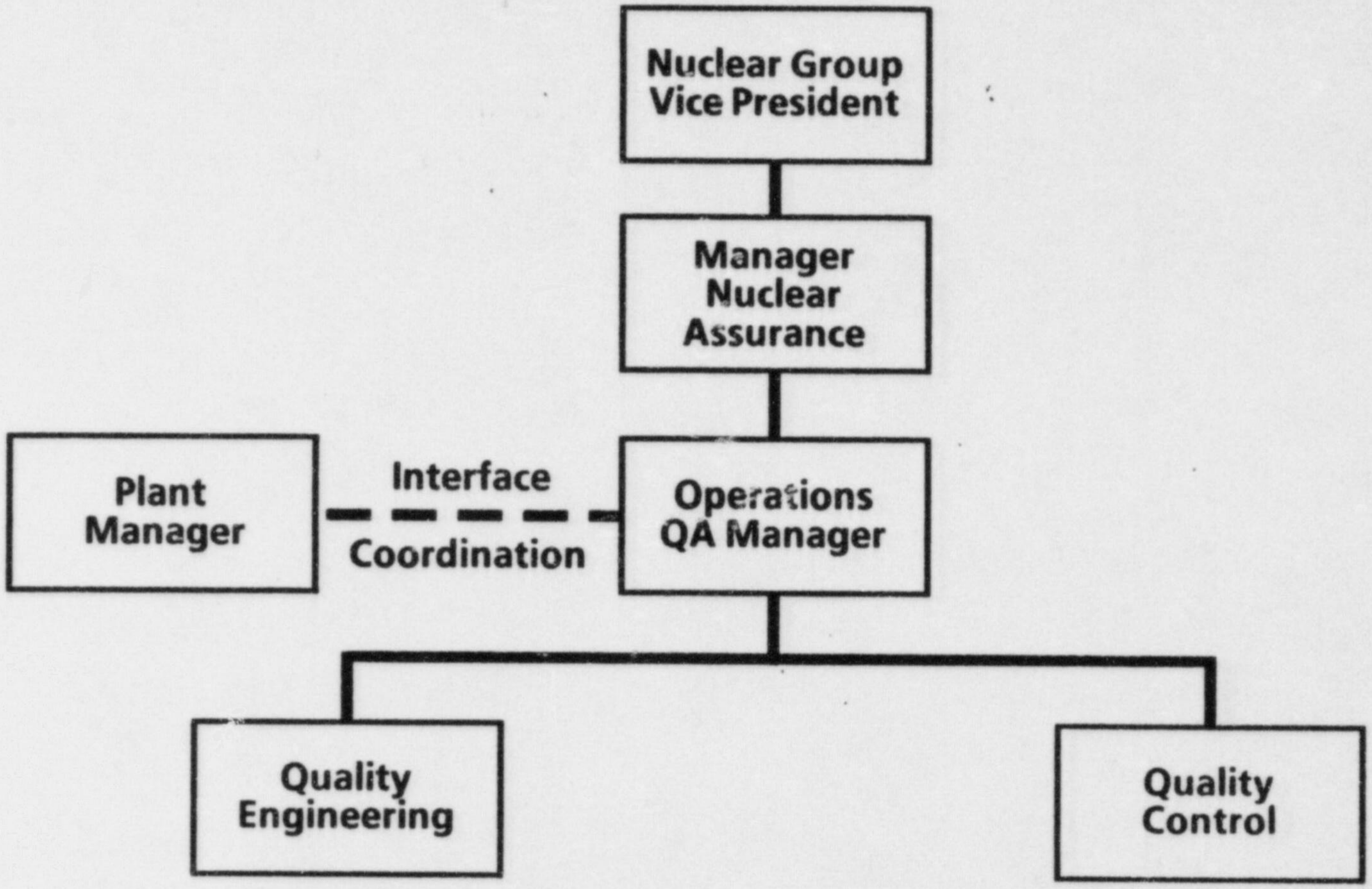
Quality Engineering

- **Procedures**
- **Audits / Surveillances**
- **Trend Analysis**

Quality Control

- **Inspection / Witness**

OPERATIONS QUALITY ASSURANCE



- Procedures
- Audits / Surveillances
- Trend Analysis

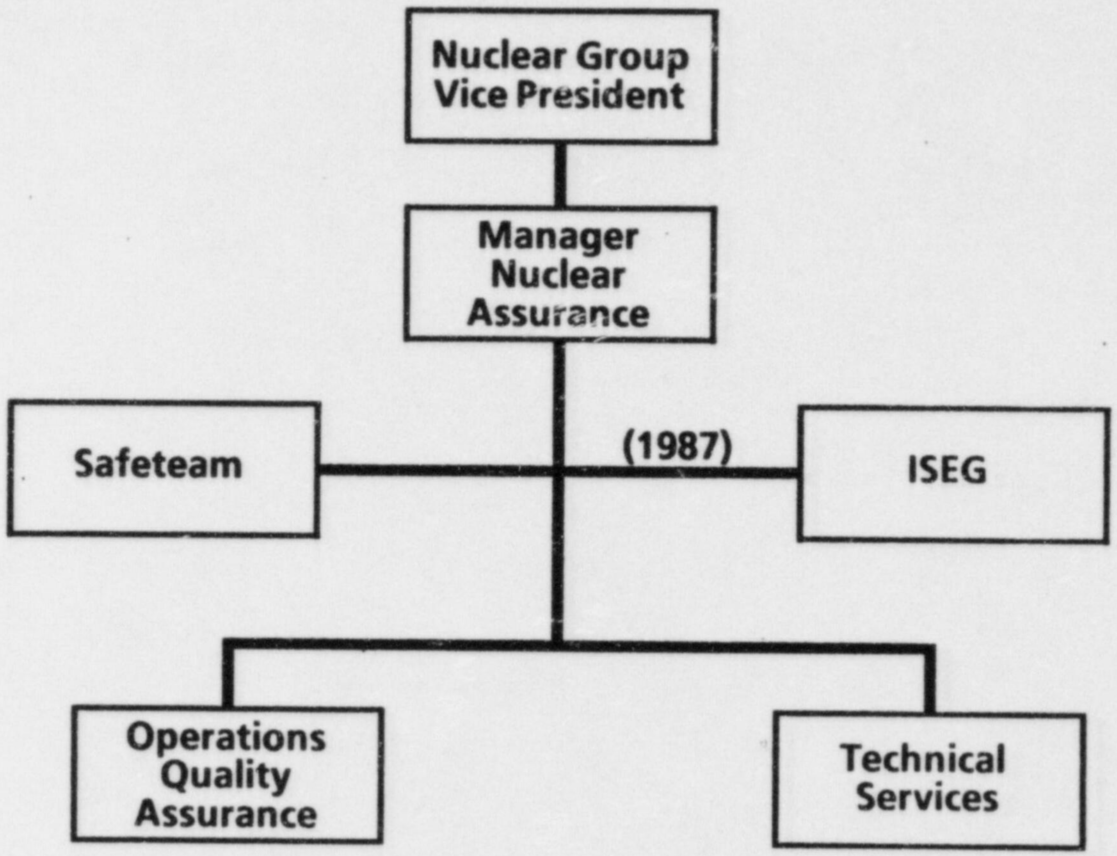
- Inspection / Witness

STP

OPERATIONS QUALITY ASSURANCE

Operations QA Manager	=	23 years nuclear QA experience 10 years operations experience professional engineer
32 People	=	101 years operations experience 16 of 32 have been reactor operators
6 Supervisors	=	32 years operations experience 4 of 6 are degreed 4 of 6 have been military plant operators

STP NUCLEAR ASSURANCE



INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

- To be created in 1987
- Staffed by 5 senior level, experienced engineers
- Responsibilities
  - Provide continuing systematic and independent assessment of plant activities, including maintenance and modifications
  - Perform observations of plant operations and maintenance activities for additional verification of proper conduct

STP

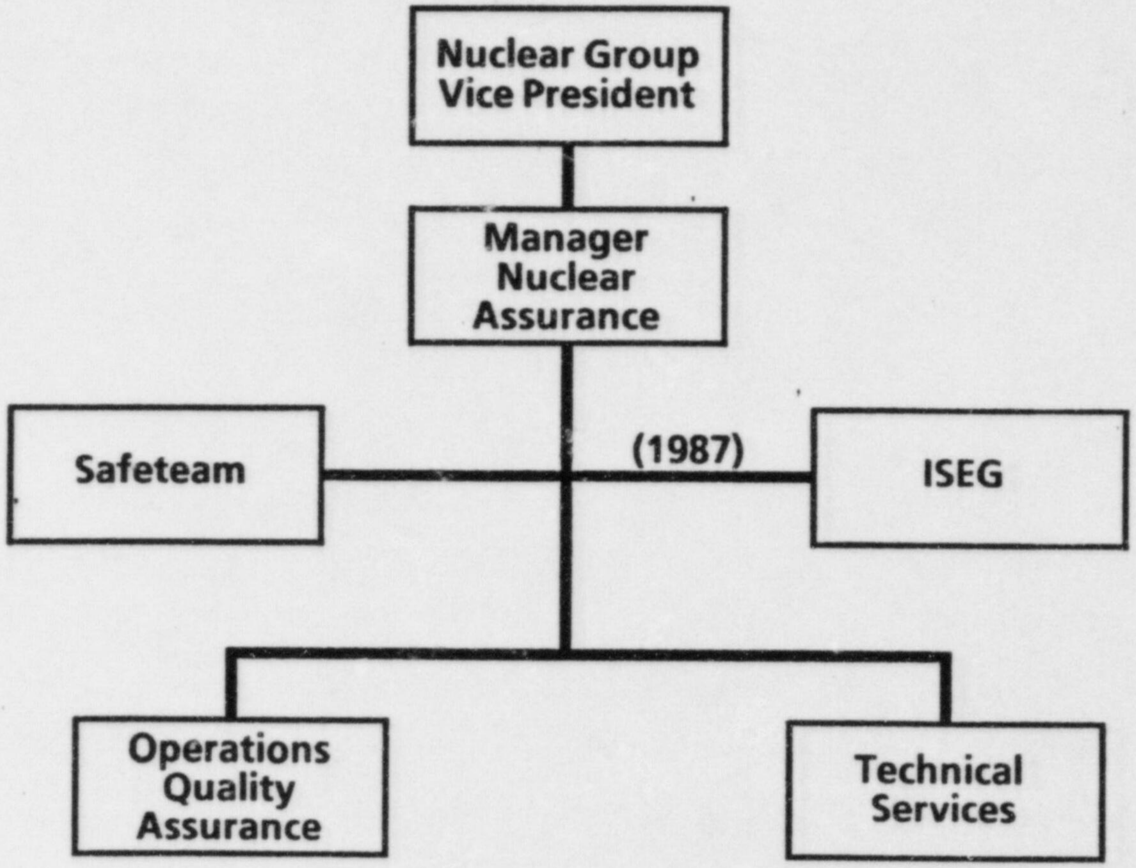
INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

- Perform review/analysis of selected modifications

- Perform review/analysis of selected problems from other plants

- Perform root cause analysis of selected problems at STP

STP NUCLEAR ASSURANCE



SAFETEAM PROGRAM

Definition:

An administrative program for the purpose of providing a forum for STP employees to identify any concern they may have in the areas of nuclear safety or quality

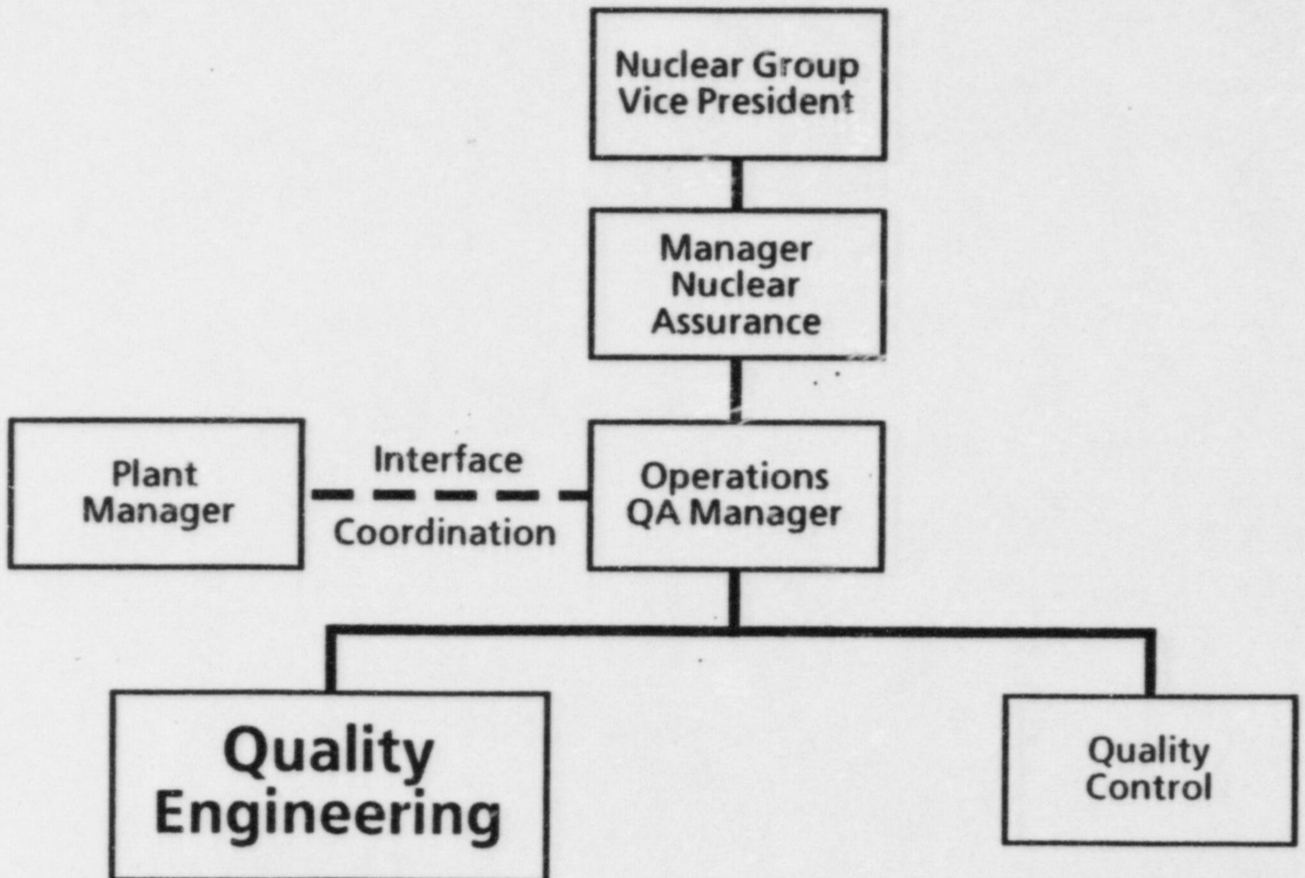
SAFETEAM PROGRAM

Objective:

To encourage employees to come forward with their concerns now, so that we may investigate the concerns, correct any deficiencies, and report back to them

Confidential - Anonymous

OPERATIONS QUALITY ASSURANCE



- Procedures
- Audits / Surveillances
- Trend Analysis

- Inspection / Witness

STP

ENGINEERING ASSURANCE PROGRAM

Purpose: Provide on-going real time review of design activities

- To influence future design activities
- To confirm the adequacy of the design and design process
- Established 1982
- Performed first review in 1983
- Substantial participation by Stone and Webster Engineering Co. personnel
- 35,000 manhours expended

ENGINEERING ASSURANCE

Complete

- Soil-structure interaction analysis and seismic design
- Design control and design verification
- ASME pipe stress analysis
- Containment analysis
- Environmental qualification of equipment
- ASME III pipe support design
- Control room HVAC system
- Component cooling water system
- Offsite AC power and medium voltage AC & DC battery power supply systems
- High-energy line break analysis
- Class I ASME pipe stress analysis including handling of Westinghouse loads by Bechtel

1986

- Separation and fire protection criteria
- System walkdowns to assess system interactions, seismic II/I, etc.

