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MARCH 4-6, 1982
WASHINGTON, DC

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

February 23, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
263RD ACRS MEETING
MARCH 4-6, 1982
WASHINGTON, DC

Thursday, March 4, 1982, Room 1046, 1717 H Street, NW, Washington, DC

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

ACRS Chairman's Report (Open)
1.1) Opening Statement
1.2) Items of Current Interest

2) 8:45 A.M. - 12:00 Noon

Byron Station Units 1 and 2 (Open)
2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee (PGS/EI)
2.2) 9:15 A.M.-12:00 Noon: Presentations by and discussions with NRC Staff and applicant regarding an operating license for this station.
(Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.)

3) 12:00 Noon - 12:30 P.M.

LER Reporting Requirements (Open)
3.1) Report of ACRS Subcommittee meeting on proposed changes in 10 CFR 50.73 (WMM/DWM/RKM)

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

LUNCH

4) 1:30 P.M. - 2:15 P.M.

Quality Assurance Programs at Nuclear Plants (Open) -
4.1) 1:30 P.M. - 2:00 P.M.: Presentation by NRR/I&E representatives regarding improved quality assurance program requirements at nuclear power plants
4.2) 2:00 P.M.-2:15 P.M.: Questions and Discussion

5) 2:15 P.M. - 4:15 P.M.

Waterford Steam Electric Station Unit 3 (Open)
5.1) 2:15 P.M. - 2:35 P.M.: Report of ACRS Subcommittee (DAW/SKB/GRQ) regarding outstanding OL issues regarding this unit

5.2) 2:35 P.M.-4:15 P.M.: Presentations by and discussions with representatives of the NRC Staff and the licensee.
(Portions of this session will be closed as necessary to discuss Proprietary Information related to this project.)

6) 4:15 P.M. - 5:45 P.M.

Qualification Program for Safety Related Equipment (Open)

6.1) 4:15 P.M.-4:45 P.M.: Report of ACRS Subcommittee regarding the proposed NRC program for environmental qualification of safety related equipment (JJR/PAB)

6.2) 4:45 P.M.-5:45 P.M.: Presentations by and discussions with representatives of the NRC Staff regarding proposed requirements for qualification of electrical equipment

7) 5:45 P.M. - 6:30 P M.

ACRS Subcommittee Activities (Open)

7.1) Reports of ACRS Subcommittees regarding:

7.1-1) Indian Point Nuclear Generating Station Unit 3 - proposed plan for evaluation of systems interactions (DO/JMG)

Friday, March 5, 1982, Room 1046, 1717 H Street, NW, Washington, DC

8) 8:30 A.M. - 12:30 P.M. — Marck III Containment Program and the Clinton Units 1 and 2 (Open)

- 8.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee on the MK III Containment Program (MSP/PAB)
- 8.2) 9:00 A.M.-9:30 A.M.: Report of ACRS Subcommittee regarding Clinton Power Station operating license (WK/RS)
- 8.3) 9:30 A.M.-12:30 P.M.: Presentations by and discussions with representatives of the NRC Staff and the applicant

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

12:30 P.M. - 1:30 P.M. — LUNCH

9) 1:30 P.M. - 2:00 P.M. — ACRS Subcommittee Activities (Open)

- 9.1) Zimmer Nuclear Station - quality control deficiencies during construction (MB/GRQ)
- 9.2) Regulatory Activities - proposed changes in NRC Regulatory Guides (CPS/SD)

10) 2:00 P.M. - 2:45 P.M. — Alternate Materials for Waste Disposal Containers (Closed)

- 10.1) 2:00 P.M.-2:30 P.M.: Report of ACRS Subcommittee (PGS/EI) and consultants regarding the contractor recommended and the program proposed for evaluation of alternate materials for radioactive waste disposal containers.
- 10.2) 2:30 P.M.-2:45 P.M.: Discussion with representatives of the NRC Staff

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

11) 2:45 P.M. - 3:45 P.M..

ACRS and ACRS Subcommittee Activities (Open)

11.1) Report of ACRS Subcommittee on seismic research applicable to the east coast of the United States (DO/RS)

11.2) Report regarding testimony before the House Subcommittee on Energy and the Environment (PGS/CPS/RFF)

11.3) Future ACRS Activities

11.3-1) Anticipated ACRS Subcommittee activities

11.3-2) Proposed ACRS activities

11.4) ACRS Report on Reactor Safety Research - comments regarding unusual operational problems following a severe earthquake (DO/SD)

12) 3:30 P.M. - 4:00 P.M.

Reactor Pressure Vessel Water Level Indicators (Open)

12.1) Briefing by ACRS Staff member regarding performance of differential pressure cells as level measuring devices (JE/JAM)

13) 4:00 P.M. - 5:30 P.M.

Proposed ACRS Reports to NRC (Open/Closed)

13.1) Discuss proposed ACRS reports to NRC on:

13.1-1) Byron Station Units 1 & 2

13.1-2) Waterford Station Unit 3

13.1-3) Qualification Programs for Safety Related Equipment

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

Saturday, March 6, 1982, Room 1046, 1717 H Street, NW, Washington, DC

14) 8:30 P.M. - 10:30 A.M.

ACRS Reports to NRC (Open/Closed)

14.1) Discuss proposed ACRS reports to NRC regarding:

- 14.1-1) Clinton Station Units 1 and 2
- 14.1-2) Program and contractor for alternate waste disposal package materials
- 14.1-3) Byron Station Units 1 & 2
- 14.1-4) Waterford Station Unit 3

(Portions of this session will be closed as necessary to discuss Proprietary Information and information that will be involved in an adjudicatory proceeding.)

15) 10:30 A.M. - 11:30 A.M.

Miscellaneous ACRS Activities (Open/Closed)

15.1) Discuss ACRS activities related to:

- 15.1-1) Appointment of a new ACRS member
- 15.1-2) Development of improved SER's and SAR's for support of ACRS activities

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

16) 11:30 A.M. - 12:30 P.M.

ACRS Reports to NRC (Open/Closed)

16.1) Complete preparation of ACRS reports to NRC regarding items discussed during this meeting.

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

consider ways to evaluate the program further before agreeing to an action resulting in definitive operational guidance. *M-81-62.* Concurs. *M-81-63.* Concurs.

Amtrak, Feb. 10, R-78-39. Has completed a survey of all equipment operated over Amtrak lines in the Northeast Corridor to determine the extent of use of cab signal and automatic train control among this equipment. Amtrak NEC Timetable Order No. 1562-A1 was issued specifically covering the operation of non-Amtrak vehicles in the NEC which are not equipped with ATC apparatus. Service in the NEC would be disrupted significantly were Amtrak to further prohibit the operation of vehicles which are not equipped with ATC.

Note.—Single copies of reports, recommendation letters, and responses are free on written request, identified by recommendation or report number, to: Public Inquiries Section, National Transportation Safety Board, Washington, D.C. 20594. (Multiple copies of reports are obtainable from National Technical Information Service, U.S. Department of Commerce, Springfield, Va. 22161.)

H. Ray Smith, Jr.,

Alternate Federal Register Liaison Officer.

February 19, 1982.

[FR Doc. 82-4990 Filed 2-24-82; 8:45 am]

BILLING CODE 4910-58-M

NUCLEAR REGULATORY COMMISSION

[redacted]

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on March 4-6, 1982, in Room 1046, 1717 H Street NW., Washington, DC. Notice of this meeting was published in the Federal Register on February 17, 1982.

The agenda for the subject meeting will be as follows:

Thursday, March 4, 1982

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

8:45 a.m.-12:00 Noon: Byron Station Units 1 and 2 (Open).—The Committee will hear the report of its Subcommittee and consultants who may be present regarding the request for an operating license for this nuclear station. The Committee will also hear and discuss presentations by representatives of the NRC Staff and the applicant regarding this matter.

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

12:00 Noon-12:30 p.m.: Licensee Event Reporting System (Open)—The Committee will hear and discuss comments of the ACRS Subcommittee Chairman and members regarding proposed changes in NRC requirements (10 CFR 50.73) for licensee event reports. Representatives of the NRC Staff will participate, as appropriate.

1:30 p.m.-2:15 p.m.: Quality Assurance Programs at Nuclear Plants (Open)—The Committee will hear a presentation by officials of the NRC Staff regarding improvements in quality assurance programs at nuclear power plants.

2:15 p.m.-4:15 p.m.: Waterford Steam Electric Station Unit 3 (Open)—The Committee will hear the report of its Subcommittee regarding outstanding safety related issues applicable to the proposed operation of this unit. Representatives of the NRC Staff and the licensee will participate, as appropriate.

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

4:15 p.m-5:15 p.m.: Qualification Program for Safety Related Equipment (Open)—The Committee will hear the report of its Subcommittee regarding the proposed NRC equipment qualification program for safety related equipment in nuclear power plants. Representatives of the NRC Staff and the nuclear industry will participate, as appropriate.

5:15 p.m.-6:30 p.m.: ACRS Subcommmittee Activities (Open)—The Committee members will hear and discuss the reports of designated ACRS subcommittees regarding safety related matters including the proposed review plan for evaluation of systems interactions at the Indian Point Nuclear Station Unit 3, quality control deficiencies at the Zimmer Nuclear Station, and seismic research applicable to the east coast of the United States.

Friday, March 5, 1982

8:30 a.m.-12:30 p.m.: Clinton Power Station Units 1 and 2 and Mk-III Containment (Open)—The Committee members will hear the reports of its subcommittee chairmen regarding the request for an operating license for the Clinton Power Station Units 1 and 2, and resolution of outstanding questions related to the Mk-III type of dynamic containment.

The Committee will hear and discuss presentations by the NRC Staff and the applicant regarding this matter.

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

1:30 p.m.-2:00 p.m.: ACRS Subcommittee and Full Committee Activities (Open)—The Committee members will hear and discuss the report of its subcommittee chairman regarding proposed changes in NRC regulatory guides. The Committee will also discuss anticipated subcommittee activity and proposed full Committee activities.

2:00 p.m.-2:45 p.m.: Alternate Materials for Waste Disposal Containers (Closed)—The Committee will hear the report of its Subcommittee and consultants who may be present regarding a proposed NRC program and contractor qualifications to evaluate alternate materials for radioactive waste disposal containers. Representatives of the NRC Staff will participate, as appropriate.

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

2:45 p.m.-3:15 p.m.: Reactor Pressure Vessel Liquid Level Indication (Open)—The Committee will hear a presentation from an ACRS staff member regarding the performance of differential pressure cells as water level indicators.

3:15 p.m.-5:30 p.m.: ACRS Reports to NRC (Open/Closed)—The Committee members will discuss proposed reports to the Nuclear Regulatory Commission and comments to the NRC Executive director for Operations regarding topics discussed during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to the topics being considered and information which will be involved in an adjudicatory proceeding.

Saturday, March 6, 1982

8:30 a.m.-10:30 a.m.: ACRS Reports to NRC (Open/Closed)—The Committee members will discuss proposed reports to the Nucler Regulatory Commission and comments to the NRC Executive director for Operations regarding topics discussed during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to the topics being considered and information which will be involved in an adjudicatory proceeding.

10:30 a.m.-11:30 a.m.: Miscellaneous Activities (Open)—The members will discuss miscellaneous topics related to the conduct of ACRS activities including testimony before the U.S. House of Representatives Committee on Interior and Insular Affairs, the format and scope of improved safety analysis reports and safety evaluation reports, and qualilfications for new ACRS members.

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

11:30 a.m.–12:30 p.m. ACRS Reports NRC (Open/Closed)—The Committee members will discuss proposed reports to the Nuclear Regulatory Commission and comments to the NRC Executive director for Operations regarding topics discussed during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information applicable to the topics being considered and information which will be involved in an adjudicatory proceeding.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 30, 1981 (46 FR 47903). 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 statememts should notify the ACRS Executive Director as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements. Use of still, motion picture and television cameras during this meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by a telephone call to the ACRS Executive Director (R. F. Fraley) prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the ACRS Executive Director if such rescheduling would result in major inconvenience.

I have determined in accordance with Subsection 10(d) Pub. L. 92–463 that it is necessary to close portions of this meeting as noted above to discuss information of a personal nature where disclosure would constitute unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)), information which will be involved in an adjudicatory proceeding (5 U.S.C. 552b(c)(10)), and Proprietary Information (5 U.S.C. 552b(c)(4)) applicable to the matters being discused.

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

Dated: February 22, 1982.

John C. Hoyle,

Advisory Committee Management.

[FR Doc. 82–5070 Filed 2–24–82; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–184]

Availability of Draft Environmental Statement for National Bureau of Standards Reactor

Notice of hereby given that a Draft Environmental Statement (NUREG–0877) has been prepared by the Commission's Office of Nuclear Reactor Regulation related to the license renewal and power increase for the National Bureau of Standards (NBS) research reactor. This reactor is located on the 576-acre NBS site near Gaithersburg in Montgomery County, Maryland about 20 miles northwest of the center of Washington, D.C.

This Draft Environmental Statement (DES) addresses the aquatic, terrestrial, radiological, social and economic costs and benefits associated with normal station operation. Also considered are station accidents, their likelihood of occurrence or their consequences. Finally, the statement presents an updated discussion of a need for the facility since the construction permit application.

This DES it available for inspection by the public in the Commission's Public Document Room at 1717 H Street N.W., Washington, D.C. 20555. Requests for copies of the DES (NUREG–0877) should be addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Technical information and Document Control.

Interested persons may submit comments on this DES for the Commission's consideration. Federal, State, and specified local agencies are being provided with copies of the DES (local agencies may obtain these documents upon request).

Comments by Federal, State, and local officials, or other members of the public received by the Commission will be made available for public inspection at the Commission Public Document Room in Washington, D.C.

After consideration of comments submitted with respect to the DES, the Commission's staff will prepare a Final Environmental Statement, the availability of which will be published in the **Federal Register**. Comments are due by April 12, 1982.

Comments on this report from interested members of the public should be addressed to the Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Md. this 18th day of February 1982.

For the Nuclear Regulatory Commission.

James R. Miller,

Chief, Standardization and Special Projects Branch, Division of Licensing.

[FR Doc. 82–5063 Filed 2–24–82; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–320]

Metropolitan Edison Co., Jersey Central Power & Light Co., Pennsylvania Electric Co. and GPU Nuclear Corp; Issuance of Amendment to Facility Operating License

The Nuclear Regulatory Commission (the Commission) has issued Amenmdment No. 19 to Facility Operating License No. DPR–73, issued to GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company (collectively "the licensee"). Operating License No. DPR–73 formerly authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI–2) located in Dauphin County, Pennsylvania, but that authorization was suspended by an Order for Modification of License, limiting the authorization to maintaining the facility in its present safe shutdown condition 44 FR 45271 (August 1, 1979). This amendment effects changes to the Appendix B Technical Specifications attached to and incorporated in License No. DPR–73 by reflecting that the positions of Manager—Generation Engineering and Manager—Operational Quality Assurance no longer exist. In addition, the amendment clarifies the responsibility for review of changes related to Appendix B Technical Specifications and their implementation.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The

Issue Date:

August 30, 1982

CERTIFIED

MINUTES OF THE
263RD ACRS MEETING
MARCH 4-6, 1982
WASHINGTON, DC

The 263rd meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N.W., Washington, DC, was convened by Chairman P. G. Shewmon at 8:30 a.m., Thursday, March 4, 1982.

[Note: For a list of attendees, see Appendix I.]

The Chairman noted the existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act and the Government in the Sunshine Act, Public Laws 92-463 and 94-409, respectively. He noted that no requests had been received from members of the public to present either oral or written statements to the Committee. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC.

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

I. Chairman's Report (Open to Public)

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

The Chairman informed the Committee that provision will be made during the meeting to accommodate requests by the Power Authority of the State of New York (PASNY) to make an oral statement regarding its systems interactions study of the Indian Point Nuclear Generating Unit 3. In addition, the Chairman reported the appearance of himself and C. P. Siess before the House Committee on Interior and Insular Affairs (Congressman M. K. Udall, Chairman) on March 2, 1982; and C. P. Siess on March 3, 1982 before the House Subcommittee on Energy Conservation and Power. Discussion of the Congressional Hearing testimony was scheduled for later in the meeting.

II. Review of Byron Station Units 1 and 2 (Open to Public)

[E. Igne was the Designated Federal Employee for this portion of the meeting.]

A. Subcommittee Reports

P. G. Shewmon, Chairman of the Byron Subcommittee, indicated that the presentation at this meeting will cover principally issues such as the difference of opinion over a fire plan and the discussion of the size of a sink hole under the water supply pipes that come from the river a couple of miles from the plant. An issue was raised as to whether the Braidwood Plant would be considered in the review and subsequent Committee report. Mentioned was the utility's favorable attitude toward accommodation of the new hydrogen rule.

B. NRC Presentation

S. Chestnut, NRC Licensing Project Manager, defined the scope of the presentation and discussion of the NRC safety review as contained in the Safety Evaluation Report, NUREG-0876 published in February, 1982.

After listing the principal review matters considered by the Staff, design similarities between the Byron Station and the Zion Station, Commanche Peak, D. C. Cook, and Indian Point plants were pointed out (see Appendix IV). The waste management system at Byron was described as similar to that at McGuire and Watts Bar. D. W. Moeller questioned why Watts Bar has twice as many of gas decay tanks as Byron. The Committee discussed the matter, but the Staff was unable to present a satisfactory explanation. J. Ebersole questioned why the Byron Plant has loop stop valves while other plants do not. He noted the potential problem involving protection against pumps running within a closed pair of loop isolation valves. After discussion by the Staff and Applicant, J. Ebersole suggested that the Staff investigate the potential for failure of pump casings when pumps are inadvertently running between two closed loop isolation valves and report back to the ACRS on this matter.

S. Chestnut indicated that the Staff review resulted in identification of 17 open items, only two of which have an element of disagreement between the Staff and Applicant. The first item concerned the pipeline foundation information which the Staff is still reviewing. The Staff is reviewing the pipeline design as well as the soil foundation support sections. The Staff will require additional information in the form of drillings to determine the exact nature of the soil and bedrock underneath portions of the pipeline. C. Mark questioned whether the location of the new Madrid Fault source could affect the seismic input for the Byron site. The staff was unable to answer the question. The Byron design for inadequate core cooling instrumentation was discussed. C. Mark inquired whether heated junction thermocouples, one of the three basic features of their system, would be acceptable to the Staff if installed. R. L. Tedesco, NRC Staff, explained that the issue is being handled on a generic basis to consider all of the vendor's proposals for this issue. S. Chestnut believed that the system would be improved as a result of this generic review.

The Fire Protection Program was discussed - one of two items where resolution between the Applicant and the Staff was not complete. There were two basic issues:

- Location of a fixed water suppression system in the cable spreading rooms, and

- Oil collection system for the reactor coolant pumps

C. Mark pointed out that putting water on electrical fires was not allowed before the Browns Ferry accident and now appears to be a requirement whether needed or not in an electrical fire. V. Benaroya, NRC Staff, explained that fire protection programs are evaluated more thoroughly than they used to be and it has been found that water is one of the cheapest and most effective agents for removing heat from a fire. J. Ebersole pointed out two reasons for nonuse of water in electrical fires: electrical shock from high voltage equipment; and the fact that electrical apparatus will stop functioning. He pointed out that a critical aspect to be considered is whether the plant has remote control capability of shutting the plant down irrespective of the loss of the spreading room or control room. J. Ebersole questioned whether the Byron plant had a remote shutdown center that could tolerate the complete obliteration of the control room and spreading rooms. V. Benaroya indicated that the Byron Plant had an alternate shutdown system in case of loss of the spreading room and control room. J. T. Westermeier, Commonwealth Edison, explained that Byron plant has an automatic Halon system, manual CO_2 system and manual standpipe water besides fire extinguishers in the upper cable spreading room, and an automatic CO_2 system and manual standpipe water with extinguishers in the lower spreading room. Commonwealth Edison noted that it considered this a "very adequate system".

S. Chestnut indicated that Appendix R requires an oil collection system for reactor coolant pumps such that in the event of a leak of the lubricating system, the oil would be collected and diverted away from areas with hot pipes or materials which could generate a fire. He indicated that the Applicant's position is that they would not like to have an oil collection system due to their operating experience at Zion where it adds a significant cost to maintenance and ALARA problems. Commonwealth Edison maintained that there is not much of a fire hazard to the reactor coolant pumps. The Staff position, based on operating experience at other plants, is that oil leaking from the oil system on the reactor coolant pumps has been involved in some fires. S. Chestnut indicated that the Staff is willing to listen to the Applicant's concerns regarding ALARA and maintenance problems.

W. L. Stiede, Assistant Vice-President for Commonwealth Edison, commented on two open items - fire protection and solution basins.

W. L. Stiede indicated that new materials have been used to jacket cables in the upper and lower spreading rooms. This material is hypolon, a material which begins to burn at a higher temperature than previous jacket materials. There is concern for water damage as a result of a false trip of an automated fire protection system which would cause significant down time and cleanup cost. Therefore, the Byron Plant relies on halon as the primary system with a manual CO_2 backup system in the upper cable spreading room. With respect to oil as a problem, W. L. Stiede indicated that the reactor coolant piping and the pump bowls are covered with a reflective insulation material whose outside surface temperature is such that it is below the combustion or flash point of the oil. He indicated more concern for the potential 20 man-rems of exposure per year to maintain the oil collection system than the small chance of flashing caused by oil leaks.

With regard to the solution basins (i.e., sink holes) issue, W. L. Stiede indicated that their investigations showed that the chance of a solution basin greater than 50 feet in diameter is extremely unlikely. He noted that the latest set of questions from J. T. Chen of the NRC Staff has been received and will be examined and answered.

C. Mark asked the Applicant to clarify a statement read into the record at the subcommittee meetirg that bentonite was used in the grouting operation at Byron. J. T. Westermeier indicated that bentonite was used in potable water well drilling only and not in the grouting operations.

M. Bender asked V. Benaroya for a clarification about the Appendix R requirements for sprinklers. V. Benaroya indicated that if three hour barriers exist between divisions, there is not a specification for sprinklers. Sprinklers are specified where a high concentration of burnable materials like cables exists. V. Benaroya discussed the use of hypolon in fire mitigation and transient combustibles as potential sources of fires.

C. Presentation by Commonwealth Edison

1. Brief Discription of Corporate Organization

The Committee determined that the organization at Byron is comparable to that of other Commonwealth Edison nuclear plants. B. Querio, Plant Superintendent at Byron, presented a chart showing experience levels of the station staff in terms of years of commercial nuclear experience, years of military nuclear experience, as well as fossil related experience. When it was determined that most of the people had experience at other Commonwealth Edison nuclear plants, further discussion was waived.

B. Querio did note that the staffing level planned was 477 individuals, in answer to an inquiry by M. Bender. He indicated that the Byron Plant was staffed by initial hiring into a common work pool for bargaining unit employees.

2. Training

Commonwealth Edison chose to develop a job position entitled Station Control Room Engineer (SCRE) to meet the NUREG-0737 requirement for Shift Technical Advisor (see Appendix V, page 4 Training Section). In response to a question by D. W. Moeller, B. Querio explained that the abnormal operating events portion of the SCRE training also includes evaluation of LERs. R. E. Joetberg, Director of Nuclear Safety for Commonwealth Edison, described the process of inhouse review of LERs on site. He indicated that there are 12 engineers engaged full time in screening LERs. A centralized site specific specialized Byron training program for maintenance training was described. This program was divided into mechanical, electrical instrumentation and welding skills training. M. Bender asked how the Byron training program compares with the minimum requirements of INPO of which Commonwealth Edison is a participating member. T. Higgins, Commonwealth Edison, indicated that the Byron training program far exceeds the minimum requirements of INPO.

3. Secondary Water Chemistry

B. Querio indicated that the Byron Station, which has a closed condenser cooling system with natural draft cooling towers, must operate with low condenser leakage and have the capability to quickly identify, respond to, and recover from a condenser leak. He added that the Byron Station has established a policy for operation of the secondary system. This policy is to implement an aggressive program aimed at minimizing the quantity of corrodents and corrosion products transported through the secondary system to the steam generators. The long-term effects of this policy will be primarily to reduce secondary steam generator leakage and minimize personnel radiation exposure. This program is based primarily on successful operation of the Zion plant. Three action levels were defined for taking remedial action when monitored parameters are observed and confirmed to be outside the normal operating values consistent with long-term system reliability. M. Bender questioned the experience at Zion with the plugging of steam generator tubes. R. E. Shannon, of the Nuclear Safety Division and Maintenance Department of Commonwealth Edison, indicated that some 25 tubes out of 12,000 have been plugged after 8 years of operation. In answer to a question by P. G. Shewmon, R. E. Shannon indicated that the majority of the tubes were in the first or second row. He said that this condition had been identified as a generic problem.

4. D.C. Power Systems Reliability

B. Querio discussed the d.c. power system at Byron and the station response to a loss of all a.c. power. The Committee had several informational questions concerning the self-contained battery systems. J. J. Ray questioned whether there was a policy for line restoration system-wide to guide the system operator as to the priority for restoration of power to the nuclear stations.

W. L. Stiede indicated that there was an emergency restoration procedure for load dispatchers at Glen Bar, but he could not tell specifically that there was a procedure that stated that nuclear stations would have power restored first in the case of a loss of a portion of the grid. J. J. Ray suggested that CE evaluate its policy for restoration of power following a blackout, with priority to power restoration at its nuclear stations.

B. Querio stated that CE does not consider a complete loss of a.c. power a credible accident or a design basis accident. There are, however, design capabilities and adequate procedures to cope with and recover from such an event. He described four major categories of actions: restoration of 5500 kv diesel generators; protection of the reactor core; protection of plant equipment; and preparation for final plant recovery. He indicated that these actions are performed concurrently. In the event of total loss of a.c. power, the control room operator takes certain actions to isolate the reactor coolant system to minimize inventory loss and keep the core covered for at least 10 hours, assuming cooldown depressurization and possible degraded reactor coolant pump seals.

L. Bowen of the Byron Engineering Staff, in answer to a question about diesel reliability, indicated that for the two auxiliary feedwater pumps on each Byron Unit, one is motor-driven and one is diesel-driven with the diesel-driven pump completely a.c. independent, including the cooling systems.

5. Probabilistic Risk Assessment and Systems Interactions Studies

L. Bowen pointed out that the Zion Station Probabilistic Risk Assessment Study had been completed and submitted for review to the NRC. He indicated that since the Zion and Byron designs share the same strong design characteristics, Commonwealth Edison is not planning to conduct a Byron PRA study.

L. Bowen explained that Commonwealth Edison had identified potential series of interactions during the course of design, construction, and licensing. The study of these interactions concerned the equipment necessary to put the plant in a safe shutdown condition or to mitigate an accident. L. Bowen indicated that Westinghouse had identified four systems that would potentially lead to control system faulty operation. Only these four potential systems interactions were reviewed for the Byron plant, and the conclusion reached was that those interactions did not pose a problem to the plant. D. Okrent questioned whether the study of systems interactions was done on a limited basis in other areas. L. Bowen indicated that it was done comprehensively, as in the area of fire evaluation, with regard to safe shutdown, the effects of flood or the effects of system pipe breaks on a generic basis. He explained that it was the area of control systems interactions that was a limited scope interactions study. L. Bowen added that Commonwealth Edison looked at some specific common caused events (see Appendix V, Agenda Item 3.4, slide 2).

D. Okrent expressed the belief that PRA analyses should be done comprehensively and not only to buttress or support a particular applicant position. He expressed interest in the depth of the systems interactions study in the seismic area. L. Bowen indicated the depth of review to be in-office evaluation of drawings of non-safety related seismic items that could adversely impact pieces of equipment that are required for safe shutdown. He indicated that Byron engineers are looking more at a target of physical impact in the seismic area.

6. Auxiliary Feedwater Reliability

L. Bowen indicated that when Commonwealth Edison evaluated the reliability characteristics of the auxiliary feedwater system, using the methodology data of NUREG-0611, a problem occurred in the area of main feedwater, in the loss of offsite power case. Byron had to take credit for a cross tie between the units to provide an unavailability number low enough to meet the NRC Staff criteria which was publisned in the recent edition of the Standard Review Plan.

7. Emergency Planning

P. G. Shewmon pointed out that the controlling factor in the area of Emergency Planning is the State and County and their limited staff capabilities to deal with events at operating plants before considering non-operating plants such as Byron. Therefore, the discussion of Emergency Planning was deferred.

8. Defective Residual Elements on Reactor Pressure Vessels and Concerns With High Strength Bolts

J. T. Westermeier, Commonwealth Edison, indicated that the Bryon plant has high strength bolting materials in the NSSS component supports, in pipe whip restraints, and certain equipment holddown bolting materials. These were tested to ASTM requirements per the material specifications. P. G. Shewmon and J. T. Westermeier discussed the basic testing requirements for certification of these bolts. R. J. Netzel, Sargent and Lundy, indicated that Charpy tests are performed on the bolts to determine adequate fracture toughness but not yield strength. He indicated that tensile testing is required to determine yield strength. P. G. Shewmon questioned why a tensile test was not conducted on the bolts. R. J. Netzel indicated that they rely on the supplier to provide bolts that conform to the hardness of the ASTM specifications. H. Silver, NRC Staff, reported that the Lawrence Livermore Laboratory has performed a bolting study, NUREG-2467, "Lower-Bound K_{Iscc} Values for Bolting Materials - Alternative Study", which has had limited distribution. Shewmon requested a copy of this bolting study. R. J. Netzel indicated that the steam generators were held down by bolts with a material designation of SA-540 and a tensile strength of 165 ksi.

9. Inadequate Core Cooling

J. A. Ainger, Commonwealth Edison, explained that the Inadequate Core Cooling Instrumentation System at Byron is a three element system consisting of subcooled margin indication, a reactor vessel water level indicator, and core exit thermocouples. J. Ebersole questioned why Commonwealth Edison had chosen Combustion Engineering level detectors instead of a Westinghouse system. He questioned why they preferred using these heated junction thermocouples.

J. A. Ainger explained that the alternate Westinghouse system had considerable electronics associated with temperature compensation that made it complicated from a maintenance point of view, more complicated than the Combustion Engineering system. H. W. Lewis pointed out that, in his opinion, the system will not measure core inventory. He questioned whether Commonwealth Edison had warnings to the operators in its emergency procedures concerning the fact that under conditions in which there is voiding in the system, these instruments could not be believed. W. L. Stiede indicated that the matter of cautions to operators is under consideration; that Commonwealth Edison is closely reviewing the statements made in Combustion Engineering and Westinghouse Owners Group presentations being made to the ACRS; and will incorporate cautions for proper use of the instruments, if necessary. C. Mark questioned whether the Applicant thought he needed this system. J. A. Ainger indicated that the system was installed to comply with NRC licensing requirements.

10. Concluding Comments by Commonwealth Edison

W. L. Stiede explained, in answer to a previous question from J. J. Ray regarding loss of off-site power, that the first priority on the load dispatcher's list is to maintain the high voltage system intact, and then as a next priority, to restore the power to the nuclear stations. D. W. Moeller asked a question about air cleaning and ventilating systems for the control room. R. C. Ward of Commonwealth Edison indicated that the initial preoperational test program at the site involves tests by components, and then integrated tests covering the system. W. L. Stiede asked the ACRS to address both the Byron and Braidwood units in its report recognizing the differences in the sites which would affect emergency plans. P. G. Shewmon spoke for a consensus of the Committee, that the ACRS could write a letter on Byron, but would like to leave the Braidwood discussion for a later date.

III. Waterford Steam Electric Unit 3 (Open to Public)

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

A. Report of ACRS Subcommittee

D. A. Ward indicated that it was the consensus of the Subcommittee that Louisiana Power and Light (LP&L) had been very responsive to the concerns of the ACRS as expressed in its interim letter of August 11, 1981 regarding orgainizational readiness to operate Waterford 3. He added that this consensus was tempered by some concern that LP&L plans to continue to develop their organization as they approach startup.

It was indicated that the Staff was still formulating requirements and guidelines concerning the issue of feed and bleed for cooling of the Waterford Plant and other similar Combustion Engineering plants. Staff PRA studies by the Office of Research and the

Reliability Risk Assessment Branch at NRR reached contradictory conclusions such that it is not obvious at the present time that there is an overwhelming benefit to requiring feed and bleed for CE plants such as Waterford. B. Sharon of the Reactor Systems Branch indicated that the latest conclusion was that the existing auxiliary feedwater system was reliable enough for decay heat removal from the standpoint of risk of a core melt, and any improvement in that number would not substantially reduce the risk of a core melt from other causes. D. A. Ward cited the LP&L presentation at the Subcommittee Meeting describing efforts to prepare for installation of the heated junction thermocouple. He indicated that LP&L has not established an independent position nor made up their minds exactly whether they will install it or not.

B. NRC Discussion of Its Review of Waterford Management

R. Benedict, NRC Staff, indicated that the Applicant had made a very noticeable turnaround in organization and staffing since the SER was issued and the ACRS had considered the Waterford application in August, 1981 (see Appendix VI). He illustrated his point by mentioning certain facts:

- Nuclear Operation Department staffed to 75% of its authorized level
- Enough RO and SRO candidates in training to provide adequate number of licensed operators at fuel load

C. Louisiana Power and Light Presentation

1. Corporate Overview

J. Wyatt, LP&L, explained how LP&L had been responsive to the ACRS comments concerning management competence in its interim report of August 11, 1981. Among other items, he indicated that the Waterford training section had been assigned to the Vice-President for Nuclear Operations and a Training Director by the name of Z. A. Sabri had been hired. He indicated that the Quality Assurance Section had been reassigned to create an overall LP&L quality assurance approach. The Safety Review Committee has been augmented with three outside members in addition to the Vice-President of Nuclear Activities at Middle South Services.

2. Management Commitment

G. D. McLendon, LP&L, presented a corporate management chart for LP&L (see Appendix VII) and pointed out two significant changes made in the organization since August, 1981. These changes involved reassignment of the nuclear group and its elevation to departmental status, and the supervision of quality assurance directly by the senior vice-president, as an independent, corporate quality assurance group.

3. Nuclear Operations Department Organization

L. V. Maurin, LP&L, listed the qualifications of the managers of plant operations, nuclear administrative services, nuclear project support, and nuclear training areas under his control. He indicated that an effective team could not be forged without a management control system whose principal ingredient will be an LP&L program manual, constructed from industry standards, regulatory guides, codes and the law. The overall program will be one of management by objective.

4. Plant Staff Organization

D. B. Lester, LP&L, described changes in the staffing of the plant organization. Mentioned was an aggressive effort to recruit the required engineering personnel for operations and maintenance managerial possitions. The need for contractor personnel in some areas was indicated. J. Ebersole requested that the Applicant comment on important differences between the security effort at nuclear plants and fossil plants, switch yards, and distribution systems. D. B. Lester explained that nuclear plants are more heavily manned by a higher class of guard-type individual. He pointed out that J. Slager, Jr., an ex-Marine officer, was in charge of the security for Waterford 3.

D. B. Lester described the startup and test organization that is being set up in three phases: a prerequisite test phase, a preoperational test phase, and an integrated test phase. Prerequisite and preoperational test phases are involved with tests of components and systems. The integrated testing phase is involved with tests of the plant operations. Mentioned was the effort at integration of contract and LP&L employees during the preoperational test program. D. B. Lester stressed that LP&L was selective in the individuals from different contractors that were employed on the project. M. Bender suggested that it might be more effective to have groups of individuals, acting as a team, from one consulting firm, rather than individuals from several contractors.