ENGINEERING ASSURANCE

Conclusion:

An additional step taken by HL&P to increase our level of confidence in the adequacy of the design and design process for the STP

Implemented during Construction Phase; carried forward into Operations Phase

STP NUCLEAR ASSURANCE

Summary

- Operations QA staff in place
- Technical Services provides support to implement a comprehensive QA program
- Safeteam in place
- ISEG plans formulated and scheduled
- Plan to integrate the Engineering Assurance function into Quality Engineering underway

FITNESS FOR DUTY

- Based on Edison Electric Institute (EEI) guide
- Ten key program elements

FITNESS FOR DUTY

- Written policy
- Top management support
- Policy communication
- Behavioral observation training
- Implementation training
- Union briefing
- Contractor notification
- Law enforcement liaison
- Chemical testing
- Employee assistance programs

STP

FITNESS FOR DUTY

- Written policy
- Top management support
- Policy communication
- **Behavioral observation training**
- Implementation training
- Union briefing
- Contractor notification
- Law enforcement liaison
- Chemical testing
- Employee assistance programs

FITNESS FOR DUTY

- Written policy
- Top management support
- Policy communication
- Behavioral observation training
- Implementation training
- Union briefing
- Contractor notification
- Law enforcement liaison
- **Chemical testing**
- Employee assistance programs

STP

FITNESS FOR DUTY TOTALS
MAY, 1986

Total Number of Personnel Sampled	5,343
Total Number of Personnel Failing	130

STP

FITNESS FOR DUTY SUMMARY

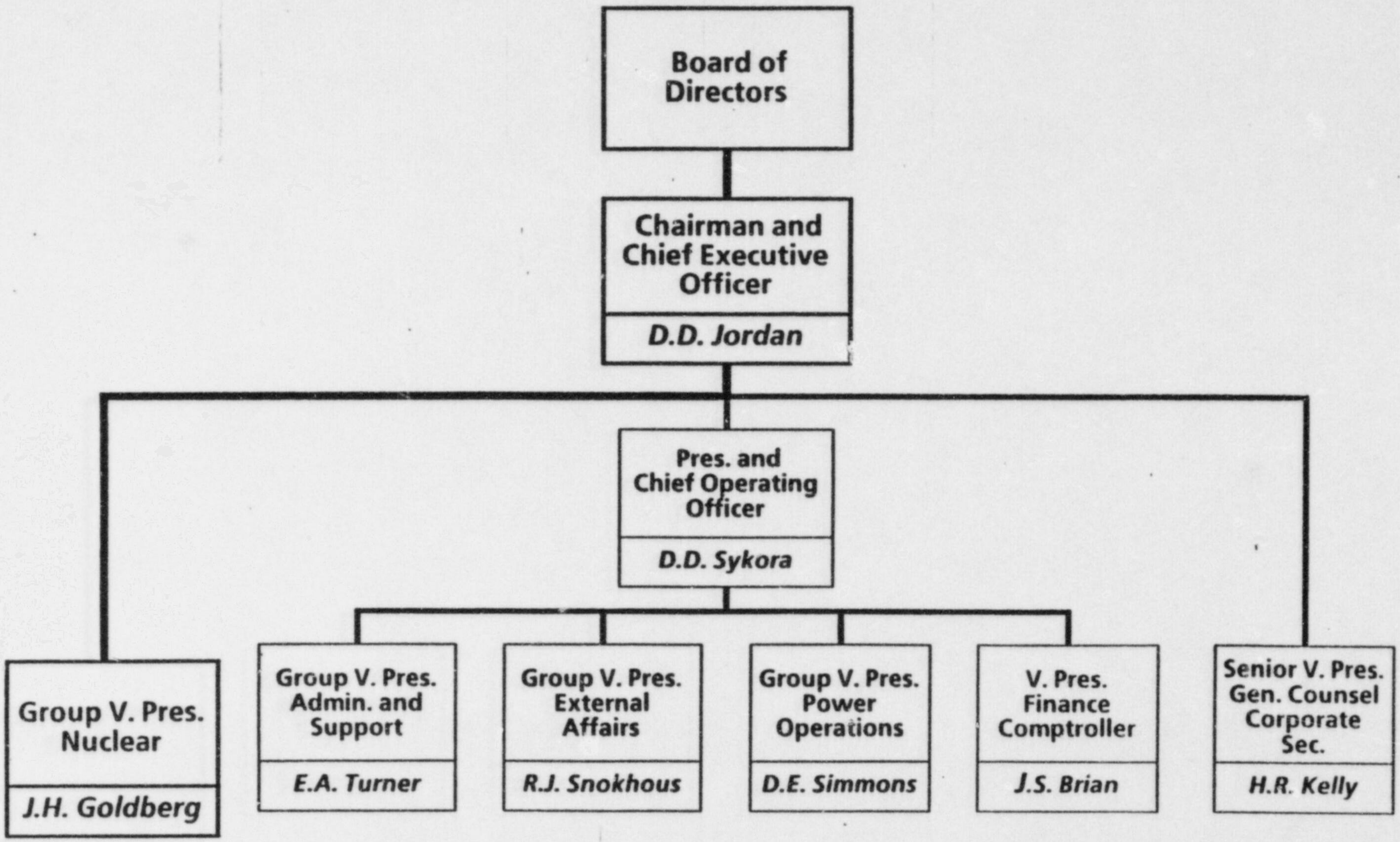
- Based on EEI guide
- Implemented late 1985
- Chemical testing began 1-1-86

OPERATIONAL PHASE ORGANIZATION

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

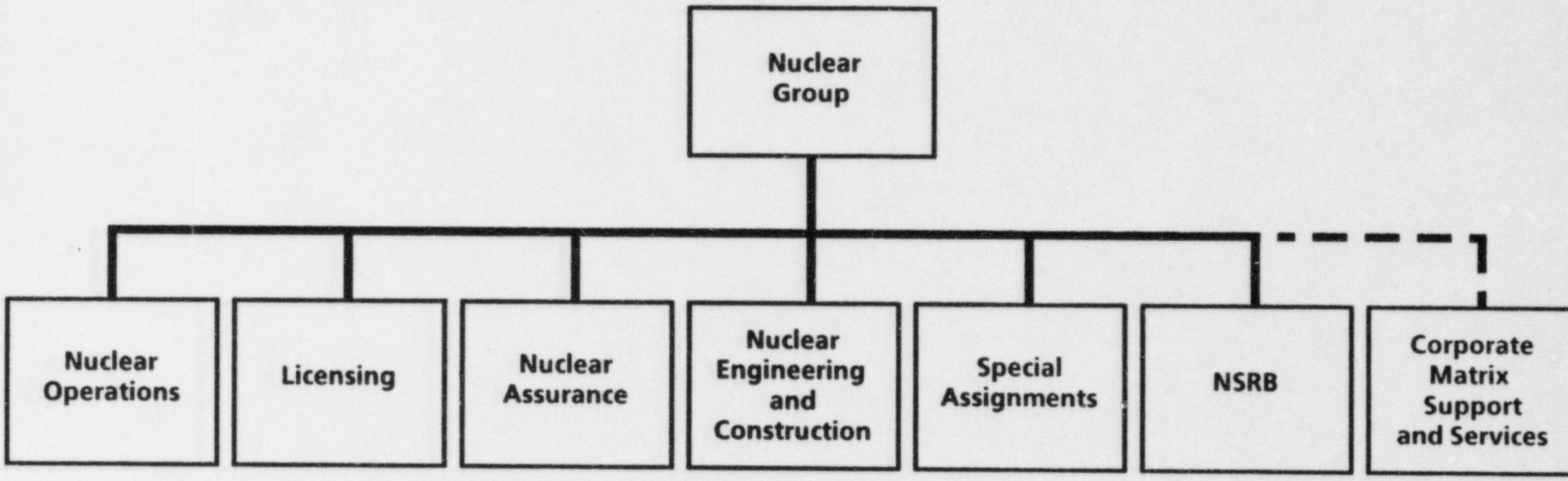
APPENDIX IX
OPERATIONAL PHASE ORGANIZATION

HOUSTON LIGHTING & POWER COMPANY
Senior Executive Organization



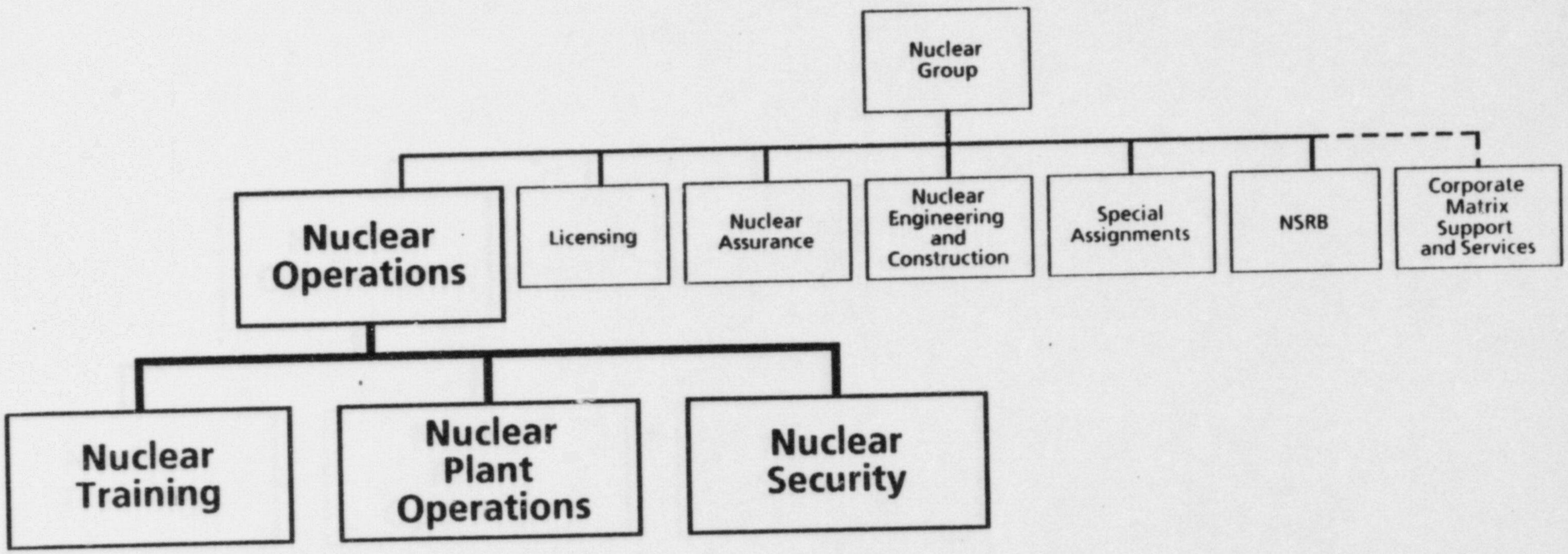
STP

NUCLEAR GROUP ORGANIZATION



STP

NUCLEAR GROUP ORGANIZATION



STP

NUCLEAR TRAINING DEPARTMENT

Vice President Nuclear Operations

Manager Nuclear Training

Operations Training Division

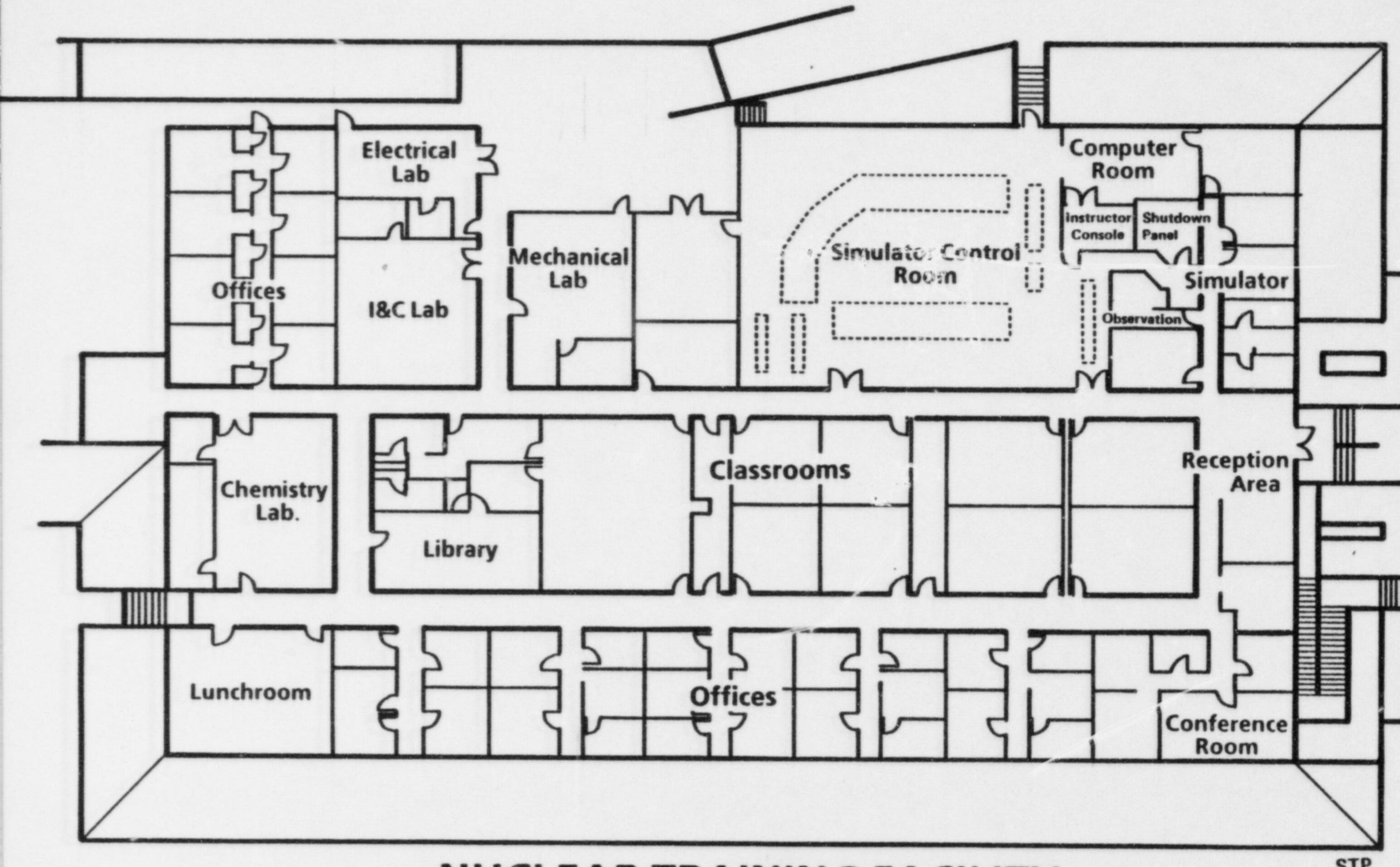
- Licensed Operations Training
- Non-licensed Operator Training
- STA
- Simulator Training
- Requalification Training

Staff Training Division

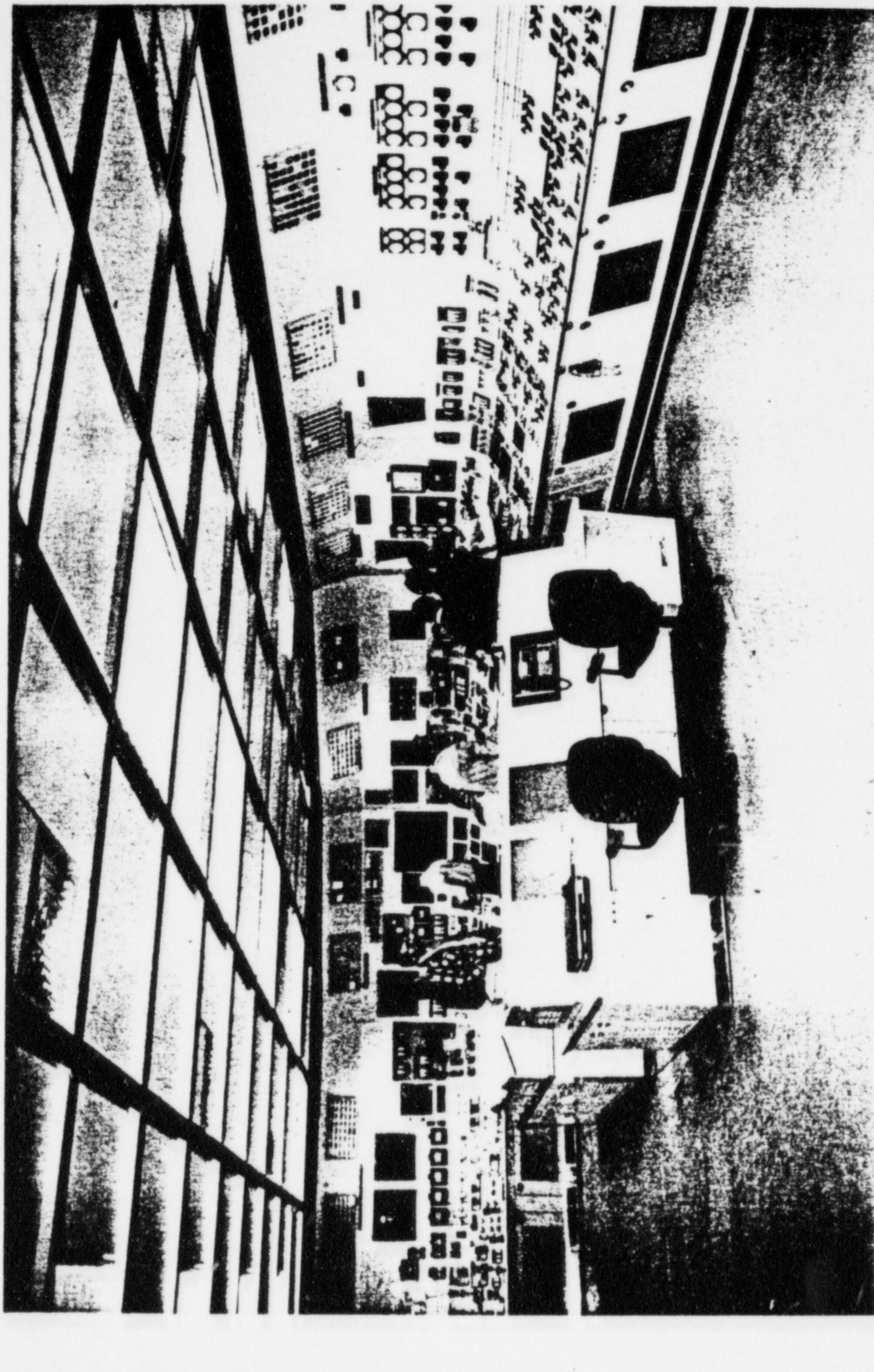
- General Training
- Technician Training
- Maintenance Training
- Engineering Training

Program Design and Evaluation Section

- Program Academic Soundness
- INPO Accreditation
- Instructor Certification
- HRD Coordinator



NUCLEAR TRAINING FACILITY



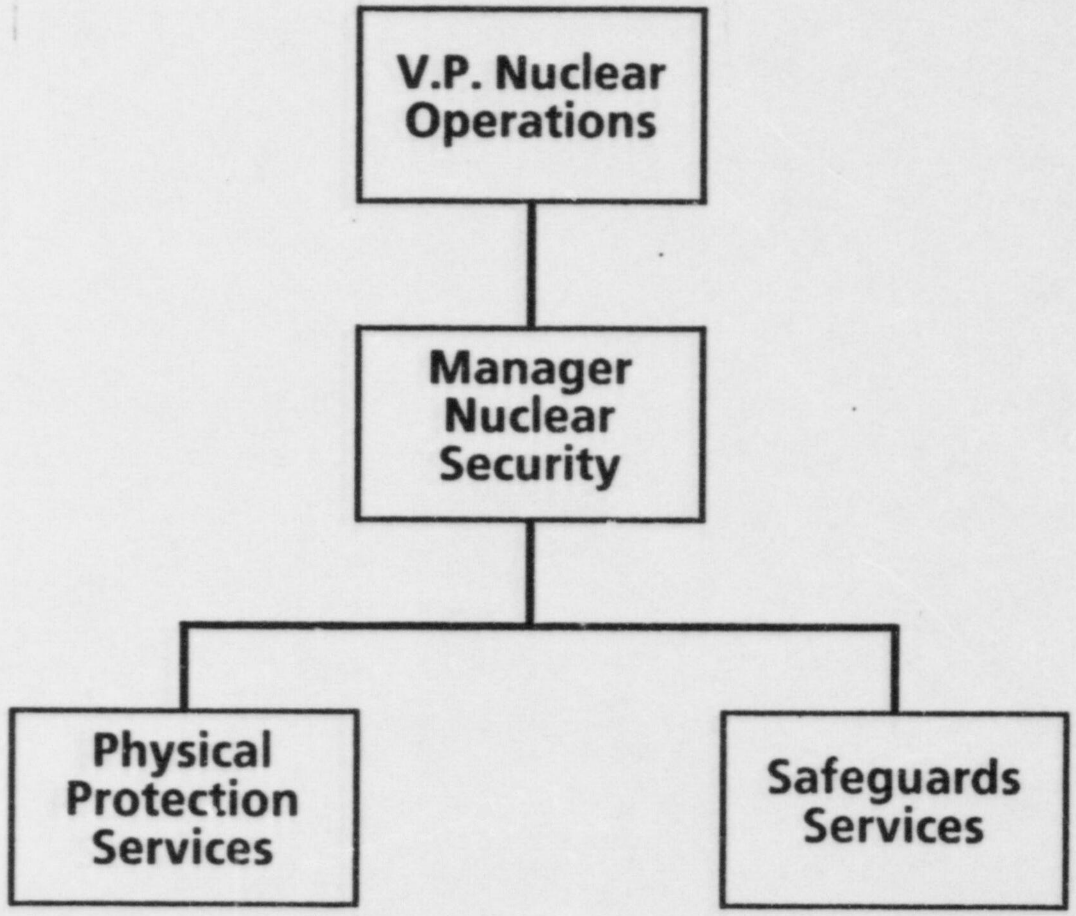
HL&P NUCLEAR TRAINING PROGRAMS

1. Non-licensed Operator
   - Plant Operator
   - Chemical Operator
2. Reactor Operator (RO)
3. Senior Reactor Operator (SRO/SS)
4. Shift Technical Advisor (STA)
5. I & C Technician
6. Electrical Maintenance
7. Mechanical Maintenance
8. Chemical Technician
9. Radiation Protection Technician
10. Technical Staff and Managers
11. Requalification*
12. Supervisory Skills*
13. Simulator Training*
14. General Employee Training
15. Instructor Certification
16. Fire Brigade Leader
17. Performance Technician
18. Emergency Plan Training
19. QA Training
20. Engineering Support Training
21. RMS Training
22. PWR Familiarization