D. B. Lester indicated that the shift complement during commercial operation would not use contract personnel. M. Bender questioned how LP&L was responding to the Staff policy of having at least one person experienced with commercial operation of a reactor system on each shift. D. B. Lester indicated that LP&L has six previously licensed RO or SRO PWR license holders, three from a Westinghouse unit, two from a B&W unit and one from a Combustion Engineering unit in addition to a seventh, an STA from a commercial BWR.

5. Project Support Organization

F. J. Drummond, LP&L, explained changes in his organization since the August, 1981 interim review. W. Kerr questioned how safety people would be involved in the operation as contrasted with licensing interface with the Commission after the operating license is granted.

F. J. Drummond discussed a safety program consisting of four elements: the Safety Review Committee, the Onsite Safety Review Group, the Quality Assurance Organization, and the Plant Operations Review Group. M. Bender pointed out that the Onsite Safety Review Group did not seem to have persons with expertise in the materials and chemistry control area. F. J. Drummond mentioned A. D. Adams, a PhD in radiochemistry, and J. A. Ainger who has a Bachelors and Masters degree in metallurgy and a PhD in nuclear engineering. R. C. Axtmann asked LP&L how their industrial safety record, during the construction phases of Waterford 3, compared with that at other nuclear projects. L. Warren, Project Manager for the Applicant, offered to provide a written response to the Committee at a later date. W. Kerr expressed concern about reporting procedures within the plant organization for the Onsite Safety Review Committee.

6. Training

Z. A. Sabri, LP&L, explained the philosophy behind the training commitment of the Applicant. She pointed out that since there is a shortage of experienced people in the nuclear industry, the Waterford training program has been designed with flexibility in its training materials to accommodate the variations in the background of trainees. J. Ebersole questioned whether the Waterford training program teaches operators to be deliberate and requires that they identify elements that integrate into a final action. Z. A. Sabri indicated that LP&L will be relying upon simulator training and symptom oriented procedures.

The Committee agreed that they would be able to write a favorable report on the Waterford application.

IV. Mark III Containment Program and the Clinton Units 1 and 2 (Open to Public)

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

A. Subcommittee Report on the Mark III Containment

M. S. Plesset reported on the results of two meetings of the Fluid Dynamics Subcommittee. Of particular importance, especially for the Grand Gulf Station, was the dynamic loads from water striking the hydraulic control unit floor. At the second of the two subcommittee meetings in January, 1982, GE and the Staff compromised on a 50 ft. per second water velocity which was accepted by the Applicant to resolve the issue. With regard to the Clinton application, even though the Clinton hydraulic control unit is slightly nearer the pool (19 ft. at Clinton vs. 21 ft. at Grand Gulf) this does not make a serious difference in the loads and the installation will be acceptable from the point of view of loads on the floor on the hydraulic control unit itself. M. S. Plesset expressed his belief that both Grand Gulf and Clinton have been accepted as far as the fluid dynamics is concerned, particularly with regard to waterproofing of the hydraulic control unit.

D. Okrent pointed to questions raised by ACRS consultants, Theodore Y. Woo and Theofanis G. Theofanous. M. S. Plesset indicated that T. Woo complained about the overconservative methods used by the Staff with respect to very large and unrealistic drag coefficients from the point of view of steady state flow. It was pointed out, however, that this is not a case of steady state flow and the impact is difficult to handle. M. S. Plesset pointed out that T. Woo and T. Theofanous both thought there were areas that needed further study.

B. Report of the Clinton Subcommittee

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

W. Kerr indicated that the Clinton plant is likely to be the first plant to go into operation with GE's Nuclenet control system. He indicated that heat rejection is to a man-made cooling lake which has within it a seismically qualified, smaller lake that serves as the ultimate heat sink. Mentioned was the long list of open items among which is the QA Program that the NRC considers unsatisfactory, which should be of special interest to the Committee. W. Kerr suggested that a seismic reanalysis is being required as a result of assumptions about the New Madrid earthquake and should be an item of some interest. The Mark III containment issue has apparently been resolved to the satisfaction of the Staff and the Applicant. Other items of interest mentioned were Nuclenet and the proposed Remote Shutdown System. He indicated that there is some concern on the part of the Staff as well as the Subcommittee regarding electrical installation of the proposed remote shutdown system. W. Kerr mentioned a summary provided by R. Savio which has additional information regarding the above mentioned issues (see Appendix VIII).

C. Mark questioned the reason for a stop-work order on Clinton. W. Kerr explained that it had to do with QA deficiencies which were being remedied by retraining of QA people and craft people before the restart of work. D. Okrent questioned whether there was a difference between the hydrogen system used on Clinton from that used on Grand Gulf. He noted that this issue had become a licensing condition for Grand Gulf instead of an outstanding issue for Clinton. W. Kerr expressed the belief that the difference between the two situations was not in the use of igniters or equipment but in the fact that the Staff has accepted this as an interim measure that is appropriate for Mark III containments since the Grand Gulf review. R. L. Tedesco, NRC Staff, explained that as far as Grand Gulf is concerned, the Staff has the system under evaluation and will return to the Committee once it is resolved, probably in April or May, 1982.

C. NRC Staff Report

J. H. Williams, NRC Project Manager for Clinton, gave a brief status report of the Clinton Power Plant project. A list of 20 outstanding issues were presented, of which 16 were currently active. Each was described and discussed (see Appendix IX). J. H. Williams indicated that items which still are most difficult to resolve include the control room habitability, the remote shutdown system, the QA organization, the containment purge, and the steady state vibration acceptance criteria. These are all items where the Staff and the Applicant have or did have differences of position.

C. Grimes discussed the Mark III containment issue which came up at the time of the Grand Gulf ACRS OL review concerning a disagreement between the Staff and GE regarding the character and definition of froth impact, the character of the load which would be used to evaluate both the floor structure and the hydraulic control unit equipment response. C. Mark addressed a question to C. Grimes concerning an ambiguity in the SER with respect to water level instrumentation being adequate to insure detection of an approach to inadequate core cooling. R. L. Tedesco indicated that a meeting in January, 1982 resulted in GE promising a report by July, 1982 which would treat the entire subject of inadequate core cooling, and deal with the question of incore thermocouples. He did not deny that there appeared to be some confusion between certain sections in the SER.

J. Ebersole pointed out that control rod drive tubing is in direct proximity to some of the large high pressure steam lines which are hypothetically supposed to break in an accident. He suggested that it might be possible for a few rods to stick out of the core and prevent the reactor from being brought to shutdown. He questioned the Staff but R. L. Tesdesco suggested that the Applicant would be the more appropriate party to answer that question.

The Committee discussed the functions of liquid control systems during a BWR LOCA. Mentioned was the destruction of one train of the water level instrumentation and disruption of pressure sensing instrumentation. R. L. Tedesco promised a presentation on the loss of water level instrumentation.

J. Gilray, NRC Staff, described his review of the Clinton Project Quality Assurance Program. He mentioned a request for additional information by the Staff which consisted of 21 outstanding concerns. W. Kerr made a distinction between operational QA problems and QA problems in construction which have come up at Clinton. J. Gilray explained that it is I&E's responsibility to ensure the safety of the plant during design and construction.

W. S. Little, NRC Staff, provided a brief summary statement as to Region III's involvement in the Clinton project and I&E's views as to the adequacy of their Quality Assurance Program with respect to obtaining an operating license. Mentioned was the recently found quality assurance breakdown in the area of electrical work at the site. J. Ebersole questioned whether improper installation or incorrect routing of cables would have to be extensively corrected by remedial construction action. W. S. Little was not able at this time to indicate how much remedial work needed to be done.

G. V. Giese-Koch, NRC Staff, discussed the Staff recommendation of a conservative safe-shutdown earthquake of intensity 8, which translates into a 0.26g acceleration. He indicated that the Applicant assumed the seismic design spectra of NRC Regulatory Guide 1.60 of 0.26g. During the process of evaluation of the seismic design at the operating license stage, the Staff requested the Applicant to reassess the seismic design of Category I structures because the technique used to derive the foundation spectra for the seismic design was no longer acceptable. The Applicant elected to develop site specific spectra from representative earthquake records, demonstrate that the Category 1 structures were adequately designed to satisfy the seismic design spectra which would envelope the appropriate 84 percentile of the site specific spectra and demonstrate that Category 1 structures were adequately designed to withstand ground motion effects from New Madrid-type earthquakes. G. V. Giese-Koch presented the Staff conclusion that a site specific spectrum developed by the Applicant is an appropriate representation of the 84 percentile ground motion of an MB 5.8 ground magnitude earthquake.

D. Okrent mentioned that the deconvolution method of analysis was not used here and that site specific spectra do not give the same design basis as you would get if you took a new plant and applied 0.25g and the Regulatory Guide 1.60 for the required analysis. Discussed was the fact that 0.25g in Regulatory Guide 1.60 was not a requirement at the time Clinton got a construction permit. L. Reiter, NRC Staff, discussed with the Committee the relative merits of site specific spectra vs. the .25g Regulatory Guide requirement. D. Okrent commented that neither the magnitude approach in site specific spectra or the Regulatory Guide requirement makes a judgment as to what is an appropriate degree of conservatism in these analyses, whether the analyses are too conservative.

N. Chokshi, NRC Staff, summarized the results of a study of the seismic analyses of structures with respect to a Staff position on soil structure interaction. J. P. Knight, NRC Staff, indicated that the staff reviewed the response spectra for comparison of spectra on all locations inside containment in the auxiliary fuel building with the recommendation that characterization of the seismic input should be more severe than was used during design.

C. Grimes presented prepared written responses to questions that were raised by the ACRS during the presentation by GE on GESSAR (see Appendix X). Also mentioned was a handout with responses to questions that came up during the Clinton Subcommittee Meeting (see Appendix XI). C. Grimes explained that the Staff has waived the inservice inspection requirement for control rod drives for BWRs because of the difficulty in entering the confined region to perform the inspection, and taking account of the preservice inspection that was performed. Personnel exposures and dollar costs were also factored in from a consequence standpoint. D. Okrent expressed concern with the Staff reasons for this exemption, especially regarding leakage. R. L. Tedesco said that a very careful analysis was done of the consequence of leaks in an area that may not be accessible or amenable to inspection. The rationale of the Staff was that leak detection capability would be adequate to assess the condition well before any catastrophic failure. J. Ebersole expressed reservation with a response to one of the ACRS subcommittee questions concerning the first single active failure following an accident. He contended that it is invalid to take failures which are a direct result of accidents and enumerate them as a single failure randomly occurring. It is an improper interpretation of the single failure criteria. P. G. Shewmon expressed concern over the bad record with respect to personnel exposure at BWRs. He questioned whether the Staff has any criteria on the size of cleanup systems recommended or required on these plants to keep primary coolant activity at a reasonable level.

D. Presentations by the Applicant

L. J. Koch, Illinois Power, showed a chart of the current organizational structure of the Company relative to Clinton Power Station (see Appendix XII). He indicated that consultants have been used only sparingly in areas other than startup groups. Major packages of work have not been subcontracted to others except the primary contracts, General Electric for the nuclear steam supply system, Sargent and Lundy for the architect/engineering work, and Baldwin Associates for construction. In this transition phase from construction and startup to operation, Illinois Power Co. is encouraging Sargent and Lundy to maintain within its organization, a dedicated Clinton Staff to assist in integration of their procedures and standards into the Clinton Engineering Department. L. J. Koch singled out the Quality Assurance Program with regard to IP taking over some of the functions performed by the contractor's QA Department such as procurement and receiving QA, and establishing an ongoing program in operations QA. With regard to Staffing, L. J. Koch indicated that about 30% of the Clinton Staff are long-term employees of Illinois Power and about 70% are new hires from outside organizations. He pointed to the special unique relationship that Illinois Power has with Commonwealth Edison Co., with Sargent and Lundy, and with the University of Illinois. Some special training programs are now conducted with the University of Illinois as a part of their Staff Technical Advisor Program.

J. D. Geier presented a chart of staffing levels and commitments of the Engineering Department and a chart of technical personnel experience (see Appendix XII). He mentioned that the training organization is departmentally organized in Illinois Power.

T. Plunkett, Illinois Power Co., explained that one of the goals of his organization, which includes startup and permanent plant staff, is to obtain and utilize personnel with previous commercial nuclear power plant experience. In response to a question by D. W. Moeller, T. Plunkett detailed the qualifications and background of the Radiation Detection Supervisor. T. Plunkett indicated that with the Illinois Power Co. support, his organization was able to write all of its own procedures, including operating procedures and emergency procedures. P. G. Shewmon expressed interest in Illinois Power Co.'s method for controlling oxygen at startup and during operation. L. Brodsky, IP, indicated that during shutdown, the Clinton Plant has condenser deaeration, using auxiliary steam to remove oxygen. Tightness of the system

is used to preserve oxygen levels during operation as well as use of deep bed and mixed bed condensate polishing. L. Brodsky indicated that deep bed and mixed bed polishing demineralizers were used in the reactor water cleanup system for crud control. T. Plunkett indicated that IP decided to go after Nuclear Navy plant experience when they were no longer able to attract commercial nuclear power plant personnel. He indicated particular difficulty in recruiting radiation protection technicians. IP

IP recruited Nuclear Navy personnel, exposed them to the Clinton training program, and then lent them out to contractors who used them at other utilities for refueling outages. He felt that this gave them about the best experience possible.

P. G. Shewmon requested information on the training of instrument and control technicians. T. Plunkett indicated that considerable balance of plant training is going on but most of the control and instrument work for startup is being done by outside contractors. In answer to a question by M. W. Carbon, T. Plunkett indicated that Illinois Power is working on a program to upgrade the shift supervisory staff to take over the function of Shift Technical Advisor.

W. C. Gerstner indicated that the present quality program consists of an Illinois Power QA organization and an outside contractor/consultant, Baldwin Associates. W. C. Gerstner explained that A. J. Budnick, the Director of the IP QA organization, as well as the Baldwin and Associates Quality and Tech Services Group, report directly to him as Executive Vice-President. A. J. Budnick has reporting to him supervisors of engineering quality assurance, construction quality assurance, operations quality assurance, operations quality control, and a training coordinator.

C. C. Wheeler reviewed Mark III containment issues concerning the SRV loads, the LOCA loads, and the Clinton Power Station design (see Appendix XII). C. Grimes, NRC Staff, indicated that the Staff intended to treat the hydrogen question for Clinton and Grand Gulf generically when the Staff brought Grand Gulf back for ACRS review of the issue.

J. Ebersole, D. Okrent and M. S. Plesset expressed concern about water impact on hydraulic control units (HCUs). C. Grimes indicated that the matter came up during the CP review of Grand Gulf, Clinton, Allens Creek and similar BWR Mark IIIs. D. Okrent suggested that the Staff resolve this matter and report to the ACRS before the Committee repeats its OL review of Grand Gulf.

J. Ebersole questioned the statement made earlier about the low damage potential of a large LOCA to pressure manifolds and instrumentation. C. C. Wheeler indicated that this is being evaluated as part of the Sargent and Lundy continuing jet impingement break analyses and has not been completed.

J. G. Cook, Illinois Power, briefly described the remote shutdown panel, the Nuclear Net, and human factors engineering as it applies at Clinton. J. Ebersole asked a question about fire exposure at the hot shutdown or remote shutdown panel. He suggested that J. Cook restudy and report to the ACRS regarding the capability of the remote shutdown panels at Clinton Units 1 and 2 and the wiring and equipment with it to handle hot shorts and other malfunctions in the cable spreading rooms and the control room. J. C. Cook indicated that the Nuclear Net Panel in the control room has 10 CRT displays. It was indicated that the Nuclear Net is backed up by safety grade visual displays. (see Appendix XII). Also mentioned was the fact that the Safety Parameter Display Systems (SPDS) has been incorporated as one of the CRT displays in the Nuclear Net. In answer to a question by J. Ebersole, J. C. Cook indicated that the panel is safety grade, and has two different power supplies which gives double redundancy. D. W. Moeller expressed interest in the fact that the layout of the control room was such that operators did not have to leave the common ventilation system in order to utilize sanitary or eating and sleeping facilities if they should be required.

A. L. Rowe, Clinton Supervisor of Electrical Engineering, described the design of offsite and onsite power systems (see Appendix XII). J. J. Ray questioned whether the system would remain stable upon the loss of two power lines simultaneously. A. L. Rowe indicated that load flow studies conducted by Illinois Power indicate that power flows will not exceed the thermal limits of the transmission lines and that there is adequate margin between the unit's steady state stability limits and systems requirements.

L. Brodsky, Illinois Power, indicated that in the case of a black-out, special procedures would direct the operator to emergency systems reactor core isolation cooling to restore and maintain reactor vessel level. Operators outside the control room would manually isolate the containment using the outboard isolation valves. J. Ebersole inquired concerning the separation scheme for the onsite d.c. busses near power supplies and control circuitry with respect to protecting them from exposure fires. A. L. Rowe indicated that the control circuits are separated by at least three to five feet even in the cable spreading room. After the discussion with the Applicant, J. Ebersole asked the Staff if it has overcome the problem of fire protection rationale for the Clinton plant design. R. L. Tedesco indicated that fire protection is still an open issue and that there are design requirements that still require study. J. Ebersole expressed concern about the viability of using solenoid-operated valves to maintain depressurization of the plant during the early stages of a loss of a.c. power event. J. J. Ray, also concerned about failure of solenoids, questioned the redundancy in the safety relief valves. L. Brodsky indicated that there is redundancy through the use of 2 separate solenoid control circuits for each valve and 7 ADS valves out of 16 safety relief valves. L. Brodsky discussed a novel method of containment heat removal through the RHR heat exchangers using diesel-driven fire pumps with lake water as a water source. In answer to a previous question by J. J. Ray concerning the priority of restoration of power to the Clinton Station, L. Brodsky indicated that procedures for system blackout set a high priority for restoration of auxiliary power to Clinton.

D. Okrent questioned whether IP had a symptom-oriented procedure for the case of a severe accident. L. Brodsky indicated that they had developed an event-oriented procedure for an earthquake, and symptom-oriented procedures which address protection of the plant itself, such as core coverage and containment protection.

A. K. Singh, Illinois Power and Light, indicated that IP has performed a state-of-the-art seismic reevaluation of the plant to show that the current design meets the present NRC requirements. This included the development of a site specific spectrum for an MBE 5.8 earthquake, using methodologies adopted and used in the Midland and Sequoya plants studies. Also performed was a soil structure interaction analysis which included soil property variation and no deconvolution. D. Okrent expressed the belief that the Committee should consider the NRC's policy of reevaluation of the seismic design basis at the operating license stage. He stated that the impression given by the Staff is that the basis for design at the CP stage of an acceleration plus Regulatory Guide 1.60 was unnecessarily conservative, such that the use of site specific spectra is used as an alternate at the OL stage to show that the plant is acceptably safe. D. Okrent expressed the belief that the reevaluation being reported for Clinton has adequately covered the larger equipment and shown that there were not very big deviations in the design basis from the CP stage, but that the Applicant should look again at the ability of smaller equipment to function ably in an earthquake. J. Ebersole requested that the NRC Staff study the seismic design of pendant-type pumps that are used for service water uptake, determine whether they will operate at the limits of the amplitude of movement, and report to the ACRS regarding recently described problems with bearing degradation and bearing failures in such pump-types.

D. W. Moeller questioned the Applicant as to whether they had checked their hydrogen or oxygen monitors for operability. He commented on the review of LERs from operating BWRs over the past four years that had shown a surprising number of failures of hydrogen monitors. J. P. O'Brien of Illinois Power indicated that their hydrogen monitoring system had been purchased to the specifications of Regulatory Guide 1.97, Rev. 2, is in duplicate in the containment, but has not yet been checked for operability. C. Mark questioned whether the Subcommittee or the Staff had discussed matters of security, sabotage, personnel security, or plant security with the Applicant. J. H. Williams, NRC Staff, indicated that a very intensive review of security had been conducted and that it remains an open item to be closed out in the next couple of months.

V. Quality Assurance at Nuclear Power Plants (Open)

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

R. DeYoung, Director, I&E, expressed the concern of Chairman Palladino for the lack of performance on quality assurance throughout the nuclear industry and recent efforts to improve it. He indicated that the NRC had requested that INPO put together a program to look at quality assurance at plants under construction and then, perhaps, later at plants licensed for operation. R. DeYoung explained the problem as a gap between the NRR review of proposed design and codes used at the construction permit phase, and I&E is not looking at the implementation of how you got from the criteria, the specifications for the design, to the hardware. There is a quality assurance gap between this construction permit license and the I&E review during which the utility has constructed the plant in accordance with what they understood to be the agreements and criteria. C. P. Siess questioned the use of quality assurance and the general term quality interchangably. W. Kerr pointed out that quality implies more than Appendix B quality assurance. He and C. P. Siess expressed concern that NRC does not think enough of quality assurance to have a quality assurance program of its own.

E. Jordan, NRC Staff, discussed two sets of initiatives. The first of these concerned improving quality assurance at construction sites (see Appendix XIII). The second set of initiatives was entitled, Additional Approaches Under Consideration, items which the Staff could identify and strongly recommend to the Commission for initiative action. Initiative to improve QA at construction sites included the following:

- NRC consolidation of Staff resources
- Third party audits
- Enforcement sanctions
- Training qualification and certification of QA personnel

C. Mark asked for a definition of the "parties involved". E. Jordan indicated that the first party is the utility with strong involvement by corporate upper management; the second party, the Quality Assurance Staff employed by the utility; and, the third party, someone independent of the utility but not just the NRC. C. Mark expressed concern that chief executive officers will be too involved with high level management problems to be worried about quality assurance. E. Jordan suggested that the NRC wishes to involve the top executive director level person in a commitment to quality, and it is quality assurance that is the pathway to achieving quality in the nuclear industry. The Committee discussed the differences and similarities between quality assurance and quality. J. Ebersole questioned whether the setting of standards is a part of this quality assurance or quality problem. E. Jordan indicated that the NRC is looking to industry to provide the standards for quality, even though that is a separate problem from the implementation of the quality or quality assurance standards that currently exist.

The second set of initiatives entitled, Additional Approaches Under Consideration, included the following:

- Quality Verification Program for problem facilities
- Design Management
- Revised Inspection Program
- Approved Bidders List

E. Jordan explained that this second list is in the formative stages and the Staff is not expecting to make a recommendation in these areas at this time. The first of the alternate initiatives would involve a reinspection program through the use of independent testing to a greater extent. NRC would use contractors to do design audits at selected sites with regard to the second alternate initiative.

C. P. Siess questioned whether the NRC was building in a feedback mechanism to test whether these new initiatives were effective. He questioned what sort of yard stick would be used for measuring the level of effectiveness in improving quality. E. Jordan indicated that a first measure might be based on the frequency of identifying specific problems through the inspection program and looking at the utility's QA organization and its ability to attack the problems. M. Bender suggested that the new NRC program is hardly more than a records investigation checking for discrepancies in records. With regards to the Diablo Canyon problems, H. Denton indicated that the reverification process was to be handled by a firm with no previous financial involvement with Pacific Gas and Electric Co. He suggested that top quality firms would be deterred by the strict conflict of interest criteria, leaving mediocre firms to do the work, firms interested only in the financial aspects. H. W. Lewis expressed concern that safety might be compromised in this search for quality. Reviewers might be influenced by considerations other than vendor safety when questioning the acceptability of all parties in a reverification effort especially with regard to independence and capability of the reverifier for performing that review.

VI. LER Reporting Requirements (Open to Public)

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

W. M. Mathis indicated that the purpose of the subcommittee meeting just held the previous day was to continue the discussion with AEOD and interested parties of SECY 82.3, Proposed Addition to 10 CFR 50.73 Establishing the Licensee Event Report (LER) System. He reminded the Committee that it generally approved of the subcommittee recommendation in December to release the rule for public comment. A handout consisting of the proposed summary of the subcommittee meeting of March 3, 1982 included several questions raised by individual Committee Members concerning whether the rule should go out to the public for comment (see Appendix XIV).

W. Mathis described the LER rule as "just a data gathering mechanism". He indicated that sequence coding and search capability will be backfitted to include some of the past LERs. As proposed, the LER rule will require more narrative reporting by the licensees. (Other points of interest that arose from the subcommittee discussions are contained in Appendix XIV.) (Commissioner Ahearne's comments are also included in Appendix XIV.) The first of Commissioner Ahearne's comments referred to the inability to properly utilize the full potential of the narrative description of events for development of a general data matrix for detecting generic problems. The second item referred to the omission of a provision for a base for future expansion into the above mentioned data matrix. The third item questioned whether the rule should be approved before it has been demonstrated that the data could provide a more meaningful analytical tool for the future. D. A. Ward was in favor of publication of the rule. He did, however, note that the NPRDS system, which will act as a framework for the LER reports, is entirely directed toward failures of components and equipment, with no provision at the present time for reporting of human or software failures.
The Committee agreed to draft a report on the LER rule that goes beyond

endorsing release for public comment into describing the evolutionary changes and shortcommings in the proposed system. K. Bissel, NRC Staff, suggested that the ACRS not proceed with the review of the LER rule because premature placement might exclude the possibility for modifying and correcting a system that is not well defined or designed.

In answer to a question by D. Okrent that concerned the possible acquisition of AEOD reports related to analyses of LER events, C. Michelson described the LER review process as consisting of three steps:

. Screening of LERs by computer

. Engineering evaluation

. Case study and recommendations.

D. Okrent requested that C. Michelson add his name to the distribution list for LER engineering evaluation memoranda which are developed in AEOD. C. Michelson agreed.

VII. Qualification Program for Safety Related Equipment (Open to Public)

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

A. Report of ACRS Subcommittee

J. J. Ray defined the first objective of the February 10, 1982 meeting of the Subcommittee on Qualification Program for Safety Related Equipment as an overall review of the NRC Equipment Qualification Program Plan. The second objective was a discussion of a series of questions by M. Bender relating to Standard IEEE 373-1974, IEEE Standard for Qualifying Class 1 E Equipment for Nuclear Powered Generating Stations. Mentioned were five principle elements of the program:

. Equipment Qualification Standards Development Unit under the RES Organization

. Equipment Qualification Test Program

. Environmental Qualification Review Groups

. Seismic and Dynamic Qualification Reviews

. Equipment Survivability and Hydrogen Environments

J. J. Ray indicated that full implementation of the program for operating plants will not be completed until 1989. He explained that the proposed rule for laboratory accreditation has been prepared by the Staff, was discussed by the Regulatory Activities Subcommittee on March 3rd, and has been referred to the Subcommittee for followup and review after receipt of public comments. Under this plan, the IEEE will accredit testing organizations under a contract with the NRC similar to the ASME accreditation program for the N-Stamp.

J. J. Ray summarized the opinions of the Subcommittee that the NRC Equipment Qualification Program appears to be comprehensive, and proceeding in the right direction. The Subcommittee was, however, disappointed that the program would not be implemented on operating plants before 1989.

M. Bender was skeptical that the program could be effectively administered, that it may be too ambitious and its expectations unrealistic, especially with respect to continual changes in plant equipment. J. Ebersole expressed his belief that a program to finally test environmentally qualified equipment in containment is long overdue. Testing of this equipment will finally establish whether it will perform as anticipated or not.

B. NRC Presentation

P. Shemanski, NRC Staff, responded to the first of several questions that were posed by M. Bender on IEEE 323-1974 requirements. The first question concerned whether environmental qualification really demonstrates an ability to function under severe environmental conditions such as flooding or a seismic event (see Appendix XV). P. Shemanski, in answer to a comment by M. Bender, cited the unavailability of adequate test chambers and the resulting limitation with regard to the environmental testing of large components. M. Bender pointed out that it is probably economically more practical to rely on well-conducted analysis to confirm reliability of large pieces of equipment rather than experimental testing.

J. Ebersole expressed concern that the tendency to put sensitive equipment in a hostile environment has proliferated and should be reversed. The trend should be to remove sensitive equipment such as electrical systems from the hostile environments and use impulse lines to hydraulic and pneumatic valves which could remain in the containment.

P. Shemanski discussed the qualification steps taken to deal with the concern for a weak link element in a very complicated piece of sensitive equipment (see Appendix XV). He discussed components of the items being qualified or examined for weak links and how reliance on redundancy and physical separation, coupled with qualification testing, allows for a judgment as to whether reasonable assurance exists that the equipment can perform its intended function during an accident and post-accident environmental conditions.

P. Shemanski described the concept of margin in the answer to a question which dealt with a type-testing of selected components and equipment to provide a reliable test for production equipment (see Appendix V).

G. Bagchi, NRC Staff, answered the question concerning whether the specific location of environmentally qualified equipment in the containment building influences the potential for unanticipated or extreme environmental conditions and the potential for interaction with systems not covered by the qualification program. G. Bagchi gave several specific examples (see Appendix XV) of influences of specific location within the containment and interactions of environmentally qualified equipment with unqualified systems. H. Etherington asked about the Staff's acceptance of Westinghouse sine beat testing to simulate actual earthquake motion. The Committee discussed the limitations of the Westinghouse test concept.

G. Bagchi demonstrated by example equipment reliability achievement through IEEE 323-1974 requirements through survey of a new power plant. He listed several recommendations which resulted from the survey (see Appendix XV). M. Bender expressed doubt of the effectiveness of this qualification program and indicated that the burden of proof rests on the Staff to show that the program will be effective if ever implemented.

VIII. ACRS Subcommittee Report regarding Indian Point Systems Interactions Study

[Note: J. M. Griesmeyer was the Designated Federal Employee for this portion of the meeting.]

D. Okrent summarized the February 26, 1982 meeting of the Subcommittee on Safety Philosophy, Technology, and Criteria (see Appendix XVI). The Subcommittee members were concerned about the scope of the proposed Systems Interactions Study for Indian Point 3 by the Power Authority of the State of New York (PASNY). D. Okrent indicated that the Subcommittee advised the Staff and PASNY that they should be selective in reviewing systems interactions for systems that have already been covered in previous safety analysis and safety evaluation reports. With regard to a generic program that the Staff also discussed at this Subcommittee meeting, it was not clear whether there were early, obvious benefits from a walk-through on operating plants, or whether there was a feedback mechanism in place to advise operating plants of generic problems that emerge from limited studies. A memorandum from J. Conran to J. M. Griesmeyer (see Appendix XVII) and a memorandum from R. Savio to the Committee (see Appendix XVIII) were mentioned but not discussed.

J. Conran, NRC Staff, indicated that looking at all systems without pre-screening was not considered a viable alternative because of PASNY's limited resources and recognition that safety systems had already been addressed in other forums. PASNY did not intend to repeat that sort of review. He stated that PASNY's approach to concentrate on interaction between safety and nonsafety systems was compatible with that of the Staff which was to examine, comprehensively, nonsafety related component and systems failures as sources of interactions with safety systems.

V. Kishinevsky, a PASNY representative, restated PASNY's position that safety system interactions had been adequately addressed at the time of the original plant design and plant qualifications associated with later NRC regulations. He indicated that it is PASNY's obligation to make stringent application of a single failure criteria. Secondly, he stated that it is PASNY's purpose for its systems interactions study to prevent systems interactions from occurring. He requested ACRS input concerning how PASNY should treat nonsafety system control failures and their impact on safety related systems, and its obligation with regard to the single failure criteria.

J. Ebersole indicated that he did not wish PASNY to duplicate previous efforts, but, he doubted that a complete job of safety vs. safety interactions studies had been done at power plants. The Committee discussed limited treatment in the area related to nonsafety related system control failure review. M. Bender was in favor of the Indian Point 3 study proceeding as planned. It was his opinion that the study would go a long way toward satisfying the interest of the ACRS. D. Okrent suggested that the Committee defer commenting on a generic approach to systems interactions to its April meeting. He suggested that PASNY do control system failure analyses on an exploratory basis but need not concentrate their efforts in this area.

J. Conran suggested that it was appropriate for PASNY to check nonsafety control system failures of the kind pointed out in Bulletin 7927 and its effects as propogated through a nonsafety control system. V. Kishinevsky added that PASNY was not going to exclude interactions pointed out in Bulletin 7927 and 7922.

V. Kishinevsky pointed out that one primary issue was whether PASNY should apply a more liberal single failure criteria to systems which operate during normal operation. D. Okrent brought up a third point as to whether safety systems should be considered as sources for systems interactions as well as targets.

IX. Zimmer Quality Control Deficiencies During Construction (Open to Public)

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

M. Bender discussed the ACRS subcommittee meeting on the Zimmer Power Station held on February 18, 1982 concerning shortcomings in the quality assurance effort at the Zimmer plant uncovered by NRC inspections. The problems were principally involved with insufficient records. NRC informed Cincinnati Gas and Electric (CG&E) to beef up its quality control inspectors from a force of 18 to more that 200.

As a result of an I&E study in July 1980 and allegations which included acceptance of faulty welds on safety-related piping systems, CG&E was forced to redo radiographic analyses of welds for which a correct status and history was not available. CG&E was also conducting a 100% reinspection of Kaiser Engineers and other site contractors. The NRC is now satisfied with efforts by CG&E, even though the utility has still not developed a training organization for the increase in QA staffing. J. Ebersole pointed out that even though there was disorder in the records, only minor defects concerned with improper installation and documentation for pipe hangers, restraints and snubbers were found. M. Bender indicated that the Subcommittee had agreed to follow the situation, but he concluded that there is no need for full Committee review of the matter. The Committee agreed not to write a report on Zimmer quality assurance at this time.

X. Alternate Materials for Waste Disposal Containers (Closed to Public)

[Note: E. Inge was the Designated Federal Employee for this portion of the meeting.]

A. Report of the ACRS Subcommittee and Consultants

P. G. Shewmon discussed the question of 1000 year integrity of waste disposal containers as now covered by a proposed rule which would allow the waste to leach out at a very slow rate after maintaining its containment integrity for 1000 years. He explained that the question that came up at the Subcommittee was how one could use short term tests over a period like one year to predict the performance of materials for 1000 years. He mentioned four ACRS consultant reports (see Appendix XIX) which evaluated the competence of the Staff-selected proposed contractor of its five year, $1 million per year testing contract and program proposed for evaluation of alternate materials for radioactive waste disposal containers. P. G. Shewmon indicated that the proposed contractor's proposal showed a lack of familiarity with Dept. of Energy contract work being done in the Battelle Columbus Laboratory. The Subcommittee conclusions were that the contractor was competent or capable of becoming competent within the five years of the program, but the Subcommittee did urge the Staff to set up a management procedure to allow it to stay ahead of the contractor and remain better informed regarding the kinds of work they thought the contractor should do.

R. C. Axtmann expressed concern that the task might not be achievable. He expressed his impression that the consultants were particularly concerned about unreasonable techniques used for metallurgically aging the containers. Several Members expressed a lack of enthusiasm for the selected contractor as well as the potential program to be undertaken.

B. Discussion with the NRC Staff

F. Arsenault, NRC Staff, indicated that it was not the intention of NRC to demonstrate compliance with the performance objective of containment for 1000 years or predict the results of 1000 year containment with this contract, but to provide additional information for improving the understanding of the phenomena involved and work with the contractor as the NRC improves its understanding to formulate and reformulate the approach to the problem. C. Mark suggested that if one could show a leach rate of 10^{-5} per year starting at year 1 instead of 1000 years from now could be achieved without any package, would this not be an acceptable means of storing the waste - without a package at all? F. Arsenault indicated that if one had to demonstrate compliance with the current rule, then it would not be acceptable because the current rule requires the performance of the package over some period of time be demonstrated to be in compliance. The Committee discussed the legalistic aspects of the proposed rule. F. Arsenault indicated that the issue was one of demonstrating adequate waste isolation which has two aspects to it. One component involves the question of technology in providing quantitative data to back up the assertion that the depository will meet the standard; the other element is that of achieving confidence that that demonstration is adequate. He indicated it was his belief that no matter what standard was promulgated, it would be susceptible to challenge. He hoped that by building into the process the ability to demonstrate that the NRC had taken account of sources of uncertainty and introduced conservatism, both quantitative and procedural, to achieve that confidence, court challenges would be avoided. The Committee agreed to write a report concerning this matter.

XI. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. ACRS Recommendation Regarding Revision 3 to Proposed Regulatory Guide 1.28 "Quality Assurance Program Requirements (Design and Construction) (Task No.RS 002-5)"

The Committee endorsed a memorandum from the ACRS Executive Director to the EDO which states that the ACRS has considered the recommendations of its Subcommittee on Regulatory Activities regarding Revision 3 to Regulatory Guide 1.28 and has agreed to defer further consideration and action until it has been reviewed by the Committee to Review Generic Requirements (CRGR). It is recommended that the CRGR consider in its review the generic question relating to the manner in which the NRC Staff utilizes voluntary consensus standards by endorsement in a regulatory Guide with exceptions and/or additions and makes mandatory those guidelines that are nonmandatory in the endorsed standard (see Appendix XX).

2. Proposed Rulemaking: Proposed Amendment to 10 CFR Part 50, Section 50.49a "Accredidation of Qualification Testing Organizations"

The Committee endorsed the recommendations of the Regulatory Activities Subcommittee regarding issuance of the proposed rule for public comment and assignment of review of the proposed rule to the Subcommittee on Qualification Programs for Safety-Related Equipment (see Appendix XX).

3. NRC Qualification Program for Safety Related Equipment

The Committee discussed the 5 year program planned by NRC for development of environmental qualification criteria for safety related equipment but was unable to reach agreement on a proposed memorandum to the EDO. The matter was referred back to the Qualification Program for Safety-Related Equipment Subcommittee for further work.

4. Systematic Evaluation Program (SEP) Reviews

A new subcommittee, the Systematic Evaluation Program Subcommittee, was approved by the Committee to handle the coordination of SEP project reviews with C. P. Siess as the appointed chairman. Membership will consist of the subcommittee chairmen of specific projects to be reviewed in this program as follows:

SEP Project Reviews (RKM/HA)....

. Siess, Kerr, Lewis, Mathis, Moeller, Ward (Consultants: D. Fitzsimmons, others as needed)

5. Use of PORVs on Combustion Engineering (CE) Plants

D. A. Ward noted the intent of the Decay Heat Removal Systems Subcommittee to schedule a briefing at a future subcommittee meeting regarding the need for PORVs on Combustion Engineering Reactor Plants.

6. Clinch River Breeder Reactor (CRBR) Subcommittee and Working Groups

The Committee discussed proposed changes by M. W. Carbon in the CRBR Subcommittee and Working Groups including the formation of an additional working group called Site Suitability. Additional changes in the assignment of members are anticipated and the final form of the CRBR Subcommittee is expected shortly.

7. Safeguards and Security Subcommittee Meeting on March 23, 1982

D. Okrent requested that a transcript or very detailed minutes be kept of the forthcoming March 23, 1982 Safeguards and Security Subcommittee Meeting. The Subcommittee Chairman agreed that a transcript should be kept of this meeting and that a copy will be made directly available to D. Okrent.

B. ACRS Reports, Letters, and Memoranda

1. Report on Clinton Power Station Unit 1

The Committee prepared a report to the Commissioners of its review of the Clinton Power Station Unit 1 recommending, subject to due consideration of recommendations in the body of the report and satisfactory completion of construction, staffing, and preoperational testing, the granting of a license to operate the plant at full power.

2. Report on the Byron Station Units 1 and 2

   The Committee prepared a report to the Commissioners of its review of the full power operating license for the Byron Station Units 1 and 2. The recommendation is for full power operation subject to certain issues requiring final resolution noted in the body of the report and subject to satisfactory completion of construction and preoperational testing.

3. Report on the Waterford Steam Electric Station Unit 3

   The Committee prepared a report to the Commissioners regarding the continuation of its review of the application of Louisiana Power and Light Company (LP&L) for a license to operate the Waterford Steam Electric Station Unit 3. The ACRS believes that the Applicant has effectively responded to concerns regarding organization and management expressed in the Committee's August 11, 1981 report. If due consideration is given to other matters in the August 11, 1981 report as well as continued dedication of LP&L management, satisfactory completion of staffing, and the planned program for training, the recommendation is for approval of full power operation of the plant.

4. Report on Systems Interaction Study for Indian Point Nuclear Generating Unit 3

   The Committee prepared a report to the Commissioners of its review of the proposal of the Power Authority of the State of New York (PASNY) to perform a systems interactions study of the Indian Point Nuclear Generating Station Unit 3. The ACRS believes that the PASNY proposal is generally responsive to prior ACRS recommendations made in letters dated July 13, 1978 and October 12, 1979 and believes that it is reasonable in this study to place emphasis on the interactions between nonsafety systems and safety systems.

5. Report on the Long-Term Performance of Materials Used for High-Level Waste Packaging

   The Committee prepared a report to the Commissioners of its review of the NRC's Contract Review Panel recommendations for the selection of a contractor to develop a methodology for predicting Long-Term Performance of Materials Used for High-Level Waste Packaging. Concern was expressed about the rationale for the extraordinarily high standards for long-term survival of these waste containers.

6. Report on the Licensee Event Report Rulemaking

The Committee prepared a report to Commissioner Ahearne regarding the status of its consideration of the proposed Licensee Event Report (LER) Rulemaking. While the ACRS believes the proposed rule represents a natural evolution in the state-of-the-art in data gathering, and supports its publication for comment, ultimate goals for such a system include better reporting, analysis, and evaluation of human errors and computer software errors and perhaps the development of a system for more effectively identifying precursors and systems interactions in addition to revisions revealed by subsequent experience.

C. Generic Safety Items

1. Status of LOFT Research Program

M. S. Plesset requested an hour at the 264th ACRS Meeting (April) for a Staff briefing regarding the future status of the LOFT Program.

2. NRC-Industry Steering Panel on Steam Generator Tube Degradation Reports (SERs)

The NRC Staff is organizing a Steering Panel to coordinate an NRC-industry effort to resolve problems associated with ubiquitous steam generator tube degradation. Consistent with the request of the NRC Chairman for ACRS participation, the Committee endorsed having P. G. Shewmon address metallurgical/chemical engineering concerns and J. Ebersole address plant design/operations aspects of the problem.

3. Improved Summaries in Safety Analysis Reports (SARs) and Safety Evaluation Reports (SERs)

The Committee discussed the development of improved SERs and SARs for support of ACRS activities in regard to a February 12, 1982 memorandum from the EDO responding to suggestions from the Committee on this matter. The Committee designated M. C. Gaske as liaison for the ACRS regarding this effort.

4. Reactor Pressure Vessel Water Level Indicators

The Committee was briefed by J. A. MacEvoy, ACRS Fellow, regarding performance and characteristics of differential pressure cells as reactor pressure vessel water level measuring devices. A discussion of heated junction thermocouples was deferred for consideration by the Electrical Systems Subcommittee. The presentation slides used during the briefing are reproduced in Appendix XXI entitled, Reactor Vessel Differential Pressure Level Instruments.

Various sources of error and ambiguity inherent in the use of differential pressure reactor vessel level instrumentation for BWRs were revealed. J. A. MacEvoy discussed the theory of operation of the differential pressure instrumentation system and how the various internal mechanical components in the system function. System response to steady state and transient response conditions was also explained. Proposed PWR level instrumentation systems were briefly discussed.

D. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 264th ACRS Meeting, April 1-3, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

The 263rd meeting of the Advisory Committee on Reactor Safeguards was adjourned on Saturday, March 6, 1982 at 11:25 a.m.

APPENDIXES
TO
MINUTES OF THE 263RD ACRS MEETING
MARCH 4-6, 1982

ACRS-1962

ATTENDEES
263RD ACRS MEETING
MARCH 4-6, 1982

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

*Member Emeritus

ACRS STAFF

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

NRC ATTENDEES

263RD ACRS MEETING

Thursday, March 4, 1982

NUCLEAR REACTOR REGULATION

E. F. Goodwin	G. Alberthal (DSI)
H. L. Brammer	J. Holonich (DSI)
R. L. Bangart	E. Doolittle (DL)
M. Jankowski	M. Hum (DE)
S. H. Chesnut	J. Chen (DE)
K. L. Liper	J. Wermiel (DSI)
J. C. Pulsipher	O. Parr (DSI)
B. J. Youngblood	W. G. Kennedy (DE)
R. L. Tedesco	V. Benaroya (DE)
N. K. Trehan	A. Thadani (DST)
J. E. Knight	F. Coffman (DST)
D. J. Kubicki	C. D. Sellers (MTEB)
G. Bagchi	P. Easley (AEB)
J. Conran	W. Paseday (AEB)
H. Silver	J. W. Gilray (QAB)
K. Bissell	W. P. Hann (QAB)
	P. Shemanski (EQB)
	S. Black (DL)
	B. Sheron (RSB)
	W. Johnston (DE)
	Z. Rosztoczy (DE)
	H. Silver

EXECUTIVE LEGAL DIRECTOR

R. Rowson
C. Michelson
J. Heltemes

NRC ATTENDEES

263RD ACRS MEETING

Friday, March 5, 1982

NUCLEAR REACTOR REGULATION

J. R. Miller
E. F. Goodwin
J. H. Williams
C. I. Grimes
J. B. J. Read
D. C. Jeng
D. Gupta
B. Jagannath
M. B. Fielch
N. Chokshi
D. Tondi
D. Serig
K. Demprey

DIV. OF LICENSING

R. L. Tedesco
S. Black

OFFICE OF EXECUTIVE LEGAL DIRECTOR

R. Goddard

NUCLEAR MATERIAL SAFETY & SAFEGUARDS

D. A. Kers
J. J. Davidson

REGION III

J. M. Peschel
W. S. Little

ATTENDEES - APPLICANT

263RD ACRS MTG.

Thursday, March 4, 1982

LOUISIANA POWER & LIGHT COMPANY

L. V. Maurin
K. R. Iyevgan
Z. Sabri
F. Drummond
C. A. Wells
G. McLendon
J. M. Wyatt
J. Sleger
R. Prados
J. C. Scott
D. Lester
S. Alleman
R. Kenning

EBASCO

R. C. Iotti
J. Costello
J. Hart
D. S. Palmer

ATTENDEES - APPLICANT

263RD ACRS MTG.

Thursday, March 4, 1982

COMBUSTION ENGINEERING

J. M. West
J. Longo
L. A. Banda
R. Newman
H. B. Mulliken
L. Banda

COMMONWEALTH EDISON

L. DelGeorge
G. Klapp
R. C. Ward
S. Barrett
K. T. Weaver
J. Golden
R. E. Joetberg
D. P. Shristiana
R. E. Shannon
R. E. Sierio
D. Hunter
T. P. Joyce
J. D. Deress
T. R. Tramm
L. A. Bowen
J. T. Westermeier
W. L. Stiede
T. K. Higgins
K. A. Ainger
J. C. Blomgiren
J. R. VanLaere
D. O'Brien

SARGENT & LUNDY

F. M. Krohn
W. Cleff
R. Netzel
D. L. Leone
K. J. Green
B. G. Treece
J. E. Szupillo
E. R. Crass
A. K. Singh
R. A. Witt
R. C. Heider

WESTINGHOUSE

J. Galembush
S. Prokopovich
T. F. Timmons
E. C. Volpenhein
B. Gergos
M. Oper
M. D. Beaumont
R. Sterdis
J. Conner
W. R. Spezialetti
W. E. Kortur
N. J. Liparulo
D. C. Richardson
P. T. McManus
P. J. McManus

INVITED ATTENDEES

263RD ACRS MEETING

Friday, March 5, 1982

ILLINOIS POWER COMPANY

J. P. O'Brien
A. Ruwe
J. Cook
W. C. Girstner
L. J. Koch
L. S. Brodsky
E. W. Kant
D. L. Holtzscher
C. C. Wheeler
T. Plunkett
J. D. Geier

SARGENT & LUNDY

R. Givan
R. Witt
W. R. Peebles
A. K. Singh
R. C. Heider

GENERAL ELECTRIC

F. C. Downey
D. L. Foreman
W. M. Davis
B. P. Grim

PUBLIC ATTENDEES

263RD ACRS MTG.

Thursday, March 4, 1982

L. Knuse, Public Service of Indiana
J. Kornblith, Medill News
M. Rubin, DOD
R. Renauant, Bechtel
C. Taylor, WPPSS
E. M. Howard, KMC
H. Levin, TERA
J. Sheless, Southern Co.
J. Perry, WDSU-TV
P. Getschal, NRDC
J. McKinney, Baton Rouge State Times
J. Roth, TV Crew
R. Condello, NUTECH
R. Gamble, EDS Nuclear
V. Kishinevsky, PASNY
J. Lamberski, PASNY
E. M. Levine, Weston Geophysical
L. J. Koch, Illinois Power Company
J. Geier Illinois Power Company
D. Green, Kansas Gas & Electric
A. Buluski, Isham, Lincoln & Beale
R. W. Huston, Consumers Power
N. G. Chapman, Bechtel Power Corp.
T. Rosberg, WIFR
A. Brau, WIFR
K. Drehobl, Consumers Power
J. Pope, Public Service of Indiana
B. W. Brown, EG&G, Idaho
R. Pallaman, Ebasco Services, Inc.
J. V. Bott, Public Service of Indiana
S. J. Brewer, Public Service of Indiana
J. C. LaVallee, Power Consultants, Inc.
J. Gallo, ILB

PUBLIC ATTENDEES

263RD ACRS MEETING

Friday, March 5, 1982

R. S. Boyd, KMC
P. A. Nevins, CEI
J. J. Waldron, CEI
H. Gutmann, WRAPS
R. Leyse, Electric Power Research Inst.
F. Murphy, Westinghouse
E. F. Beckett, Nuclear Pros. Inc.
E. M. Levine, Weston Geophysical
C. Taylor, Wash. Public Power
E. Murphy, Westinghouse
E. M. Howard, KMC
E. Buzzelli, CEI
J. Pennecott, PSP&L
C. Grochnor, Stone & Webster

APPENDIX II
FUTURE AGENDA

APRIL

ACRS comments on the NRC Long Range Research Program Plan	4 hrs
Briefing by the NRC Staff regarding reactor pressure vessel liquid level indication	3 hrs
Response on Safety Goals (DO/JMG) initial session	2 hrs +2 hrs
Briefing by the NRC Staff regarding the incorporation of the Fire Protection Rule, Appendix A in the new NRC Standard Review Plan	1 hr
ACRS comments on the Proposed NRC Rule regarding Application of TMI-2 Lessons Learned to OLs (WMM/RKM)	Deferred to May
ACRS comments regarding proposed changes in seismic design methodology	Tentative
ACRS comments regarding NUREG-0799 (Draft), Criteria for Preparation of Emergency Operating Procedures	Tentative

Subcommittee Reports

. Subcommittee on Extreme External Phenomena regarding a proposed reply to Commissioner V. Gilinsky's inquiry concerning Dr. P. Jennings' suggestions regarding seismic design methodology (DO/RS)	1 hr
. Subcommittee on Metal Components regarding a proposed NRC plan to resolve bolting problems in nuclear plants (PGS/EI)	Deferred
. Subcommittee on the Diablo Canyon Units 1 and 2 regarding seismic design deficiencies (CPS/JCM)	Deferred
Subcommittee on AC/DC Power Systems Reliability regarding the reliability of DC power supplies and results of the St. Lucie Unit 1 cable surveillance program (JJR/JMG/RS)	30 min

Future ACRS Activities

ACRS comments on the NRC plan to resolve reactor pressure vessel thermal shock problem (Report of the Metal Components Subcommittee chaired by M. Bender for this discussion)	May

APPENDIX III

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MARCH

16	Decay Heat Removal Systems (Savio) - Ward, Ebersole, Etherington, Ray. Purpose: To review the status of Task Action Plan A-45 and PWR Decay Heat Removal Systems with the emphasis on the CESSAR System 80 standard design.
17	Human Factors (Fischer) - Ward, Bender, Mathis, Ray. Purpose: To review the various Safety Parameter Display System (SPDS) designs and the status of plant diagnostic systems. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," will be discussed also. Additionally, the Subcommittee will discuss the training of Shift Technical Advisors (STAs) in the areas of plant systems and transient/accident analysis, and Senior Reactor Operator (SRO) training and qualification programs.
18 & 19	Joint Reactor Operations/R.E. Ginna (Rochester,NY) (Major/Fischer) - Mathis, Ebersole, Etherington, Ray, Siess. Purpose: To discuss the 1/25/82 steam generator tube failure incident and the SEP review of Ginna.
19	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Bender, Kerr, Siess*. Purpose: To review draft Commission Policy Statement on Safety Goals.
22	Structural Engineering (Albuquerque, NM) (McKinley/Igne) - Siess, Bender, Ebersole, Shewmon. Purpose: To review Sandia's containment integrity program, including a visit to the Sandia structural laboratory.
23	Safeguards & Security (Albuquerque, NM) (Alderman/McKinley) - Mark, Ray, Shewmon, Ward, Siess, Carbon, Mathis, Plesset, Lewis. Purpose: To discuss design features in proposed standard design plants that would make sabotage by insiders more difficult.
24	ECCS (Albuquerque, NM) (Boehnert) - Plesset, Ebersole, Mathis, Ward, Mark (1/2 day). Purpose: To discuss NRC's use of LOCA/ECCS codes and aspects of the recent Ginna transient that had impact on LOCA/ECCS concerns.
25, 26 & 27	Advanced Reactors (Argonne, IL) (Igne/Boehnert) - Carbon, Mark. Purpose: To continue discussion and preparation of report to ACRS entitled, "Safety Issue and Philosophy of LMFBR."

* Part-Time

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MARCH (CONT'D)

25 (p.m.) 26	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Bender, Ebersole, Ward, Siess, Kerr. Purpose: To discuss the Zion Probabilistic Safety Study.
30	AC/DC Power System Reliability (Savio) - Ray, Ebersole, Mathis, Okrent. Purpose to review the status of Task Action Plan A-44 and the NRR implementation of the recommendations of NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants."
30 (p.m.) 31 (a.m.)	CRBR (Boehnert**/Igne) - Carbon, Bender**, Mark**. Purpose: To review the CRBR General Design Criteria.
31	Joint Electrical Systems and ECCS (Savio/Boehnert**) - Kerr**, Ebersole, Plesset**, Ray, Lewis, Bender, Etherington. Purpose: To continue review of the NRC- and Industry-sponsored research on core water level indicator instruments and the NRC and Industry implementation of core water level indicator installation requirements.
31	Nuclear Safety Research Program (Duraiswamy) - Siess, Okrent, Kerr**, Plesset**, Shewmon (tent.), Mark**, Ward, Carbon*. Purpose: To continue discussion of the NRC Long-Range Research Program Plan.

APRIL

1-3	264th ACRS Meeting
15	Systematic Evaluation Program (Major/Alderman) - Siess, Lewis, Mathis, Moeller, Ward. Purpose: To review the completion of the Systematic Evaluation Program review on Palisades.
21 & 22	Wolf Creek (Emporia, KS) (Major/Bucci) - Ray, Mark. Purpose: Site visit and review of application for an operating license.
26 & 27	WPPSS 2 (Hanford, WA) (Griesmeyer/Quittschreiber) - Plesset, Ebersole, Mark, Mathis. Purpose: To review application for an operating license.

* Part-time

** Conflict to be resolved.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

APRIL (CONT'D)

27	Transportation of Radioactive Materials (Duraiswamy) - Siess, Mark, Bender. Purpose: To continue the review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.
28	Watts Bar (Knoxville/Chattanooga area) (Beal/Quittschreiber) - Ebersole, Bender, Ward. Purpose: To review application for an operating license.
mid-April	Metal Components (Igne) - Shewmon, Ward, Axtmann, Bender, Etherington, Mathis, Plesset. Purpose: To continue the review of pressurized thermal shock.
mid-April	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Axtmann, Ray. To review control room habitability and to discuss the design basis for normal and abnormal conditions, testing, and research needs.
late April or early May	Ad Hoc Metal Components Subgroup (Igne) - Bender, Shewmon, Etherington, Okrent, Plesset, Ward, Axtmann, Mathis. Purpose: To review the NRC Staff's action plan on pressurized thermal shock before a Staff position is issued on this matter.

MAY

4 & 5 (tent.)	CRBR (Boehnert) - Carbon, Mark. Purpose: To begin Subcommittee review of the CDA energetics issue for CRBR licensing.
5	TMI-2 Action Plans (Major) - Mathis, Etherington, Lewis, Okrent. Purpose: To review the proposed 10 CFR 50 rule on, "Licensing Requirements for Pending Operating License Applications" (rule contains Basic Requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements").
6-8	265th ACRS Meeting
late May	Perry (Cleveland, OH, tent.) (Savio) - Ray, Axtmann, Bender, Okrent. Purpose: To review the OL application and to conduct a site visit.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATES TO BE DETERMINED

Wk. of June 14	ECCS (Boehnert) - Plesset, Ebersole, Etherington, Ward. Purpose: To discuss GE's request for change in Appendix K decay heat requirements; NRR code audit capability; LOFT ATWS test vendor code predictions and results; and NRR work on operator accident guidelines and procedures.
Date to Be Determined (June or July)	Joint CRBR and Site Suitability (Boehnert/Alderman) - Carbon, Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, Ray. Purpose: To begin site suitability review for CRBR.
Date to Be Determined	Class 9 Accidents (Beal/Quittschreiber) - Kerr, Axtmann, Bender, Moeller, Okrent, Siess, Ward. Purpose: To review severe accident research plan.
Fall 1982	Reactor Radiological Effects (Alderman/McKinley) - Moeller, Ray, Axtmann. Purpose: To review NUREG-0833, "Environmental Impact Statement on the Siting of Nuclear Power Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 16	Decay Heat Removal Systems	(SAVIO) Ward, Ebersole, Etherington, Ray

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: ACRS

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MARCH 17	Human Factors	(FISCHER) Ward, Bender, Mathis, Ray Cons: Arnold, Buck, Debons, Pearson, Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: To review the various Safety Parameter Display System (SPDS) designs and the status of plant diagnostic systems. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," will also be discussed. Additionally, the Subcommittee will discuss the training of Shift Technical Advisors (STAs) in the areas of plant systems and transient/accident analysis, and the Senior Reactor Operator (SRO) training and qualification programs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," dated June 1981 (for comment version).

NUREG-0899, "Revised Criteria for Preparation of Emergency Operating Procedures."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 18-19, 1982*	Reactor Operations/R.E. Ginna	(MAJOR/FISCHER) Mathis, Etherington, Ebersole, Ray, Siess Consultants: I. Catton D. Fitzsimmons

LOCATION: Plant is in Ontario, New York (15 miles northeast of Rochester, New York). Meeting room at the Sheraton Inn, 1100 Brooks Ave. Rochester, N.Y. (716) 235-6030.

BACKGROUND:

Who proposed action: W. Mathis

Purpose: The purpose of this meeting will be two fold. First the Reactor Operations Subcommittee wishes to discuss the January 25, 1982 steam generator tube failure - Site Emergency incident. Among the goals of this portion of the meeting will be to evaluate how well the emergency preparations at Ginna served the situation and examine the operators response to the incident. Secondly, Ginna is rapidly becoming tied with Palisades as the lead SEP (Systematic Evaluation Program) plant. Once at the site, those improvements which can be observed resulting from the SEP program could be viewed. An SEP "tour" of Ginna coupled with the steam generator tube rupture review could eliminate the need for another trip to Ginna as part of the SEP review, and allow Ginna's SEP meeting to be conducted in Washington.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Only a Preliminary Evaluation of Operator Actions for Ginna SG Tube Rupture Event is (1/29) available.

2. A February 5, 1982 - chronology of the January 25, 1982 incident is available - prepared by Rochester Gas and Electric Corporation.

3. The SEP Safety Evaluation (SE) is currently scheduled for release in April, however, it may be possible to proceed with a plant tour to observe SEP upgrades without the SE.

4. NUREG-0485, "Systematic Evaluation Program, Status Summary Report."

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* Should this date interfere with unit restart, a fall back date of May 19 & 20, 1982 has been picked. Participants will be notified by phone at any change to March dates.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 19	Reliability and Probabilistic Assessment	(GRIESMEYER/QUITTSCHREIBER) Okrent, Bender, Kerr, Siess

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Office of Policy Evaluation

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NUREG-0880, Safety Goals for Nuclear Power Plants: A Discussion Paper, February, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 22, 1982	Structural Engineering	(MCKINLEY/IGNE) Siess, Bender, Ebersole, Shewmon Cons.: Zudans, White

LOCATION: Sandia National Laboratory (Albuquerque, NM)

BACKGROUND:

Who proposed action: C. Siess

Purpose: To review containment integrity program at Sandia.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

None

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 23, 1982	Safeguards & Security	(ALDERMAN/McKINLEY) Mark, Ray, Shewmon, Ward, Siess, Carbon, Mathis, Plesset, Lewis, Consultant: S. Lawroski

LOCATION: Albuquerque, NM
Room A, Building 822 (Technical Area) Sandia National Laboratories

BACKGROUND:

Who proposed action: J.C. Mark

Purpose: To discuss design features in proposed standard design plants that would make sabotage by insiders more difficult.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Status report February 24, 1982 which includes report by Ken Kirby, Summary of ACRS Positions on Industrial Security" (Jan. 28, 1982).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MARCH 24	ECCS	(BOEHNERT) Plesset, Ebersole, Mathis, Ward, Mark (½ day), Cons.: Catton, Garlid, Schrock, Theofanous, Zudasn

LOCATION: Albuquerque, NM

BACKGROUND:

Who proposed action: M. Plesset

Purpose: To discuss the NRC's use of LOCA/ECCS codes and aspects of the recent Ginna transient that had impact on LOCA/ECCS concerns.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in the near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 25, 26 & 27	ADVANCED REACTORS	(IGNE/BOEHNERT) Carbon, Mark, Cons: Avery, Golden, Hartung, Lipinski, Siegal

LOCATION: Argonne, IL

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To continue discussion and preparation of "Safety Issue and Philosophy of LMFBR" report to the ACRS.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Subject Report as revised.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 25 (p.m.) 26	Reliability and Probabilisitc Assessment	(GRIESMEYER/QUITTSCHREIBER) Okrent, Bender, Ebersole, Ward, Siess, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To discuss the Zion Probabilistic Safety Study.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 30	AC/DC Power Systems Reliability	(SAVIO) Ray, Ebersole, Mathis, Okrent

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the status of the NRC work on Task Action Plan A-44 and the NRR Implementation of the recommendation of NUREG-0666.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 30 (pm) March 31 (am)	Clinch River Breeder Reactor	(BOEHNERT/IGNE) Carbon, Bender, Mark Cons: Lipinski, Kastenberg, Zudans

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: Carbon

Purpose: To review CRBR General Design Criteria.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

"CRBR Plant Principal Design Criteria" - Sent to Subcommittee and Consultants via E. Igne 2/17/82 memo.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 31	Combined ECCS/ Electrical Systems Subcommittee	(BOEHNERT/SAVIO) Kerr, Ebersole, Plesset, Ray, Lewis, Bender, Etherington

LOCATION: Washington, DC

BACKGROUND:

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
March 31	Nuclear Safety Research Program	(DURAISWAMY) Siess, Okrent. Kerr, Plesset, Shewmon (tent.), Mark, Ward, Carbon (part-time)

LOCATION: Washington, DC

BACKGROUND:

Purpose: To continue discussion of the NRC Long-Range Research Program Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Final Draft of the Long-Range Research Plan is expected to be made available to the ACRS in the middle of March.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
APRIL 15	Systematic Evaluation Program	(MAJOR/ALDERMAN) Siess, Lewis, Mathis, Moeller, Ward Cons: Fitzsimmons (others as needed)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review the completion of the Systematic Evaluation Program review on Palisades. Palisades is the lead SEP plant. This will be the test case for deciding how the Committee deals with the other ten SEP plants. Palisades will be attempting to upgrade their provisional operating license to a full-term operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The SEP Safety Evaluation Report is scheduled to be issued prior to the end of March 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
April 21-22, 1982	Wolf Creek Station	(RKM/DRB) Ray, Mark
		Consultants: J. C. Maxwell

LOCATION: Site visit - Meeting to be held in Emporia, Kansas
(Flights in are most convenient to Wichita, Kansas)

BACKGROUND:

Who proposed action: Staff and ACRS

Purpose: To visit the site and to review the application for an operating license. (WC is - 50 miles south of Topeka, Kansas)

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The plant safety evaluation report is due on April 7, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
April 26 & 27	WPPSS-2	(GRIESMEYER/QUITTSCHREIBER) Plesset, Ebersole, Mark, Mathis

LOCATION: Hanford, WA

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER due March 6, 1982 without seismic evaluation.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
April 27	Transportation of Radioactive Materials	(DURAISWAMY) Siess, Mark, Bender Cons: Zudans, Langhaar, Shappert

BACKGROUND:

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
APRIL 28 & 29 (tent.)	Watts Bar	(BEAL/QUITTSCHREIBER) - Ebersole, Bender, Ward

LOCATION: Knoxville/Chattanooga area

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for an OL and site visit.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER due April 5, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
(About mid-April)	Metal Components	(IGNE) Shewmon, Ward, Axtmann, Bender, Etherington, Mathis, Plesset Consultants: Kouts, Theofanous, Catton, Zudans, Irwin, Abbott, Binford, Fitzsimmons

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: P. G. Shewmon

Purpose: To continue the review regarding pressurized thermal shock.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The NRC Staff SER and guidance for continued operation documents are scheduled to be available in April or May.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
APRIL	REACTOR RADIOLOGICAL EFFECTS	(ALDERMAN/MCKINLEY) Moeller, Axtmann, Ray

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: To review control room habitability and to discuss the design basis for normal and abnormal conditions, testing, and research needs.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
late April early May	Ad Hoc Metal Components Subgroup	(IGNE) Bender, Shewmon, Etherington, Okrent, Plesset, Ward, Axtmann, Mathis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To review the NRC Staff's action plan on pressurized thermal shock before a Staff position is issued on this matter.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be forwarded later.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 4 & 5 (tent.)	CRBR	(BOEHNERT) Carbon, Mark

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To begin Subcommittee review of the CDA energetics issue for CRBR licensing.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Will be provided later.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 5, 1982	TMI-2 Action Plans	(Major), Mathis, Etherington, Lewis, Okrent

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: W. Mathis

Purpose: To review the proposed rule on Licensing Requirements for Pending Operating License Applications (Rule contains the Basic Requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements"). This will be the second meeting with the Staff on this rule. Public comments should have been evaluated and incorporated into the final form of the rule prior to Subcommittee meeting. The will have been reviewed by the CRGR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

The final form of the rule is expected to be available by mid to late February.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Late May	Perry	(SAVIO) Ray, Axtmann, Bender, Okrent

LOCATION: Cleveland, Ohio (tentative)

BACKGROUND:

Who proposed action: NRR

Purpose: To review the OL application and to conduct a site visit.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NRR has committed to supply an SER by May 10, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Wk. of June 14	ECCS	(BOEHNERT) Plesset, Ebersole, Etherington, Ward

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To discuss: (1) GE's request for change in Appendix K decay heat requirements; NRR code audit capability; (3) LOFT ATWS test vendor code predictions and results; (4) NRR work on operator accident guidelines and procedures.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June or July	Combined Clinch River Breeder Reactor and Site Suitability	(BOEHNERT/ALDERMAN) Carbon, Moeller, Bender, Mark, Okrent, Plesset, Shewmon, Siess, Axtmann, Ebersole, Ray
LOCATION: Washington, DC		Cons: to be determined

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To begin site suitability review for CRBR

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Site Suitability Report by the Office of Nuclear Reactor Regulation, USNRC in the matter of the Clinch River Breeder Reactor Plant, dated March 4, 1977 (to be revised in June or July).

NUREG-0833 "Environmental Impact Statement on the Siting of Nuclear Power Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To be Determined	Class 9 Accidents	(BEAL/QUITTSCHREIBER) Kerr, Axtmann, Bender, Moeller, Okrent, Siess, Ward

LOCATION: To be Determined

BACKGROUND:

Who proposed action:

Purpose: To review severe accident research plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NUREG-0900 (Draft), "Nuclear Plant Severe Accident Research Plan.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
Fall 1982	Reactor Radiological Effects	(ALDERMAN/McKINLEY) Moeller, Ray, Axtmann

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: D. Moeller

Purpose: Review NUREG-0833 "Environmental Impact statement on the siting of nuclear power plants" and obtain an update on the current NRC Staff thoughts on siting.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NUREG-0833

NRC STAFF
PRESENTATION TO THE ACRS
FOR BYRON STATION,
UNITS 1 AND 2
MARCH 4, 1982

PREPARED BY
STEPHEN H. CHESNUT
LICENS 'G PROJECT MANAGER

STAFF CONCLUSIONS

UPON FAVORABLE RESOLUTION OF OUTSTANDING MATTERS, AS DESCRIBED IN THE SER (NUREG-0876):

- o APPLICATION FOR LICENSE COMPLIES WITH NRC REQUIREMENTS
- o REASONABLE ASSURANCE THAT FACILITY CONSTRUCTION WILL BE COMPLETED AND PLANT WILL BE OPERATED IN CONFORMANCE WITH NRC REQUIREMENTS
- o REASONABLE ASSURANCE THAT ACTIVITIES AUTHORIZED BY OL CAN BE CONDUCTED IN COMPLIANCE WITH REGULATIONS AND WITHOUT ENDANGERING PUBLIC HEALTH AND SAFETY
- o APPLICANT IS TECHNICALLY AND FINANCIALLY QUALIFIED
- o ISSUANCE OF LICENSE WILL NOT BE INIMICAL TO COMMON DEFENSE AND SECURITY OR TO PUBLIC HEALTH AND SAFETY

COMPARISON OF BYRON

WITH OTHER PREVIOUSLY

REVIEWED FACILITIES

DESIGN COMPARISON

SYSTEM/CONTAINMENT	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
1. CONTAINMENT	ZION	3 BUTTRESS DESIGN (6 AT ZION)
2. REACTOR INTERNALS	INDIAN POINT	BYRON HAS 17 X 17 FUEL AND NEUTRON SHIELDS SIMILAR TO TROJAN
3. REACTOR		
o FUEL	COMANCHE PEAK	
o NUCLEAR DESIGN	COMANCHE PEAK	
o THERMAL HYDRAULIC DESIGN	COMANCHE PEAK D. C. COOK 2	
o INTERNALS	INDIAN POINT 2	BYRON HAS "TOP HAT" DESIGN
o REACTIVITY CONTROLS	COMANCHE PEAK	
4. REACTOR COOLANT SYSTEM	COMANCHE PEAK	
o REACTOR VESSEL	COMANCHE PEAK	
o REACTOR COOLANT PUMPS	COMANCHE PEAK	
o STEAM GENERATORS	COMANCHE PEAK	UNIT 2 IS TYPE D5 WITH IMPROVED MATERIAL FOR TUBE SUPPORT

DESIGN COMPARISON

SYSTEM/CONTAINMENT	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
o RESIDUAL HEAT REMOVAL	COMANCHE PEAK	
o PRESSURIZER	COMANCHE PEAK	
o VALVES	COMANCHE PEAK	BYRON HAS LOOP STOP VALVES
5. ENGINEERED SAFETY FEATURES		
o ECCS	COMANCHE PEAK	
6. INSTRUMENTATION AND CONTROLS		
o REACTOR TRIP SYSTEM	SYSTEM FUNCTIONS SIMILAR TO SNUPPS	
o ESF SYSTEMS		
o CONTROL SYSTEMS		
7. RADIOACTIVE WASTE MANAGEMENT		
o SOURCE TERMS	MCGUIRE	
o GAS WASTE MANAGEMENT	WATTS BAR	WATTS BAR HAS DOUBLE THE NUMBER OF GAS DECAY TANKS.
8. ACCIDENT ANALYSIS	COMANCHE PEAK	

DUPLICATED PLANT CONCEPT

- BYRON AND BRAIDWOOD ARE DUPLICATED PLANTS LOCATED AT DIFFERENT SITES
- FINAL DUPLICATE DESIGN APPROVAL WILL BE ISSUED AT CONCLUSION OF BYRON OL REVIEW
- DUPLICATE AND REPLICATE PLANT EVALUATIONS WILL RELY ON BYRON REVIEW FOR DUPLICATED FEATURES
- SITE-SPECIFIC FEATURES AND DIFFERENCES WILL BE CONSIDERED DURING FUTURE PLANT SAFETY REVIEWS.