* Will be accredited with licensed programs

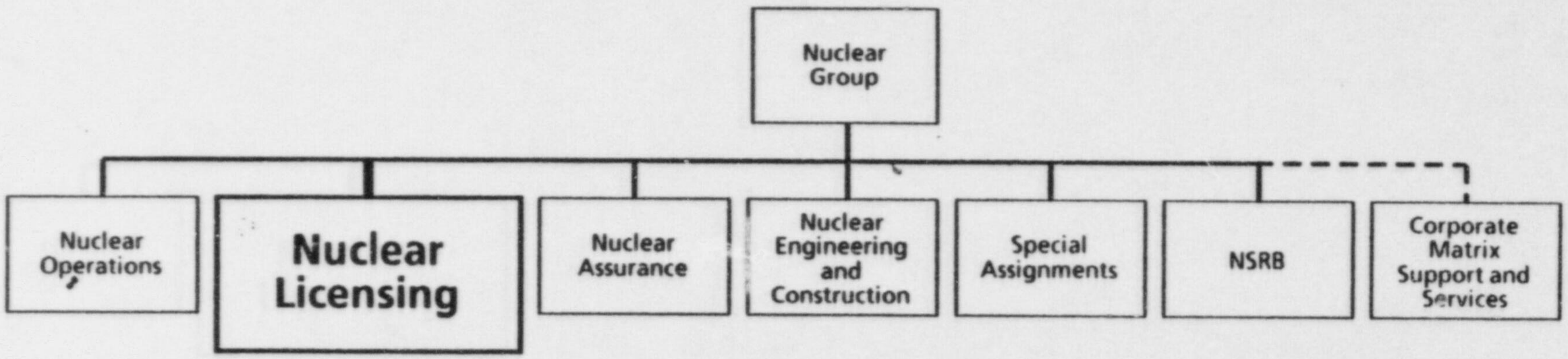
STP

NUCLEAR SECURITY DEPARTMENT



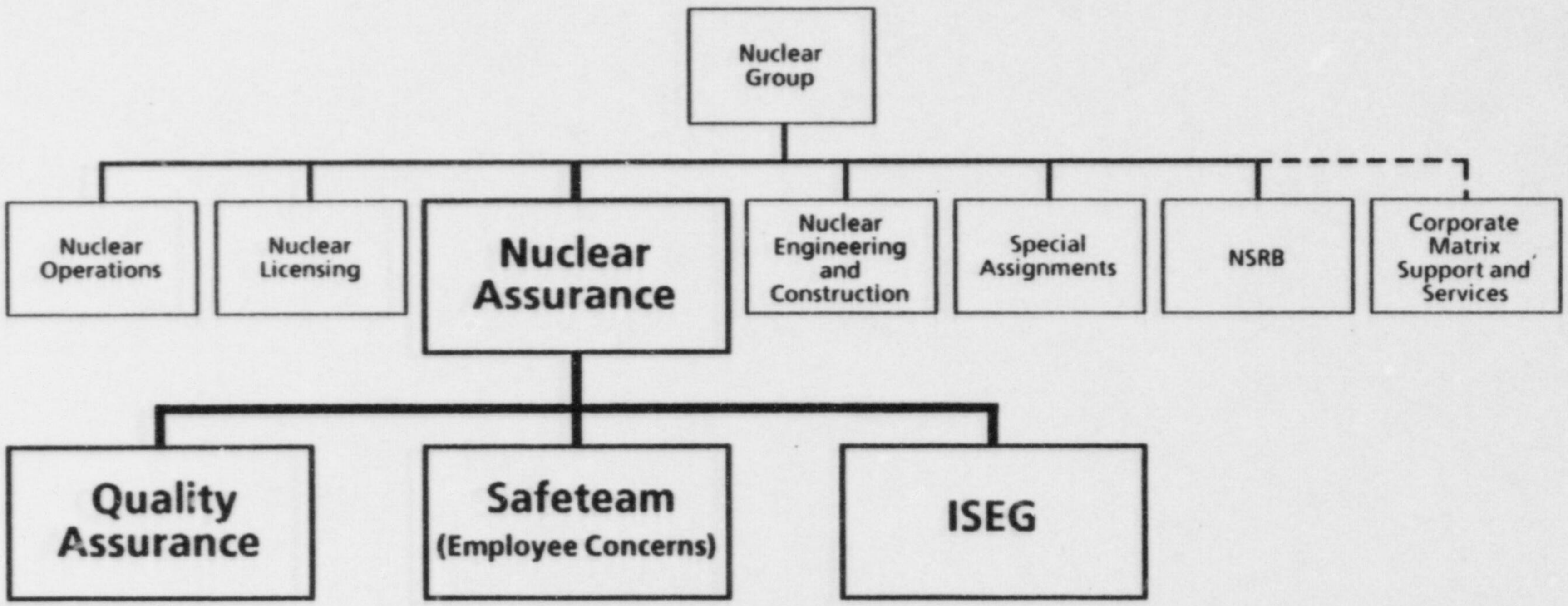
STP

NUCLEAR GROUP ORGANIZATION



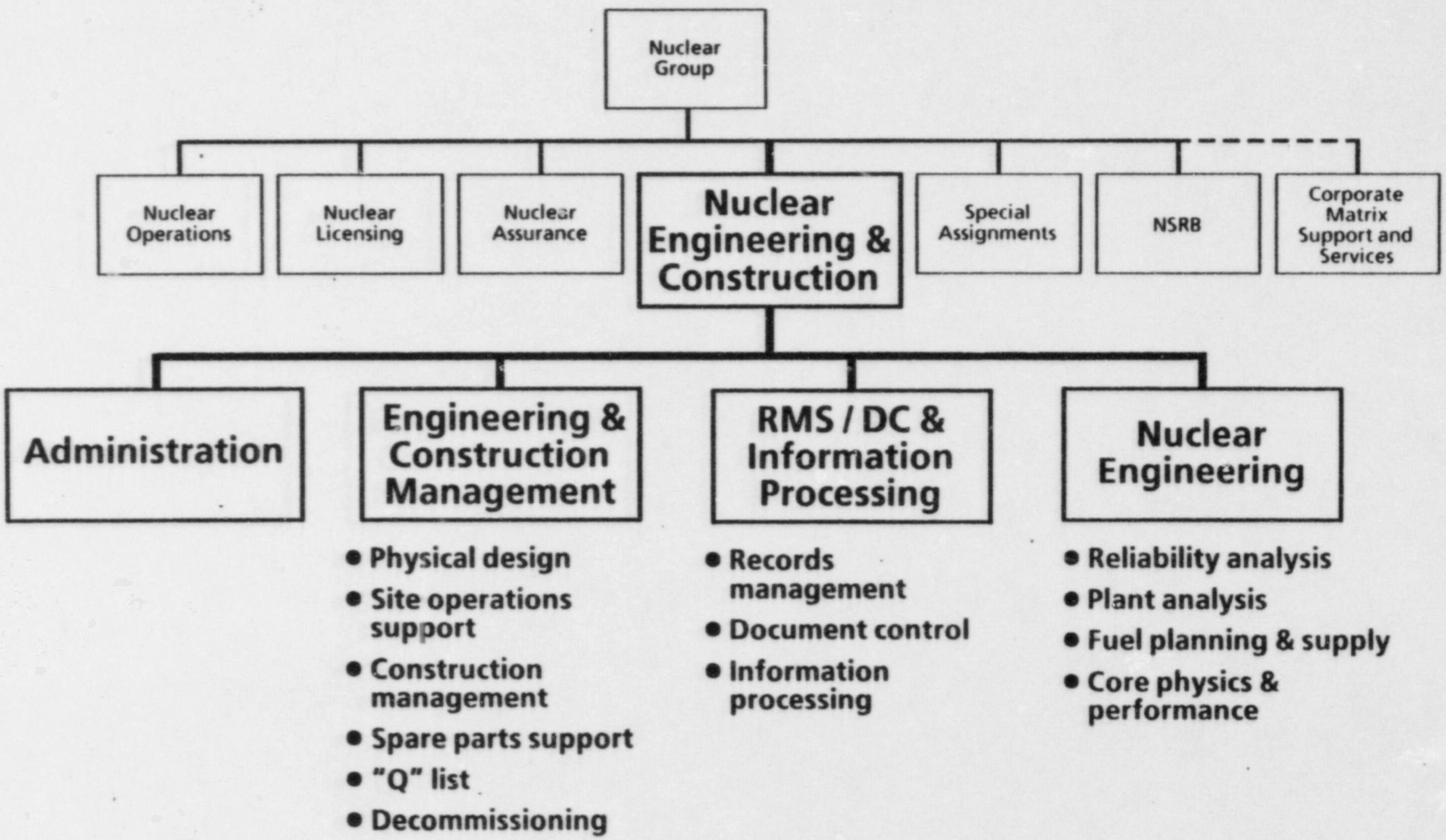
- **NRC and State of Texas licensing activities and interfaces**
- **Licensing commitment tracking**
- **Operational experience review**
- **Maintenance of FSAR**
- **Dissemination of new or revised licensing requirements**
- **Preparation of comments on proposed rules, regulations, and policies of the NRC**

NUCLEAR GROUP ORGANIZATION

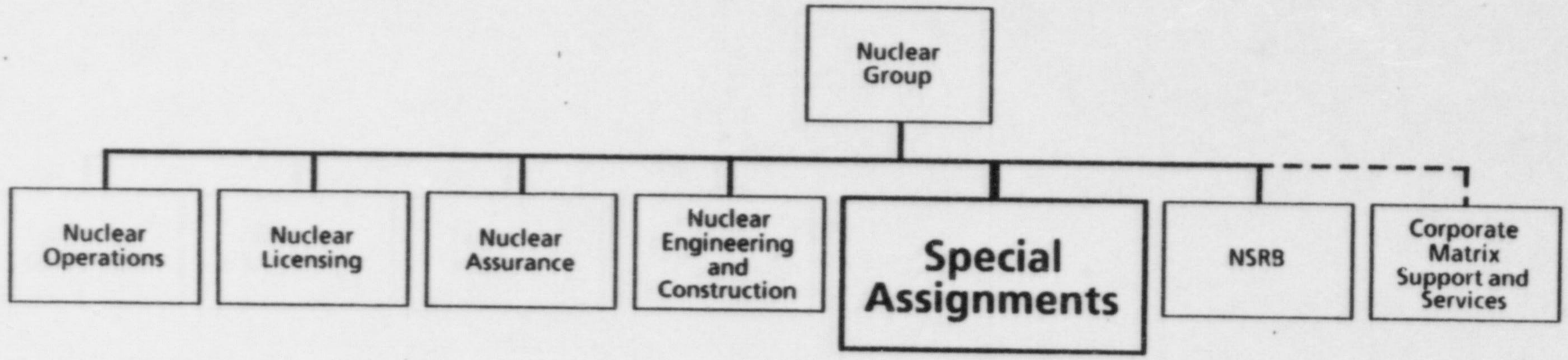


STP

NUCLEAR GROUP ORGANIZATION

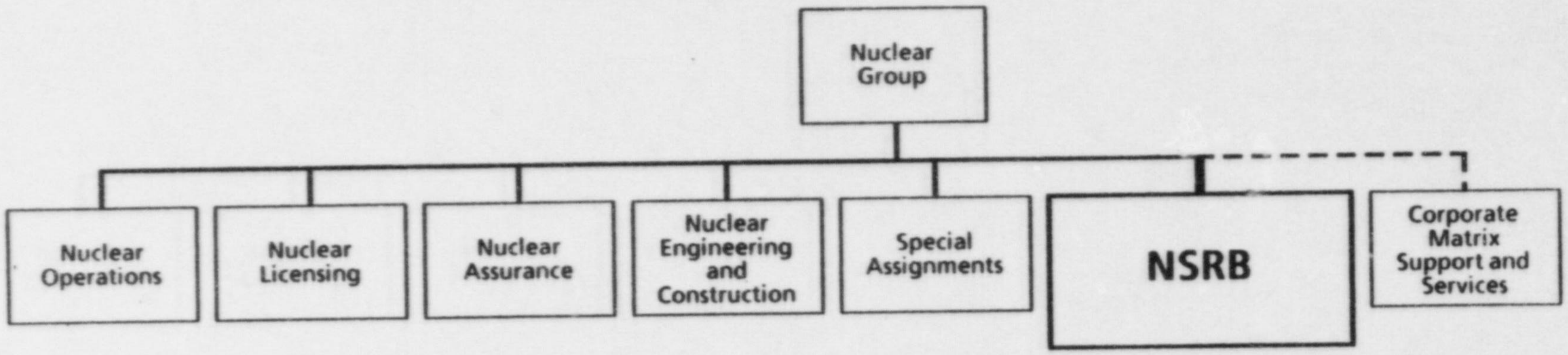


NUCLEAR GROUP ORGANIZATION



STP

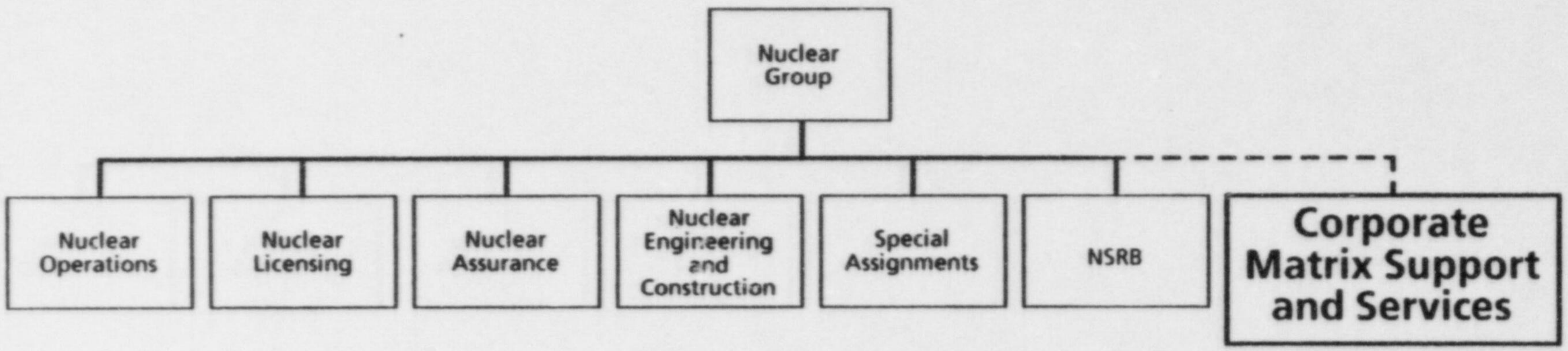
NUCLEAR GROUP ORGANIZATION



- Independent review and responsibilities are specified in tech specs
- Will be established in early 1987
- Will consist of a full time director, members from senior management, and consultants as necessary to provide expertise specified in FSAR