OPEN ITEMS FROM SER

ITEM	NEXT ACTION	EXPECTED CLOSE
1. PIPELINE FOUNDATION INFORMATION	STAFF	SPRING 1982
2. TURBINE MISSILE EVALUATION	CECO	JUNE 1982
3. HIGH/MODERATE - ENERGY PIPE BREAK ANALYSIS	CECO	MAY 1982
4. PIPE SUPPORT BASE PLATE FLEXIBILITY: EFFECT ON ANCHOR BOLT LOADS	NRC	MAY 1982

ITEM	NEXT ACTION	EXPECTED CLOSE
5. PUMP & VALVE OPERABILITY ASSURANCE PROGRAM	CECO	SUMMER 1982
6. SEISMIC & DYNAMIC QUALIFICATIONS OF MECHANICAL AND ELECTRICAL EQUIPMENT	CECO	JULY 1982
7. ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED ELECTRICAL EQUIPMENT	CECO	JULY 1982
8. IMPROVED THERMAL DESIGN PROCEDURES	CECO	JUNE 1982

ITEM	NEXT ACTION	EXPECTED CLOSE
9. TMI ITEM II.F.2 INADEQUATE CORE COOLING	CECO	JULY 1982
10. STEAM GENERATOR FLOW INDUCED VIBRATIONS	CECO	LATE 1982
11. REACTOR PRESSURE VESSEL FORCES AND MOMENTS	CECO	MAY 1982
12. EQUIPMENT AND FLOOR DRAINAGE SYSTEM FOR INTERNAL FLOOD PROTECTION	CECO	APRIL 1982

ITEM	NEXT ACTION	EXPECTED CLOSE
13. FIRE PROTECTION PROGRAM	CECO	
14. RESIDUAL MOISTURE IN DIESEL AIR START PIPING	- CLOSED -	
15. VOLUME REDUCTION SYSTEM	CECO	JUNE 1982
16. EMERGENCY PREPAREDNESS PLANS AND FACILITIES	CECO/COUNTIES	LATE 1982
17. CONTROL ROOM HUMAN FACTORS REVIEW	CECO	LATE 1982

LICENSE CONDITIONS FROM SER

STATUS

- 11 LICENSE CONDITIONS
- EXPECT SEVERAL TO BE IMPLEMENTED PRIOR TO LICENSING AND THEREFORE WILL NOT BECOME LICENSE CONDITIONS

LICENSE CONDITIONS FROM SER

- GROUNDWATER MONITORING PROGRAM
- STEAM VALVE INSERVICE INSPECTIONS
- ONSHIFT EXPERIENCE DURING START UP PHASE
- COMPLIANCE WITH APPENDIX R OF 10 CFR 50
- POST ACCIDENT MONITORING
- IMPLEMENTATION OF SECONDARY WATER CHEMISTRY PROGRAM
- TMI ITEM II.B.3 - POST ACCIDENT SAMPLING
- MASONRY WALLS
- PRESERVICE AND INSERVICE INSPECTION PROGRAM
- RESPONSE TIME TESTING

AGENDA

BYRON ACRS MEETING

	Duration	Approximate Time (A.M.)
I. Chairman's Report	15 min.	8:45 - 9:00
II. NRC Presentation		
1. Introduction, including relationship of Byron to Braidwoood	10 min.	9:00 - 9:10
2. Status of Review and Comparison of Byron with Similar PWRs reviewed by NRC Staff	10 min.	9:10 - 9:20
3. Summary of Principal Review Issues and Status of Commitments on the TMI Action Plan (dissenting NRC opinions, if any)	20 min.	9:20 - 9:40
III. Presentation by Commonwealth Edison		
1. Brief Description of Corporate Organization	10 min.	9:40 - 9:50
2. Training Program	10 min.	9:50 - 10:00
3. DC Power Systems Reliability	15 min.	10:00-10:15
4. Consideration of PRA and System Interaction Studies; including examples of changes made as a results of PRA.	15 min.	10:15-10:30
*** BREAK ***	15 min.	10:30-10:45
5. Auxiliary Feedwater Reliability	10 min.	10:45-10:55
6. Emergency Planning	15 min.	10:55-11:10
7. Effect of Residual Elements on Reactor Pressure Vessel and Concerns on High-Strength Bolts	10 min.	11:10-11:20
8. Secondary Water Chemistry	10 min.	11:20-11:30
9. Reactor Vessel Level Indication	10 min.	11:30-11:40
IV. Executive Session	20 min.	11:40-12:00
V. Adjournment		12:00

BYRON NUCLEAR POWER STATION

COMMONWEALTH EDISON

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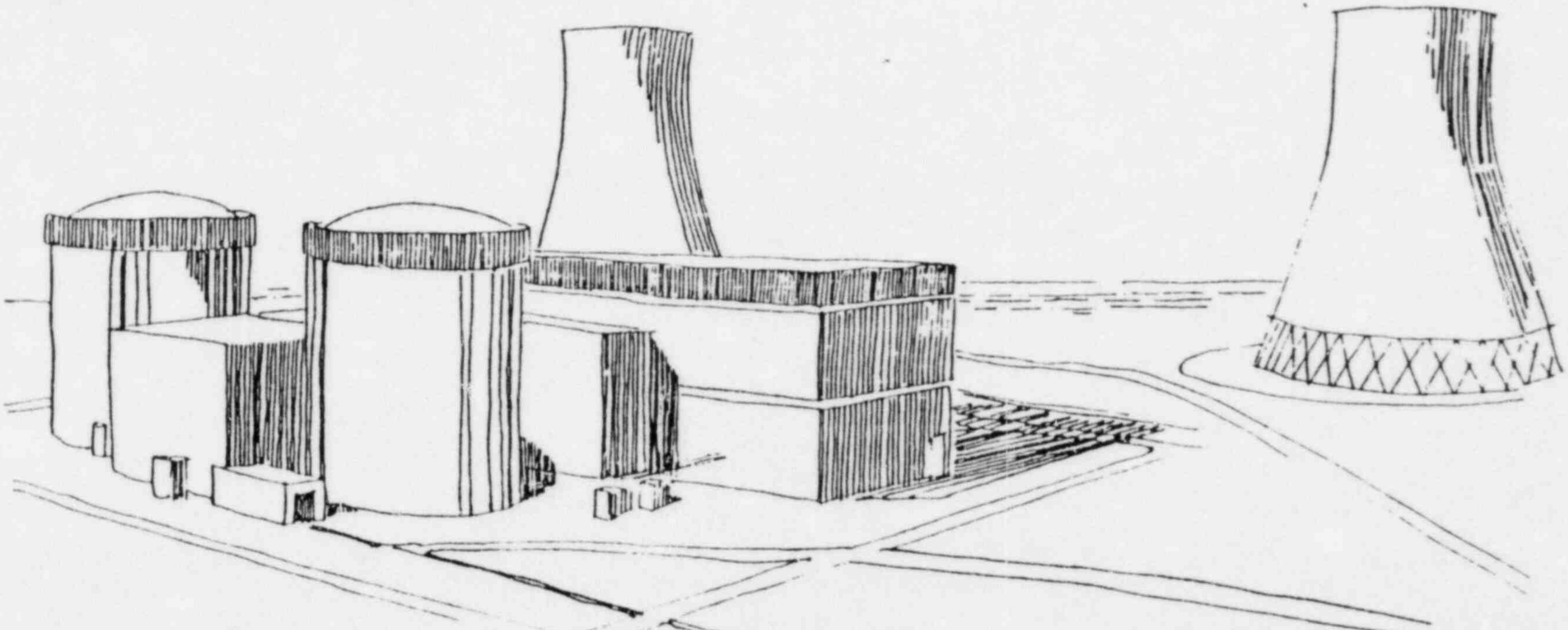

FIGURE 1

COMMONWEALTH EDISON COMPANY CORPORATE ORGANIZATION

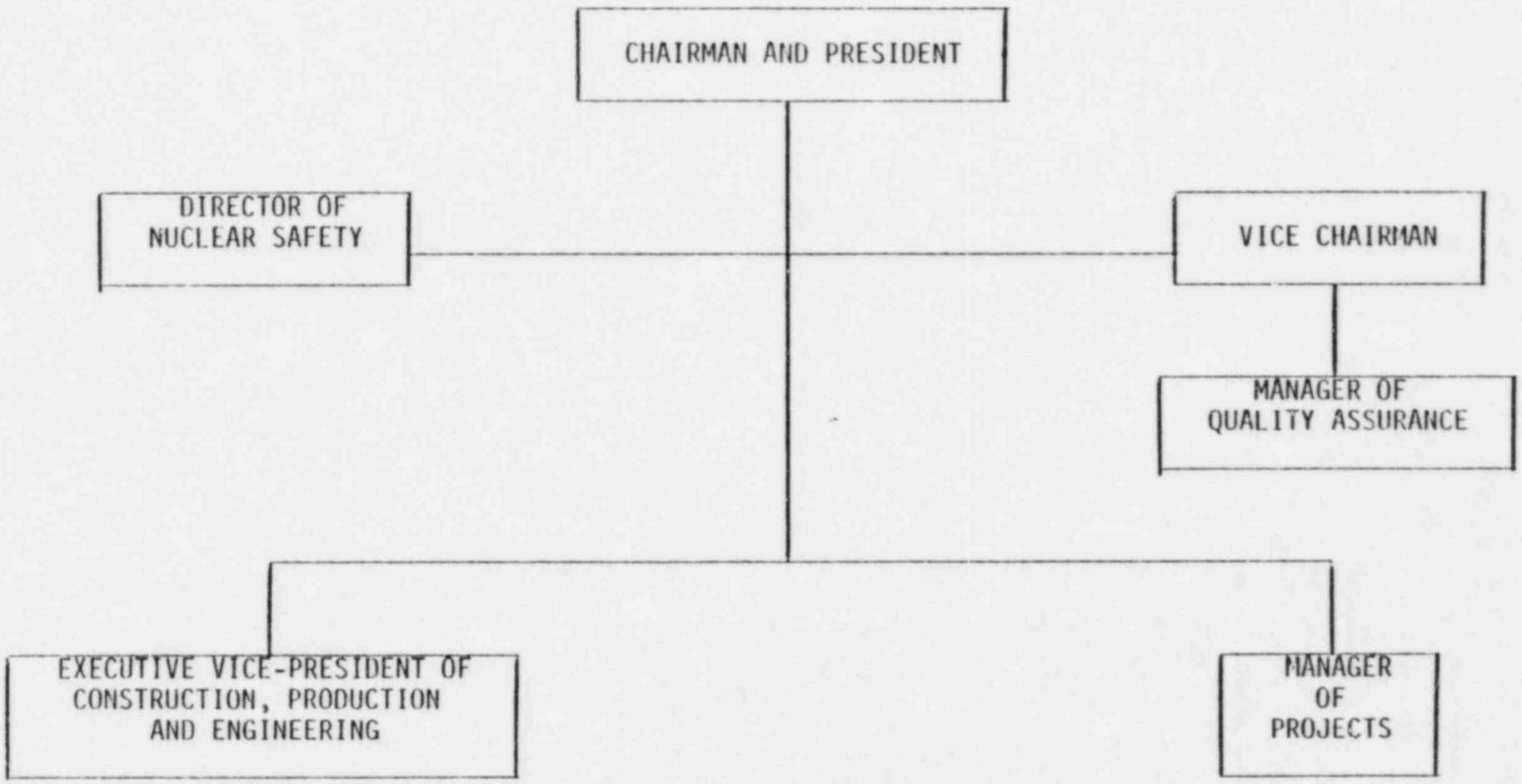

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FIGURE 2

NUCLEAR OPERATIONS ORGANIZATION

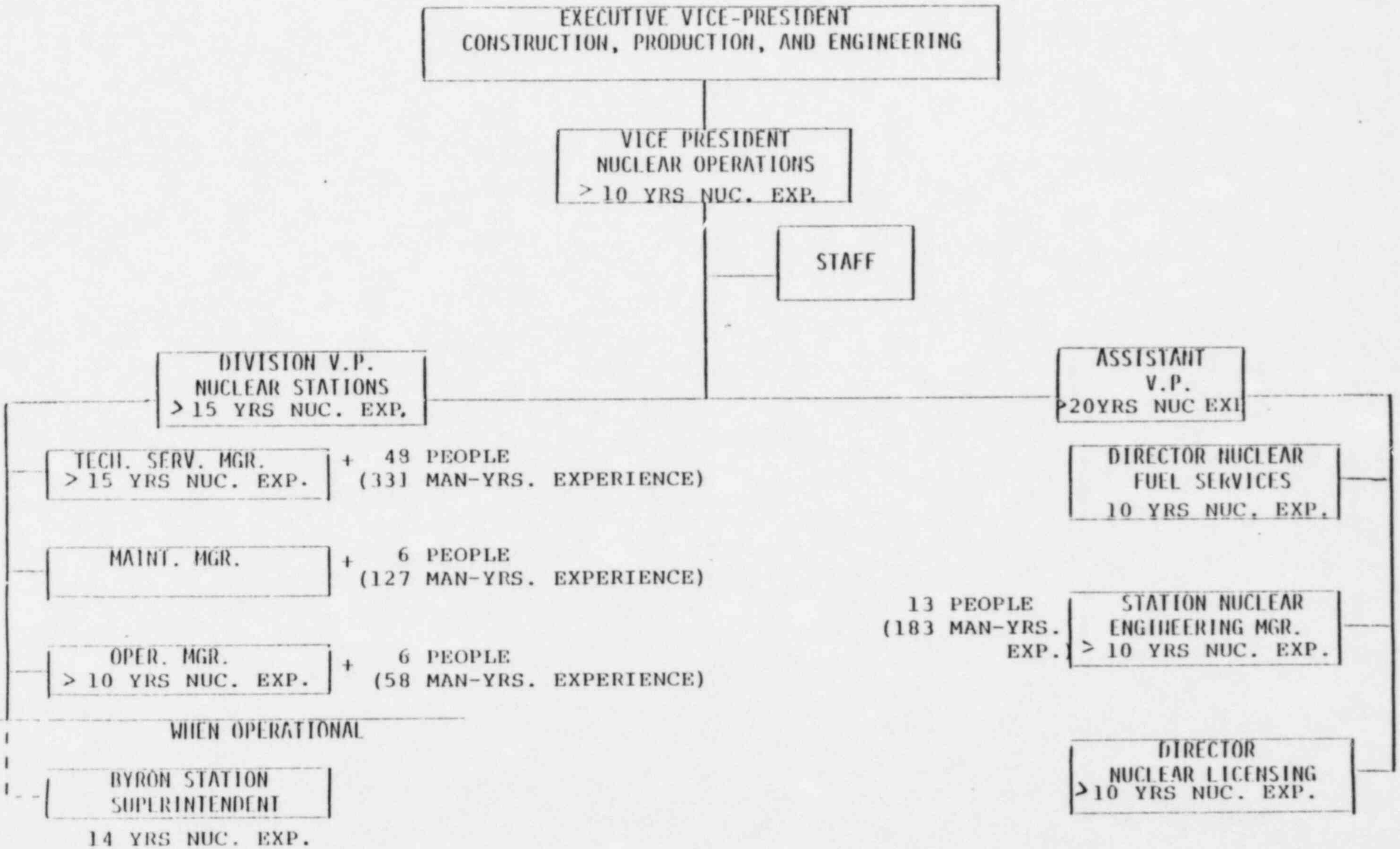

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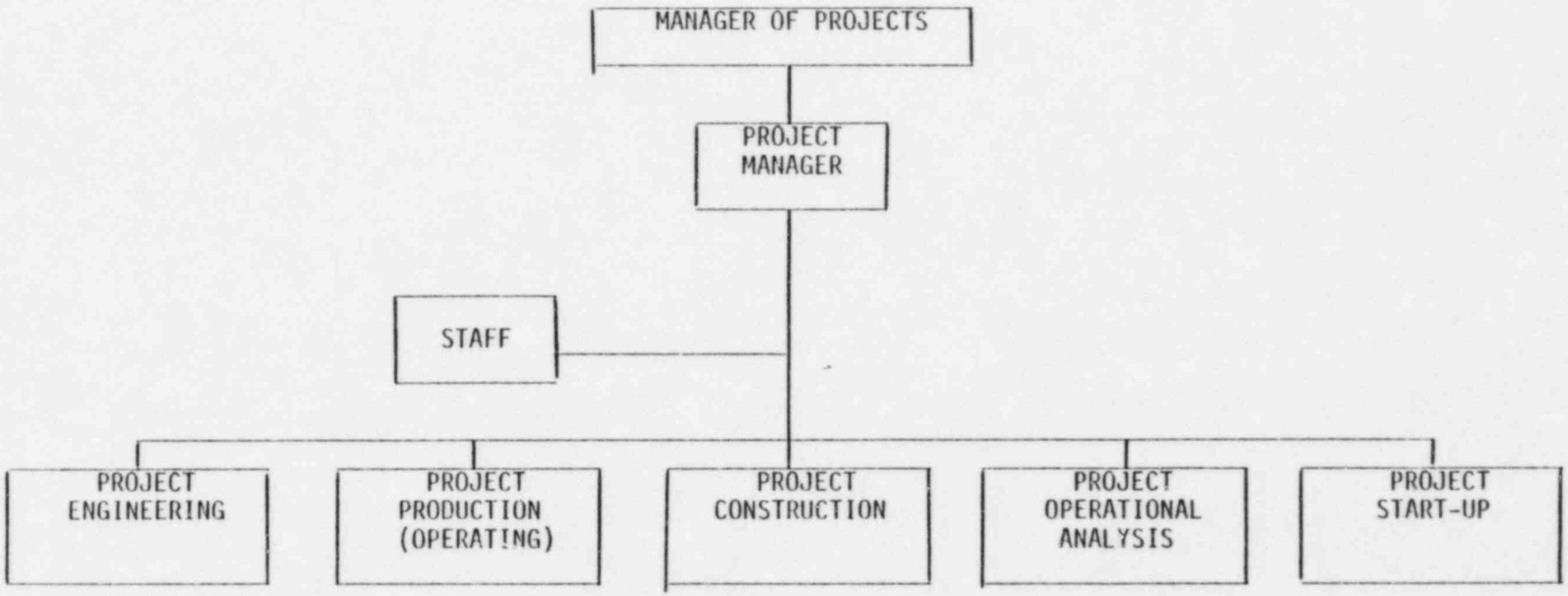
FIGURE 3

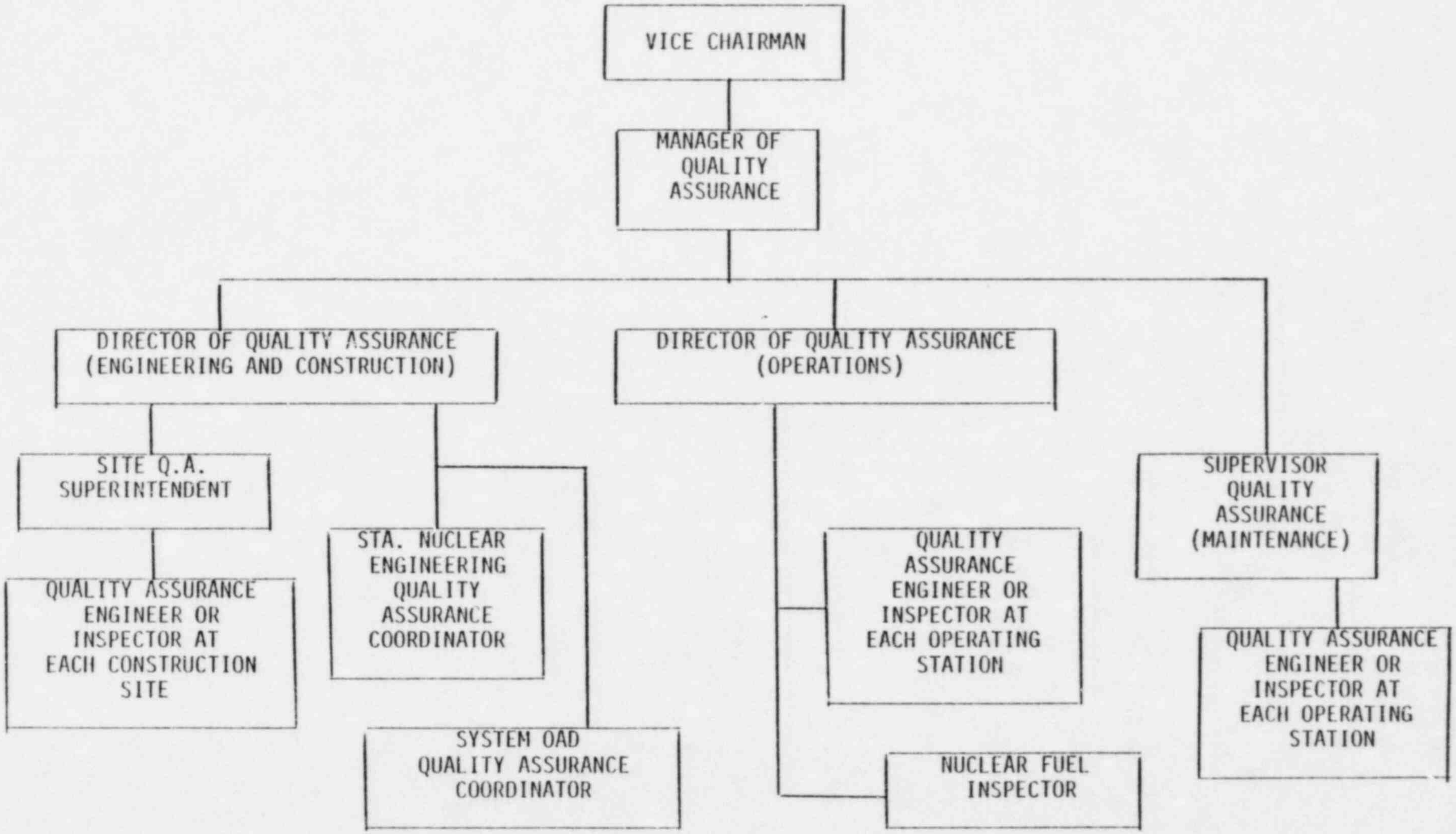
STARTUP ORGANIZATION



A-5-8

FIGURE 4

QUALITY ASSURANCE ORGANIZATION



A-5-9

FIGURE 5

NUCLEAR SAFETY ORGANIZATION

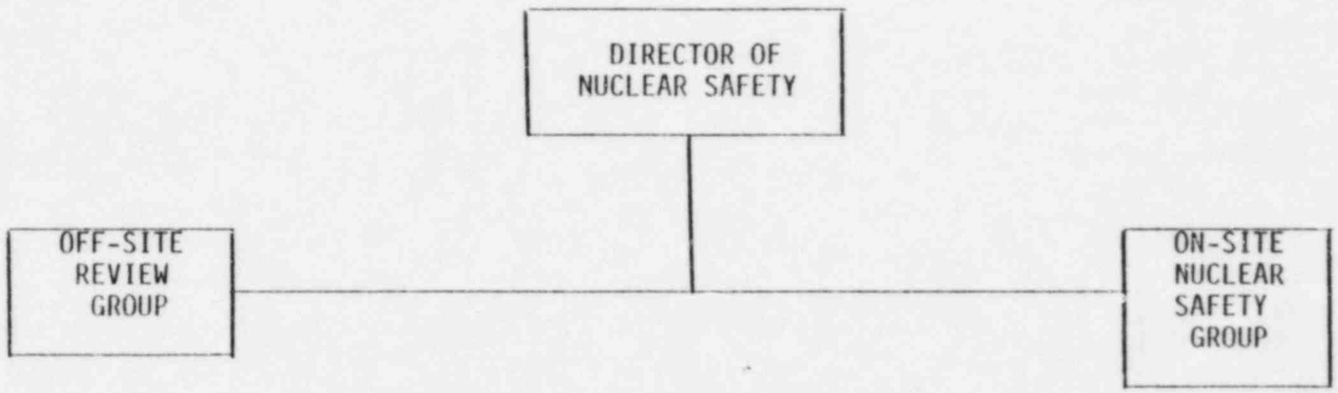


FIGURE SIX

CORPORATE LEVEL EXPERIENCE

TECHNICAL RESOURCES - MANAGERS

POSITION	DEGREE(S)	LICENSE(S)	NUCLEAR EXPERIENCE (YRS.)
VP-NUC. OPS.	BSME	SRO(BWR)-L(1)	>10
·DIV. VP -NUC. STA.	BSME	_	15
·OPS. MRG.	BSME/MBA	SRO(BWR)-L	>10
·MNT. MGR.	BSME	-	
·TECH.SERV.MGR.	BSEE	SRO(BWR/PWR)-L	>15
·ASST. VP- NUC.	BSEE/MBA	-	>20
·SNED MGR.	BSME	-	>10
·DIR. NUC.LIC.	BSES/JD	-	>10
·DIR. NFS	PUD-NE	-	10
·PROD. TRNG.MGR.	BSE/MSME/JD	EOW	10
·DIR.NUC.SAF.	BSE/MSBA	EOW	> 25
·SUPV. OSR	BSEE	-	>20
·SUPV. ONSG	BSME	SRO(PWR)-(A)(2)	>10
MGR. OF PROJ.	BSEE	-	5
·B/B PROJ.MGR.	BSME	-	> 10
·PROJ. ENG.MGR.	BS ENG./MSNE	-	> 20

(1) LAPSED
(2) ACTIVE

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FIGURE SIX (Continued)

NUCLEAR STATIONS DIVISION

MAINTENANCE - NUCLEAR

		TOTAL	6
MAN-YEARS EXPERIENCE IN DISCIPLINE			127
B.S. MECH. ENG.	1		
B.S. ELECT. ENG.	1		
B.S. METALLURGY	2		
B.S. WELD. ENG.	1		
M.S. MAT'l. SC.	1		

OPERATIONS - NUCLEAR

(OPERATIONS AND SECURITY)		TOTAL	6
MAN-YEARS EXPERIENCE IN DISCIPLINE			58
MAN-YEARS NUCLEAR PLANT EXPERIENCE			31
B.S. MECH. ENG.	1		
B.S. PHYSICS.	1		
B.S. PSYCHOLOGY	1		
B.A. LAW ENFORCEMENT	1		
MBA	2		

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FIGURE SIX (Continued)

NUCLEAR STATIONS DIVISION (CONT.)

TECHNICAL SERVICES - NUCLEAR			2
·HEALTH PHYSICS AND EMERGENCY PLANNING			13
·CHEMISTRY AND RAD WASTE SERVICES			26
·STATION SUPPORT SERVICES			7
		TOTAL	48
MAN-YEARS EXPERIENCE IN DISCIPLINE			331
MAN-YEARS NUCLEAR PLANT EXPERIENCE			125
B.S. BIOLOGY	1		
B.S. MATH	1		
B.S. FIRE PROTECTION	1		
B.S. MECH. ENG.	1		
B.S. ENG.	1		
B.S. ELECT. ENG.	3		
B.S. METEOROLOGY	1		
B.A. CHEM	1		
B.S. BIO CHEM	2		
B.S. CHEM. ENG.	2		
B.S. CHEM	10		
M.S. CHEM	2		
PH.D. CHEM	1		
PH.D. NUC. CHEM.	1		
PH.D. PUB. HEALTH	1		
M.S. NUC. ENG.	6		
M.S. ENV. HEALTH	1		

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FIGURE SIX (Continued)

STATION NUCLEAR ENGINEERING DEPARTMENT

RELIABILITY AND SPECIALIST GROUP

	TOTAL	13
MAN-YEARS EXPERIENCE IN DISCIPLINE		183
MAN-YEARS NUCLEAR PLANT EXPERIENCE		72

B.S. MECH. ENG.	3
B.S. ELECT. ENG.	1
B.S. CHEM.	1
B.S. METALLURGY	2
M.S. MECH. ENG.	2

FIGURE SIX (Concinued)

PRODUCTION TRAINING DEPARTMENT

NO. IN DEPARTMENT	YEARS NUCLEAR EXPERIENCE	DEGREES/TYPES*	LICENSES
56 MANAGEMENT	658	4 - 2 YR.	10 SRO
		23 - 4 YR.	1 RO
		17 - ADVANCED	

*DEGREE TYPES

2 YR.	4 YR.		ADVANCED
4 A.A.S.	1 B.S. CHEM	1 B.S. CHEM ENG	1 M.S. CHEM
	3 B.S. PHYSICS	1 B.S. AERO	3 M. ED.
	3 B.S. BIOLOGY	1 B.S. METAL	1 M.A. VOC.ED
	1 B.S. BIO SCI	1 B.A. SOC SCI	1 M.A. ED.
	2 B.S. ENG PHYSICS	1 B.A. SCI	2 MBA
	1 B.S. BIO CHEM	1 B.A. BUS	2 M.S.M.E.
	1 B.S. ENG	1 B.S. ACCOUNT	1 M.S.N.E.
	1 B.S. MGT	1 B.S. IND TECH	1 M.S.E.E.
	1 B.S. NUC ENG	1 B.S. ED	1 PHD N.E.
			1 NAV. ENG.
			2 PHD ED
			1 J.D.

FIGURE 7

BYRON STATION OPERATING ORGANIZATION

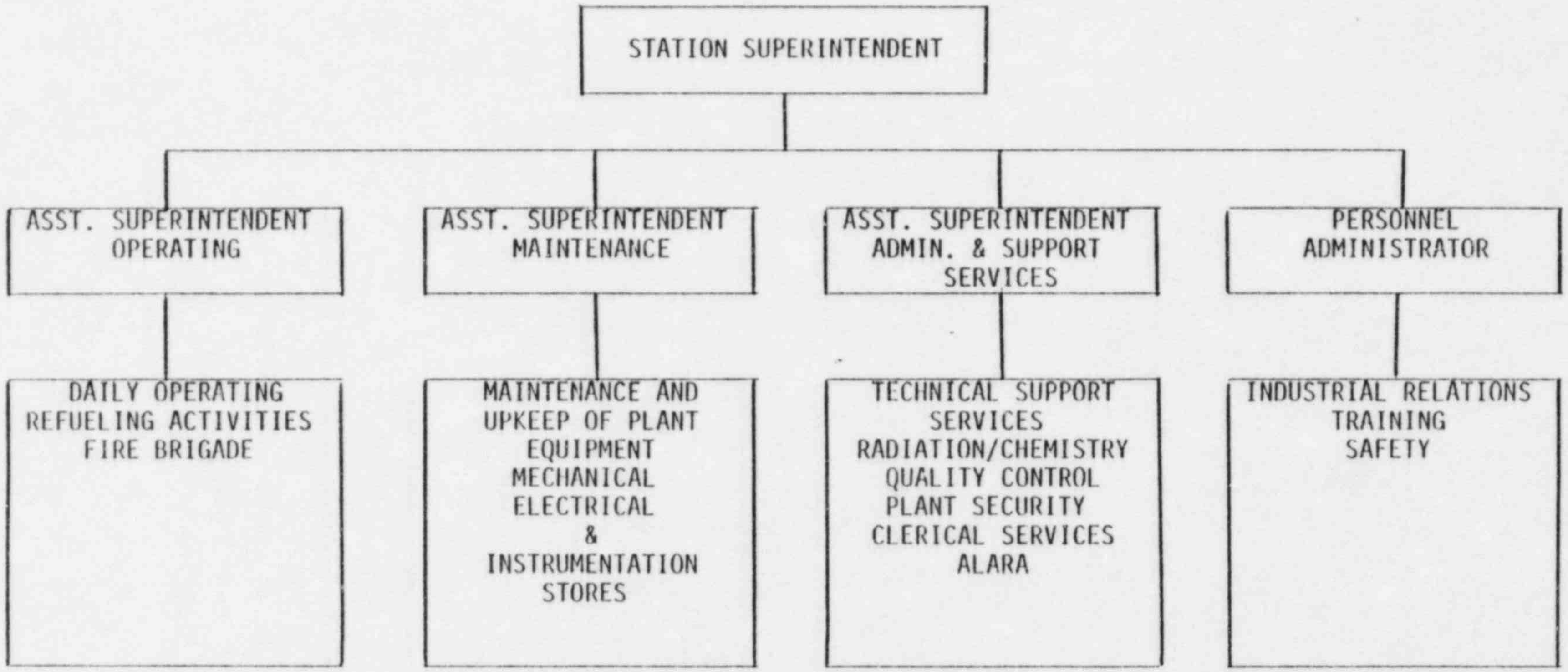

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FIGURE 8

OPERATING ORGANIZATION

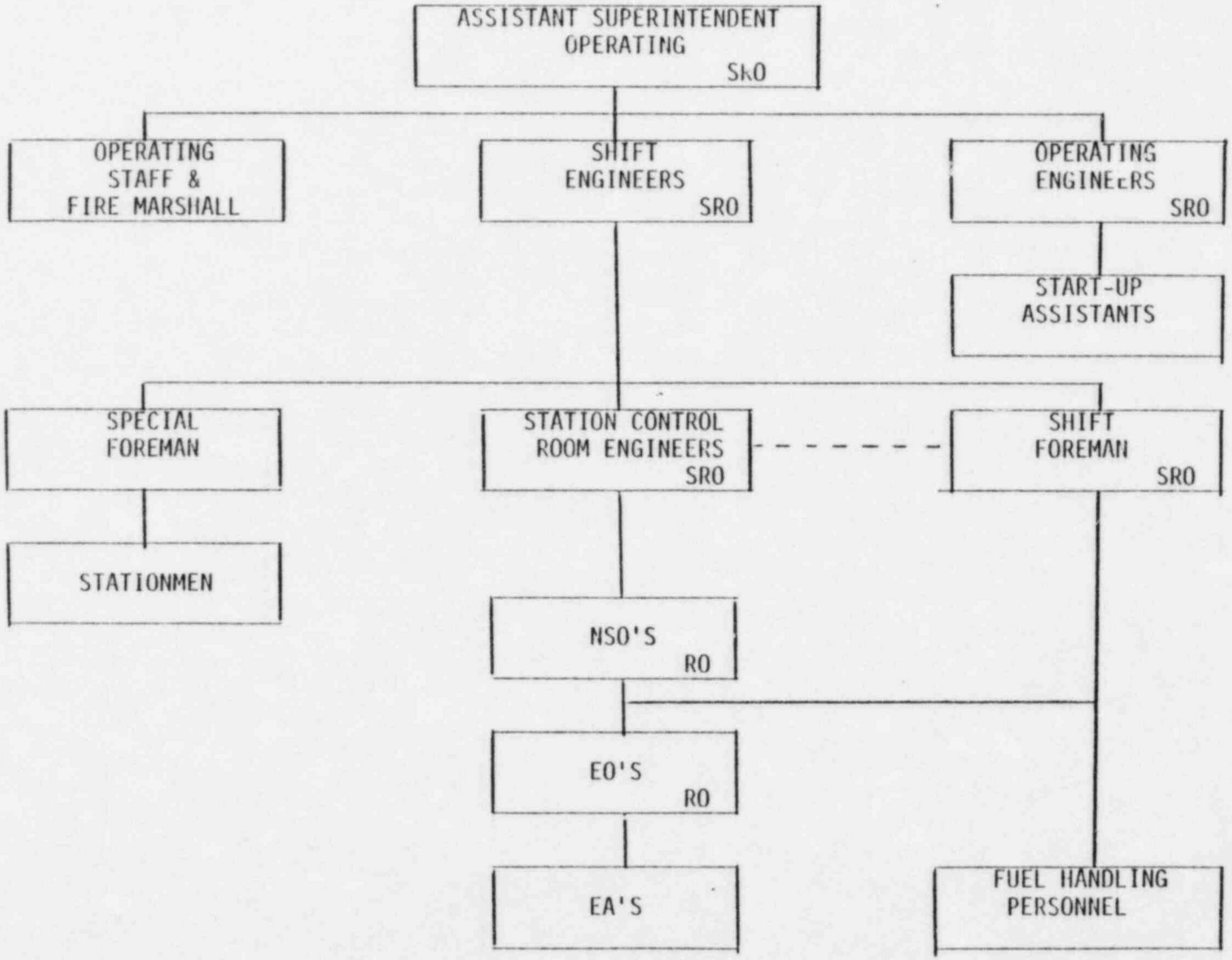

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FIGURE 9

MAINTENANCE ORGANIZATION

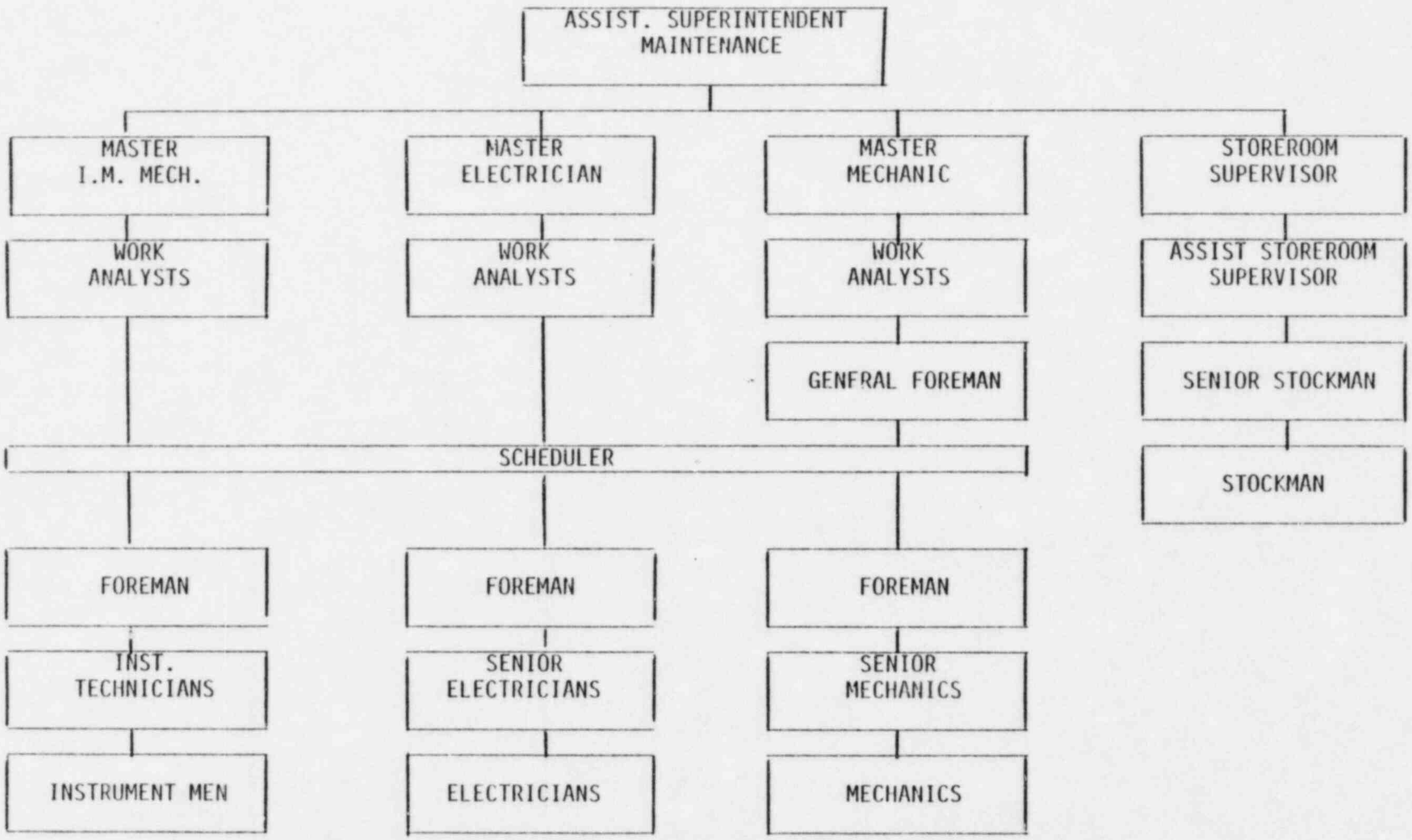

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FIGURE 10

ADMINISTRATIVE AND SUPPORT SERVICES ORGANIZATION

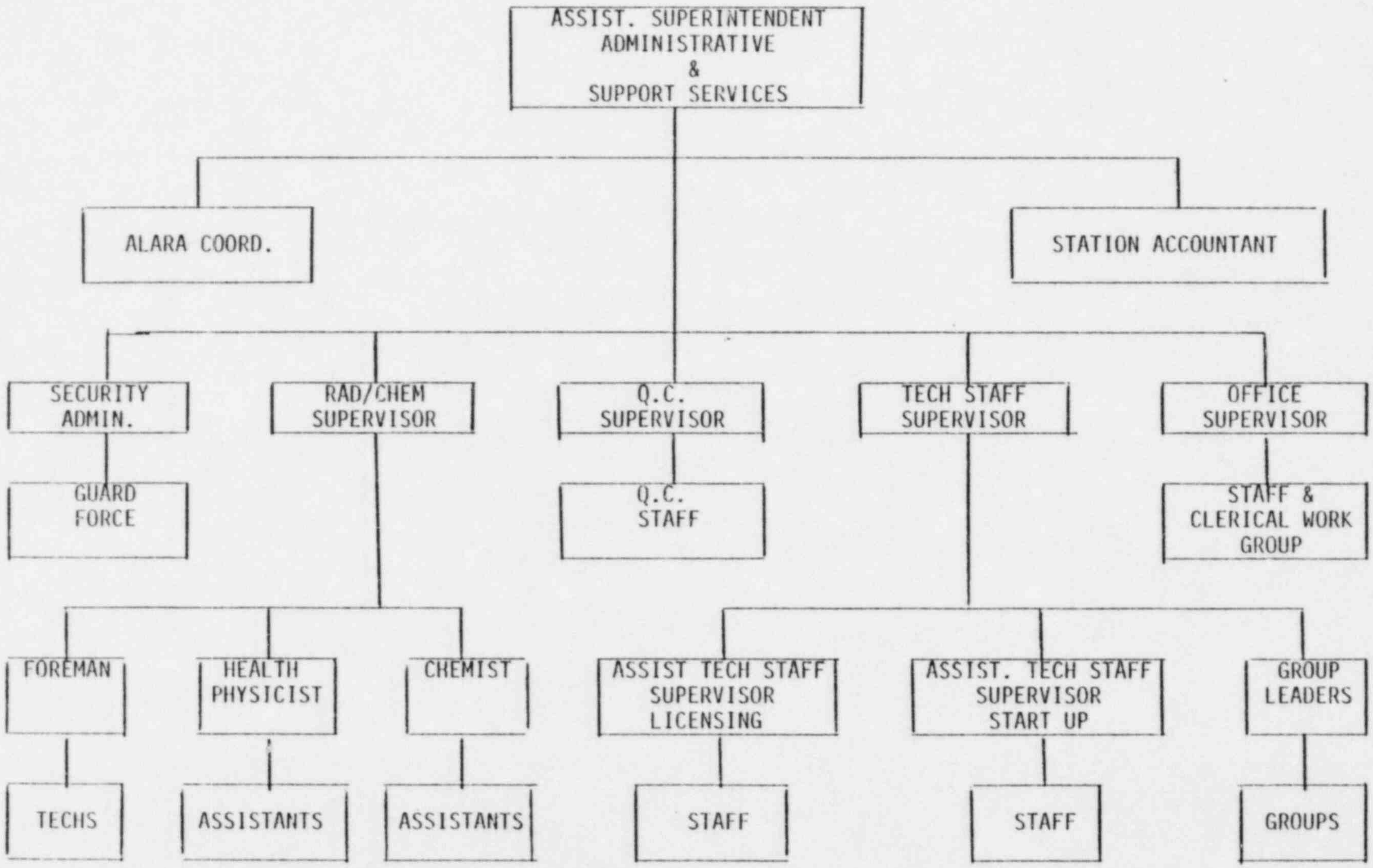

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FIGURE 11

INDUSTRIAL RELATIONS AND TRAINING ORGANIZATION

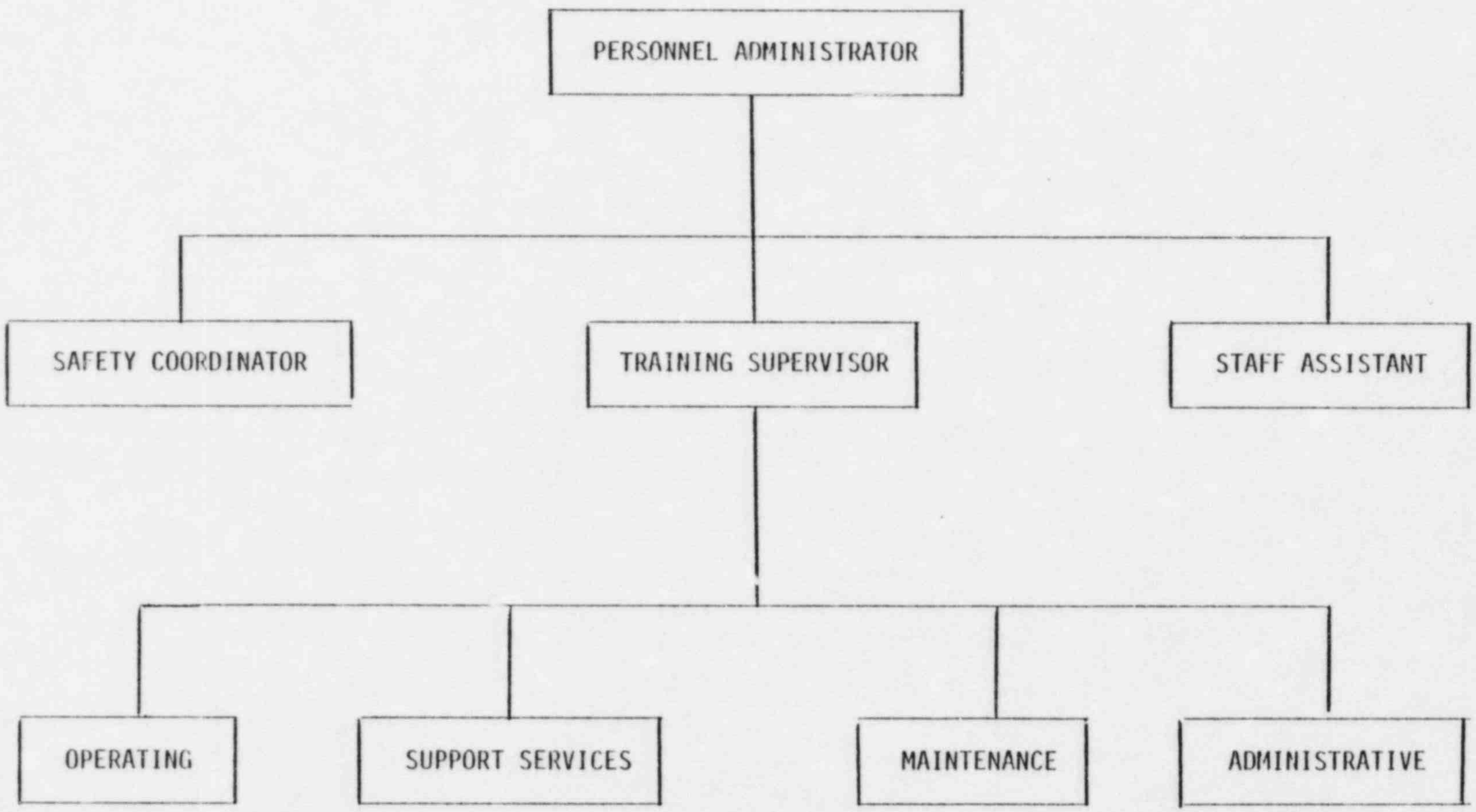

FIGURE 12

AVERAGE EXPERIENCE LEVELS IN YEARS

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Job Title		Ave. Commercial Nuc.	Ave. Military Nuc.	Total Other*	Total Experience	Total Exp. Per Job Title
Station Supt.	(1)	14	--	3	17	17
Op. Ass't. Supt.	(1)	9	6	--	15	15
Maint. Ass't. Supt.	(1)	12	--	1	13	13
Admin. Ass't. Supt.	(1)	9	6	--	15	15
Operating Eng.	(4)	9.8	--	20	59	14.8
Shift. Eng.	(6)	8.2	4.8	--	78	13
Shift Foreman	(6)	6.8	4.2	--	66	11
SCRE	(6)	5.0	1.3	3	41	6.8
NSO	(18)	4.8	--	13	100	5.6
EA	(59)	0.8	0.3	4.5	70	1.2
Op. Staff	(5)	6.6	--	83	116	23.2
Lead Mech.	(3)	6.3	--	20	39	13
Rad/Chem Supv.	(1)	6	--	2	8	8
Rad/Chem Mgmt.	(9)	4.2	--	4	42	4.7
Rad/Chem Tech.	(6)	4	--	8	32	6.4
Q.C. Supv.	(1)	11	--	1	12	12
Q.C. Staff	(5)	3.6	--	13	31	6.2
Security Admin.	(1)	6	--	3	9	9
Tech. Staff Supv.	(1)	7	--	6	13	13
Tech. Staff	(63)	2.3	0.4	36	208	3.3
Training Supv.	(1)	13	8	--	21	21
Nuclear Training Staff	(15)	3.6	2.8	60	157	10.5
Others**	(128)	3.8	--	563	1006	7.8
Totals	(342)	--	--	--	2262	--

* Indicates Power Plant Related Experience Areas
**Indicates Remainder of Plant Staff other than Clerical Staff.

The average experience level of the technical personnel in this figure is approximately 6.3 years.

REGULATIONS

10CFR55

OPERATOR'S LICENSES

ANSI 18.1(1971)

SELECTION AND TRAINING OF NUCLEAR POWER PLANT PERSONNEL

NUREG 0094

NRC OPERATOR'S LICENSING GUIDE

NUREG 0737

CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

Figure 2

OPERATOR TRAINING

EQUIPMENT ATTENDANT	HIGH VOLTAGE SWITCHING
FUNDAMENTALS	SWITCHYARD
ELECTRICAL	TRANSMISSION
MECHANICAL	IN PLANT ELECTRICAL DISTRIBUTION
1 WEEK ON SHIFT	AUXILIARY POWER
PLANT SYSTEMS	MAIN GENERATION
2 WEEKS ON SHIFT	EMERGENCY POWER
PLANT SYSTEMS	COMMUNICATION/TOURS
2 WEEKS ON SHIFT	SAFETY
•	REVIEW
•	EXAM
•	LOAD DISPATCHER REVIEW BOARD
•	
REVIEW	
WRITTEN TEST	
PLANT WALK-THRU	

LICENSE TRAINING

COLD

PHASE I

THEORY

NTR

PHASE II

PWR SYSTEMS

PHASE III

PWR OPERATION ORIENTATION

SIMULATOR CERTIFICATION

HOT

NUCLEAR SCIENCE

REACTOR SCIENCE

RADIOLOGICAL SCIENCE

THERMAL SCIENCE

NTR

PRIMARY SYSTEMS

SECONDARY SYSTEMS

SAFETY SYSTEMS

PRIMARY SUPPORT SYSTEMS

SECONDARY SUPPORT SYSTEMS

OPERATING PROCEDURES

ADMINISTRATIVE PROCEDURES

PRE-LICENSE REVIEW

NRC EXAMINATION

SCRE/STA

TECHNICAL GRADUATE
SRO LICENSED
STA TRAINING

MANAGEMENT

COMMUNICATIONS
DECISION MAKING
PROBLEM SOLVING

ENGINEERING CONCEPTS

ELECTRICAL SCIENCES
REACTOR CHEMISTRY
REACTOR THEORY
NUCLEAR I&C
NUCLEAR MATERIALS
NUCLEAR RAD. PROT. &H.P.
THERMAL SCIENCES

ADMINISTRATION

TECHNICAL SPECIFICATIONS
ADMINISTRATIVE PROCEDURES
FEDERAL REGULATIONS
GSEP

ABNORMAL OPERATING EVENTS

SIMULATOR

5

COMMONWEALTH EDISON

MECHANICAL MAINTENANCE TRAINING

1. MECHANICAL MAINTENANCE I

A. Plant Safety and Procedures
B. Store Items
C. Labeling Storage of Parts
D. Math Review
E. Tools and Their Uses
F. Welding Equipment
G. Basic Rigging
H. Fork Lift Truck Operation and Safety and Operation of Trucks

2. MECHANICAL MAINTENANCE II

A. Lathe
B. Drill Press
C. Blueprint Reading
D. Precision Tools
E. Plant Safety and Procedures

3.MECHANICAL MAINTENANCE III

A. Rigging
B. Cranes
C. Lubrication
D. Bearings
E. Gears
F. Milling Machine
G. Shaper
H. Gaskets and Packing

4. MECHANICAL MAINTENANCE IV

A. Piping and Tubing
B. Valve Maintenance
C. Pump Maintenance

5. MECHANICAL MAINTENANCE VI

A. Beginning Welding

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COMMONWEALTH EDISON
ELECTRICAL MAINTENANCE TRAINING

1. ELECTRICAL MAINTENANCE I
   - A. Math Review
   - B. Introduction to Electricity and Electronics
   - C. Batteries and D.C. Circuits
   - D. Transformers and A.C. Circuits
   - E. Electrical Measuring Instruments
   - F. Electrical Protective Devices
   - G. D.C. Equipment
   - H. Single Phase Motors
   - I. Three Phase Motors
   - J. A.C. Equipment
   - K. Electrical Trouble Shooting

2. ELECTRICAL MAINTENANCE II
   - A. Math Review
   - B. Introduction to Electricity and Electronics
   - C. Solid State I
   - D. Solid State II
   - E. Solid State III

COMMONWEALTH EDISON

INSTRUMENT MAINTENANCE TRAINING

1. INSTRUMENT MAINTENANCE I

   A. Mechanical Instruments and Mechanisms
   B. Measurement and Pneumatic Instruments
   C. Final Control Elements and Introduction to Pneumatic Controllers
   D. Pneumatic Controllers-Manual/Automatic Stations
   E. Control Loops
   F. Electrical Measuring Methods
   G. Electronic Sub-assemblies and Recorder Amplifiers

2. INSTRUMENT MAINTENANCE II

   A. Math Review
   B. Introduction to Electronics
   C. Analog Feedback Systems
   D. Introduction to Blueprint Reading
   E. Fundamental Process Control
   F. Pressure Theory
   G. Level Theory
   H. Recorder Operation
   I. Conductivity and Turbidity Theory
   J. Loop Integration

3. INSTRUMENT MAINTENANCE III-7300 A

   A. Introduction to 7300 Instrumentation
   B. Operational Amplifier Review
   C. Individual 7300 Card Theory
   D. 7300 Nuclear Cabinet Power Supply and Pneumatics
   E. Simulator Checkout

4. INSTRUMENT MAINTENANCE IV - 7300 B

   A. Cabinet Configuration 7300-B
   B. Westinghouse Symbols Explanation
   C. Delta T/Tave Loop
   D. Pressurizer Loop
   E. Steam Generator Control
   F. 7300 Card Block Diagrams

8

DEPARTMENTAL TRAINING

BYRON STATION INDOCTRINATION

COMPANY ORIENTATION

IPPO-WESTINGHOUSE SIMULATOR

INTRODUCTION TO POWER PLANT OPERATIONS

BYRON SYSTEMS GENERAL TRAINING

N-GET NUCLEAR GENERAL EMPLOYEE TRAINING

CPR CARDIOPULMINARY RESUSCITATION

INTRODUCTION TO EDISON QUALITY ASSURANCE MANUAL

PROPER USE OF OUT-OF-SERVICE & PROTECTIVE CARDS

RADIATION/CHEMISTRY TECHNICIAN TRAINING

MANAGEMENT TRAINING

TECHNICAL STAFF TRAINING

CONTRACTOR TRAINING

INSTRUCTOR TRAINING

FIGURE 1

125 VDC ESF (DIV. II) AND NON ESF

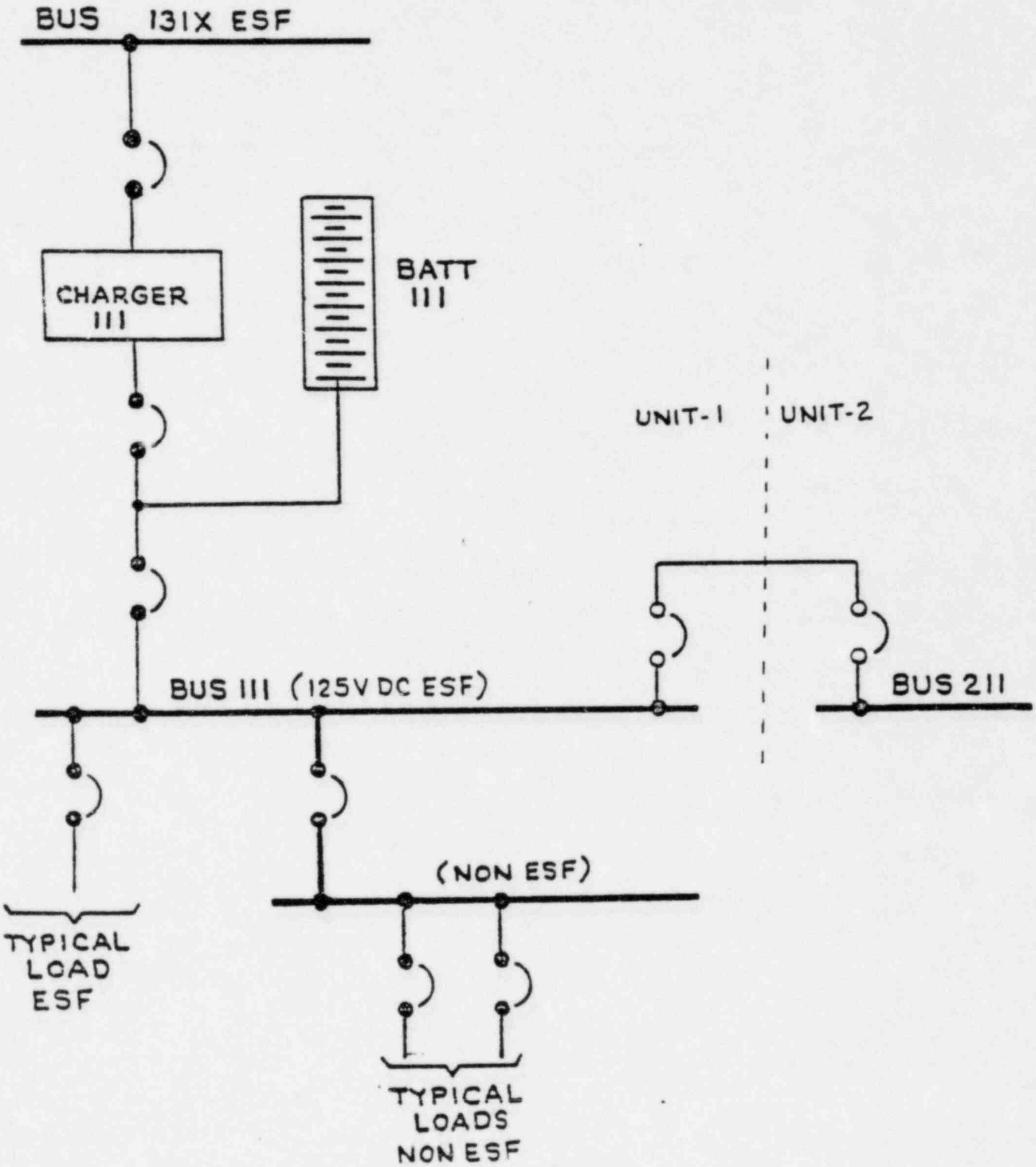

FIGURE 2

RECOVERY FROM LOSS OF ALL A.C.

- RESTORE THE ONSITE EMERGENCY DIESEL GENERATORS
- PROTECT THE REACTOR CORE BY MINIMIZING RCS INVENTORY LOSS
- PROTECT PLANT EQUIPMENT
- PREPARE PLANT FOR RECOVERY

DURING A FAILURE OF THE AC POWER SYSTEM THE ABOVE ACTIONS WOULD BE ACCOMPLISHED IN PARALLEL.

AGENDA ITEM III.4

"APPLICANT CONSIDERATION OF EXISTING PRA AND SYSTEMS INTERACTION STUDIES"

I CONSIDERATION OF EXISTING PRA STUDIES

II CONSIDERATION OF SYSTEMS INTERACTION STUDIES

SLIDE 2

SYSTEMS INTERACTION STUDIES

COMMON CAUSE EVENTS

1. FIRE
2. FLOOD
3. IDENTIFIED CONTROL SYSTEM INTERACTIONS
4. ENVIRONMENTAL
5. SEISMIC
6. HEAVY LOADS
7. INTERNALLY AND EXTERNALLY GENERATED MISSILES

SYSTEMS INTERACTION

SEISMIC

PURPOSE

TO ENSURE THAT SYSTEMS, STRUCTURES AND COMPONENTS REQUIRED FOR ACCIDENT MITIGATION OR HOT SHUTDOWN WILL NOT BE PREVENTED FROM PERFORMING THEIR INTENDED SAFETY FUNCTION AS A RESULT OF PHYSICAL INTERACTIONS WITH NON-SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

SLIDE 4

III.4

BYRON

DESIGN CHANGES

A. SEISMIC

B. FIRE

C. FLOOD

D. ENVIRONMENTAL

E. MISSILES

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SLIDE 1

III.5

BYRON AFWS RELIABILITY

	UNAVAILABILITY ON DEMAND
CASE 1: LOSS OF MAIN FEEDWATER	6.9×10^{-5} (HIGH)
CASE 2: LOSS OF MAIN FEEDWATER WITH LOSS OF OFFSITE POWER	9.2×10^{-5} (HIGH)
CASE 3: LOSS OF MAIN FEEDWATER WITH LOSS OF ALL AC	1.7×10^{-2} (*)

* APPLICANT RATES THIS AS HIGH ON THE QUALITATIVE GUIDELINES PRESENTED IN NUREG-0611.

1336b*

BYRON AFW SYSTEM

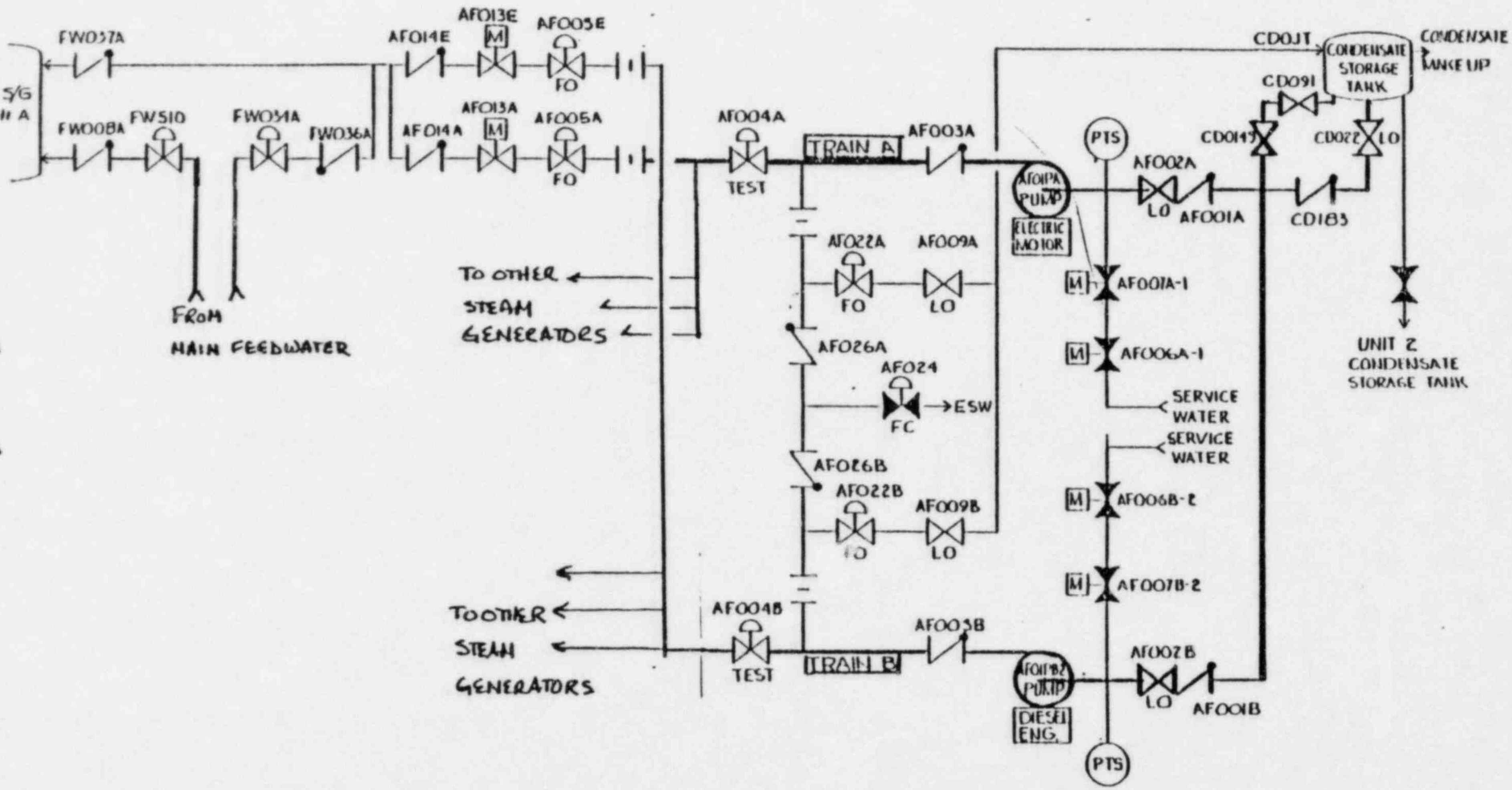
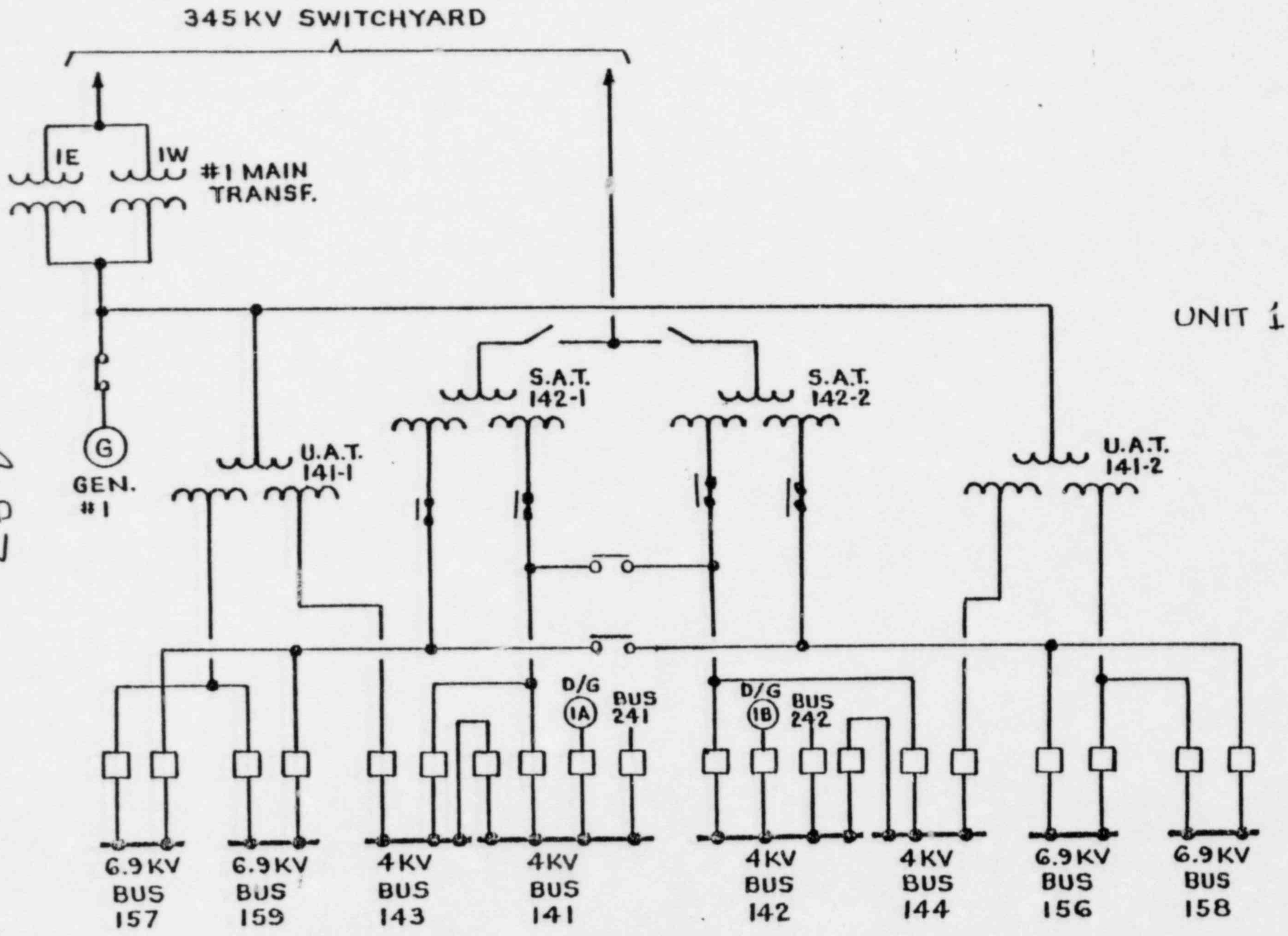


FIGURE 6



345KV SWITCHYARD
IE
IW
#1 MAIN TRANSF.
G
GEN. #1
U.A.T. 141-1
S.A.T. 142-1
S.A.T. 142-2
U.A.T. 141-2
UNIT 1
D/G
1A
BUS 241
D/G
1B
BUS 242
6.9KV BUS 157
6.9KV BUS 159
4KV BUS 143
4KV BUS 141
4KV BUS 142
4KV BUS 144
6.9KV BUS 156
6.9KV BUS 158