STP

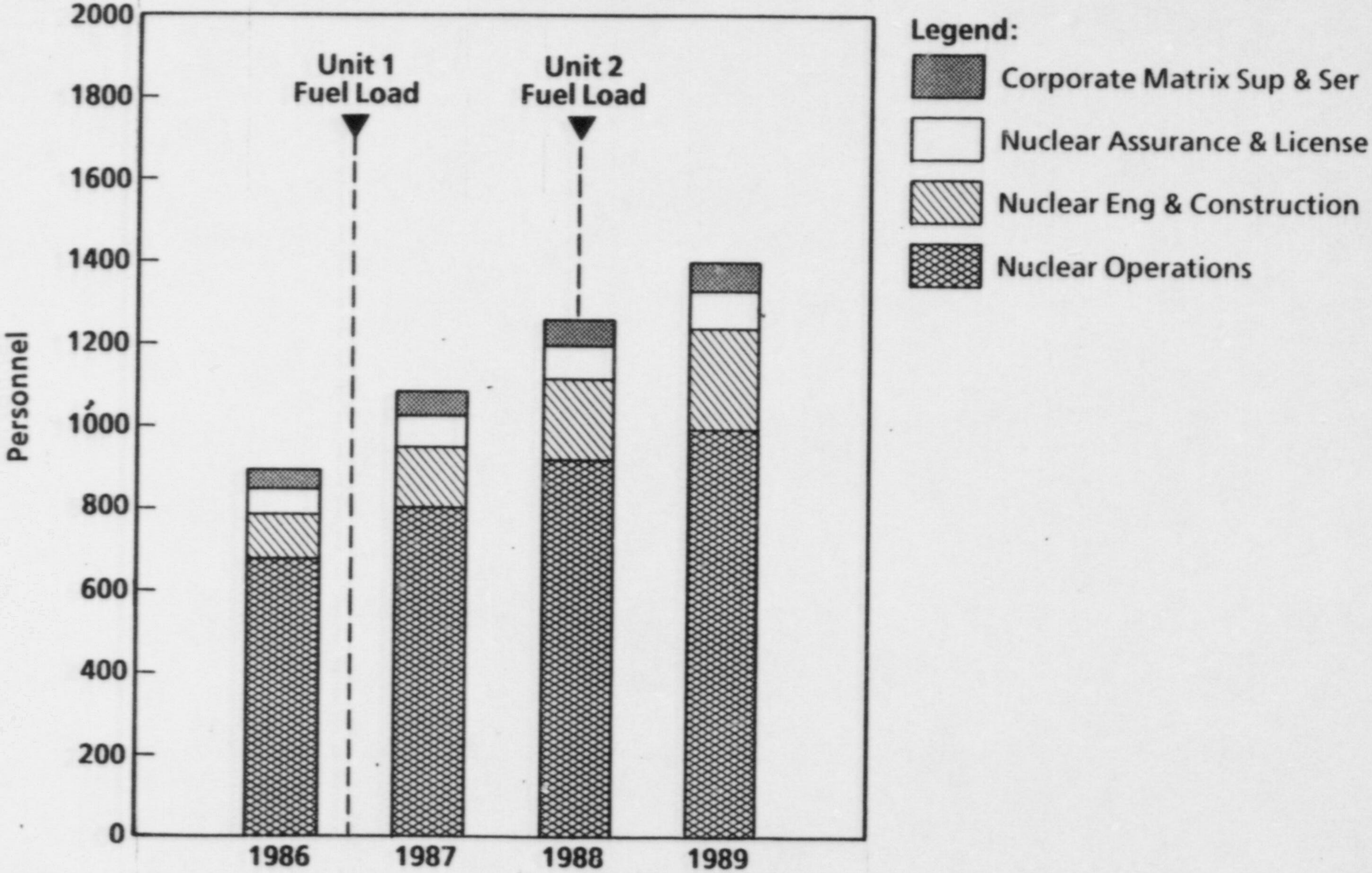
NUCLEAR GROUP ORGANIZATION



- **Purchasing**
- **Stores**
- **Accounting**
- **Human Resources**

STP

STAFFING PLAN NUCLEAR OPERATIONS PHASE



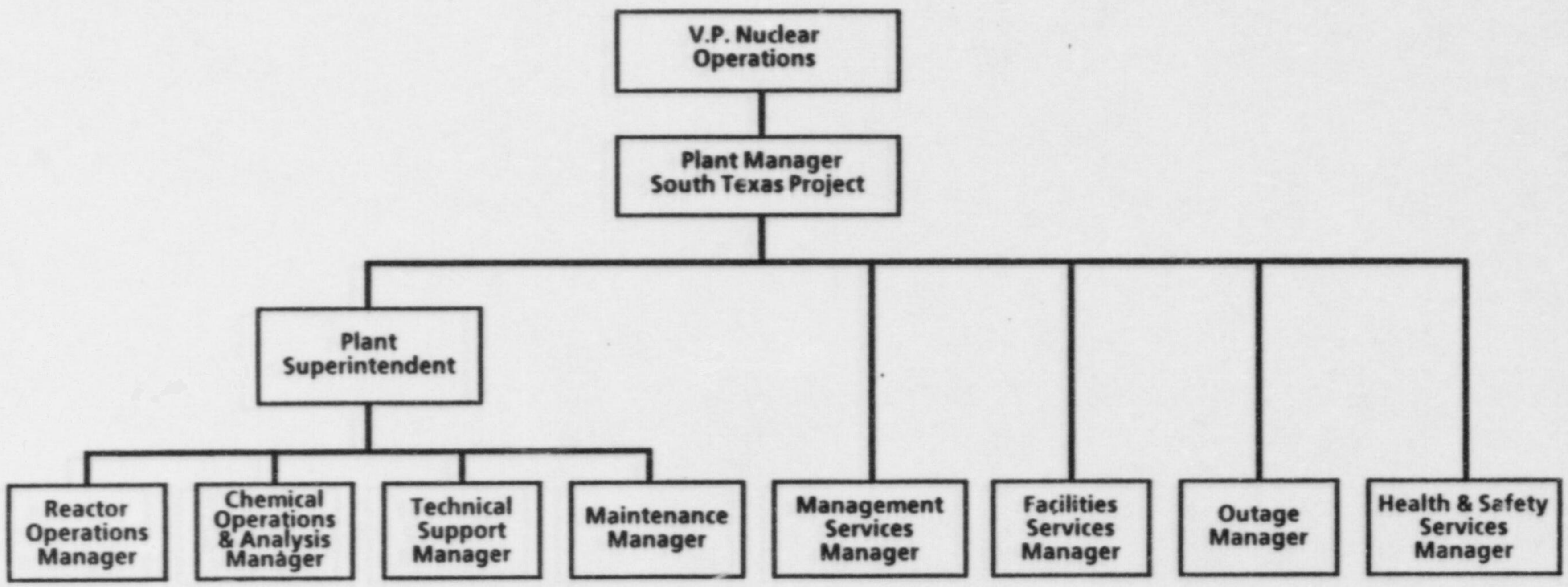
STP

OPERATIONS DEPARTMENT

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

APPENDIX X
OPERATIONS DEPARTMENT

NUCLEAR PLANT OPERATIONS DEPARTMENT



Total Department Staffing	847
Unit One O.L. Staffing	652
Current Staffing	491

STP

TO KEEP ABREAST OF INDUSTRY OCCURRENCES AND APPLY WORTHWHILE EXPERIENCES TO IMPROVE PROGRAMS

- Studied organizations of other utilities
- Reviewed studies and reports addressing weaknesses and strengths
- Developed organization providing strength and flexibility
  - Line responsibility backed by support organization
  - Production unit
  - Support divisions

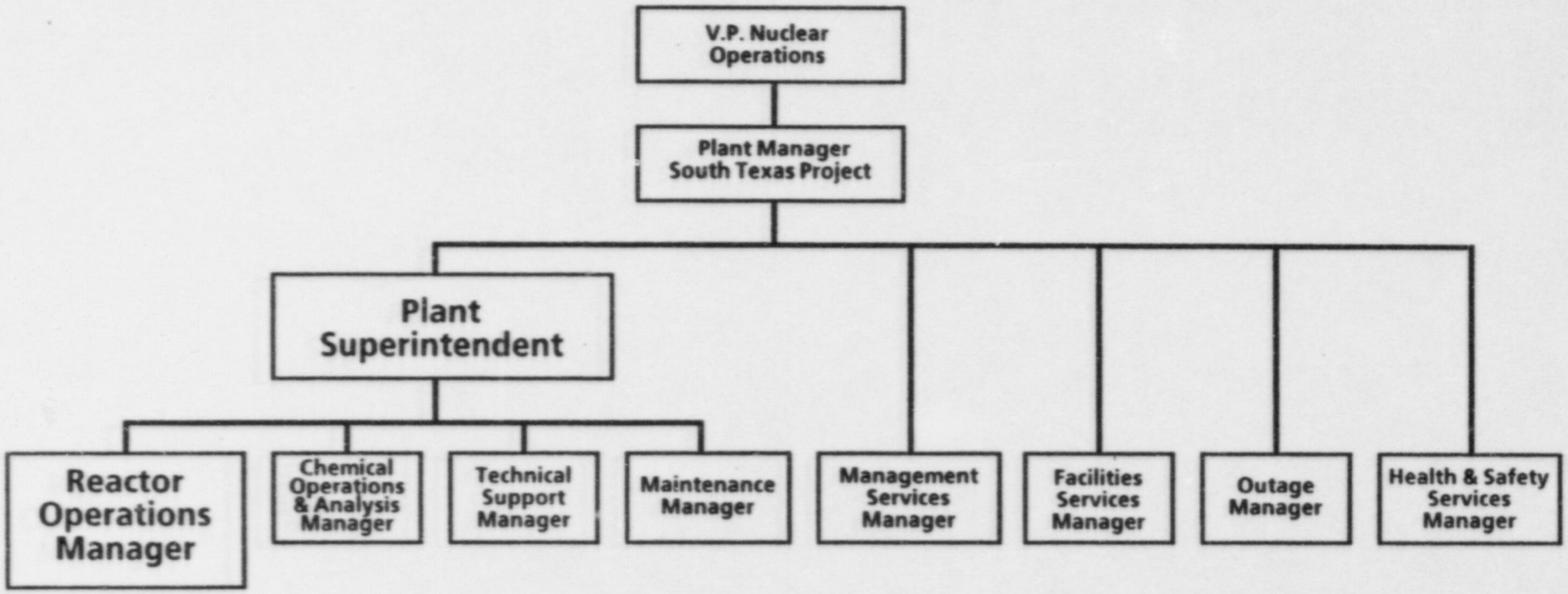
STP

PLAN FOR, DEVELOP, AND RETAIN QUALIFIED AND TRAINED PERSONNEL

HL&P recognized the need:

- To develop its operating staff early to involve employees in the construction and testing program to gain experience
- To ensure that those employees who would operate the plant would have sufficient time to develop the procedures they would use and sufficient time to learn the plant
- To minimize dependence on contract personnel

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

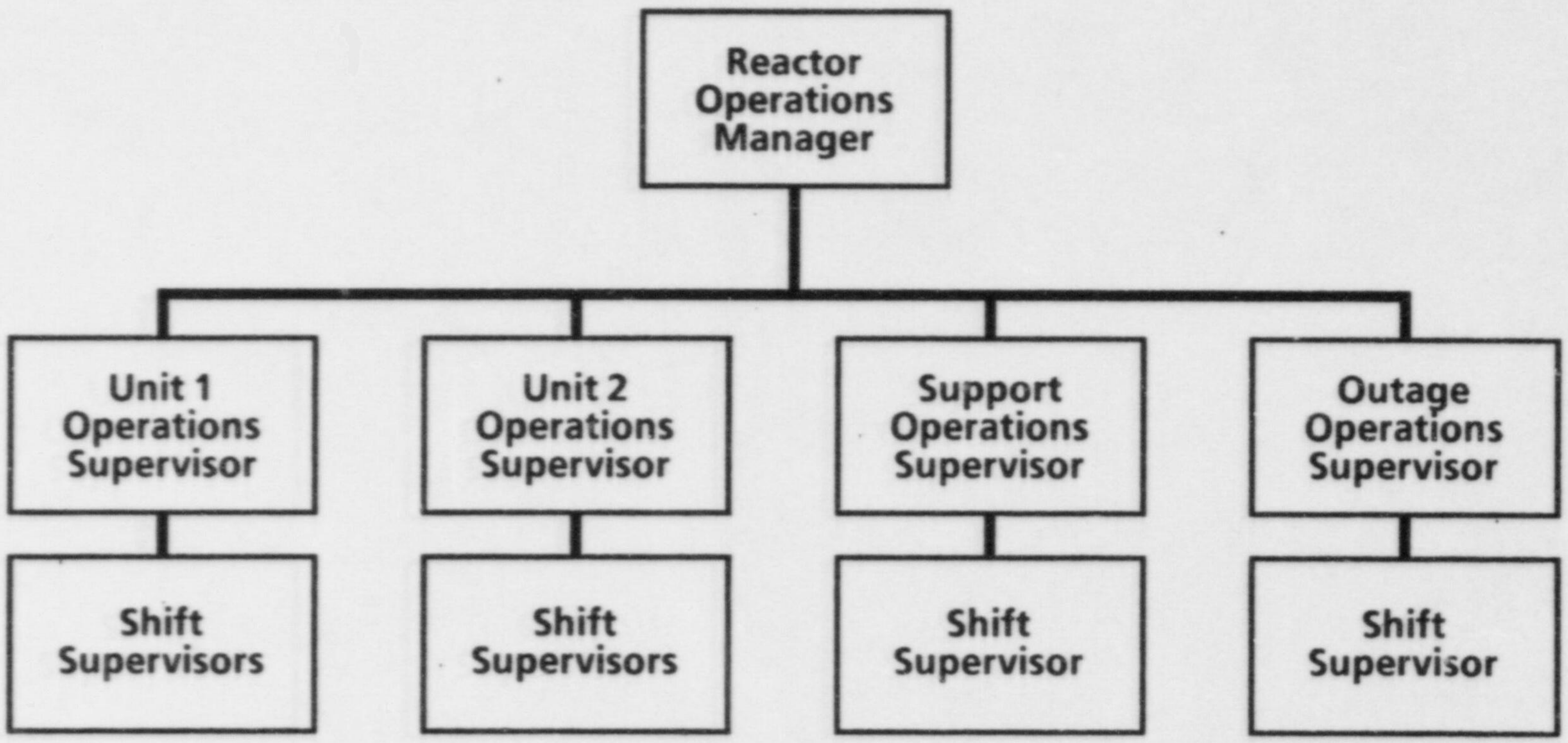
REACTOR OPERATIONS DIVISION

Responsible for the operation of:

- Nuclear steam supply system
- Safeguards systems
- Turbine-generator
- Support auxiliaries

STP

REACTOR OPERATIONS DIVISION



Total Division Staffing	125
Unit One O.L. Staffing	106
Current Staffing	97

STP

SHIFT COMPLIMENT CHART

Position	No. of Shifts	One Unit	Two Units
Shift Supervisor (Licensed SRO)	6	1	2
Unit Supervisor (Licensed SRO)	6	1	2
Reactor Operator (Licensed RO)	6	3	6
Reactor Plant Operator	6	4	8
Administrative Aide	4	1	2
Shift Technical Advisor	-	1	1

STP

REACTOR OPERATIONS DIVISION STAFF EXPERIENCE

NUCLEAR EXPERIENCE:	281 Years
COMMERCIAL EXPERIENCE:	73 Years
NUCLEAR NAVY EXPERIENCE:	30 of 39 Candidates

STP

REACTOR OPERATIONS DIVISION PREVIOUS LICENSES

Division Manager/Operations Supervisors

- SRO License on Large PWR -- 3

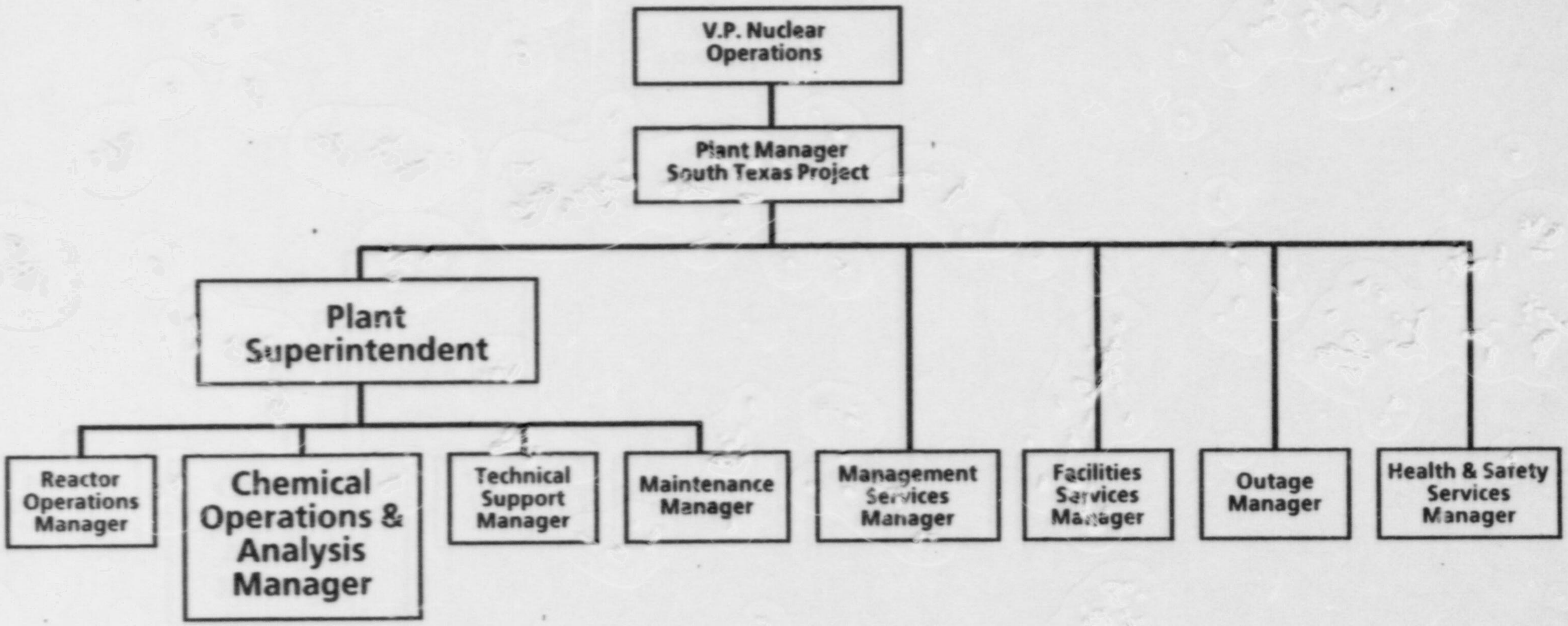
Shift Supervisor Candidates

- SRO License on Large PWR -- 6

Unit Supervisors

- RO License -- 4
- SRO Certification -- 1
- RO Certification -- 3
- SRO License Research Reactor -- 1

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

CHEMICAL OPERATIONS AND ANALYSIS DIVISION

Responsible for the operation of the:

- Water production plant
- Condensate polishers and regeneration systems
- Radwaste processing systems
- Water production support systems
- Waste processing support systems

Responsible for analyzing and maintaining chemical specifications for all plant systems

STATION RADWASTE GROUP

Responsible for ensuring that:

- Radwaste systems are operated optimally
- Other plant systems are operated by plant personnel to minimize radioactive contamination and production of radwaste

CHEMICAL OPERATIONS & ANALYSIS DIVISION ANALYSTS

Responsible For:

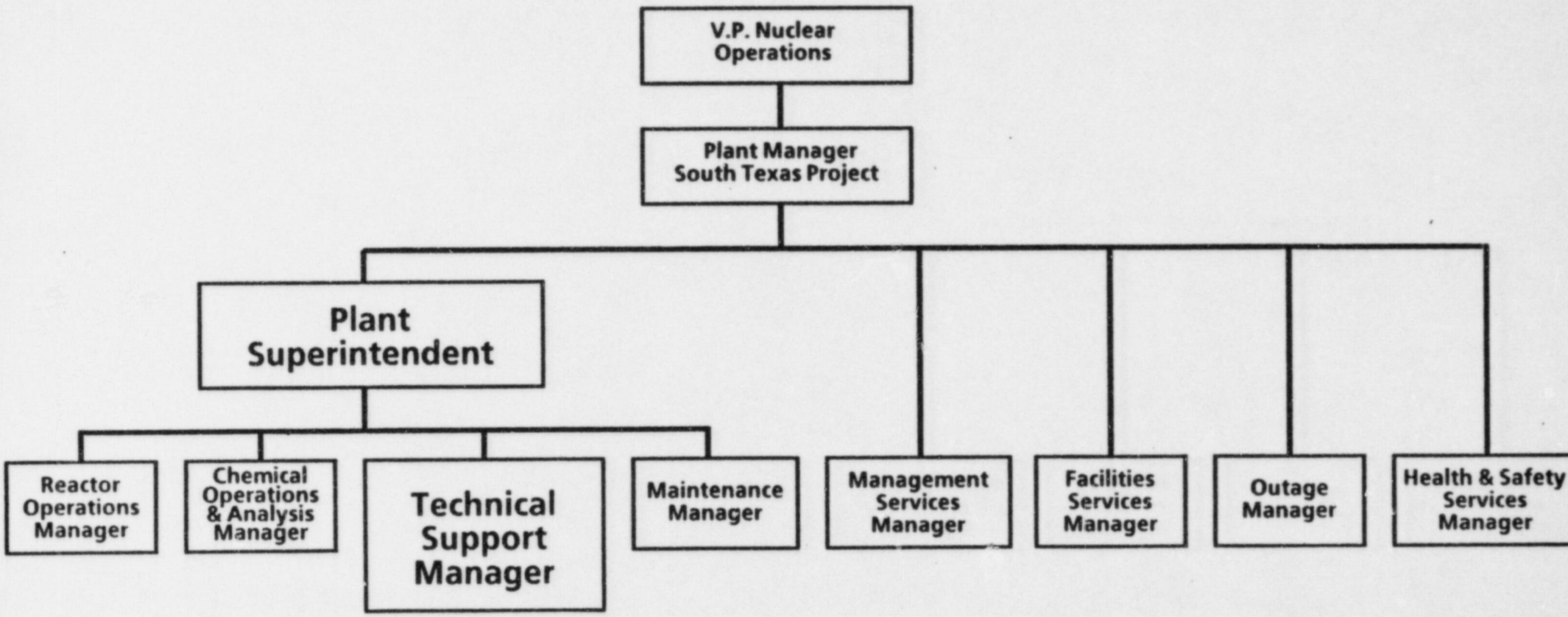
- Monitoring the chemistry parameters of all plant systems
- Providing recommendations to the reactor plant operators and chemical operators on maintaining systems within specifications
- Successfully completing three-year training program
- Preparing procedures
- Performing tests without aid of contract employees