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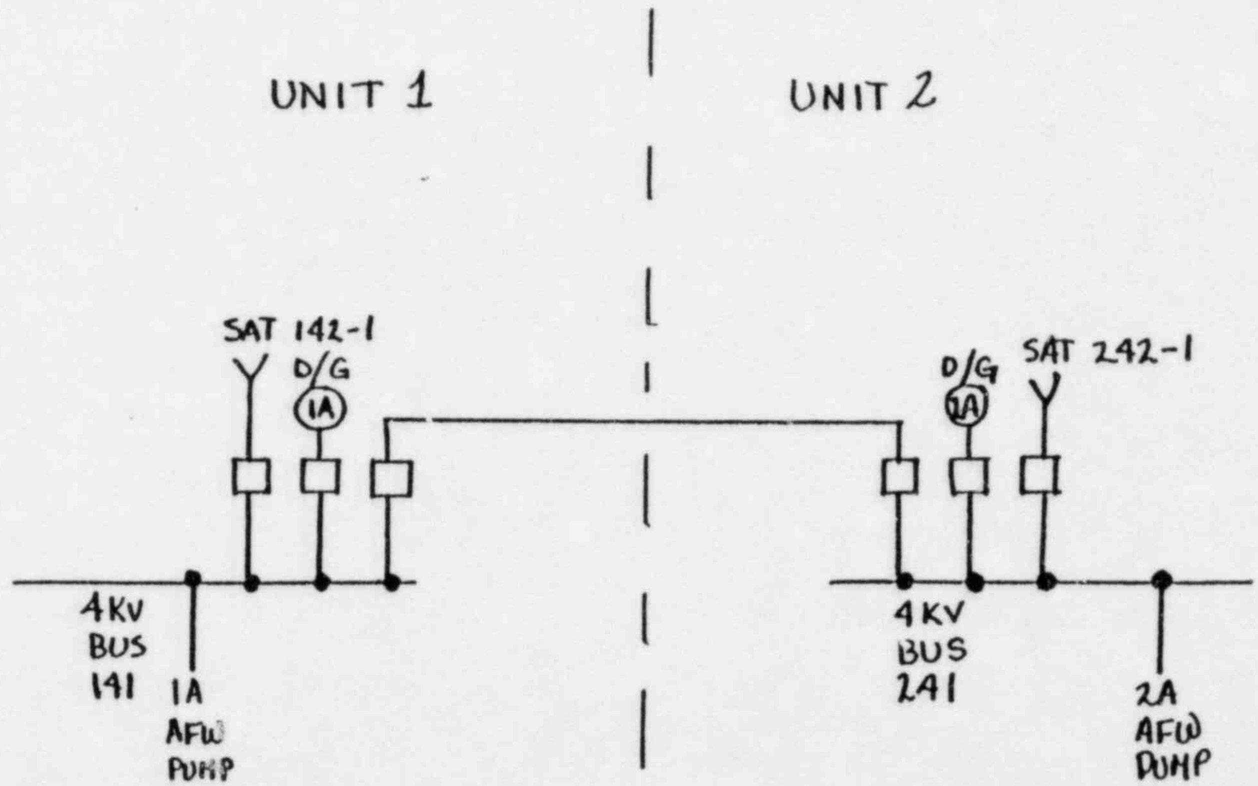

UNIT 1
UNIT 2
SAT 142-1
D/G
1A
4KV
BUS
141
1A
AFW
PUMP
D/G
2A
SAT 242-1
4KV
BUS
241
2A
AFW
PUMP

#1

COMMONWEALTH EDISON'S

Generating
Stations
Emergency
Plan

A-89

FIGURE TWO

- CONTROL ROOM
  - INITIAL CENTER OF EMERGENCY CONTROL
- TECHNICAL SUPPORT CENTER (TSC)
  - SUPPORTS CONTROL ROOM
  - ASSESSES PLANT STATUS
  - COORDINATE EMERGENCY MEASURES
- OPERATIONAL SUPPORT CENTER (OSC)
  - ASSEMBLY AREA FOR EMERGENCY PERSONNEL TO BE DISPATCHED
- CORPORATE COMMAND CENTER (CCC)
  - CENTER LOCATED IN CHICAGO
- EMERGENCY OPERATIONS FACILITY (EOF)

FIGURE THREE

EMERGENCY OPERATIONS FACILITY

- THREE PRIMARY FUNCTIONS
  - COORDINATION OF RECOVERY OPERATIONS
  - COORDINATION OF EVALUATION OF OFFSITE RELEASES
  - DISSEMINATION OF PUBLIC INFORMATION
- LOCATED IN DIXON
- DESIGNED TO REQUIREMENTS OF NUREG 0696

FIGURE FOUR

GSEP ORGANIZATION

- ONSITE GROUP

  PERFORMS FOLLOWING FUNCTIONS UNDER STATION DIRECTOR
  - REQUIRED SYSTEM OPERATIONS
  - SURVEYS AND PERSONNEL MONITORING
  - FIREFIGHTING
  - RESCUE
  - FIRST AID
  - DECONTAMINATION
  - SECURITY OF PLANT
  - MAINTENANCE
  - PERSONNEL ACCOUNTABILITY
  - RECORDKEEPING
  - COMMUNICATIONS

- OFFSITE GROUP
  - CORPORATE COMMAND CENTER
  - RECOVERY GROUP OF EOF

FIGURE FIVE

EMERGENCY MEASURES

- CLASSIFY INCIDENT
  - TRANSPORTATION ACCIDENT
  - UNUSUAL EVENT
  - ALERT
  - SITE EMERGENCY
  - GENERAL EMERGENCY
- ASSESSMENT ACTIONS
  - EVALUATE PLANT STATUS
  - RADIOLOGICAL MEASUREMENTS
  - DOSE PROJECTIONS

STATE OF ILLINOIS

I llinois
P lan for
R adiological
A ccidents

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#6

HIGH STRENGTH BOLT MATERIAL

CHARPY IMPACT TEST REQUIREMENTS

<u>NSSS</u>

1. 5 INCH DIAMETER REACTOR COOLANT PUMP COLUMN HOLD DOWN BOLTS -

   12 BOLTS PER UNIT - - MINIMUM ONE TEST

2. 5 1/2 INCH DIAMETER REACTOR COOLANT PUMP AND STEAM GENERATOR COLUMNS VERTICAL ROCKER SUPPORT BOLTS -

   56 BOLTS/UNIT MINIMUM ONE TEST

3. 2 INCH DIAMETER REACTOR COOLANT PUMP AND STEAM GENERATOR COLUMN BOLTS -

   224 BOLT PER UNIT - MINIMUM ONE TEST

4. 1 1/2 INCH DIAMETER STEAM GENERATOR COLUMN BOLTS -

   192 BOLTS PER UNIT - MINIMUM ONE TEST

<u>PIPE WHIP RESTRAINTS</u>

TESTED IN ACCORDANCE WITH ASME NF-2345 -

APPROXIMATELY 120 PIPE WHIP RESTRAINTS PER UNIT

CHARPY IMPACT TEST REQUIREMENTS

NF-2345 BOLTING MATERIAL

ONE TEST SHALL BE MADE FOR EACH LOT OF MATERIAL WHERE A LOT IS DEFINED AS ONE HEAT OF MATERIAL HEAT TREATED IN ONE CHARGE OR AS ONE CONTINUOUS OPERATION, NOT TO EXCEED IN WEIGHT THE FOLLOWING:

1 3/4 INCH DIAMETER AND LESS	1500 LB
OVER 1 3/4 INCH TO 2 1/2 INCH DIAMETER	3000 LB
OVER 2 1/2 INCH TO 5 INCH DIAMETER	6000 LB
OVER 5 INCH DIAMETER	10000 LB

SECONDARY CHEMISTRY CONTROL

FIGURE 1

COMMONWEALTH EDISON'S INDUSTRY INVOLVEMENT

- MEMBER STEAM GENERATOR'S OWNERS' GROUP (SGOG)
- MEMBER TECHNICAL ADVISORY COMMITTEE - SGOG
- MEMBER CHEMICAL CLEANING SUBCOMMITTEE - SGOG
- MEMBER SECONDARY WATER CHEMISTRY GUIDELINES COMMITTEE - SGOG
- ZION OPERATING EXPERIENCE

SECONDARY CHEMISTRY CONTROL

FIGURE 2

BYRON STATION DESIGN CONSIDERATIONS

- STAINLESS STEEL CONDENSER, FEEDWATER AND MOISTURE SEPARATOR REHEATER TUBING: ELIMINATION OF COPPER
- SHUTDOWN CONDENSATE CLEANUP CAPABILITY: REMOVAL OF IMPURITIES FROM THE CONDENSATE AND FEEDWATER SYSTEMS PRIOR TO UNIT STARTUP
- SECONDARY CHEMISTRY SAMPLING AND MONITORING SYSTEM: CONTINUOUS MONITORING OF KEY SECONDARY PARAMETERS.
- CAPABILITY FOR CONDENSER REPAIR DURING OPERATION: ISOLATION OF HOTWELL QUADRANTS FOR LEAK REPAIR AT LOW POWER
- IMPROVED CONDENSER DESIGN: REDUCE THE POTENTIAL FOR LEAKAGE

SECONDARY CHEMISTRY CONTROL

FIGURE 3

BYRON STATION'S OPERATION POLICY EFFECT

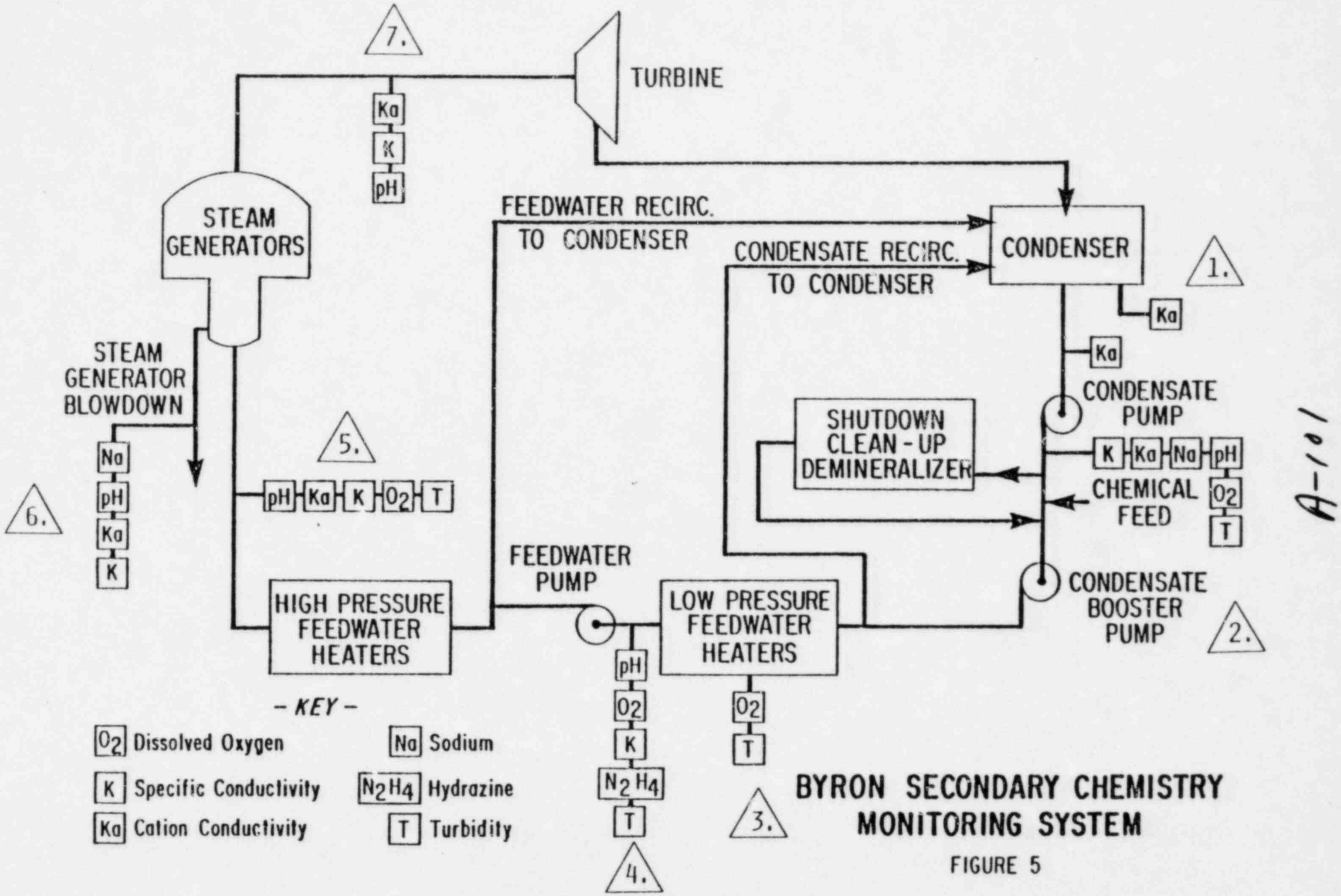
- REDUCED STEAM GENERATOR TUBE LEAKAGE
- REDUCED PERSONNEL RADIATION EXPOSURE

SECONDARY CHEMISTRY CONTROL

FIGURE 4

KEY CHEMISTRY CONTROL PARAMETERS

- OXYGEN: THE CONTROL OF OXYGEN IN THE SECONDARY SYSTEM MINIMIZES SLUDGE TRANSPORT TO THE STEAM GENERATOR.
- CATION CONDUCTIVITY: CATION CONDUCTIVITY IS THE PRIME INDICATOR OF INLEAKAGE OF WATER TO THE SECONDARY SYSTEM.
- SODIUM: SODIUM CONTROL MINIMIZES CAUSTIC INDUCED FAILURE OF THE STEAM GENERATORS.
- CHLORIDE: CHLORIDE CONTROL MINIMIZES THE POSSIBILITY OF STEAM GENERATOR TUBE FAILURE.
- PH: pH CONTROL MINIMIZES CORROSION OF THE CARBON STEEL COMPONENTS



BYRON SECONDARY CHEMISTRY MONITORING SYSTEM

FIGURE 5

SECONDARY CHEMISTRY CONTROL

FIGURE 6

BYRON STATION'S SECONDARY WATER CHEMISTRY PROGRAM ELEMENTS

- MONITORING SYSTEM SURVEILLANCE PROCEDURES
- LABORATORY ANALYSIS
- CORRECTIVE ACTION LEVELS
- ACTION LEVEL IMPLEMENTING PROCEDURES

FIGURE 7

ACTION LEVEL RESPONSES

ACTION LEVEL 1:	PROMPTLY IDENTIFY AND CORRECT CAUSE OF AN OUT-OF-NORMAL CONDITION WITHOUT POWER REDUCTION.
ACTION LEVEL 2:	REDUCE POWER WHILE IDENTIFYING AND CORRECTING OUT-OF-NORMAL CONDITION.
ACTION LEVEL 3:	UNIT SHUTDOWN WHILE IDENTIFYING AND CORRECTING OUT-OF-NORMAL CONDITION.

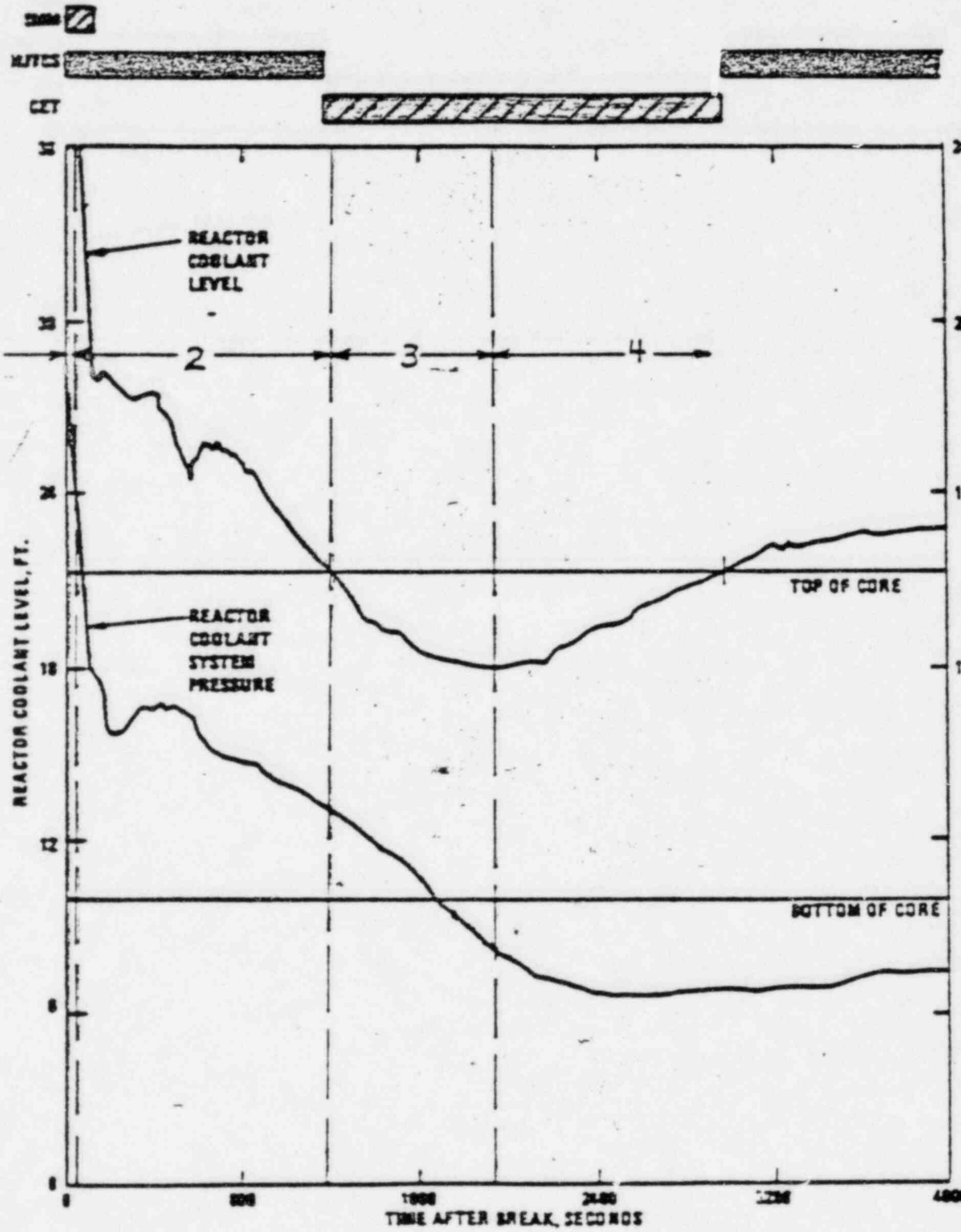
SECONDARY CHEMISTRY CONTROL

FIGURE 8

SUMMARY OF BYRON STATION'S SECONDARY CHEMISTRY PROGRAM

- RELIABILITY OF BYRON'S STEAM GENERATORS
  - CHEMISTRY
    - SURVEILLANCE PROGRAM
    - LABORATORY CAPABILITY
    - OPERATIONAL POLICY
  - MATERIALS
    - ELIMINATE COPPER
  - SECONDARY SYSTEM OPERATION
    - SAMPLING AND MONITORING SYSTEM
    - SHUTDOWN CLEANUP DEMINERALIZERS
    - CONTINUOUS STEAM GENERATOR BLOWDOWN
    - ACTION LEVELS
  - FIELD EXPERIENCE
    - ZION STATION
    - WESTINGHOUSE
    - STEAM GENERATORS OWNERS GROUP

USEFUL OPERATING RANGE OF C-E ICC MONITORING INSTRUMENTS



* SMM - SUBCOOLED MARGIN MONITOR
HJTCS - HEATED JUNCTION THERMOCOUPLE SYSTEM
CET - CORE EXIT THERMOCOUPLES

POWER SYSTEMS COMBUSTION ENGINEERING, INC.	REACTOR COOLANT PARAMETERS FOLLOWING SMALL BREAK LOCA	Figure 5

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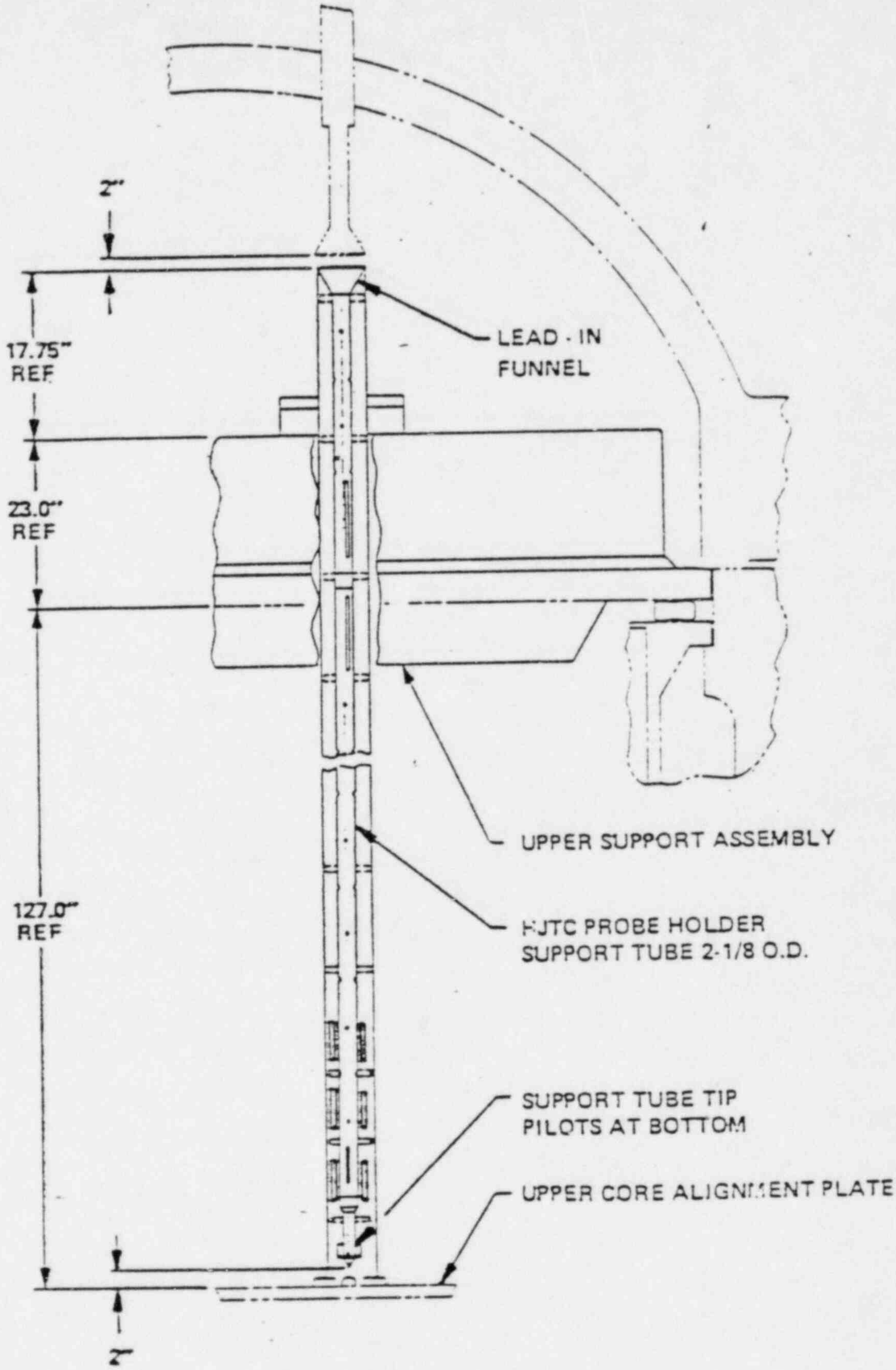
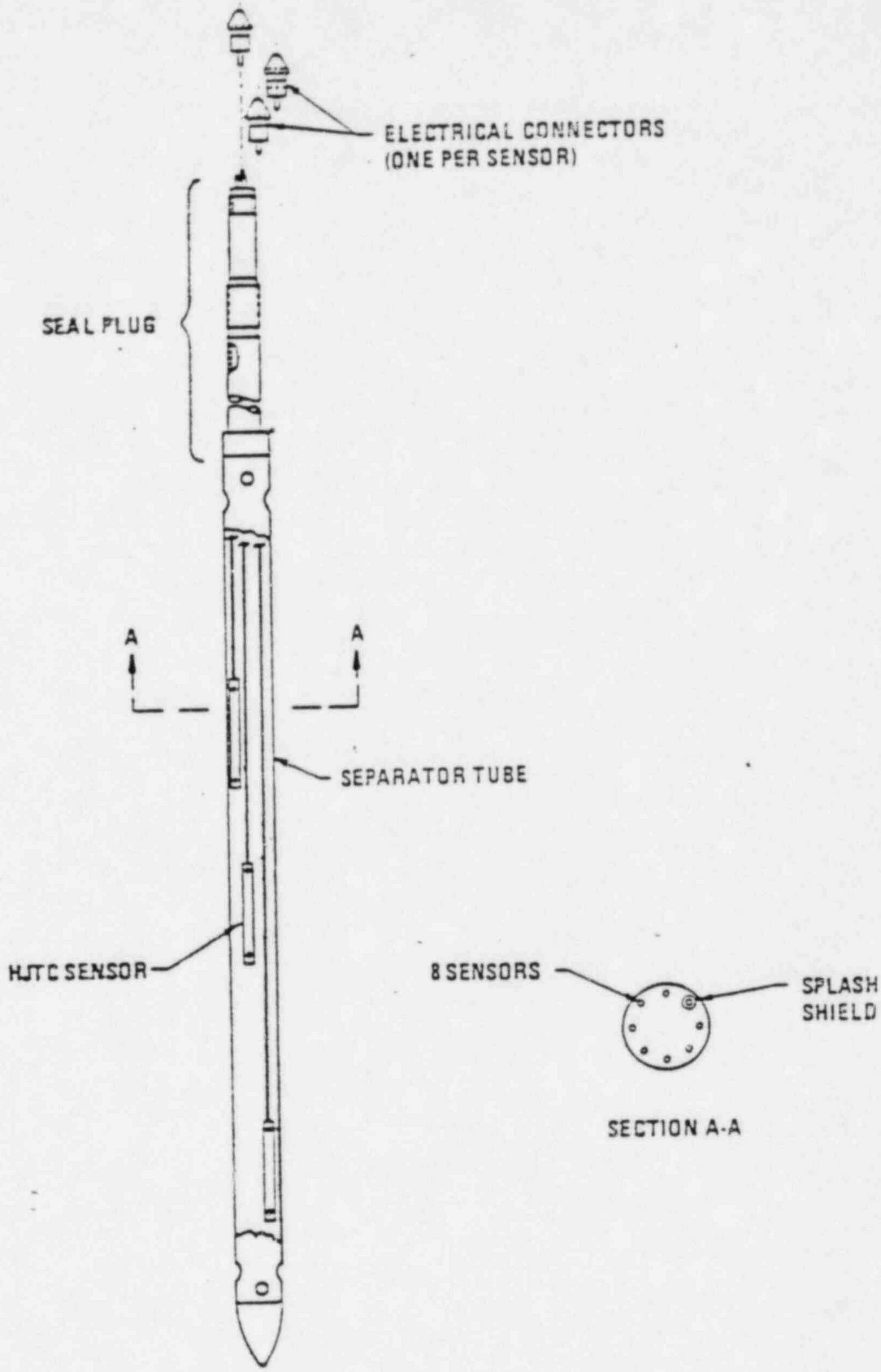


POWER SYSTEMS COMBUSTION ENGINEERING INC	HJTC PROBE HOLDER SUPPORT TUBE INSTALLATION	Figure



POWER SYSTEMS COMBUSTION ENGINEERING INC	HEATED JUNCTION THERMOCOUPLE PROBE ASSEMBLY	Figure [illegible]

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SUBCOOLED MARGIN INDICATION

INPUTS

- REACTOR COOLANT LOOP WIDE RANGE PRESSURE
- CORE EXIT THERMOCOUPLES
- WIDE RANGE REACTOR COOLANT LOOP RTD'S
- UNHEATED THERMOCOUPLES OF REACTOR VESSEL LEVEL INSTRUMENTS

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BYRON/BRAIDWOOD INADEQUATE CORE COOLING INSTRUMENTATION SYSTEM

- SUBCOOLED MARGIN INDICATOR
- REACTOR VESSEL LEVEL INDICATOR
- CORE EXIT THERMOCOUPLES

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SECONDARY CHEMISTRY CONTROL

FIGURE 1

COMMONWEALTH EDISON'S INDUSTRY INVOLVEMENT

- MEMBER STEAM GENERATOR'S OWNERS' GROUP (SGOG)
- MEMBER TECHNICAL ADVISORY COMMITTEE - SGOG
- MEMBER CHEMICAL CLEANING SUBCOMMITTEE - SGOG
- MEMBER SECONDARY WATER CHEMISTRY GUIDELINES COMMITTEE - SGOG
- ZION OPERATING EXPERIENCE

STATUS & SCHEDULES

- o 95% CONSTRUCTION COMPLETE
- o PROJECTED FUEL LOAD DATE 01/83
- o PROJECTED OL ISSUANCE 01/83
- o SAFETY REVIEW SCHEDULE
  - SER ISSUED 7/9/81
  - ACRS LETTER (LICENSEE QUALIFICATIONS ISSUE)
  - SSER #1 ISSUED 10/13/81
  - SSER #2 ISSUED 01/13/82
  - SSER #3 SCHEDULED FOR 3/82

STAFF PRESENTATION

- o LICENSEE QUALIFICATIONS

AGENDA FOR MARCH 4, 1982 ACRS

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• OVERVIEW OF NUCLEAR OPERATIONS DEPARTMENT ORGANIZATION CHANGES/ STAFFING AND RECRUITING (15 MIN.)	L.V. MAURIN - VICE PRES. NUCLEAR OPERATIONS
• PLANT STAFF ORGANIZATION CHANGES/INTEGRATION OF CONTRACTORS IN ORGANIZATION (15 MIN.)	D.B. LESTER - PLANT MGR.
• PROJECT SUPPORT ORGANIZATION CHANGES/SAFETY REVIEW PROGRAM (10 MIN.)	F.J. DRUMMOND - PROJECT SUPPORT MANAGER
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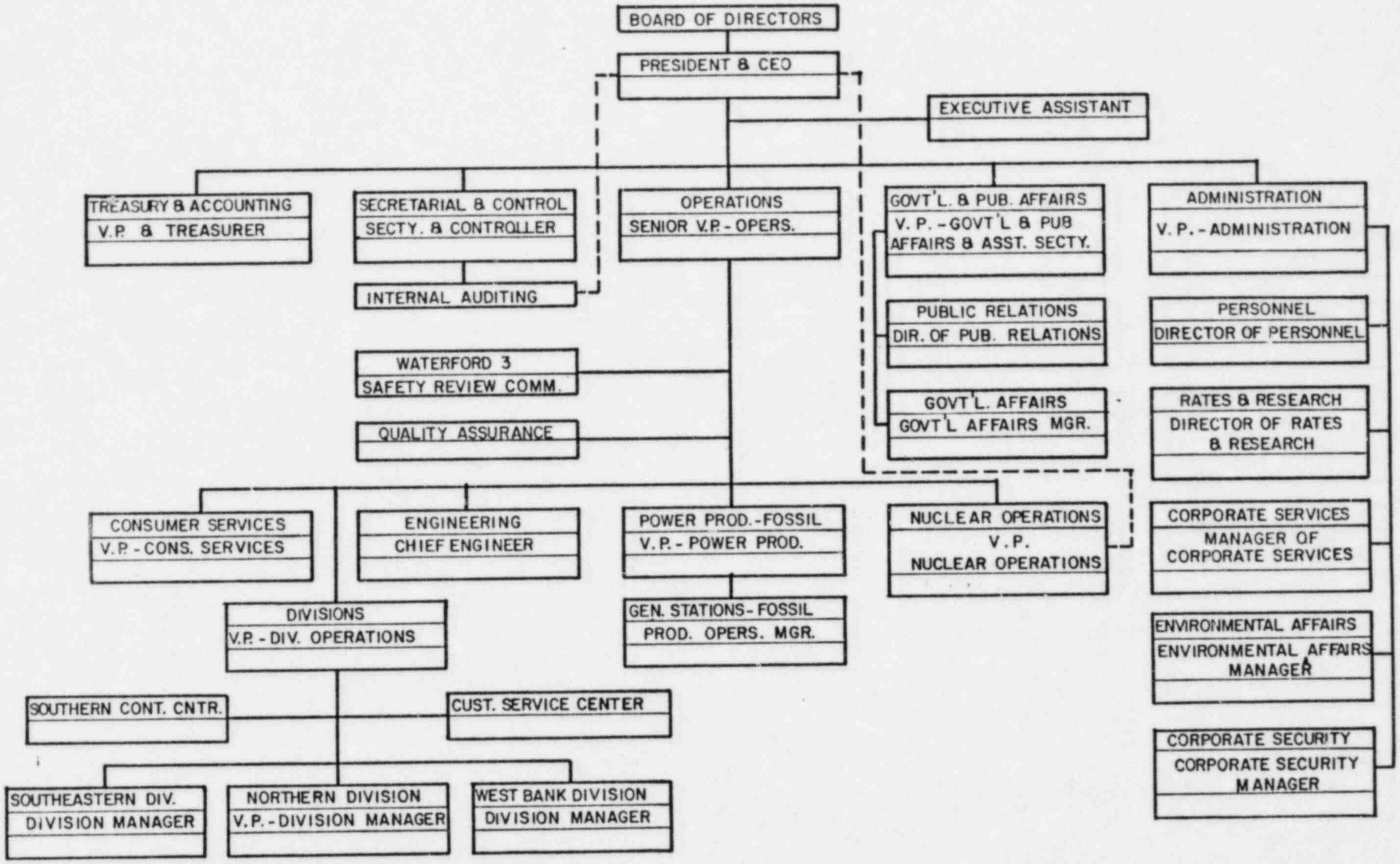
• CLOSING REMARKS (3 MIN.)	G.D. McLENDON - SR. VICE PRESIDENT - OPERATIONS