CHEMICAL OPERATIONS AND ANALYSIS DIVISION SUPPORT

Key Activites:

- Support CO&A in program development and system operation
- Program development and operation of the radiochemistry counting room
- Development of computerized chemistry parameter monitoring and trending program
- Effluent release program
- Hazardous chemical control program
- Spill prevention program
- No contract employees

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

SYSTEMS PERFORMANCE SECTION

Responsible for:

- Monitoring plant performance through:
  - Direct testing
  - Observation of normal operating parameters through plant tours
  - Review of plant maintenance work requests
- Trending plant problems
- Monitoring equipment performance
- Conducting the plant surveillance testing program
- Aiding the determination of corrective actions for malfunctioning equipment

REACTOR PERFORMANCE SECTION

Responsible for:

- Routine monitoring of core performance
- Preparation and performance of special tests
- Phase III startup testing program including fuel load and subsequent tests

Shift technical advisors will hold a senior reactor operator license

PERFORMANCE SUPPORT SECTION

Engineers

- Electrical systems
- HVAC systems
- Fire protection
- Snubber testing
- Vibration monitoring
- Operations Experience Review Program

Performance Technicians

- Performance testing
- Surveillance testing

STP

OPERATING EXPERIENCE REVIEW PROGRAM

Key elements:

- NRC Bulletins, Notices and Circulars since 1972
- INPO SOER's, SER's and O&MR's since 1980
- Screened by Nuclear Licensing Department
- Each item addressed by written Plan of Action
- Required action is tracked via computer
- QA Department verifies closure of each Plan of Action

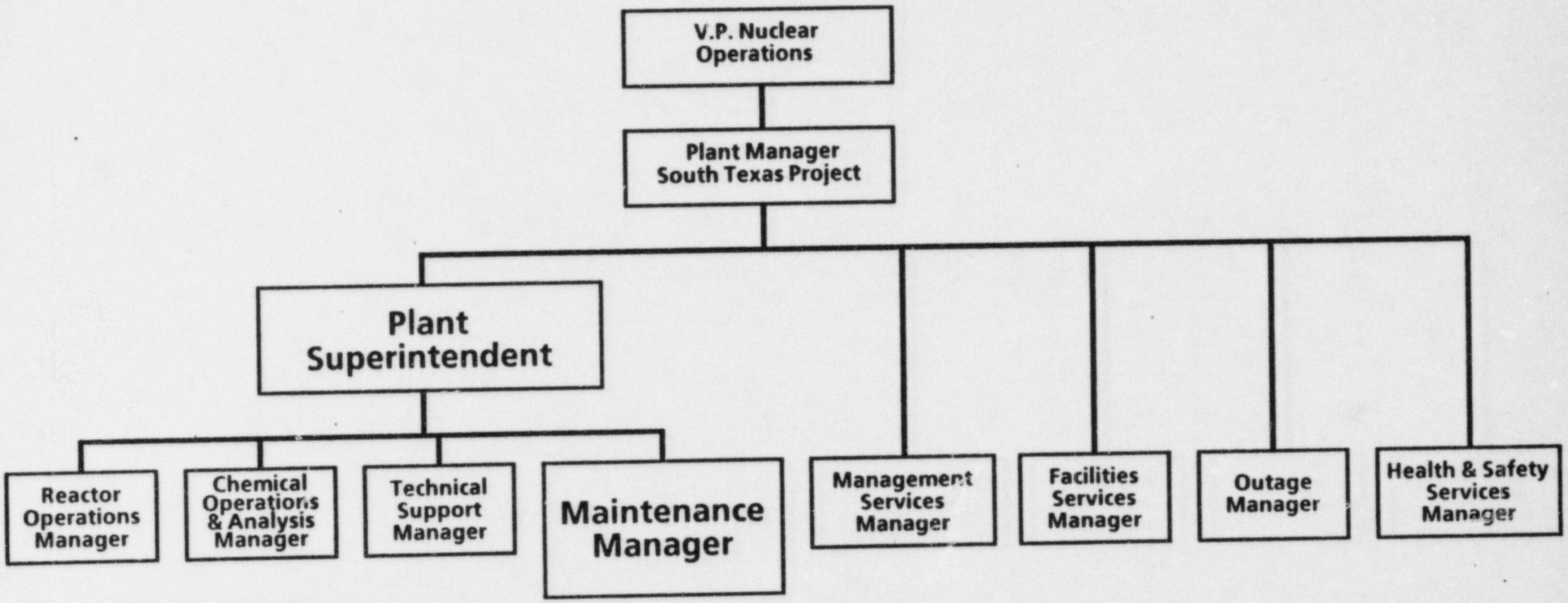
STP

COMPUTER SUPPORT SECTION

Responsible for the startup and operation of the plant computers:

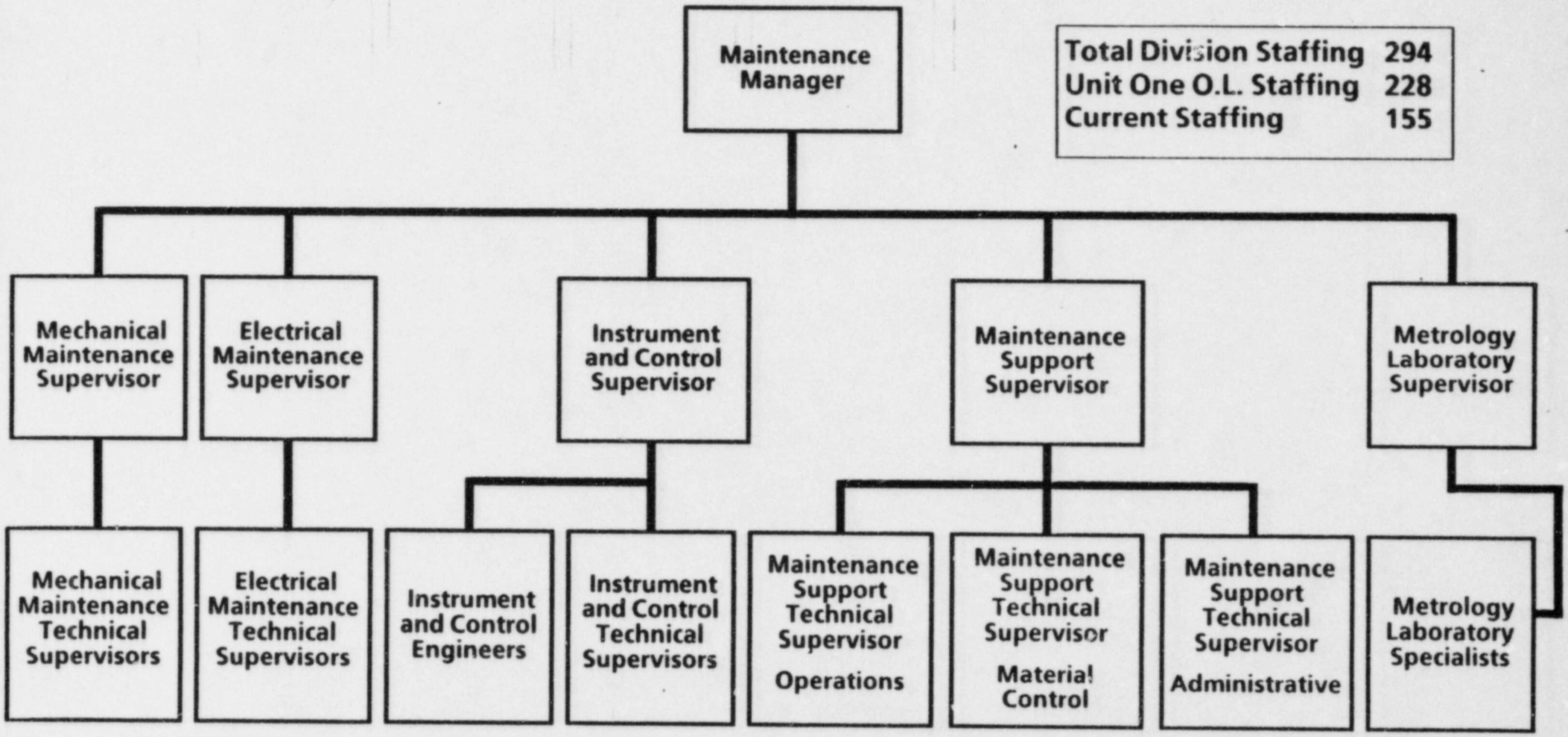
- Plant process computer
- Radiation monitoring computer
- Emergency response facilities data acquisition and display system computer
- Plant security system computer

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

MAINTENANCE DIVISION



STP

MAINTENANCE PERSONNEL QUALIFICATIONS NUCLEAR EXPERIENCE

- Journeyman with commercial nuclear experience in each craft
- 26 Electricians with over 190 years nuclear experience
- 35 Mechanics with over 206 years nuclear experience
- 36 I&C Technicians with over 180 years nuclear experience

MAINTENANCE PROGRAM HIGHLIGHTS

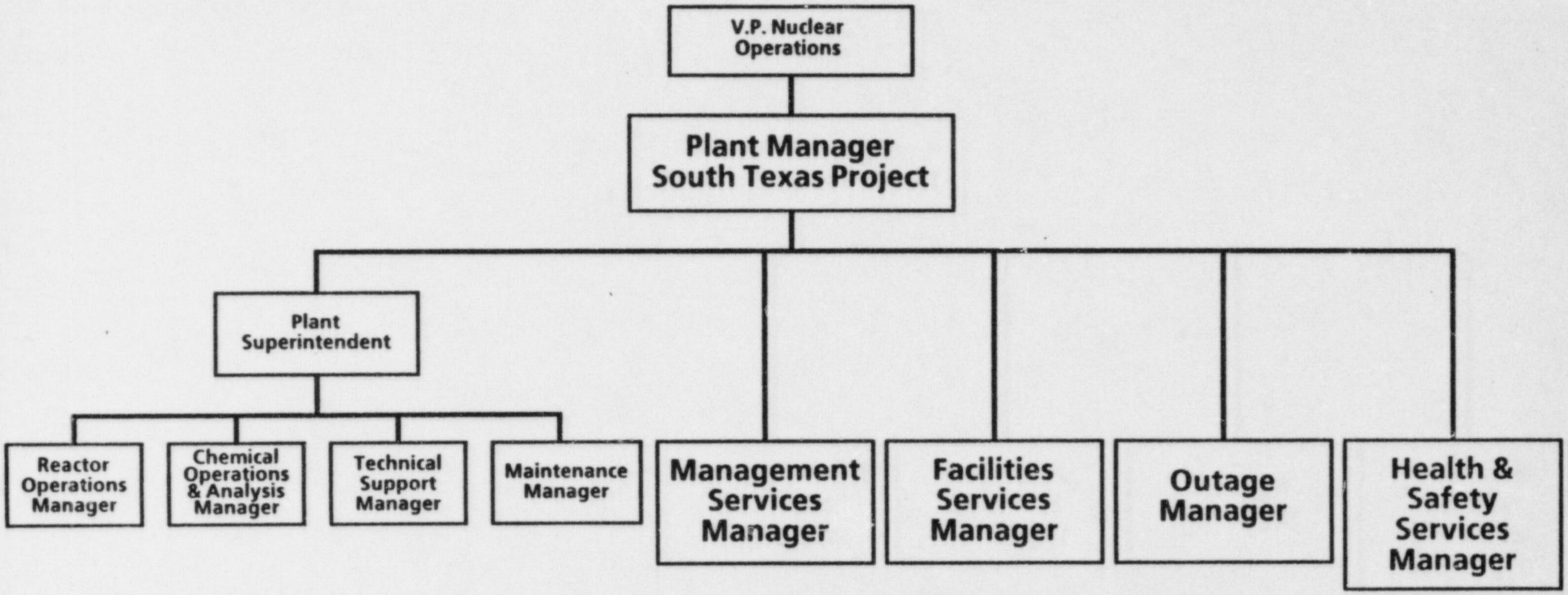
- Utilize prepared procedures for major maintenance (1400)
- Corrective maintenance work requests
- Safety related work reviewed by QA
- Broad-based preventative maintenance program

STP

MAINTENANCE PROGRAM HIGHLIGHTS

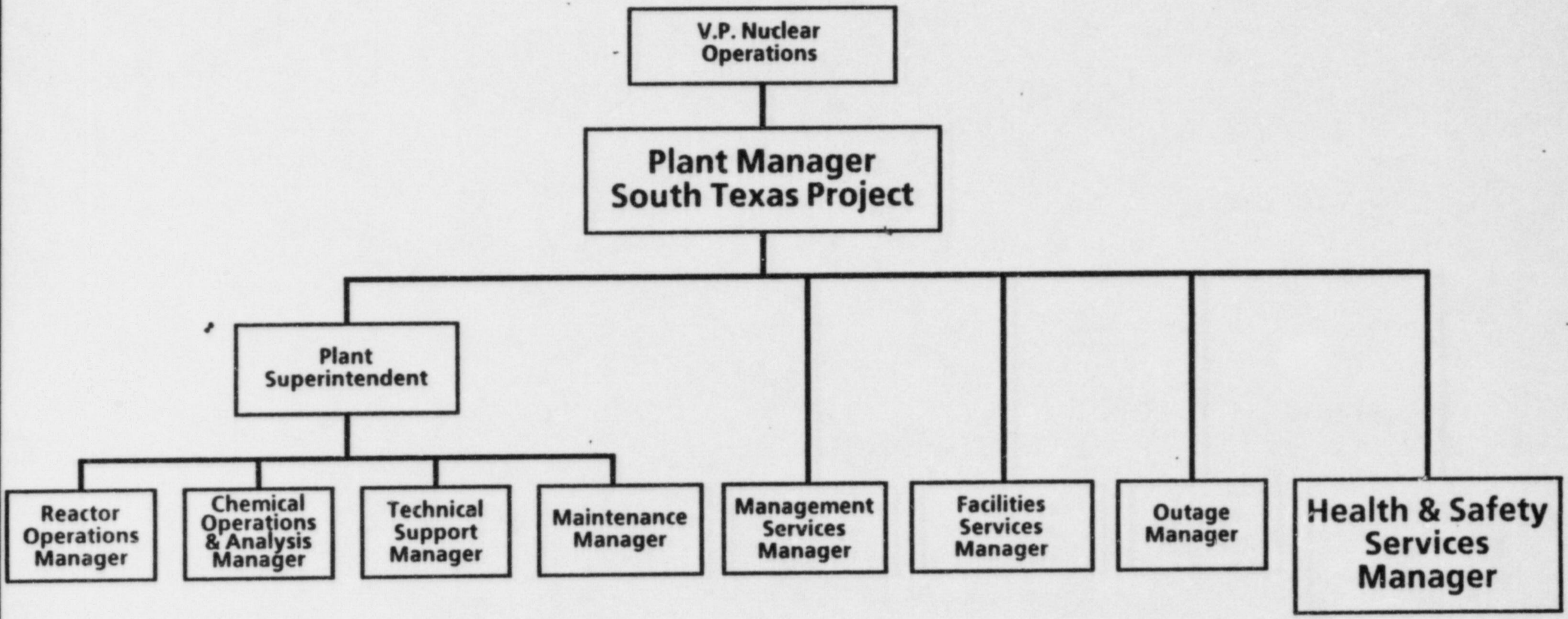
- Root cause determination
- Material control
- Work experience
- Work quality

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

NUCLEAR PLANT OPERATIONS DEPARTMENT



STP

RADIOLOGICAL LABORATORY SECTION

Responsible for:

- Radiological environmental monitoring program
- Dose assessment monitoring program

Key Activities:

- Environmental sampling program
- Offsite Dose Calculation Manual
- Laboratory and dosimetry program

RADIOLOGICAL PROTECTION SECTION

Responsibilities:

- Whole Body Counting Program
- Respiratory Protection Program
- Radiation Work Permits
- Surveys
- Calibration of Portable Monitoring Instrumentation

AS LOW AS REASONABLY ACHIEVABLE ALARA

5 Rem/year is maximum dose any individual will receive at STP

- Review of engineering design
  - Review plant design to assure that engineering organization has incorporated features that will reduce doses to workers
  - Perform walk-downs to verify that design is carried out properly in construction and to find ways to minimize system interrelations that would increase dose levels
- Effective work practices
  - Pre-job planning
  - Exposure reduction
  - Exposure usage accountability
  - Post-job review

NUCLEAR OPERATIONS PERSONNEL EDUCATION

Department	Bachelors Degrees	Advanced Degrees
Technical Support	41	2
Reactor Operations	7	0
Chemical Operations and Analysis	12	1
Health Physics	11	4
Maintenance	5	0
Management Services	19	3
Managers	2	0
TOTAL	97	10

STP

NUCLEAR PLANT OPERATIONS DEPARTMENT EXPERIENCE

Organization	Current Staffing	Total Nuclear Experience
Plant Management	2	14
Reactor Operations	94	295
Chemical Operations & Analysis	67	206
Technical Support	66	161
Maintenance	155	464
Health & Safety Services	40	142
	----------	---------
TOTAL	424	1282

STP

KEY ELEMENTS

- Licensed operators on six-shift rotation
- Other critical positions on five-shift rotation
- Maintenance performed primarily on two shifts, five days/week
- Chemical analyst coverage 24 hours/day, 7 days/week

STP

SOUTH TEXAS PROJECT
ACRS MEETING

JUNE 5, 1986

INSPECTION PROGRAM STATUS
- CONSTRUCTION
- PREOPERATION

INSPECTION RESULTS
- RIV
- CAT
- SALP

ALLEGATION STATUS

CURRENT OBSERVATIONS

PRESENTED BY:
LES CONSTABLE, CHIEF
REACTOR PROJECTS SECTION C
REACTOR PROJECTS BRANCH
USNRC, REGION IV

STP INSPECTION STATUS

TOTAL NRC SITE INSPECTION/INVESTIGATION
MANHOURS 1976 - 1986 21,731 HRS

CONSTRUCTION INSPECTION		
UNIT 1	=	10,032
UNIT 2	=	4,404
INVESTIGATIONS	=	1,768
STARTUP/OPERATIONS	=	1,479
OTHER	=	4,048*

CONSTRUCTION INSPECTION PROGRAM STATUS

UNIT 1 = 70-80% COMPLETE
UNIT 2 = 40-50% COMPLETE

PREOPERATIONAL INSPECTION PROGRAM

SYSTEM TESTING	15%
PROCEDURE REVIEWS	10%
TECHNICAL SPECIFICATIONS	20%
ORGANIZATION/STAFFING	50%
TRAINING	30%

*TRANSITION ENGINEERING OVERVIEW & MISCELLANEOUS

FUNCTIONAL AREA	PERFORMANCE CATEGORY 12/82 - 11/83	PERFORMANCE CATEGORY 12/83 - 6/85
A. SOILS AND FOUNDATIONS	3	2
B. CONTAINMENT AND OTHER SAFETY-RELATED STRUCTURES	2	2
C. PIPING SYSTEMS AND SUPPORTS	1	2
D. SAFETY-RELATED COMPONENTS	2	2
E. HVAC	NA	2
F. FIRE PROTECTION	NA	2
G. ELECTRICAL POWER SUPPLY AND DISTRIBUTION	1	2
H. INSTRUMENTATION AND CONTROL SYSTEMS	NA	NA
I. LICENSING ACTIVITIES	1	2
J. PHYSICAL SECURITY	NA	2
K. TRAINING	NA	1
L. CORRECTIVE ACTION REPORTING	3	1
M. DESIGN AND DESIGN CHANCE CONTROL	2	1
N. MATERIAL CONTROL	3	2
O. QUALITY PROGRAM AND ADMINISTRATIVE CONTROLS	3	2
P. PREOPERATIONAL TESTING	NA	NA

ALLEGATIONS

CURRENTLY OPEN	19
NEW ALLEGATIONS JUNE 85 - MAY 86	24

NRR STAFF PRESE
AC

APPENDIX XII
NRR PRESENTATION ON SOUTH TEXAS

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 & 2

DATE: JUNE 5, 1986

PRESENTER: N. PRASAD KADAMBI

PRESENTER'S TITLE/BRANCH/DIV: PROJECT MANAGER
PROJECT DIRECTORATE #5
DIVISION OF PWR LICENSING-A

PRESENTER'S NRC TEL. NO.: (301) 492-7272

SUBCOMMITTEE: FULL COMMITTEE

ACRS COMMITTEE MEETING,
SOUTH TEXAS PROJECT

SUMMARY OF SUBCOMMITTEE MTG

- THE STAFF PRESENTATION CONSISTED OF REVIEWING THE ISSUES ASSOCIATED WITH LICENSE CONDITIONS, OPEN ITEMS AND CONFIRMATORY ITEMS.

- STAFF WAS REQUESTED TO PROVIDE CLARIFICATIONS AND/OR ADDITIONAL INFORMATION ON FIVE QUESTIONS.

- STAFF WAS REQUESTED TO PROVIDE CLARIFICATIONS AND/OR ADDITIONAL INFORMATION ON FIVE QUESTIONS.

- NO OTHER COMMENTS WERE RECEIVED ON THE SAFETY EVALUATION REPORT FOR SOUTH TEXAS (NUREG-0781).

Listing of open items

Item		SER section
(1)	Internal flooding analysis	3.4.1, 9.2.7, 9.3.3
(2)	Internal missiles analysis	3.5.1, 10.4.9
(3)	Staff review of jet impingement from high energy pipe failures	3.6.1
(4)	Equipment qualification	
	(a) Seismic and dynamic qualification	3.10.1
	(b) Pump and valve operability assurance	3.10.2
	(c) Environmental equipment qualification	3.11.3
(5)	Preservice inspection/inservice inspection program review	5.2.4, 6.6.1
(6)	Design, verification, and validation of qualified display processing system	7.1.2
(7)	Acceptability of isolation between safety and non-safety systems	7.3.2.5
(8)	Conformance to RG 1.97	7.5.2.4
(9)	Test results of aluminum-sheathed and copper-sheathed cable	8.3.3.3
(10)	Maximum available fault currents at electrical penetrations	8.3.3.5
(11)	Safe and alternate shutdown systems	9.5.1
(12)	Auxiliary feedwater system reliability study	10.4.9
(13)	Emergency planning	13.3
(14)	Industrial security	13.6
(15)	Analysis for boron dilution event during modes 4 and 5	15.4.6
(16)	Use of TREAT code for small-break loss-of-coolant-accident analysis	15.6.5, 6.3.5
(17)	Review of submittals on Generic Letter 83-28	15.8.2

RELIABILITY OF FIRE DAMPERS

- THE STAFF IS AWARE OF RELIABILITY PROBLEMS (INFO. NOTICES 83-69, 84-31 AND 10 CFR 21 REPORT FROM RUSKIN)

- THE APPLICANT HAS INFORMED THE STAFF THAT ONLY RUSKIN DAMPERS USED AT SOUTH TEXAS.

- THE APPLICANT FILED REPORTS UNDER 10 CFR 50.55 (E) DESCRIBING DEFICIENCIES AND CORRECTIVE ACTIONS AT SOUTH TEXAS.

- THE APPLICANT HAS CONCLUDED THAT THE PROBLEM AS BEEN RESOLVED. THE IMPLEMENTATION IS SUBJECT TO NRC INSPECTION.

DIESEL FUEL OIL STORGE AREAS

- ACRS RAISED QUESTION IN SEPTEMBER, 1975 AND STAFF SERS IN OCTOBER, 1975 AND APRIL 1986 ADDRESSED THE ISSUE

- STAFF HAS FOUND TO BE ACCEPTABLE DESIGN CHANGES RELATED TO PROXIMITY OF FUEL STORAGE AREAS TO THE CONTROL ROOM AND THE FIRE PROTECTION IN THE STORAGE AREAS.

- ADEQUATE FIRE PROTECTION HAS NOW BEEN PROVIDED IN THE STORAGE ROOMS.

SEPARATION OF BATTERY ROOMS
FROM BALANCE - OF - PLANT.

- SEPARATION FOR FIRE PROTECTION WHICH IS REFERRED TO IN SER RELATES ONLY TO AREAS WITHIN EACH TRAIN.

- THE TERM "BALANCE - OF - PLANT" AS USED DOES NOT RELATE TO SECONDARY SYSTEMS OR TURBINE - GENERATOR SYSTEMS.

COMBUSTION TESTS ON IEEE-383
QUALIFIED CABLES

- QUESTION: DID STAFF CONSIDER SANDIA TESTS IN EVALUATING THE FIRE POTENTIAL FROM THE CABLES.

- ANSWER: THE STAFF DID TAKE THE APPLICABLE TESTS INTO ACCOUNT.

- RECENT TESTS IN CABINETS NOT APPLICABLE TO CABLE TRAYS.

RELEVANCE OF SAN ONOFRE EVENT
TO SOUTH TEXAS DESIGN

- THE SAN ONOFRE EVENT OF NOVEMBER 21, 1985 WAS CHARACTERIZED BY WATER HAMMER PHENOMENA AND CHECK VALVE FAILURE.

- STRONG DEFENSE AGAINST WATER HAMMER PROVIDED AT SOUTH TEXAS (SER SECTION 10.4.9) BY SEPARATE LINES FOR FW AND AFW.

- FEEDWATER LINE WITH ESF ACTUATED ISOLATION VALVE, IN ADDITION TO THE CHECK VALVE, PROVIDES ADDITIONAL DEFENSE.

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE-   APPENDIX XIII

POSSIBLE ATWS EVENT AT LASALLE COUNTY STATION, UNIT 2

This preliminary notification constitutes EARLY notice of events of possible safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region III staff on this date.

Facility: Commonwealth Edison Co.
LaSalle Unit 2
Marseilles, IL 61341

Docket No: 50-374

Licensee Emergency Classification:
____ Notification of an Unusual Event
XX Alert
____ Site Area Emergency
____ General Emergency
____ Not Applicable

Subject: POSSIBLE ATWS EVENT

HE
CJM
FJR

At 4:21 a.m. (CDT), June 1, 1986, while operating at about 83 per cent power, both feedwater pumps tripped during a surveillance test, causing the reactor water level to decrease. A preliminary review of this incident indicates that the water level may have decreased to about four inches above instrument zero (which is 13.7 feet above the top of the fuel), but which is below the automatic scram setpoint of 12.5 inches. Yet, no scram occurred (the normal operating water level is 36 inches above instrument zero). Control room operators, who apparently did not recognize the potential "Anticipated Transient Without Scram" (ATWS), restarted the feedwater pumps in about two minutes.

The possible ATWS was not identified until about two hours after the event when an oncoming shift engineer apparently noticed an abnormal trace on one of the reactor water level recorders. After analyzing the situation, the licensee initiated a shutdown at 2:40 p.m. (CDT), June 1, 1986, and declared an "Alert" under its emergency classification system on the basis of the potential ATWS at about 6:30 a.m. (CDT) June 1, 1986.

LaSalle Unit 1 is currently shut down for refueling.

The licensee is investigating the event to determine whether there may have been an ATWS, or whether instrumentation indicating the low water level may have been faulty.

Region III (Chicago) will issue a Confirmatory Action Letter documenting the licensees commitment to obtain the Regional Administrator's concurrence before restarting the unit. An Augmented Inspection Team (AIT) composed of regional inspectors including the Oyster Creek resident inspector (the Oyster Creek design is similar to LaSalle) and a representative from the Office of Nuclear Reactor Regulation has been dispatched to the site and is expected to arrive in the early afternoon of June 2.

The State of Illinois will be notified. Region III first learned of this event at 5:48 p.m. (CDT), June 1, 1986. This information is current as of 12:00 p.m. (CDT), June 2, 1986.

CONTACT: G. Wright
FTS 388-5695

W. Guldemond
FTS 388-5574

DISTRIBUTION:
H. St.
Chairman Palladino
Comm. Zech
Comm. Bernthal
Comm. Roberts
Comm. Asselstine
ACRS
SECY
CA
PDR

EDO NRR E/W Willste Mail: ADM:DMB
PA IE NMSS DOT:Trans only
ELD OIA RES
AEOD

SP ______ Regional Offices

INPO ______ NSAC ______
RIII Resident Office
Licensee: ____________ (Corp. Office - Reac
& Lic. Only)

C/R Dist ... PE, GE, CA, ACRS, Historian, SECY ...

Region III
Rev. November 1985

LaSalle Unit 2

PALISADES PLANT

BACKGROUND:

- SALP CATEGORY 3 - MAINTENANCE, SURVEILLANCE, QUALITY PROGRAM
  - LACK OF AGRESSIVE CORRECTIVE ACTION
  - POOR MANAGEMENT CONTROLS
- CYCLE 5 RECURRENT EQUIPMENT PROBLEMS - 1985
  - SAFETY INJECTION TANK SYSTEMS (SIT)
- MARCH 1986 STARTUP FROM REFUELING/MAINTENANCE OUTAGE
  - TWO OF FOUR PRIMARY COOLANT PUMPS WITH SEAL PROBLEMS
  - PCS LOOP CHECK VALVE LEAKAGE
  - SIT SYSTEM VALVE LEAKAGE
  - CVCS DIVERT VALVE LEAKAGE
- APRIL 10, 1986, SHUTDOWN - PCS LEAKAGE
- APRIL 11, 1986, DERATING - CONDENSATE PUMP PACKING FAILURE
- APRIL 23-29, 1986, - VALVE LEAKAGE PROBLEMS IN PCS MAKEUP SYSTEM

PALISADES PLANT - REACTOR TRIP OF MAY 19, 1986

PROBLEMS:

MULTIPLE FAILURES

- TURBINE BY-PASS VALVE FAILED TO OPEN
- 1 STEAM DUMP VALVE FAILED TO OPEN
- BACKPRESSURE REGULATOR IN LET-DOWN LINE FAILED CLOSED
- PRESSURIZER SPRAY VALVE FAILED TO FULLY CLOSE
- VARIABLE SPEED CHARGING PUMP TRIPPED 5 TIMES
- ROD BOTTOM LIGHT FAILED TO INDICATE ONE ROD FULL IN
- TURBINE LIFT PUMPS FAILED TO START AUTOMATICALLY

EXISTING OUT-OF-SERVICE EQUIPMENT

- CONDENSATE RECIRC VALVE AUTO OPERATOR INOPERABLE
- BANK OF PRESSURIZER HEATERS INOPERABLE

SIGNIFICANCE

- UNNECESSARY CHALLENGES TO SAFETY EQUIPMENT
- INCREASED BURDEN ON OPERATORS TO COMPENSATE FOR FAILED OR DEFICIENT EQUIPMENT
- IMPLICATIONS CONCERNING THE QUALITY OF MAINTENANCE AND POST-MAINTENANCE TESTING

SEQUENCE OF EVENTS

- PM ON TURBINE VALVE CONTROL CABINET FANS
- 14:15:14 TURBINE VALVES CLOSED
- REACTOR TRIP ON HIGH PRESSURIZER PRESSURE
- TURBINE TRIP
- FIRST ATMOSPHERIC DUMP VALVE OPENED, AFW PUMP P-8A STARTED
- 2ND ATMOSPHERIC DUMP VALVE OPENED
- 3RD ATMOSPHERIC DUMP VALVE OPENED
- CHARGING PUMP P-55A STARTED (55B & C ALREADY RUNNING)
- PRESSURIZER LEVEL LOW
- LAST LETDOWN ISOLATED
- CHARGING PUMP 55A TRIPPED; THIS PUMP WAS RESTARTED 4 MORE TIMES TRIPPING 30 SECONDS LATER AFTER EACH START
- 14:22:15 PRESSURIZER LEVEL NORMAL

PLANT PARAMETERS:

- PRESSURIZER PRESSURE MAX 2245 PSIA, MIN 1689 PSIA
- T/HOT MAX 594°F, MIN 535°F
- T/COLD MAX 557°F, MIN 535°F
- S/G PRESSURE MAX 1025 PSIA
- S/G LEVEL DROPPED FROM 70 TO 12 PERCENT

NRC RESPONSE:

- REGION III ISSUED A CONFIRMATORY ACTION LETTER REQUIRING
  - LICENSEE CONDUCT THOROUGH INVESTIGATION INTO CAUSE AND IMPLICATIONS OF THE MAY 19 TRIP
  - REGION III APPROVAL PRIOR TO RESTART

CONCLUSIONS:

- PERFORMANCE OF PLANT OPERATORS AND THE OPERATION OF OTHER MAJOR OR SAFETY-RELATED SYSTEMS WERE AS EXPECTED AND DESIGNED CONSIDERING THE EQUIPMENT FAILURES THAT OCCURRED.

- SIGNIFICANT WEAKNESSES EXIST IN MAINTENANCE FUNCTIONS OF DIAGNOSTICS, REPAIR, POST-MAINTENANCE TESTING. THESE WEAKNESSES WERE CONTRIBUTORY TO MOST OF THE EQUIPMENT FAILURES.

- EQUIPMENT FAILURES AND DEGRADED EQUIPMENT HAS PLACED VARYING LEVELS OF ADDITIONAL BURDEN ON PLANT OPERATORS. FOR MAY 19, 1986, TRIP, EQUIPMENT FAILURES DISTRACTED OPERATORS BUT DID NOT SIGNIFICANTLY JEOPARDIZE PLANT SAFETY.

- PLANT OPERATORS HAVE SERIOUS CONCERNS REGARDING THE ADEQUACY OF MAINTENANCE ACTIVITIES AND EQUIPMENT RELIABILITY.