LOUISIANA POWER & LIGHT COMPANY



---- DIRECT COMMUNICATIONS DURING EMERGENCIES

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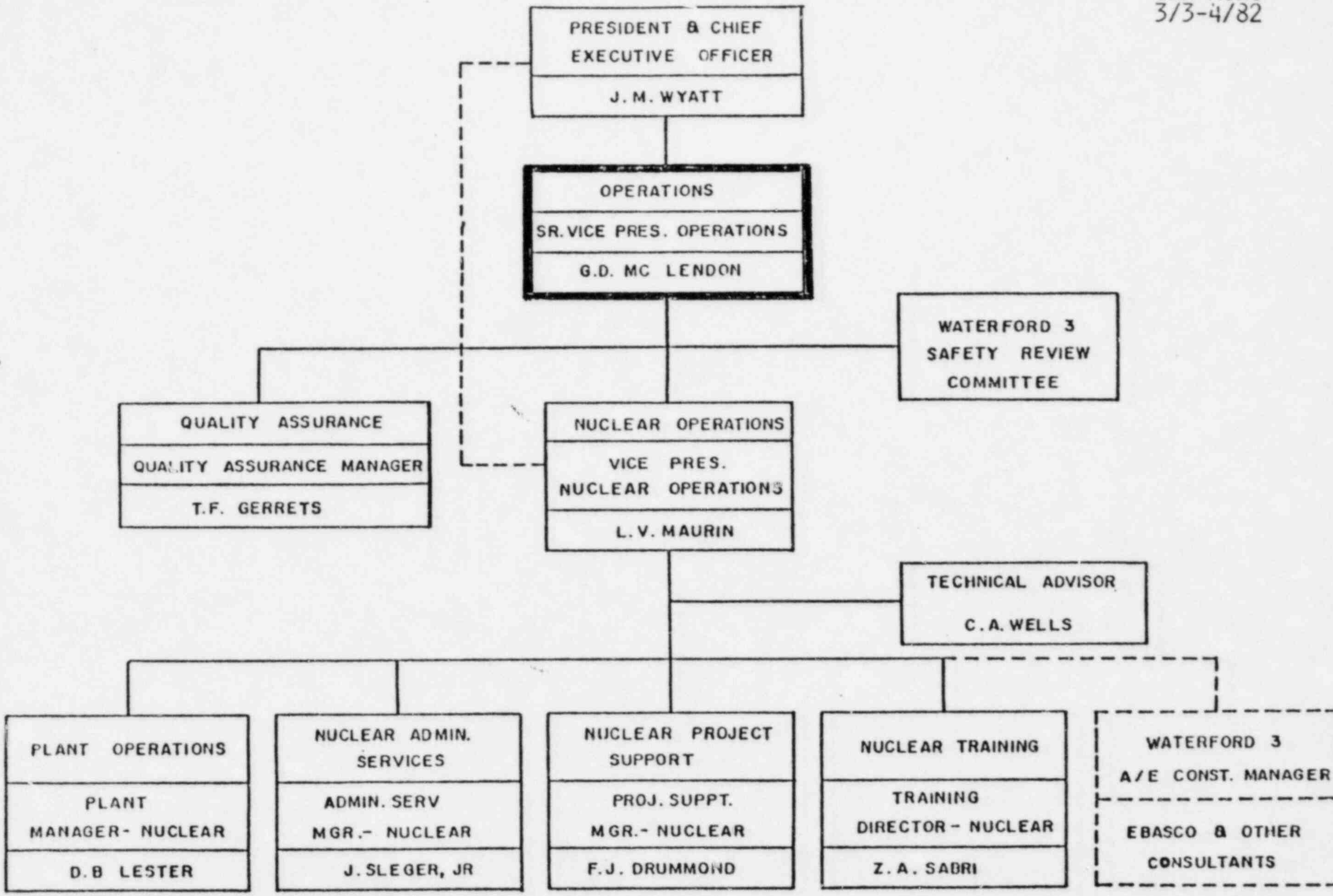
GDM-ACRS
3/3-4/82



----DIRECT COMMUNICATIONS DURING EMERGENCIES

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LOUISIANA POWER & LIGHT COMPANY
QUALITY ASSURANCE ORGANIZATION

SUMMARY

- Functional Organizational Structure
  - Independence
  - Authority

- Excellent Working Relationship
  - Nuclear Operations
  - Other LP&L Departments

- Experience
  - 20 Yrs/Person Overall
  - 8 Yrs/Person QA

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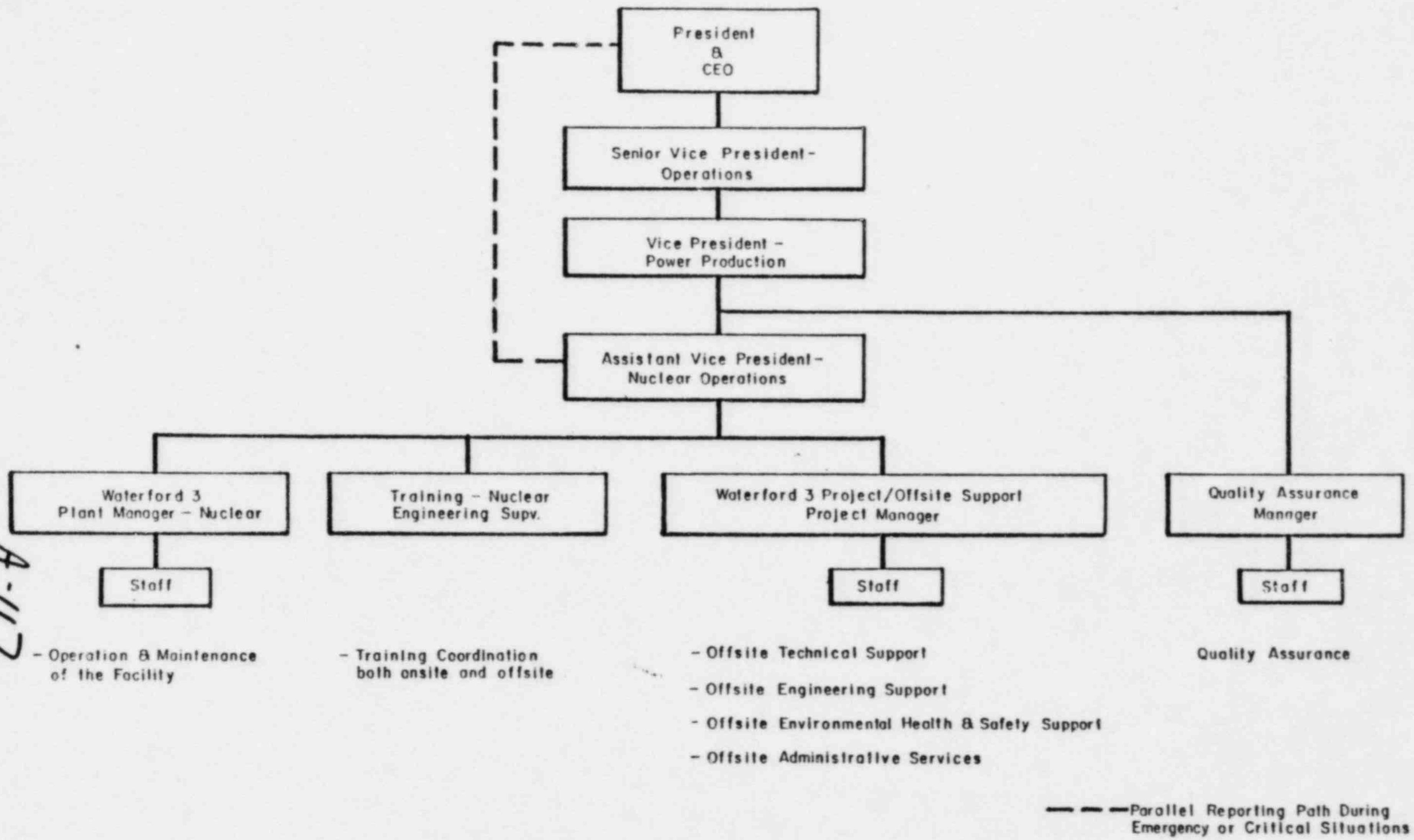
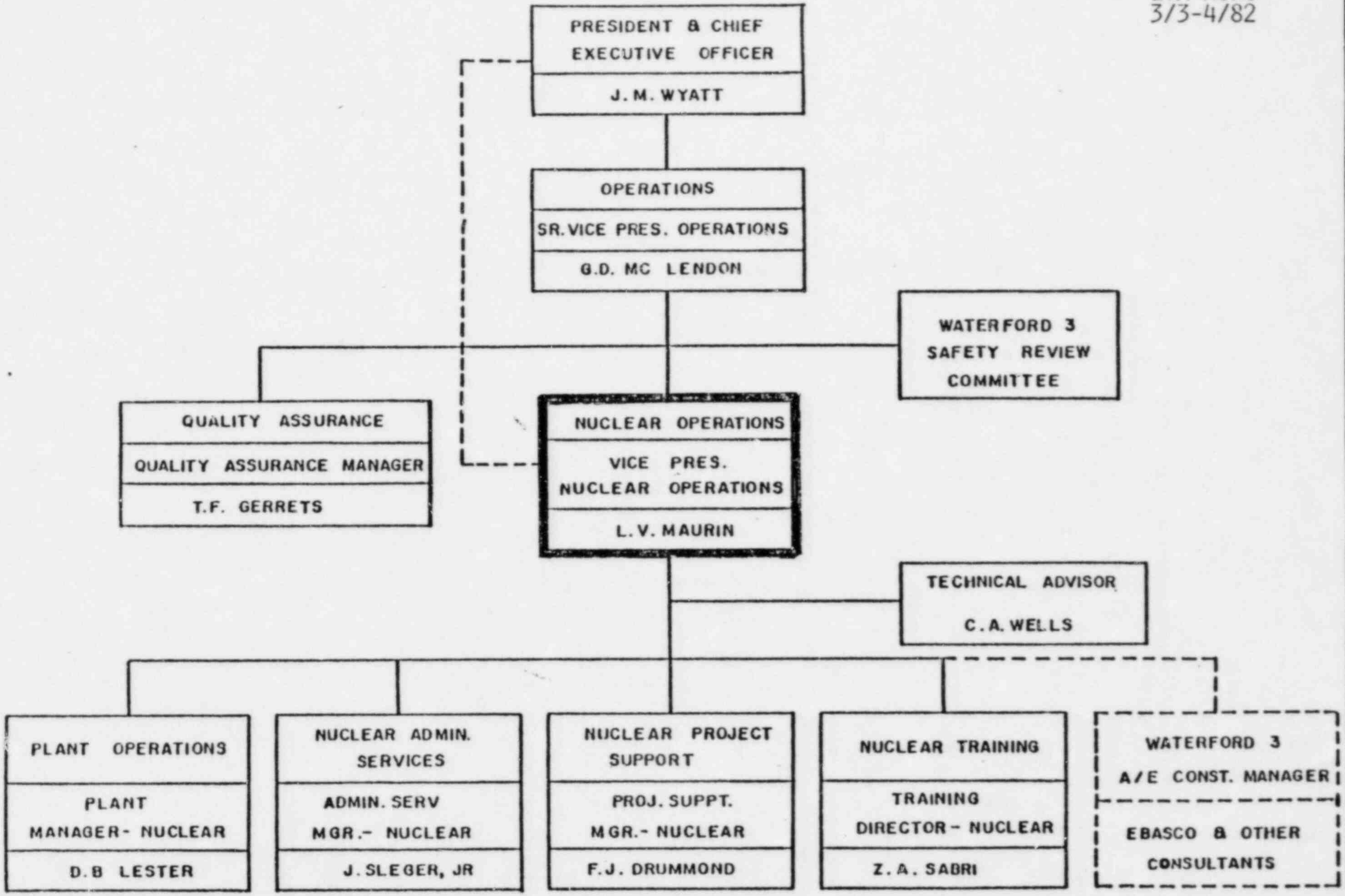
LVM
8/6/81

LP&L WATERFORD 3 CORPORATE ORGANIZATION



LVM-ACRS
3/3-4/82



----DIRECT COMMUNICATIONS DURING EMERGENCIES

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JS-ACRS
3/3-4/82

NUCLEAR ADMINISTRATIVE SERVICES

- ADMINISTRATIVE AND SUPPORT ACTIVITIES ARE ASSIGNED TO A SINGLE MANAGER
- PERMITS A MORE DIRECT LINE FOR ACCOMPLISHMENT OF ADMINISTRATIVE REQUIREMENTS AND ACTIONS
- RELIEVE TECHNICAL STAFFS OF CERTAIN ADMINISTRATIVE BURDENS
- ENHANCES THE COORDINATION WITH, AND ASSISTANCE TO BE GIVEN TO, OTHER LP&L DEPARTMENTS IN SUPPORTING WATERFORD 3

LVM-ACRS
3/3-4/82

RESUME

JOE SLEGER, JR.

- RETIRED COLONEL U.S. MARINE CORPS
- QC INSPECTOR SLINE PAINTING W-3
- QC SUPERVISOR SLINE PAINTING W-3
- QC MANAGER SLINE PAINTING W-3
- LP&L QA ASSOCIATE ENGINEER
- GENERAL SUPPORT SUPERINTENDENT W-3
- ADMINISTRATIVE SERVICES MANAGER - NUCLEAR W-3

RESUME

LVM-ACRS
3/3-4/82

DR. Z.A. SABRI

- PHD (1972) NUCLEAR ENGINEERING, MINOR IN CHEMICAL ENGINEERING, UNIVERSITY WISCONSIN
- BSC (1966) ELECTRICAL ENGINEERING, UNIVERSITY OF ALEXANDRIA
- MSC (1969) NUCLEAR ENGINEERING, UNIVERSITY OF WISCONSIN
- PROF. OF NUCLEAR ENGINEERING & DIRECTOR OF NUCLEAR SAFETY RESEARCH GROUP, ISU
- MEMBER OF IEEE STANDARDS 5.5 WORKING GROUP DEVELOPING STANDARDS ON HUMAN PERFORMANCE IN LWR
- ADVISOR TO NRC-NRR DIVISIONS OF HUMAN FACTORS SAFETY AND OPERATIONAL DATA EVALUATION
- MEMBER OF THE EXECUTIVE COMMITTEE, TECHNICAL GROUP ON HUMAN FACTORS SAFETY, ANS
- DIRECTOR OF THE HUMAN FACTORS AND NUCLEAR SAFETY ANALYSIS DIVISION, TECHNOLOGY INTERNATIONAL INCORPORATED
- ADVISOR TO INPO, CRITERIA & ANALYSIS DIVISION

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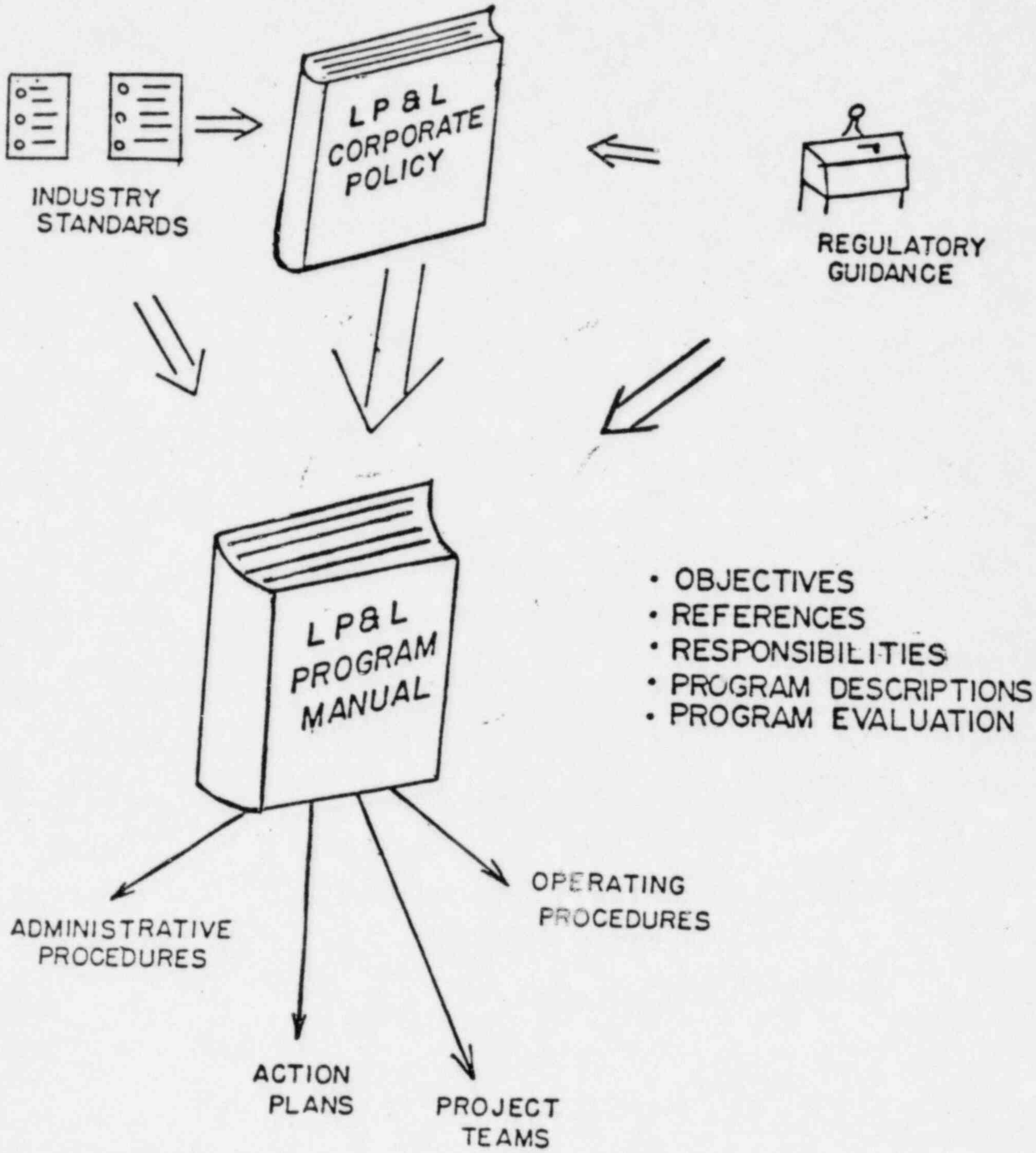
LVM-ACRS
3/3-4/82

RESUME
DR. Z.A. SABRI
(CONTINUED)

- PRINCIPLE INVESTIGATOR ON SEVERAL PROJECTS INCLUDING:
  - NSIC-ORNL EVALUATION OF HUMAN RELATED LER'S FOR LWR - SEE-IN SCREENING (NSAC)
  - NRC OPERATOR ERROR RATE EVALUATION PROJECT (NRC)
  - NRC HUMAN FACTORS IN ACCIDENT INITIATION AND MITIGATION (NRC)
  - SANDIA/DOE IMPACTS OF M&T ON LWR SAFETY
  - EVALUATION OF CONTROL ROOM DESIGN (GEORGIA POWER)

MANAGEMENT CONTROL SYSTEM PROCESS

LVM-ACRS
3/3-4/82



MANAGEMENT CONTROL PROGRAMS

TOTAL NUMBER	129
APPROVED	22

- TYPICAL PROGRAMS
  - NUCLEAR OPERATIONS DEPARTMENT POLICY AND ORGANIZATION
  - SAFETY REVIEW COMMITTEES
  - INDEPENDENT SAFETY ENGINEERING
  - STATION MODIFICATION
  - OPERATION ASSESSMENT
  - OPERATION ORGANIZATION AND ADMINISTRATION
  - TRAINING GROUP ORGANIZATION
  - ENGINEERING TRAINING
  - SECONDARY CHEMISTRY
  - MAINTENANCE ORGANIZATION AND ADMINISTRATION
  - NUCLEAR ENGINEERING SUPPORT TO OPERATIONS
  - DANGER AND CAUTION TAG CONTROL

LVM-ACRS
3/3-4/82

WATERFORD 3 NUCLEAR PROJECT

RECRUITING STATISTICS

• NUMBER OF CANDIDATES INTERVIEWED	869
• NUMBER OF JOB OFFERS MADE	304
• NUMBER OF JOB OFFERS ACCEPTED	162

ABOVE FIGURES REFLECT ACTIVITIES FROM NOVEMBER 1, 1980 THROUGH FEBRUARY 28, 1982.

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LVM-ACRS
3/3-4/82

NUCLEAR OPERATIONS STAFFING

	APPROVED STAFF	PERSONNEL HIRED AS OF 8/6/81
NUCLEAR OPERATIONS STAFF	2	2
PLANT OPERATIONS *	267	174
PROJECT SUPPORT	33	14
QUALITY ASSURANCE	10	8
TOTALS	312	198

* AS OF 8/6/81, TRAINING AND ADMINISTRATIVE SERVICES WERE INCLUDED IN PLANT OPERATIONS.

NUCLEAR OPERATIONS STAFFING

	APPROVED STAFF	PERSONNEL HIRED AS OF 2/28/82	PERCENT STAFFED
NUCLEAR OPERATIONS			
STAFF	2	2	
PLANT OPERATIONS			
STAFF	10	10	
STARTUP	23	12	
MAINTENANCE	76	63	
OPERATIONS	67	65	
TECHNICAL SUPPORT	47	32	
HEALTH PHYSICS	27	14	
QUALITY CONTROL	6	6	
TOTAL	256	204	
PROJECT SUPPORT			
STAFF	2	2	
CONSTRUCTION ENGRG.	12	7	
OPERATIONAL ENGRG.	26	12	
TECHNICAL SERVICES	15	9	
ONSITE SAFETY REVIEW	9	4	
LICENSING	13	6	
TOTAL	77	40	
TRAINING			
STAFF	11	6	
QUALITY ASSURANCE			
STAFF	21	17	
ADMINISTRATIVE SERVICES			
STAFF	59	49	
TOTALS	426	316	

ACQUISITION OF VITAL PERSONNEL

LVM-ACRS
3/3-4/82

PLANT STAFF	COMMITTED 4/20/81	HIRED AS OF 2/28/82	PERCENT STAFFED
OPERATIONS SUPT.	1	1	
ASST. PLANT MGR., O & M	1	1	
PLANT ENG. DEPT. SUPV.	1	-	
GENERAL SUPPORT SUPT.	1	-	
NUCLEAR OPERATIONS SUPV.	6	6	
NUCLEAR AUXILIARY OPERATOR (COLD LICENSE)	10	10	
NUCLEAR AUXILIARY OPERATOR (HOT LICENSE)	10	11	
PLANT UTILITY ENG.	3	3	
STA ENGINEERING SUPV.	1	-	
PLANT ASSOC. II/I ENG.	5	5	
OFFSITE SUPPORT			
ONSITE SAFETY REVIEW ENG. SUPV.	1	1	
ONSITE SAFETY REVIEW ENG.	1	-	
OFFSITE TRAINING SUPV.	1	-	
NUCLEAR TRAINING DIR.	1	1	
CONSULTANTS			
TECHNICAL ADVISOR TO VICE PRES. NUCLEAR OPERATIONS	1	1	
TECHNICAL ADVISOR TO PLANT MANAGER	1	1	
TOTALS	45	41	

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LP&L TOTAL EXPERIENCE

	TOTAL NUCLEAR	COMMERCIAL NUCLEAR	COMMERCIAL OPERATIONS
PROJECT SUPPORT	199	128	15
NUCLEAR OPERATIONS	1141	632	190
QUALITY ASSURANCE	150	81	4
TOTAL	1490	841	209

NOTE: ALL THE ABOVE NUMBERS IN MANYEARS

CONCLUSION

LP&L HAS STRUCTURED AN ORGANIZATION WHICH:

- REALIZES THE MAGNITUDE OF THE PROJECT
- CAPABLE OF MANAGING AND CONTROLLING ALL ASPECTS OF WATERFORD 3 IN A SAFE AND EFFICIENT MANNER INCLUDING:
  - CONSTRUCTION COMPLETION
  - PRE-OPERATIONAL TESTING
  - PLANT START-UP
  - PLANT OPERATION
- EMPHASIZES IMPORTANCE OF TRAINING
- HAS THE REQUISITE EXPERIENCE TO OPERATE WATERFORD 3 SAFELY.
- DEMONSTRATED DURING DEC. 1981 AUDIT "THAT LP&L'S MANAGEMENT CAPABILITIES ARE ADEQUATE TO DIRECT AND SUPPORT SAFE OPERATION OF WATERFORD 3 AND THAT MANAGEMENT IS COMMITTED TO ASSURING SAFE OPERATION OF WATERFORD 3."

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. CLOSING REMARKS (3 MIN.)	G.D. McLENDON - SR. VICE PRESIDENT - OPERATIONS

DBL-ACRS
3/4/82

PLANT OPERATIONS

- ORGANIZATION
- STARTUP/OPERATIONS CONTROL

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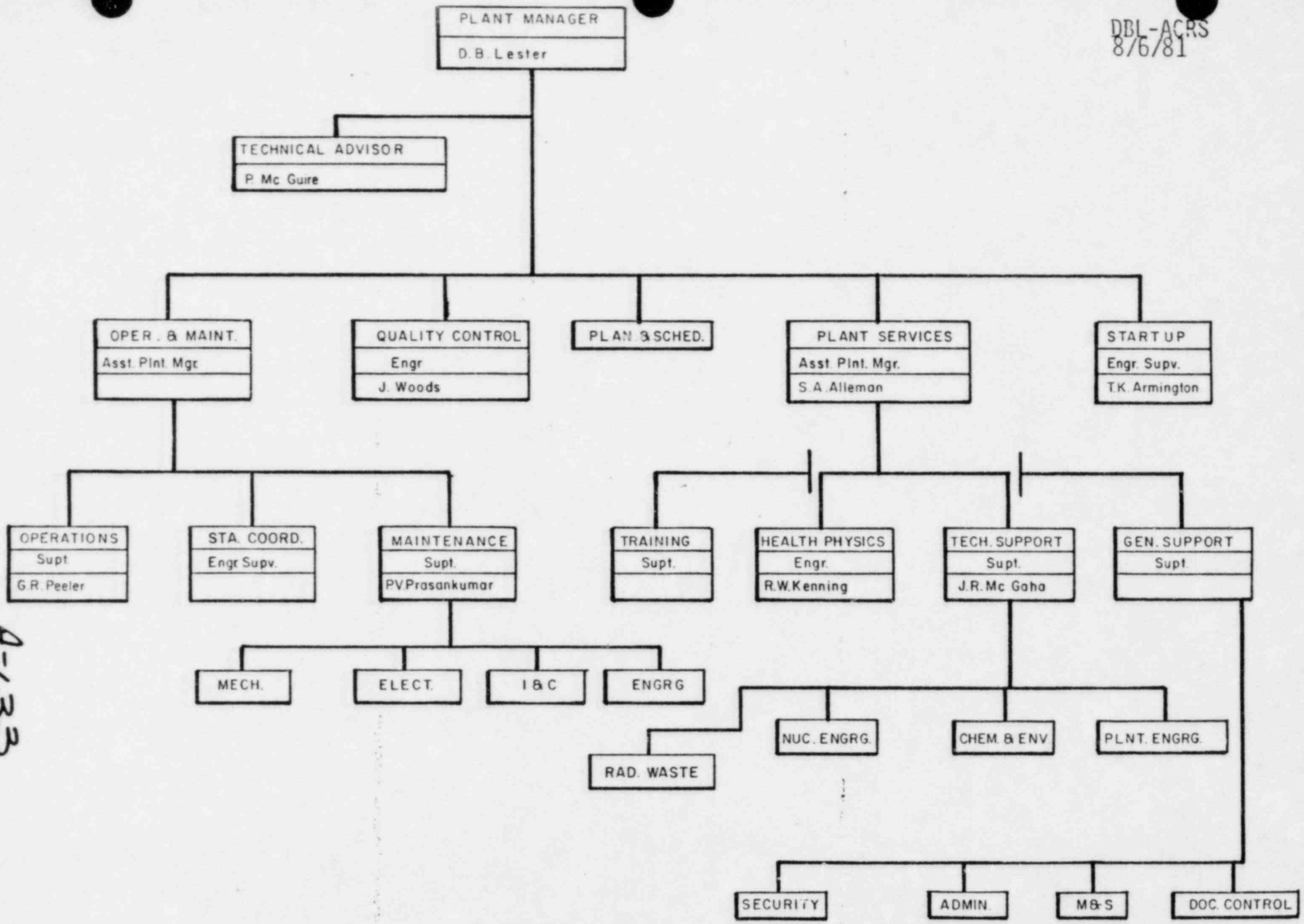
DBL-ACRS
8/6/81



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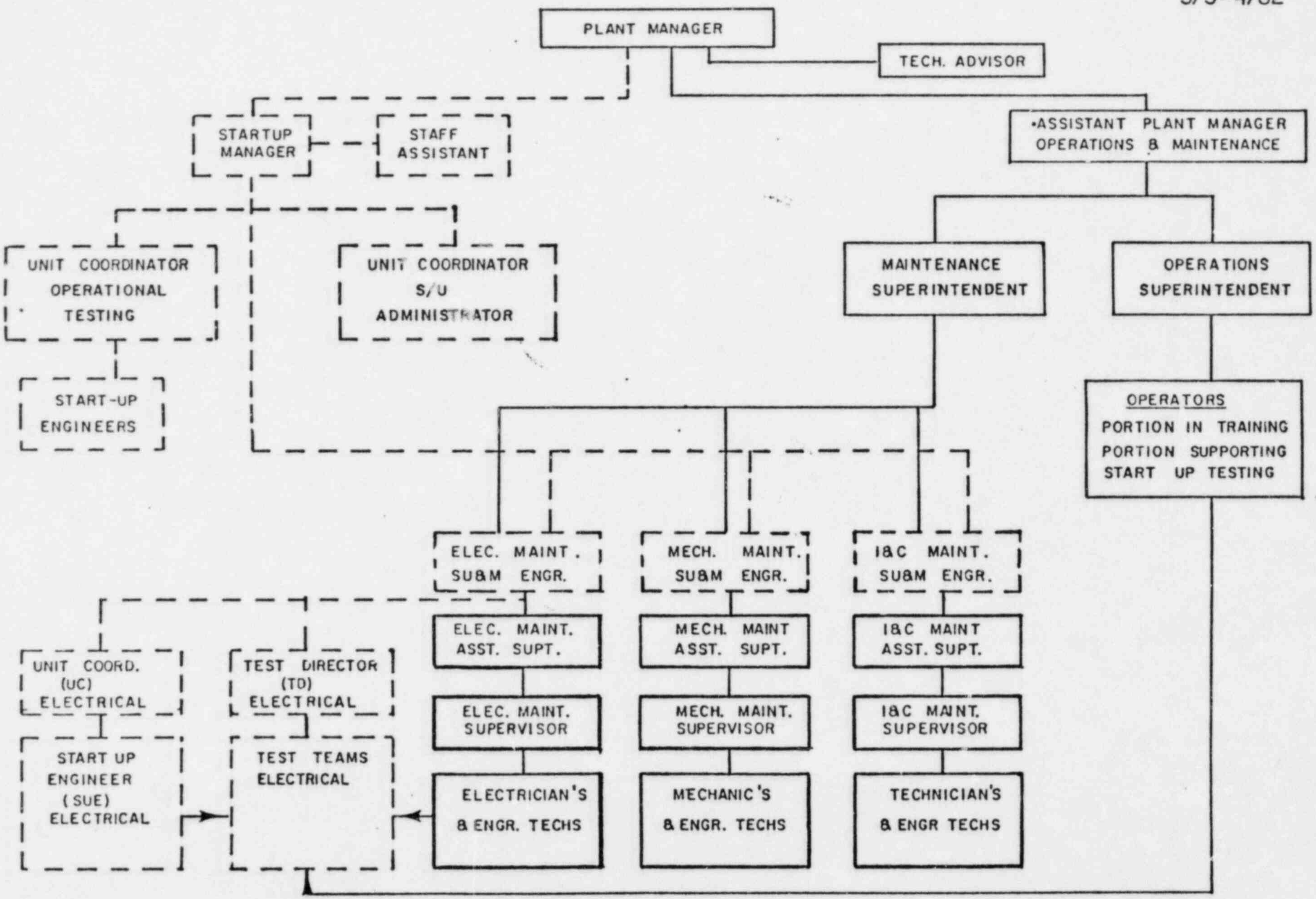
LP&L PREQUISITE & PREOPERATIONAL TEST ORGANIZATION

DBL
ACRS
3/3-4/82



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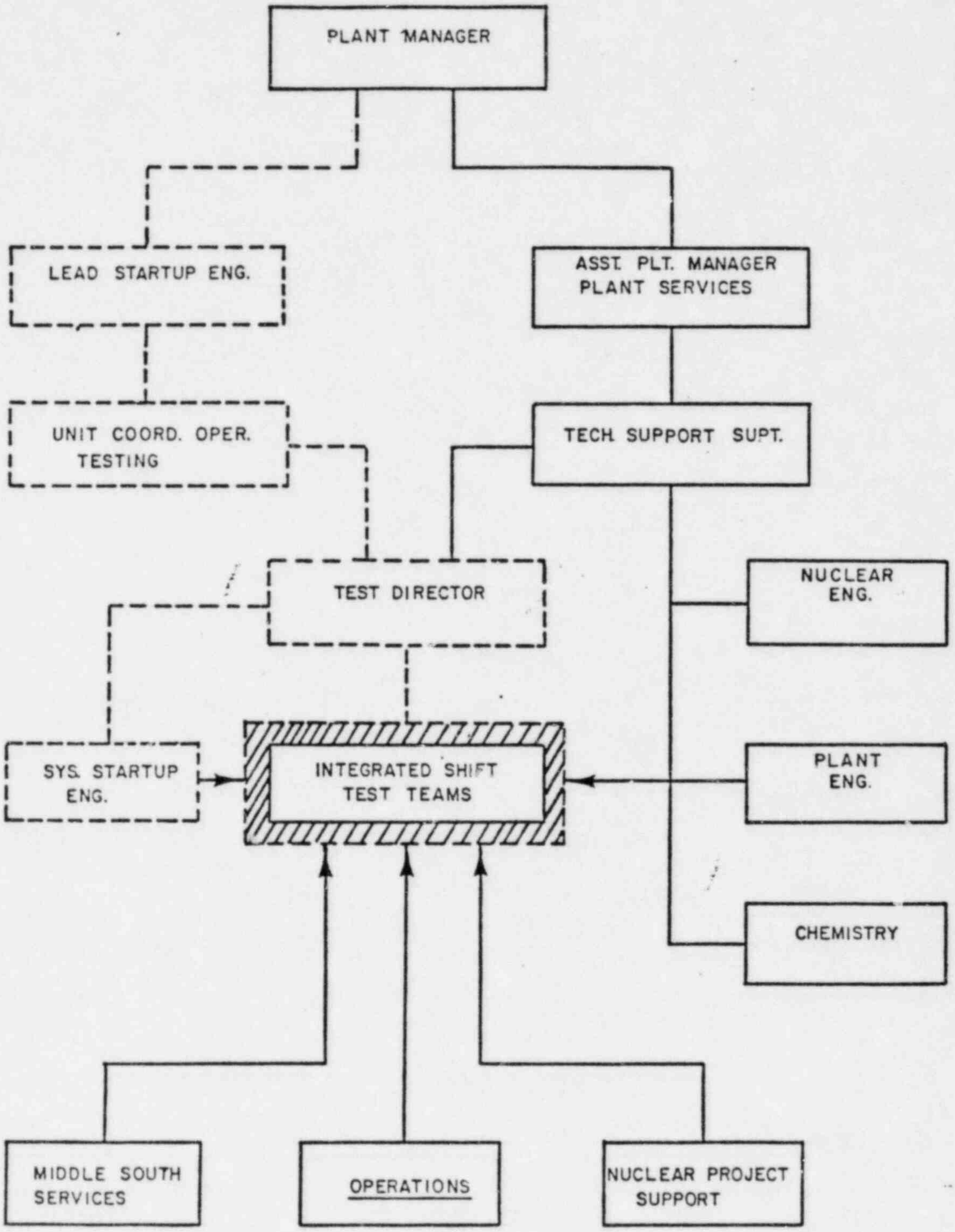
LP&L INTEGRATED TESTING ORGANIZATION