TROJAN - REPEATED SNUB

JUNE, 1985 (T. CH

PROBLEM: STEAM GENERATOR HYDRAULIC SNUBBERS LOCKING UP WHEN NOT DESIRED

SIGNIFICANCE: OVERSTRESSING PORTIONS OF THE RCS PIPING

BACKGROUND:

- 1985 FEBRUARY - ISSUANCE OF SNUBBER TECHNICAL SPECIFICATIONS
- APRIL - SNUBBERS TESTED FOR THE FIRST TIME. 2 SG HYDRAULIC SNUBBERS TESTED; BOTH FAILED, ALL 16 SG SNUBBERS DECLARED INOPERABLE. FAILURES ATTRIBUTED TO INAPPROPRIATE ACCEPTANCE CRITERIA FOR CONTROL VALVES
- APRIL - HOT LEG (TO SG "B") PIPE WHIP RESTRAINT LATERAL MEMBER WAS FOUND PULLED FROM THE WALL

- 1986 JANUARY - LER 85-13 STATES THAT SNUBBER LOCKUP MIGHT HAVE CAUSED OVERSTRESSING OF "B" SG HOT LEG ELBOW.
- APRIL - OUTAGE INSPECTION REVEALS 11 OF 16 SG SNUBBERS FAILED TESTS; ATTRIBUTED TO CONTROL VALVE DESIGN DEFICIENCY

FOLLOW-UP:

- PT PERFORMED ON "B" SG ELBOW TO PIPE WELD. NO INDICATIONS FOUND
- LICENSEE, NRR AND REGION V WALKED DOWN RCS PIPING; EVIDENCE OF RESTRAINED THERMAL GROWTH OBSERVED
- UT PERFORMED ON ALL 4 HOT LEG ELBOWS. NO INDICATIONS FOUND
- SNUBBER CONTROL VALVES REPLACED WITH A NEW DESIGN
- LICENSEE TO MONITOR THERMAL GROWTH DURING HEAT-UP AND OPERATION
- NRR TO REVIEW LICENSEE'S STRESS AND FATIGUE ANALYSES FOR RCS PIPING

Agenda for ACRS
Meeting on June 6, 1986
1:00 p.m.
Room 1046, H Street

RECENT SIGNIFICANT EVENTS

Date	Plant	Event	Presenter/Office telephone	Page
5/19/86	Pilgrim	Single Failure Could Disable All Redundant RHR Pumps	E. Weiss, IE 492-9005	2
6/85	Trojan	Repeated Snubber Failure	T. Chan, NRR 492-7136	5
5/19/86	Palisades	Reactor Scram AIT on site as of 5/22/86	W. Hehl, Reg III 312-790-5552	Not In Package

PILGRIM - SINGLE FAILURE COULD DISABLE ALL REDUNDANT RHR PUMPS
MAY 19, 1986 (ERIC WEISS, IE)

PROBLEM:

SINGLE FAILURE OF MINIFLOW LOGIC COULD DISABLE ALL REDUNDANT RHR PUMPS DURING SMALL OR INTERMEDIATE SIZE BREAK LOCA

SIGNIFICANCE:

- POTENTIAL SINGLE FAILURE CAUSES LOSS OF MULTIPLE SAFETY FUNCTIONS
- POTENTIAL FOR NO LONG TERM COOLING FROM SAFETY SYSTEMS

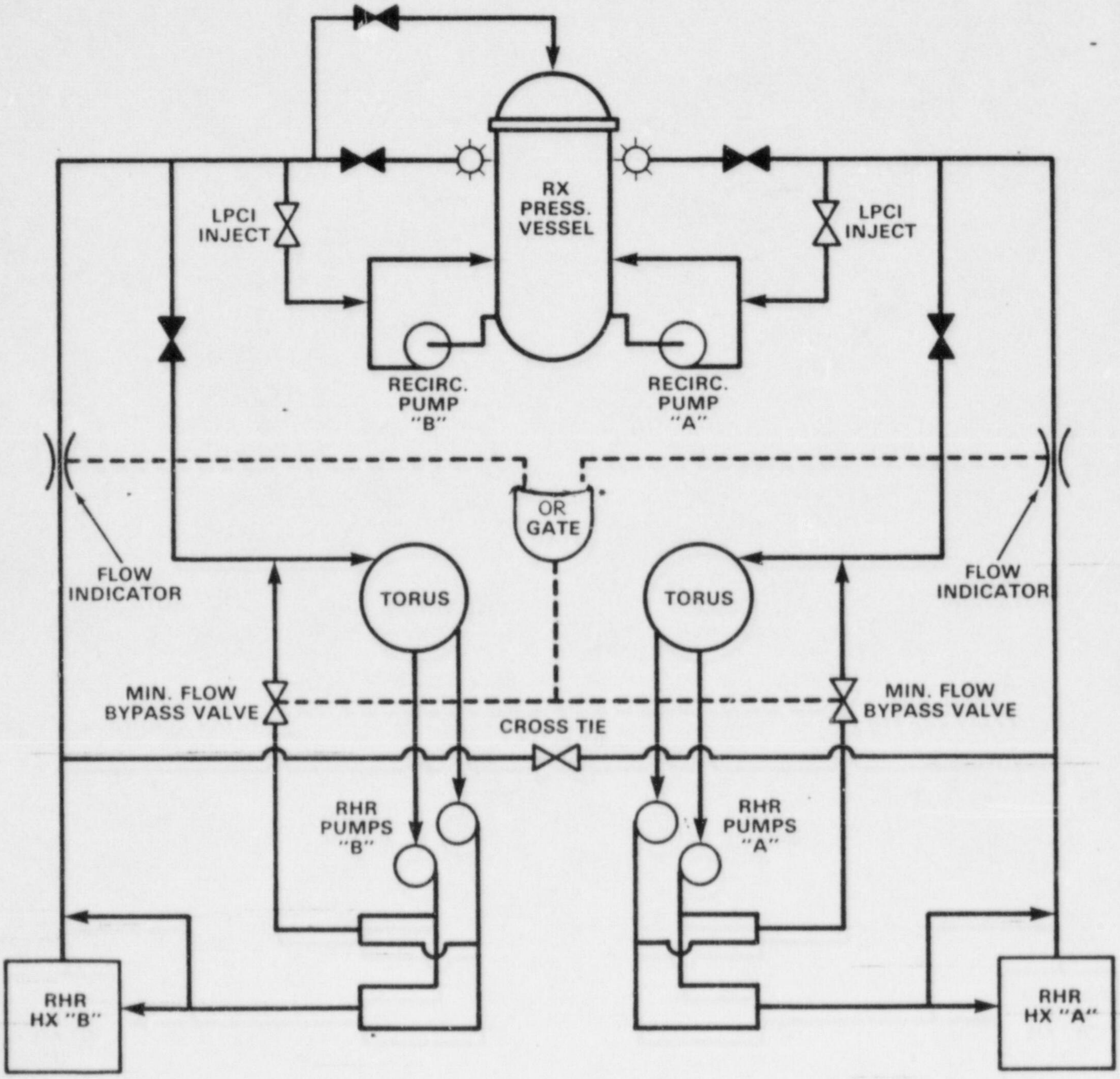
CIRCUMSTANCES:

- LICENSEE REVIEW (PROMPTED BY INFO NOTICE 85-94) DISCOVERED THAT SINGLE FAILURE OF EITHER MINIFLOW SWITCH COULD PREVENT ALL AUTOMATIC LOW FLOW PROTECTION FOR ALL RHR PUMPS; PUMPS COULD BURN UP IF MANUAL ACTION NOT TAKEN IMMEDIATELY
- DURING SOME ACCIDENTS OR SPURIOUS ACTUATIONS, RHR PUMPS WOULD BECOME DEAD HEADED FOR EXTENDED PERIOD
- CURRENT MINIFLOW LOGIC DESIGNED TO BE CONSISTENT WITH LOOP SELECT LOGIC FOR LPCI
- WHEN FLOW DETECTORS IN EITHER LOOP SENSE ADEQUATE FLOW, BOTH RHR MINIFLOW LINE VALVES CLOSE
- CONSEQUENCE OF RHR PUMP LOSS IS LOSS OF LONG TERM COOLING WITH RHR HEAT EXCHANGERS, AND OTHER FUNCTIONS INCLUDING:
  - -SHUTDOWN COOLING MODE
  - -LOW PRESSURE COOLANT INJECTION
  - -HEAD SPRAY (REMOVED FROM PILGRIM)
  - -CONTAINMENT SPRAY
  - -TORUS SPRAY
  - -SUPPRESSION POOL COOLING WHICH EVENTUALLY WOULD CAUSE LOSS OF:
    - -LOW PRESSURE CORE SPRAY
    - -HIGH PRESSURE COOLANT INJECTION
    - -REACTOR CORE ISOLATION COOLING
- GE FIX IS TO ELIMINATE "CLOSE" SIGNAL TO MINIFLOW VALVES; COULD INCREASE PEAK CLAD TEMP 50°F IN SOME BREAK SIZES; NRC CONSIDERS THIS TO BE INTERIM ACTION

FOLLOW-UP

- IE BULLETIN 86-01 ISSUED 5/23/86
- IE AND GE ARE DETERMINING GENERIC SIGNIFICANCE
- NRR WILL REVIEW RESOLUTION FOR PLANTS WITH PROBLEM, INCLUDING TECHNICAL SPECIFICATION ISSUES

SIMPLIFIED DIAGRAM OF PILGRIM MINIMUM FLOW FOR RHR PUMPS



*EITHER SENSOR DETECTING FLOW WILL CAUSE MINIMUM FLOW VALVE TO GO CLOSED

TROJAN - REPEATED SNUBBER FAILURES
JUNE, 1985 (T. CHAN , NRR)

PROBLEM:

STEAM GENERATOR HYDRAULIC SNUBBERS LOCKING UP DUE TO DESIGN INADEQUACY.

SIGNIFICANCE:

- DAMAGE TO HOT LEG PIPE WHIP RESTRAINT (1985)
- OVERSTRESSING OF HOT LEG ELBOWS
- PREVIOUSLY UNACCOUNTED FOR MOVEMENT IN THE PRESSURIZER SURGE LINE (1982-1985)

CIRCUMSTANCES:

- NRC RECENTLY LEARNED THAT RCS HOT LEG PIPE RESTRAINT HAD PULLED FROM WALL IN 1985
- LICENSEE, NRR, AND REGION V WALKED DOWN RCS PIPING
- DYE PENETRANT TEST PERFORMED ON "B" SG ELBOW. NO INDICATIONS FOUND.
- W PERFORMED UT ON ALL 4 HOT LEG ELBOWS AND FOUND NO INDICATIONS
- CRUSHED GRAPHITE SHIMS FOUND ON 3 OF 4 HOT LEG PIPE WHIP RESTRAINTS INDICATING HOT LEG TO RESTRAINT BINDING
- 11 OF 16 SG SNUBBERS FOUND TO HAVE FAILED AGAIN IN SAME WAY

BACKGROUND:

- 1982 - LICENSEE REMOVED THE THERMAL SLEEVE ON THE PRESSURIZER SURGE LINE; HOWEVER, SURGE LINE DID NOT SETTLE OVER NEXT FEW CYCLES, AS HAD BEEN EXPECTED IN W ANALYSES; MOVEMENT CONTINUED
- 1985 - LICENSEE HIRED IMPELL TO REVIEW THE SURGE LINE MOVEMENT; UNABLE TO ACCOUNT FOR CONTINUED MOVEMENT
- 1985 - A HOT LEG (TO SG "B") PIPE WHIP RESTRAINT HORIZONTAL SUPPORT WAS FOUND PULLED FROM THE WALL

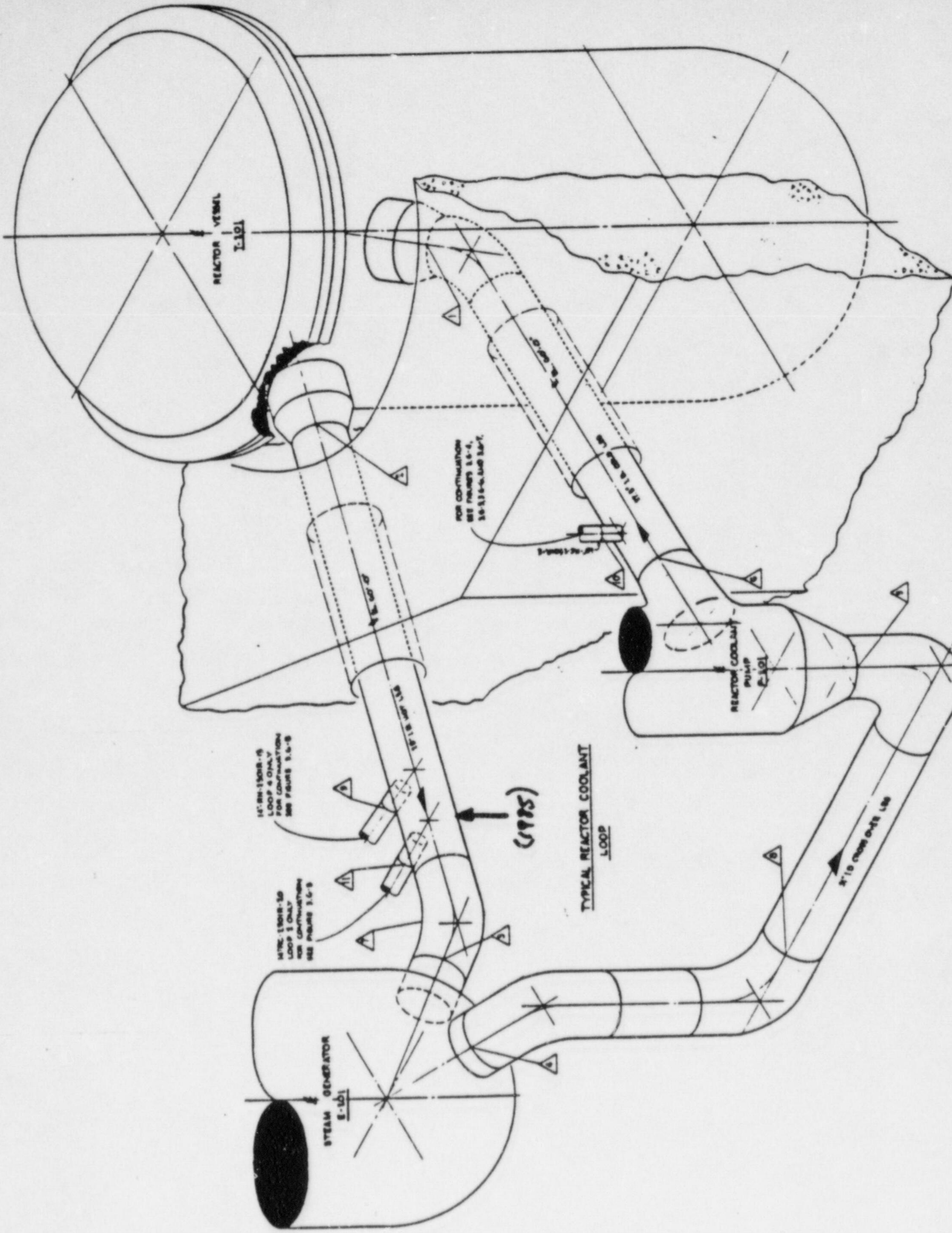
TROJAN - REPEATED SNUBBER FAILURES
JUNE, 1985 (T. CHAN, NRR), (CON'T.)

BACKGROUND, (CON'T.)

- 1985 - SNUBBERS TESTED PER NEW TS REQUIREMENTS
  - 2 OF 16 SG HYDRAULIC SNUBBERS WOULD NOT RESPOND TO 100 KIP LOAD; SHOULD HAVE RESPONDED AT < 10 KIP; ALL 16 WERE DECLARED INOPERABLE AND REBUILT
  - SNUBBER FAILURE ATTRIBUTED TO CLOGGED HYDRAULIC LINES; CLEANED
- WHEN ASSUMED THAT ALL SG SNUBBERS WERE INOPERABLE, IMPELL ANALYSES WAS ABLE TO ACCOUNT FOR THE SURGE LINE MOVEMENT AND THE DAMAGE TO THE PIPE WHIP RESTRAINT
- THE LICENSEE CLAIMED (1985) THAT ALTHOUGH HOT LEG STRESSES EXCEEDED ASME SECTION III ALLOWABLES, STRAIN IS WITHIN 1% LIMIT, WHICH WAS NRC-APPROVED LIMIT FOR SONGS-1 ON SEISMIC CRITERIA AND METHODOLOGY

FOLLOW-UP:

- SNUBBER CONTROL VALVES TO BE REPLACED WITH NEW DESIGN
- LICENSEE TO PERFORM PRE-STARTUP WALKDOWN OF RCS IN A HOT CONDITION
- NRR TO REVIEW RCS PIPING STRESSES AND APPLICABILITY AND ACCEPTABILITY OF LICENSEE'S ANALYSIS


REACTOR VESSEL
T-101
REACTOR COOLANT PUMP
P-101
TYPICAL REACTOR COOLANT LOOP
(1995)
STEAM GENERATOR
E-101

TROJAN

NOTE:

CONTROL VALVE IS FURNISHED WITH (4) OUTLETS (2 EA. SIDE) TO ALLOW MOUNTING (2) OR MORE SUPPRESSORS IN PARALLEL.

THE OUTLETS ARE PROVIDED WITH CHECKS, SPRING BIASED NORMALLY OPEN.
REF. - A1, A2, B1 & B2.

SEE FIG. 4a, PAGE 12a FOR TYPICAL CONTROL CIRCUIT.

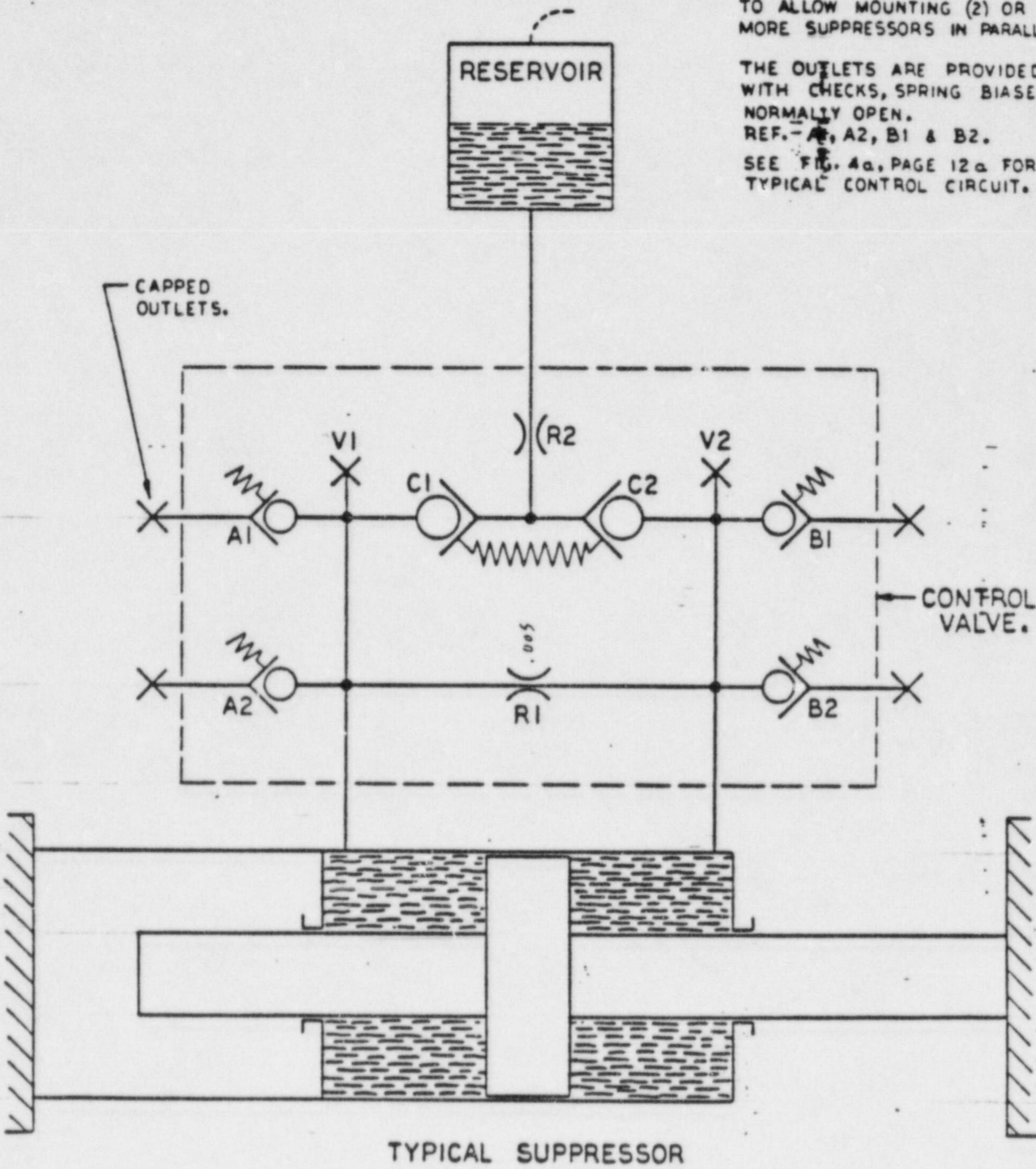


FIG. 4. SCHEMATIC DIAGRAM OF CONTROL VALVE 900K SUPPRESSOR

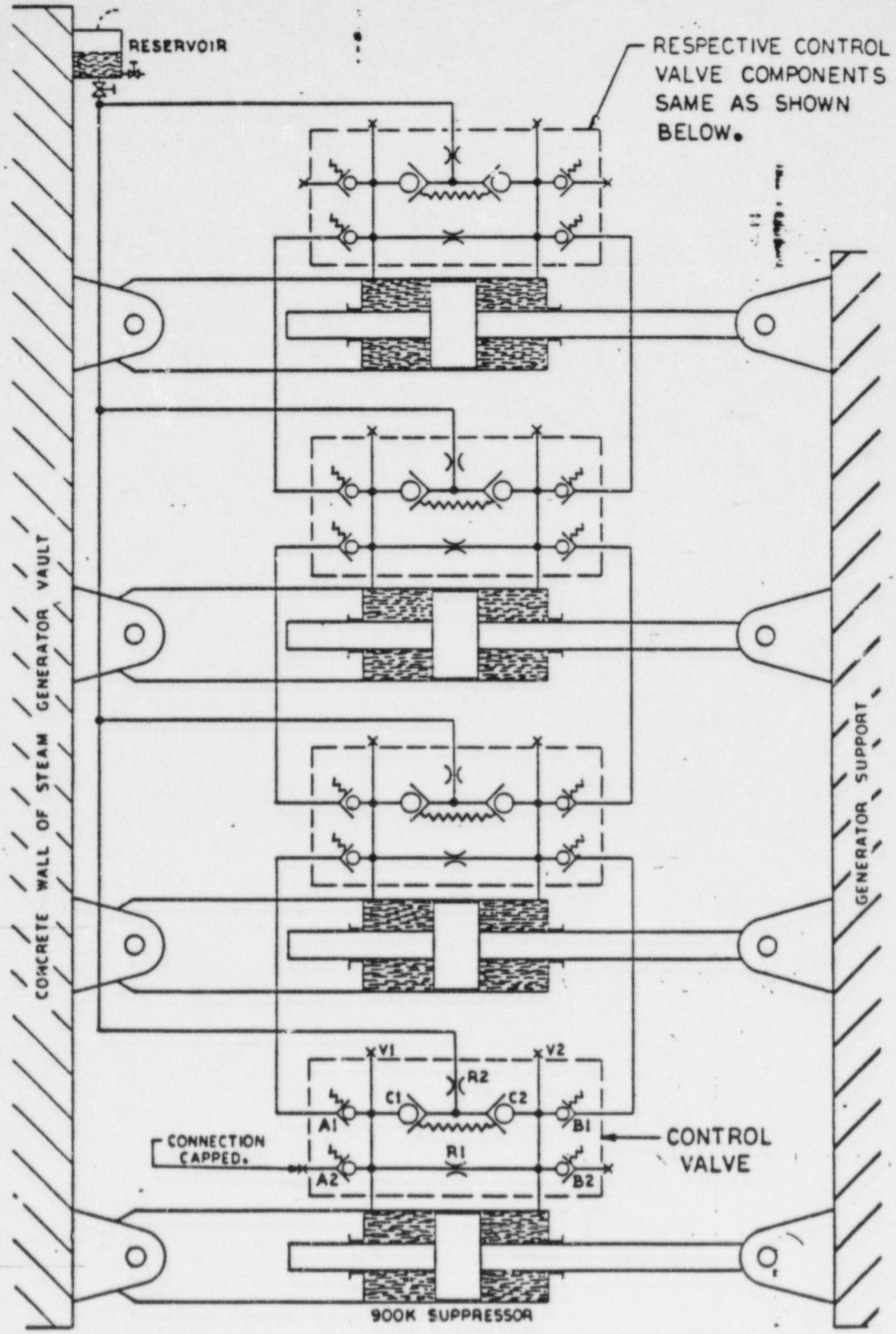


FIG. 4a. SCHEMATIC DIAGRAM OF CONTROL CIRCUIT

ACRS MEETING

ON

NUREG-0956 FINAL REPORT

JUNE 6, 1986

M. SILBERBERG

J. MITCHELL

OFFICE OF NUCLEAR REGULATORY RESEARCH

APPENDIX XVII

NUREG-0956 - FINAL REPORT STAFF BRIEFIN

MAJOR CHANGES IN NUREG-0956

1. Added Technical Information
2. Upgraded Code Suite
3. Performed New Sequence Analyses
4. Removed Material On Risk and Containment
5. Added Discussion of Public Comments
6. Reflected Revised Severe Accident Research Plan
7. Improved the Statement of Conclusions

Table ES.1 Major advances in source term technology since the WASH-1400 Reactor Safety Study

Area of Improvement
1. Treatment of chemical forms of iodine and other fission products
2. Mechanistic analysis of fission product retention in reactor coolant system
3. Improved data base for in-vessel melt progression, hydrogen generation, and control rod behavior
4. Mechanistic treatment of aerosol behavior in containment, including the effects of suppression pools and ice compartments
5. Greatly enlarged data base for in-vessel fission product release from fuel
6. Data base and mechanistic treatment of core-concrete interaction and related radionuclide release
7. Improved models for analysis of containment pressure loads

UNIT
NUCLEAR REGU
ADVISORY COMMITTEI
WASHING

December 12, 1985

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON NUREG-0956, "REASSESSMENT OF THE TECHNICAL BASES FOR ESTIMATING SOURCE TERMS -- DRAFT REPORT FOR COMMENT"

During its 306th meeting, October 10-12, 1985, the Advisory Committee on Reactor Safeguards discussed NUREG-0956 with representatives of the NRC Staff, and we completed our deliberations during the 308th meeting, December 5-7, 1985. This report had previously been reviewed by a Subcommittee in meetings on May 2, August 1 and 2, and September 27, 1985. We also had the benefit of the documents referenced in 1-5 and discussed the report in Reference 6. 1

We conclude that:

(1) Although the report is a useful description of progress that has been made in the NRC's Severe Accident Research Program, it provides only a part of the information likely to be needed in deciding whether and how to restructure existing regulations to deal with accidents beyond the current design basis accidents. 2

(2) Since much of the motivation for the severe accident research program came from observations made after the TMI-2 accident, some of which led several investigators to conclude that source terms previously used to describe severe accident consequences were much too large, we believe the report should either state that information developed to date indicates a significant difference compared to the predictions of WASH-1400, or that no significant difference is now believed to exist. The report is ambiguous on this point. 3

(3) The report is cast in a framework which depends on the use of a suite of codes to describe the course of severe accidents. Reference is made to the considerable uncertainty that exists in the results that the codes predict. No guidance is given as to how to take this uncertainty into account in making decisions related to licensing or regulation. Since dealing with this uncertainty is one of the more difficult parts of the decision process, more attention needs to be given to approaches for dealing with it. 4

(4) The suite of codes that forms much of the basis for the report deals with containment in a rather preliminary fashion. It 5

appears to us that a much less ambiguous method for taking account of containment performance is needed, especially in light of the wide variety of containment types that exist in operating plants.

(5) Many of the phenomena and the processes described in this report have also been studied in some detail by those responsible for the Industry Degraded Core Rulemaking (IDCOR) Program. It would be valuable, in considering the results of and the conclusions drawn from NRC's research programs, to have some discussion of the differences and the similarities of the conclusions reached by the IDCOR group compared to those of this report. 6

Additional comments on these points and other features of the report are given in what follows. 7

It was recognized, following the TMI-2 accident, that more attention must be given to the risk posed by accidents beyond what were then being considered as design basis accidents. It was also known that new information and new understanding had been developed since the publication of WASH-1400, Reactor Safety Study. Accordingly, the NRC Staff undertook to collect, evaluate, and publish in NUREG-0772, Technical Bases for Estimating Fission Product Behavior, the best information then available concerning fission product release and transport during and following a severe core damaging accident. 8

On the basis of that collection, and of an evaluation of the information that would be needed by the NRC Staff as it prepared to deal with the severe accident issue, the Office of Nuclear Regulatory Research (RES) formulated a research program aimed at improving the accuracy with which the source term could be predicted. NUREG-0956 reports the results of that research. 9

The report was described by staff members of the RES Office as containing the scientific bases from which source term calculations could be made. It places major emphasis on the assembly of a set of computer codes which have been used for computing source terms for five reference plants. Several steps were taken to improve the validity of the codes: a validation study of the constituent computer codes done at the Oak Ridge National Laboratory, a quantitative uncertainty study performed by the Sandia National Laboratories, and an independent review of the results of the NRC's source term research by a study group of the American Physical Society. 10

Much of the research that forms the basis for this report was stimulated by the investigations associated with the TMI-2 accident. Several investigators concluded, primarily as a result of the radioactive iodine estimated to have been released to the containment atmosphere during the accident, that the source terms calculated and used in WASH-1400 were much 11 larger than should be expected if one considered the release and transport of fission products in the light of a more careful investigation of the chemistry and the physics of the various processes involved.

Many of those who concluded that the source terms used in the WASH-1400 calculations were too large also predicted that when more appropriately chosen source terms were used, the calculated risk from severe accidents could be shown to be several orders of magnitude smaller than those previously calculated. One might therefore have expected this report to contain some conclusions concerning the risks to be expected when this newly developed set of codes, incorporating the new data resulting from an extensive research program, are applied to the analysis of severe reactor accidents. Comments in the report on this question are at best tentative. For example, the report notes in the section on Risk Insights that a "comparative risk appraisal" (using WASH-1400 accident frequencies, but source terms calculated from the new set of codes) indicates a reduction in risk. The report concludes that the reduction (early fatalities are about a factor of ten lower -- delayed, about a factor of four) is about equally divided between that resulting from a change in the treatment of fission product release and transport, and that resulting from a different approach to describing containment behavior. In other cases the comments are ambiguous. For example, "New source terms have been calculated for selected accident sequences for five reference plants that represent major reactor containment types in operation in the United States. These selected sequences have provided a sufficient test of the capabilities of the computer codes." What was the "sufficient test"? How was the adequacy of the codes developed? One attempting to judge the merits of the code set, or to ascertain whether the risks predicted in light of the new information that has been developed are indeed smaller, would find more information helpful. 12 a b

On the basis of our examination of the report, and of our extensive discussions with the Staff, we conclude that the report can best be characterized as a status report for a task well begun but far from conclusion. 13

In our efforts to evaluate the adequacy of the report we repeatedly raised the question of how and for what purpose the material in the report will be used. Several possible applications were mentioned, but we were told that details of usage will be developed by those who are to use it -- that this report contains primarily the science that has been developed, and not its application. This response is understandable, given the compartmentalization of the Staff that exists, but as a result, as uses are developed, questions will arise that are likely to require further investigation or additional explication of the material that has been gathered. We commend for consideration of the Staff the proposition that applied research is not completed until it is used. 14

We conclude that this report, and further investigations likely to be required in order to make its application to severe accident analysis feasible, can be understood only as part of a package made up of several identifiable components. This report is one of those. It includes, or refers to, the new information that has been developed concerning fission product release from fuel (both in and outside of the vessel), and its transport into containment. It also incorporates the suite of codes developed (as described in BMI-2104) for modelling the course of severe accident sequences following the onset of core damage. 15

The new risk calculations to be carried out for six selected plants and to be reported in NUREG-1150, Risk Perspectives and Rebaselining, form another component. The accident sequence initiator frequencies to be used in this set of calculations will presumably come from the Accident Sequence Evaluation Program. Presumably the modelling of containment performance to be used in the calculations will come from the Severe Accident Risk Reduction Program, although this is not clear. 16

The incorporation, yet to be accomplished, into one coherent method, of the various approaches being developed to describe containment performance is another, and an extremely important component. The formulation of methods for carrying out a detailed severe accident analysis for each operating plant, cited in the Severe Accident Policy Statement, is another. 17

Judged in this context we believe the report is a useful addition to the earlier information on fission product release and transport, and to the methods that have been used in the past to model the behavior and the consequences of severe accidents. However, we conclude that the codes, in their present form, should not be given much weight in making decisions. 18

For example, the report observes that considerable uncertainty exists in the results to be expected when the constituent codes are employed. Reference is made to further work to be done in defining uncertainties. However, no guidance is given to the prospective user on how to account for or how to deal with uncertainties. Nor is there any comment on whether the uncertainty to be expected from employment of the suggested new approach is greater than or less than that which might be expected if, say, the WASH-1400 approach is used. More information on the effects of the identified uncertainties is needed. Guidance on how to deal with existing uncertainties should be provided if the results of the report are to be used for making decisions. Furthermore, the description given to us by the Staff, of work which is planned to provide more nearly quantitative estimates of uncertainty, leads us to believe that what is proposed would be better described as a sensitivity analysis. 19

One of the "Source Term Insights" given in the report is that, "For most accident sequences, the largest single factor affecting source terms is containment behavior. A delay of several hours in containment failure 20

746

will reduce source terms significantly." We agree with both statements. However, the guidance on containment behavior modelling is confusing. Appendix A gives some general discussion of containment types, and of their behavior in accident situations. Appendix B claims to be a summary of a Sandia National Laboratories' report which treats "Containment Event Analysis." It is intended to "provide a containment matrix for the risk perspective for the Surry plant and to discuss the containment behavior of the other plants analyzed" in BMI-2104. However, the discussion and the conclusions are laced with caveats, and the reader is warned that the evaluations are preliminary. The material in Appendix B also seems to be at variance with other NRC work related to containment behavior. For example, in Appendix B, in several places, there is reference to in-vessel steam explosions in a context which indicates that they are thought by the Staff to be a possible significant contributor to the likelihood of early containment failure. However, the report of a review by the Steam Explosion Review Group convened by the NRC Staff (NUREG-1116) indicates a consensus that the likelihood of early containment rupture caused by in-vessel steam explosion is so low as to be negligible. There is also a comment in NUREG-0956 indicating that steam generator tube rupture may be an important containment bypass mechanism. No guidance is given as to how to deal with it. We conclude that in light of the importance attributed to containment system performance, and in view of the preliminary status of current models, much more work is needed in this area. We emphasize, as we have in other comments on methods for severe accident analysis and decision making, that development of more elaborate computer codes is not the only way or even necessarily the best way to proceed. Some well defined method for describing containment behavior is needed.

Bearing in mind that early comments concerning the contribution of iodine gave impetus to much of the research on fission product chemistry that has occurred, and observing that the report points to better fission product chemistry as one of the major improvements that has been produced, we asked what changes in risk could be identified as a result of the changes in the way iodine is treated. We were told that the Staff had not attempted to identify these changes. We suggest that, especially in light of the designation of the report as a scientific document, efforts to identify the changes in risk due to differences in the treatment of a few key contributors would add considerably to the understanding and to the utility of the results. We also believe it would be valuable to identify and to discuss areas of agreement and of disagreement (with more discussion of the latter) between the Source Term Package reported upon here and other relevant work, the IDCOR approach, for example. 21 a b

There are several key areas in the modelling of severe accident progression as described in the report, about which we have some reservations. The transport and the retention of radionuclides in the primary system are tightly coupled to the temperature distribution in the primary system. This in turn is likely to be a strong function of the buoyancy driven recirculation in the primary system. This phenomenon is not 22 treated in the models described in NUREG-0956. Work by other groups suggests that it could have an important bearing on temperatures in the primary system. For example, some investigators have suggested that it might lead to transport and condensation of fission products in the steam generator tubes sufficient to produce tube rupture in some postulated high pressure accident sequences. It is also predicted by some that this mechanism may lead to a sufficiently high temperature of the upper level components of the primary system in PWRs, that rupture will occur, in high pressure sequences, before, for example, the postulated expulsion of molten core material from the bottom of the reactor vessel, leading to severe containment atmosphere heating, occurs. This possibly important mode of heat transfer deserves further investigation.

Fission product release from the fuel is highly temperature dependent. Core melt progression and core melt temperatures are based on the MARCH code. Even in its present form, the code provides only a crude representation of the physical processes it is meant to predict. As a result, the molten core temperature is subject to considerable uncertainty. This uncertainty is reflected in calculations of fission product release. Better understanding of the resultant uncertainties is needed. 23

Ex-vessel release of fission products from the melt is strongly dependent upon the melt temperature, and this in turn is highly dependent on the core-concrete interaction. Some investigators interpret the results of the Beta tests at the Karlsruhe Nuclear Research Center (Federal Republic of Germany) to indicate that the heat transfer to concrete is higher than that predicted by the code used to model the core-concrete interaction in this package. Because much of the fission product release following late containment failure is currently calculated to come from the nonvolatiles released during core-concrete interactions, this possible discrepancy deserves further investigation. 24

The report is based upon work described in a large number of documents, some not readily available. Because of the importance of a thorough understanding of the bases of the results reported and conclusions drawn, it is vital that care be taken to identify the documents to which a user can go to obtain further information. We emphasize the importance of complete documentation of the foundation reports from which NUREG-0956 is drawn. 25

We have commented in a letter to the Executive Director for Operations, dated August 13, 1985, that we believe the representative risk calculations to be carried out and to be reported in NUREG-1150, as well as the methods developed for analysis of individual plants, should take account of external initiators. 26

We express our appreciation to the Staff for providing us with thorough, well organized presentations on this report, and for their efforts in 27 responding to a number of questions which we posed during the course of our discussions.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Reassessment of the Technical Bases for Estimating Source Terms - Draft Report for Comment," USNRC Report NUREG-0956, dated July 1985
2. U. S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," USNRC Report WASH-1400 (NUREG-75/104), dated October 1975
3. U. S. Nuclear Regulatory Commission, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," USNRC Report NUREG-0772, dated June 1981
4. Battelle Columbus, "Radionuclide Release Under Specific LWR Accident Conditions," Vols. I-VII, BMI-2104, dated July 1983 - February 1985
5. U. S. Nuclear Regulatory Commission, "A Review of the Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions," USNRC Report NUREG-1116, dated June 1985
6. U. S. Nuclear Regulatory Commission, "Risk Perspectives and Rebaselining for Six Reference Plants," USNRC Report NUREG-1150, to be published

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' ___

1. Testimony of H.W. Lewis, Subcommittee on Energy and the Environment, House Interior Committee, June 10, 1986

2. Letter, J.C. Ebersole, ACRS Chairman to N.J. Palladino, NRC Chairman, ACRS Report on the Hope Creek Generating Station, December 18, 1985.

3. NUREG-0979, Supplement No. 5, Safety Evaluation Report related to the final design approval of the GESSAR II BWR/6 Nuclear Island Design, May 1986.