DBL
ACRS
3/3, 4/82



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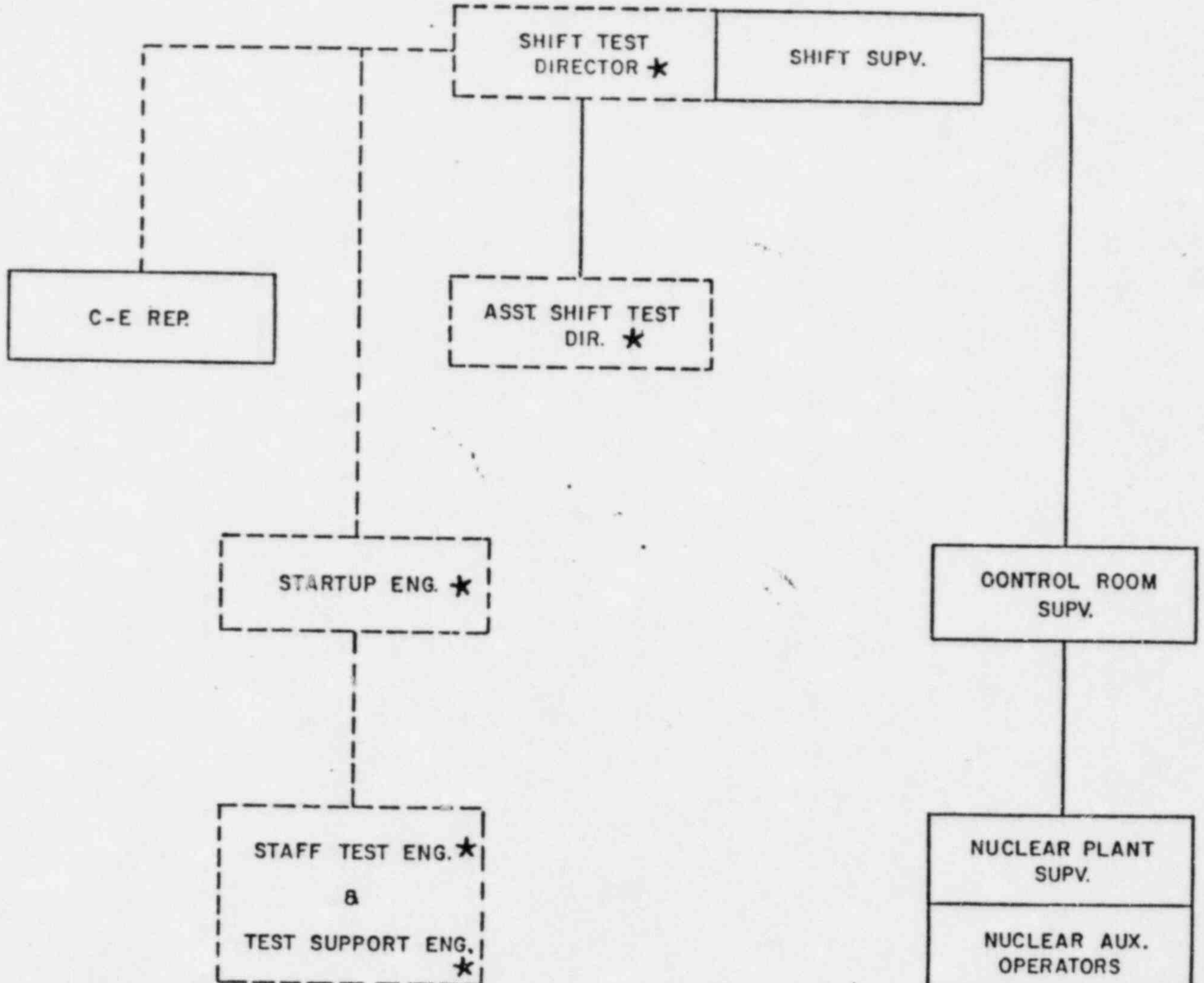
DBL
ACRS
3/3, 4/82

SHIFT TEST TEAM
INTEGRATED TESTS



★ POSITIONS TO BE HELD BY QUALIFIED PERSONNEL FROM THE STARTUP GROUP, PLANT STAFF AND SUPPORT STAFF

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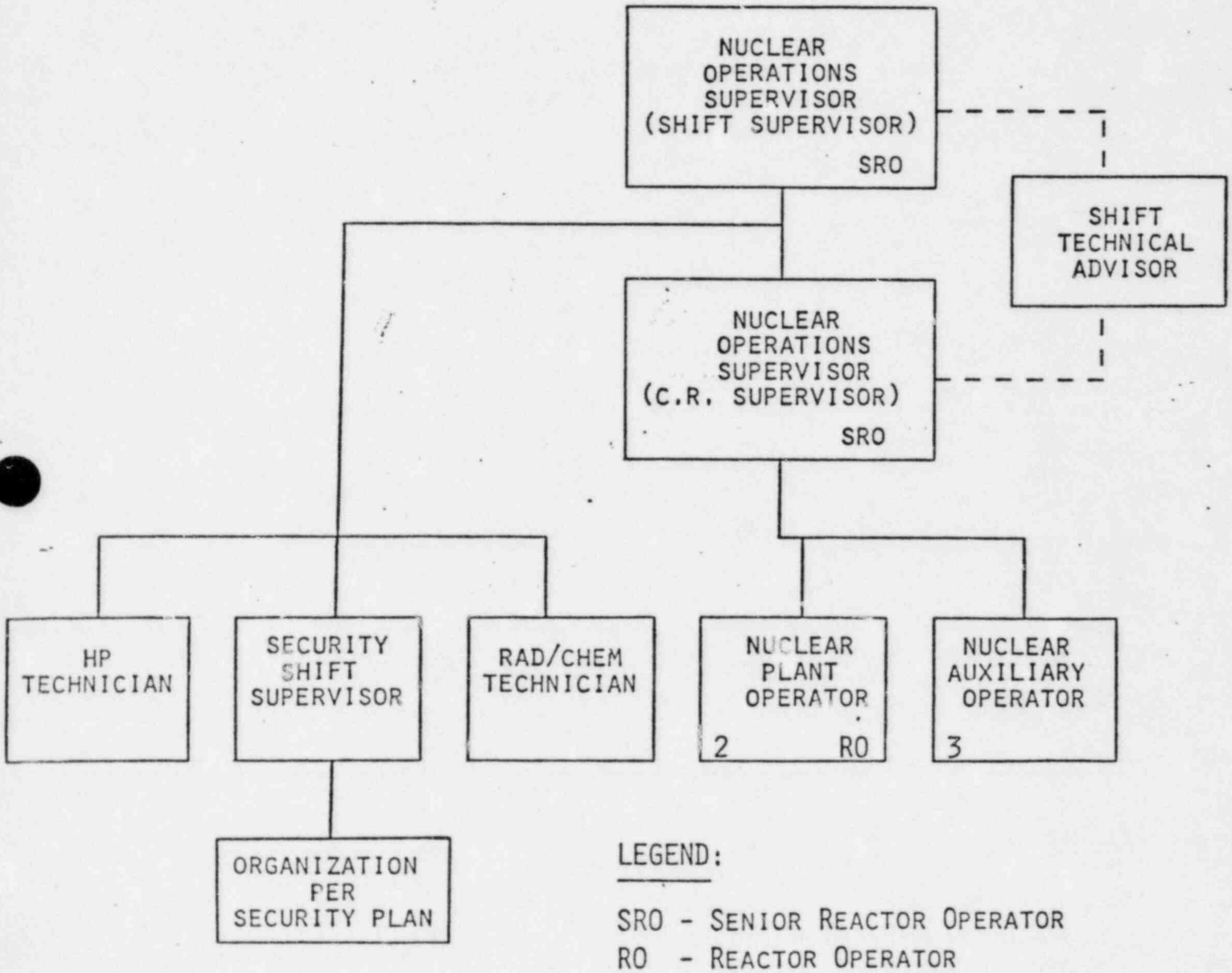
DBL-ACRS
3/3-4/82

COMMERCIAL OPERATION SHIFT ORGANIZATION



AGENDA FOR MARCH 4, 1982 ACRS

• INTRODUCTION (2 MIN.)	J.M. WYATT-PRESIDENT CHIEF EXECUTIVE OFFICER
• MANAGEMENT COMMITMENT (5 MIN.)	G.D. McLENDON - SR. VICE PRESIDENT - OPERATIONS
• OVERVIEW OF NUCLEAR OPERATIONS DEPARTMENT ORGANIZATION CHANGES/ STAFFING AND RECRUITING (15 MIN.)	L.V. MAURIN - VICE PRES. NUCLEAR OPERATIONS
• PLANT STAFF ORGANIZATION CHANGES/INTEGRATION OF CONTRACTORS IN ORGANIZATION (15 MIN.)	D.B. LESTER - PLANT MGR.
• PROJECT SUPPORT ORGANIZATION CHANGES/SAFETY REVIEW PROGRAM (10 MIN.)	F.J. DRUMMOND - PROJECT SUPPORT MANAGER
• TRAINING (10 MIN.)	Z.A. SABRI - NUCLEAR TRAINING DIRECTOR
• CLOSING REMARKS (3 MIN.)	G.D. McLENDON - SR. VICE PRESIDENT - OPERATIONS

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3/4/82

NUCLEAR PROJECT SUPPORT ORGANIZATION

- ORGANIZATION
- FUNCTIONAL RESPONSIBILITIES
- STAFFING
- SAFETY REVIEW AND AUDIT

FJD-ACRS
3/3-4/82

NUCLEAR PROJECT SUPPORT ORGANIZATION

TRANSITION

- FROM CONSTRUCTION . . . TO OPERATIONS PHASE
  - RETAIN PERSONNEL FROM DESIGN/CONSTRUCTION PHASE
  - ALL SUBGROUP SUPERVISORS IN PLACE
  - INCREASE EMPHASIS ON OPERATIONAL SUPPORT
  - FOCUS ON PLANNING ACTIVITIES
- FROM CONSTRUCTION . . . TO LP&L TECHNICAL MANAGEMENT
  - INCREASE STAFFING
  - MOLD FUNCTIONAL GROUPS
  - CONTRACT TEMPORARY EXPERTISE FOR INTERIM WORKLOAD
  - TRAINING
- FROM CONSTRUCTION . . . TO SUPPORT ORGANIZATION
  - EMPHASIZE TECHNICAL EXPERTISE STAFFING
  - REDUCE BURDENS OF ADMINISTRATIVE TASKS
  - ACCELERATED RECRUITING EFFORTS
  - TRAINING
  - FOLLOW NUREG - 0731 AND INDUSTRY GUIDANCE

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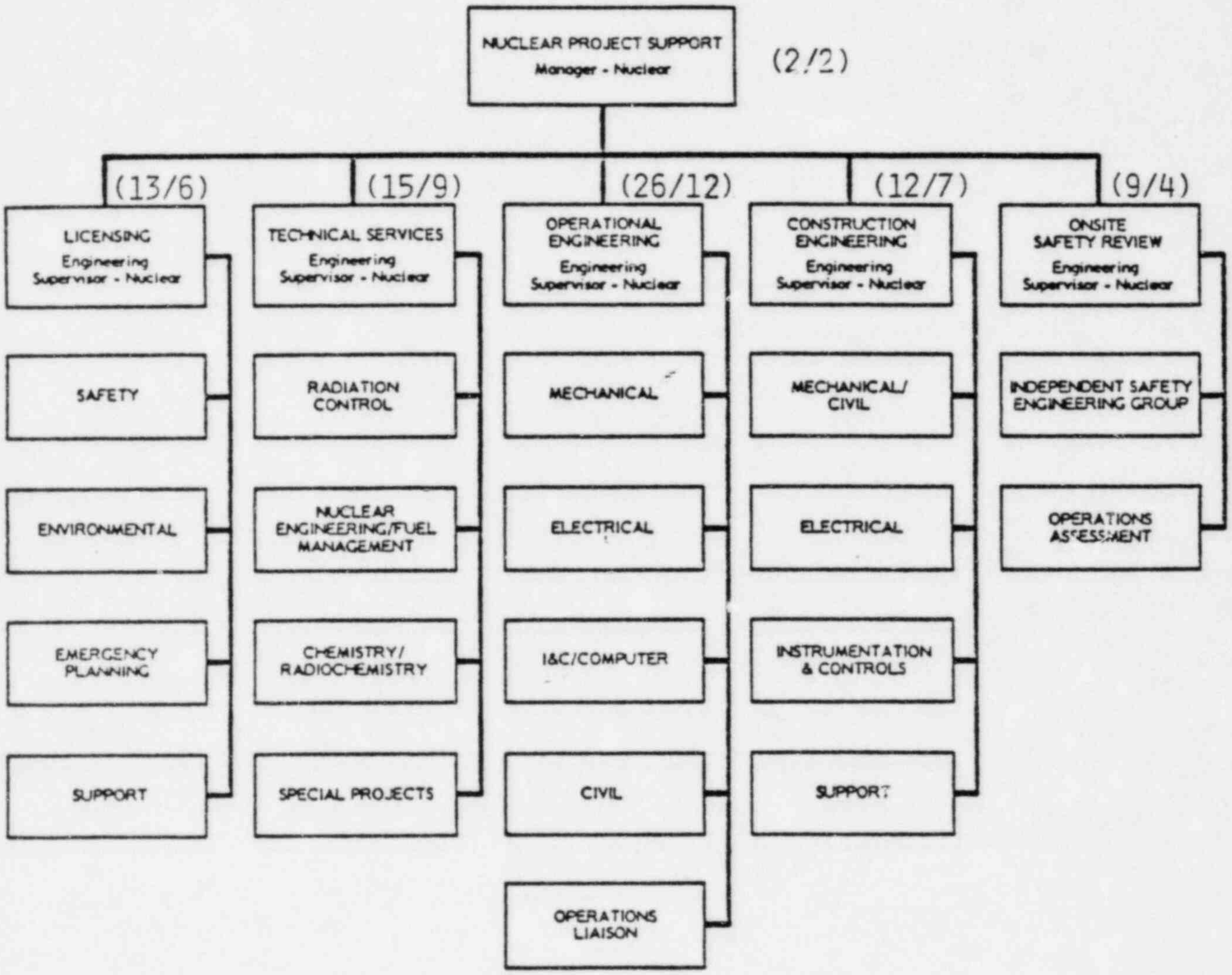

( ) NUMBER OF PERSONNEL (AUTHORIZED/HIRED)

TOTAL AUTHORIZATION - 77

TOTAL HIRED - 40

NUCLEAR PROJECT SUPPORT ORGANIZATION

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3/3-4/82

SAFETY REVIEW AND AUDIT PROGRAM

- SAFETY REVIEW COMMITTEE
  - REPORTS TO SENIOR VICE PRESIDENT - OPERATIONS
  - 12 MEMBERS - POSITIONS FILLED
  - 3 OUTSIDE NATIONALLY RECOGNIZED EXPERTS
  - ASSESSES POTENTIAL RISKS AT WATERFORD-3
  - EXAMINES EFFECTIVENESS OF PORC, ONSITE SAFETY REVIEW AND QUALITY ASSURANCE
- PLANT OPERATIONS REVIEW COMMITTEE
  - REPORTS TO PLANT MANAGER
  - 8 MEMBERS FROM PLANT STAFF
  - ADVISES PLANT MANAGER ON MATTERS RELATED TO NUCLEAR SAFETY
- QUALITY ASSURANCE
  - INDEPENDENT REVIEW AND AUDIT OF OPERATION, MAINTENANCE AND TESTING
  - REPORTS TO SENIOR VICE PRESIDENT - OPERATIONS
- ONSITE SAFETY REVIEW SUBGROUP

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3/3-4/82

ONSITE SAFETY REVIEW SUBGROUP

- REPORTING RESPONSIBILITY
  - REPORTS TO THE MANAGER, NUCLEAR PROJECT SUPPORT GROUP
- FUNCTIONAL RESPONSIBILITY
  - EVALUATE SAFETY-RELATED PROCEDURES FOR TECHNICAL ACCURACY, ADEQUACY AND CLARITY
  - REVIEW PLANT OPERATIONS FROM A SAFETY PERSPECTIVE
  - EVALUATE QA PROGRAM EFFECTIVENESS
  - COMPARE WATERFORD OPERATING EXPERIENCE WITH OTHER PLANTS
  - ASSESS PLANT PERFORMANCE REGARDING CONFORMANCE TO SAFETY REQUIREMENTS
  - ASSESS PLANT SAFETY PROGRAMS
- ORGANIZATION
  - INDEPENDENT SAFETY ENGINEERING SECTION
  - OPERATIONS ASSESSMENT SECTION
- DISCIPLINES
  - ELECTRICAL
  - MECHANICAL
  - I & C
  - NUCLEAR
  - RADIATION PROTECTION

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3/3-4/82

TRAINING GROUP

- LP&L APPROACH TO TRAINING AND TRAINING PHILOSOPHY
- ORGANIZATION TO IMPLEMENT LP&L TRAINING CONCEPTS
- STATUS OF TRAINING PROGRAMS

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ZAS-ACRS
3/3-4/82

TRAINING

- MANAGEMENT SUPPORT TO TRAINING
- KEY FEATURES OF THE TRAINING PROGRAMS
  - BEST FEATURES OF BS ENGINEERING
  - PLANT SPECIFIC
  - DEVELOPING MECHANISMS FOR TIMELY UPDATING OF PROGRAMS TO REFLECT
    - OPERATION EXPERIENCE (LERs, SEE-IN, ETC.)
    - WATERFORD 3 MODIFICATION
    - PRA - FMEA RESULTS
    - REGULATORY AND INPO GUIDES
    - SAFETY REVIEW COMMITTEE
  - CLOSE INTERACTION WITH DIFFERENT DEPARTMENTS
  - MEASURING EFFECTIVENESS - MODIFICATIONS
  - FLEXIBILITY - VARIATION IN BACKGROUND OF TRAINEE
  - ASSURING THAT EVERY INDIVIDUAL HAS A VIVID VISUALIZATION AND REALIZATION OF THE CONSEQUENCES OF HIS ACTIONS
  - HANDS ON EXPERIENCE AND MAXIMUM INVOLVEMENT OF TRAINEES - GENERATION OF THE RIGHT "WHAT IF's?"
- SIMULATOR TRAINING AND PLANNED TRAINING CENTER
  - LOCATION AND SCHEDULING
  - CHARACTERISTICS OF THE PLANNED SIMULATOR
- SELECTION AND SCREENING AND CAREER PATHS

ZAS-ACRS
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- V.P. Nuclear Operations
  - Training Director - Nuc.
    - ★ Training Manager
      - Operations Training
      - Simulator Training
      - Electrical, Controls, Computer Training
      - Chemistry & Health Physics Training
      - General Employee Training
      - Mechanical Maintenance Training
      - QA Training
      - Emergency Plan Training
      - Engineering Training
      - Offsite Training
      - Training Support and Cost Control
    - ★ Simulator /Training Facilities Procurement
      - Plant System Simulation
      - Configuration Control
      - Facilities
      - Project Adminst.
      - Simulator Facilities Operation
      - H.P. & Chem.
      - I/C
      - Mechanical
      - Electrical
    - ★ Training Supervisor
      - SOAR Program/Products and training update (inputs: PRA, H.F., FMEA, PORP)
      - OE&A Interface
      - Plant Manager and PORC Interface
      - Safety Review Committee Interface
      - Licensing/INPO Interface
      - Onsite Safety Interface

★ Staffing Levels are being developed by Training Director for discussion with upper Management

Functions:

- W-3 plant specific Inhouse "Think Tank"
- Immediate Implementation of the inputs from the SOAR Program and other inputs into training programs
- Identify items for considerations by SRC, PORC, PM, etc.
- Engineering Cadet Training

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3/3-4/82

CURRENT TRAINING RESOURCES

	LP&L	CONTRACT	TOTAL NUCLEAR EXPERIENCE	TOTAL COMMERCIAL EXPERIENCE	COMMERCIAL OPERATING EXPERIENCE
MANAGEMENT	1	0	14	5	0
OPERATIONS	3	8	120	82	62
ELECTRICAL	0	3	23	4	1
INSTRUMENT & CONTROLS	1	0	10	2	0
CHEMISTRY	1	0	1	14	0
HEALTH PHYSICS	1	0	4	4	3
GENERAL EMPLOYEE	0	2	23	2	0
MECHANICAL	1	1	64	6	0
ENGINEERING	1	3	21	2	11
TRAINING SUPPORT	1	8	80	44	13
TOTALS	10	25	360	165	90

*EXPERIENCE IN YEARS

SUMMARY .

- LP&L HAS MADE SIGNIFICANT PROGRESS IN ITS NUCLEAR ORGANIZATION SINCE LAST AUGUST BY:

    - NUCLEAR OPERATION DEPARTMENT FORMATION

    - VICE PRESIDENT - NUCLEAR OPERATION DEPARTMENT BEING LOCATED ON SITE TO GIVE CORPORATE DIRECTION AND TIMELY DECISION MAKING

    - PUTTING TOGETHER AN INTEGRATED LP&L AND CONTRACT TEAM WITH REQUISITE EXPERIENCE TO TEST AND START-UP WATERFORD-3

    - A SIGNIFICANT INCREASE IN NUCLEAR EXPERIENCE WITHIN LP&L ORGANIZATION

    - OUR AGGRESSIVE RECRUITING PROGRAM BEING SUCCESSFUL IN HIRING QUALITY PERSONNEL

    - HAVING OBTAINED A HIGHLY QUALIFIED PROFESSIONAL TO DIRECT LP&L's TRAINING PROGRAM

    - INVOLVEMENT OF LP&L MANAGEMENT AND BOARD OF DIRECTORS TO ENSURE AN APPRECIATION FOR THE MAGNITUDE OF WATERFORD-3

    - ADDITION OF RECOGNIZED EXPERTS TO THE SAFETY REVIEW COMMITTEE

WE ARE DEDICATED TO SAFELY MANAGE AND OPERATE WATERFORD-3 AND WILL CONTINUE TO EVALUATE THE NEEDS OF OUR ORGANIZATION SUCH THAT AN IN-DEPTH TEAM OF QUALIFIED PERSONS ARE MAINTAINED AT LP&L. WE BELIEVE WE HAVE REACTED TO YOUR AUGUST 1981 REPORT IN A RESPONSIBLE AND EXPEDITIOUS MANNER.

APPENDIX VIII
CONSULTANT'S REPORT - TECHNICAL REVIEW
OF CLINTON PLANT VISIT

University of Pittsburgh

SCHOOL OF LIBRARY AND INFORMATION SCIENCE
Interdisciplinary Department of Information Science

TO: Mr. Richard Savio, NRC

FROM: Anthony Debons, Ph.D.
Consultant

DATE: March 3, 1982

SUBJECT: Technical Review of Clinton Plant Visit
Decanteur, Illinois, February 25-26, 1982.

1.0 General Assessment

Visit did not allow for the detailed study of the many aspects of human factors that could be applied to total plant operations (including training, career development, habitat, etc.). To best of available knowledge there is no Human Factors representation at the staff level of the plant organization. Human Factors is accounted through services of a Human Factors consultant. There is a need for a well formulated research program for the test and validation of present display formats developed by the applicant. The Remote Control "shut down" Display Panel requires careful study both in terms of console configuration and location. Training program needs careful examination of underlying career assumptions.

2.0 Human Factors Representation

The magnitude and importance of human involvement with technology and plant procedures suggest that human factor (ergonomic) principles can be extended to the following areas:

2.1 Main control room display design and operations. This includes the Remote "shut down" Control Display Console.

2.2 Quality control in the present construction program and subsequent maintenance requirements.

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2.3 To the environmental conditions projected for operational status. This includes such factors as general habitat, lighting-heating conditions under prolonged work schedules, noise, vibrations, etc.

2.4 Training.

The impact of career ladder structure on motivation and ultimately on productivity. These considerations need to be carefully integrated into the present' policies for training of personnel, if not presently included.

3.0 Control Display Console

The applicant is proud of present display formats included in the Control Display Console. The display formats, however, appear crowded. At present it is not possible to correlate the effectiveness of display formats with human performance requirements.

The Remote "shut down" Control Display Console seems primitive. It leaves the impression that it was included as an after-thought in the design of the overall plant configuration.

Variations in electric source output can severely influence the resolution of data on displays particularly those which require high levels of visual acuity. On the other hand information presented by the applicant suggest that this factor has been acknowledged.

4.0 Quality Control

Numerous incidences reflecting inadequate quality control in the construction activities were reported. Quality control practices by the contractor were defended. Specific instances of good quality control were cited in contrast to the many other quality control violations. In most of these cited instances there was a recognition of symptoms rather than causes. In some instances of inadequate quality control cited in the literature, failure in quality control can be traced to poor acknowledgment of human factors parameters (i.e. poor lighting, poor work habits, fatigue, motivation, etc.).

5.0 Environmental Conditions

Plant configuration and mission suggest that operational personnel will be shielded during work hours from the environment outside of the plant. Further, it is reasonable to speculate that the nature of the work environment can promote possible states of uncertainty and stress. These conditions can lead to a sense of isolation and at times, depression - the extent of which should be identified and studied. Other factors include the influence of prolonged noise, heat and other ambient conditions which could seriously impact on human performance (e.g. tracking, vigilance, etc.).

6.0 Training

The training program seems well formulated. Liaison with local educational institutions has been established. Career ladders - for progressive career development and training - are available. The following aspects underlying the training program however need to be examined.

6.1 The requirement for college level achievement to meet specific work levels.

6.2 Retraining concepts based on technological state-of-the-art developments.

6.3 Criteria for validating training effectiveness.

6.4 Inclusion of Human Factors study in current college level curriculum.

7.0 Recommendations

7.1 Develop a specifically defined and well delineated research program for the study of display formats for the Control Room Display Console.

7.2 A study should be initiated on Human Factors problems related directly to quality control aspects. The findings of this study should clearly suggest a program for the generating of policies which insure that Human Factors are sufficiently acknowledged in instances of quality control non-compliance and breakdown.

CLINTON POWER PLANT

OCTOBER 30, 1973	APPLICATION DOCKETED FOR UNITS 1 & 2
OCTOBER 1974	FES ISSUED
MARCH 1975	SER ISSUED (NUREG-75/013)
JUNE 17-JULY 3, 1975	ASLB HEARINGS
SEPTEMBER 30, 1975	ASLB PARTIAL INITIAL DECISION
OCTOBER 1975	LIMITED WORK AUTHORIZATION ISSUED
DECEMBER 1975	SER SUPPLEMENT 1 ISSUED
JANUARY 7-8, 1976	ASLB HEARINGS
FEBRUARY 20, 1976	ASLB DECISION
FEBRUARY 1976	CPPR-137 AND CPPR-138 ISSUED
JULY 29, 1976	ASLAB AFFIRMED ASLB DECISION
SEPTEMBER 25, 1978	SOYLAND POWER COOPERATIVE, INC AND WESTERN ILLINOIS POWER COOPERATIVE, INC BECAME APPLICANTS
DECEMBER 1, 1979	APPLICATION FOR OL'S FOR UNITS 1 & 2 TENDERED
SEPTEMBER 8, 1980	APPLICATION DOCKETED
OCTOBER 30, 1981	ILLINOIS POWER COMPANY REQUESTED SEPARATION OF UNITS 1 & 2 PROCEEDINGS
NOVEMBER 13, 1981	ASLB GRANTED APPLICANT'S REQUEST
DECEMBER 1981	DES ISSUED
FEBRUARY 1982	SER ISSUED
APRIL 1982	FES ISSUE
MARCH 1982	SSER ISSUE
AUGUST 2, 1982	ASLB HEARINGS
JANUARY 1983	APPLICANTS ESTIMATED FUEL LOAD

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OUTSTANDING ISSUES

- TRANSPORTATION ACCIDENTS
- ~~UNIT 2 EXCAVATION~~
- ~~SEISMIC ANALYSIS~~
- ~~INTERNALLY GENERATED MISSILES~~
- POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT
- STEADY-STATE VIBRATION ACCEPTANCE CRITERIA FOR BOP PIPING
- ENVIRONMENTAL AND SEISMIC QUALIFICATION TEST PROGRAMS
- PRESERVICE AND INSERVICE INSPECTION PROGRAMS
- CONTAINMENT PURGE, ISOLATION, BYPASS LEAKAGE, AND LEAKAGE TESTING
- CONTROL ROOM HABITABILITY
- ENGINEERED SAFETY FEATURES RESET CONTROLS
- REMOTE SHUTDOWN SYSTEM
- CAPABILITY FOR SAFE SHUTDOWN FOLLOWING LOSS OF BUS SUPPLYING POWER TO INSTRUMENTS AND CONTROLS
- CONTROL SYSTEM FAILURES RESULTING FROM HIGH-ENERGY LINE BREAKS OR COMMON POWER SOURCE OR SENSOR MALFUNCTIONS
- ~~SEPARATION OF THE RPS AND MSIV SOLENOID CIRCUITS AND PGCC CIRCUITS~~
- ORGANIZATION AND STAFFING
- EMERGENCY PLAN
- SECURITY
- QA PROGRAM
- POOL DYNAMIC LOADS

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1. TRANSPORTATION ACCIDENTS (2.2)

DESCRIPTION:

THE ILLINOIS CENTRAL GULF RAILROAD PASSES 0.75 MILES NORTH OF THE STATION: BASED ON 1976 AND 1980 TRANSPORTATION DATA OBTAINED FROM THE RAILROAD THE APPLICANT HAS IDENTIFIED SEVERAL MATERIALS REQUIRING FURTHER ANALYSIS. THE HAZARDS ASSOCIATED WITH TOXIC AND EXPLOSIVE MATERIALS IS STILL BEING EVALUATED.

<u>STATUS</u>:

THE STAFF EXPECTS TO RECEIVE ADDITIONAL INFORMATION ON THIS ITEM IN ~~MARCH~~.
JUNE

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2. EFFECTS ON UNIT 2 EXCAVATION
(2.4.2.2, 2.4.6, 2.4.8, 2.6.3.3, 2.6.3.7)

DESCRIPTION AND STATUS

THE EXCAVATION COULD CAUSE PONDING OF WATER AGAINST THE WALL OF UNIT 1 THAT IS EXPOSED BELOW PLANT GRADE. THE APPLICANT HAS COMMITTED TO CERTAIN IMPROVEMENTS TO REMEDY THIS CONDITION (2.4.2.2). THIS INFORMATION MUST BE INCORPORATED IN THE FSAR.

THE APPLICATION HAS USED ELEVATION 730 FT MSL AS THE DESIGN GRADE FOR HYDROSTATIC LOADINGS. WITH THE OPEN EXCAVATION AND STEEP GRADIENT TOWARDS THE COOLING LAKE ON THE OPPOSITE SIDE OF THE PLANT WILL TEND TO KEEP GROUNDWATER LEVEL BELOW 730 FT MSL. NO PERMANENT UNDER DRAIN OR GROUNDWATER DEWATERING SYSTEMS ARE INSTALLED. THE STAFF CONCLUDED THAT THE DESIGN BASIS GROUNDWATER LEVEL OF 730 FT MSL SATISFIES GDC 2 FOR HYDROSTATIC LOADING PROVIDED THE OPEN EXCAVATION REMAINS AS IS. IN THE EVENT THE EXCAVATION IS TO BE FILLED IN OR UNIT 2 IS CONSTRUCTED FURTHER VERIFICATION IS REQUIRED FROM THE APPLICANT THAT EITHER 730 FT MSL IS THE MAXIMUM GROUNDWATER LEVEL ATTAINABLE OR THAT THE PLANT MAY SAFETY WITHSTAND HIGHER GROUNDWATER LEVELS. (2.4.6)

THE APPLICANT IS INVESTIGATING THE ADDITIONAL SETTLEMENT WHICH WOULD BE EXPERIENCED BY UNIT 1 AS A RESULT OF THE LOAD FROM UNIT 2 IF IT IS CONSTRUCTED AT A LATER DATE. (2.6.3.3)

THE APPLICANT HAS BEEN REQUESTED TO EVALUATE THE EFFECTS OF THE OPEN EXCAVATION ON THE VALIDITY OF HIS DESIGN ASSUMPTIONS FOR UNIT 1.

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3. SEISMIC ANALYSIS (2.5.2, 3.7.1, 3.7.2)

DESCRIPTION AND STATUS

IN THE SER THE STAFF CONCLUDED THAT THE DESIGN BASIS EARTHQUAKE IS ONE OF INTENSITY I_{MM}=VII AND THE CORRESPONDING FREE-FIELD VIBRATORY GROUND MOTION IS 0.25G ACCELERATION ANCHORED TO THE REG. GUIDE 1.60 STANDARD RESPONSE SPECTRUM.

(2.5.2) THE APPLICANT SUBMITTED A REPORT BY WESTERN GEOPHYSICAL ON SITE SPECIFIC SPECTRA ON FEBRUARY 17, 1982. AFTER SATISFACTORY COMPLETION OF ITS REVIEW, THE STAFF MAY BE ABLE TO CONCLUDE THAT THE MORE APPROPRIATE DESCRIPTION OF THE EARTHQUAKE (M_b = 5.8) IS A SET OF 84TH PERCENTILE SITE-SPECIFIC SPECTRA APPROXIMATELY EQUIVALENT TO 0.20G ACCELERATION REG. GUIDE 1.60 STANDARD RESPONSE SPECTRA.

IN THE EVENT THAT THE SITE SPECIFIC SPECTRA ARE SOMEWHAT HIGHER THAN THE 0.17G REG. GUIDE 1.60 SPECTRA, ADDITIONAL ANALYSIS TO QUALIFY THE PLANT DESIGN FOR THE HIGHER SITE SPECIFIC SPECTRA MAY BE REQUIRED (3.7.1)

IT IS THE STAFF'S POSITION THAT THE SOIL-STRUCTURE INTERACTION SHOULD INCLUDE BOTH ELASTIC HALF-SPACE AND FINITE DEMENT APPROACHES FOR ALL CATEGORY I STRUCTURES FOUNDED IN SOIL. THE APPLICANT USED THE FINITE ELEMENT METHOD AND DEMONSTRATED THAT THE RESULTS ENVELOP THE RESULT OF "HALF-SPACE" ANALYSIS BASED ON A SEISMIC EVENT DEFINED BY REG. GUIDE 1.60 SPECTRA ANCHORED AT 0.17G. THE FINAL RESOLVATION OF THIS MATTER IS DEPENDENT UPON THE ACCEPTANCE OF THE SITE SPECIFIC SPECTRA. (3.7.2)

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4. INTERNALLY GENERATED MISSILES (3.5.1)

DESCRIPTION:

THE STAFF REQUESTED INFORMATION ON SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS INSIDE AND OUTSIDE CONTAINMENT PROTECTED FROM INTERNALLY GENERATED MISSILES. THE INITIAL RESPONSE TO THIS REQUEST WAS NOT COMPLETE.

STATUS:

ADDITIONAL INFORMATION HAS BEEN PROVIDED AND EVALUATED. THIS ITEM IS NOW SATISFACTORY AND CONSIDERED CLOSED.

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5. POSTULATED PIPING FAILURES (3.6.1, 3.6.2)

DESCRIPTION:

THE APPLICANT HAS NOT YET COMPLETED THE ANALYSIS FOR THE EFFECTS OF JET IMPINGEMENT. THIS WILL BE SUBMITTED AFTER COMPLETION OF THE NEW LOADS (BWR MARK III HYDRODYNAMIC AND ANNULUS PRESSURIZATION) EVALUATION PROGRAM (3.6.1)

THE APPLICANT HAS NOT PROVIDED THE RESULTS OF THE ANALYSIS FOR JET IMPINGEMENT FOR HIGH AND MODERATE-ENERGY FLUID SYSTEMS. ADDITIONALLY, THE APPLICANT HAS NOT PROVIDED THE RESULTS OF THE PIPE RUPTURE ANALYSIS FOR THE REACTOR WATER CLEANUP SYSTEM AND THE CONTROL ROD DRIVE SYSTEM. (3.6.2)

STATUS:

THE STAFF IS AWAITING INFORMATION ON BOTH OF THE ABOVE ITEMS.

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6. STEADY-STATE VIBRATION ACCEPTANCE CRITERIA FOR BOP PIPING (3.9.2)

DESCRIPTION:

FOR THE BALANCE OF PLANT (BOP) PIPING, THE APPLICANT IS PROPOSING TO USE 80% OF THE ALTERNATING STRESS INTENSITY, S_a AT 10^6 CYCLES, AS DEFINED IN THE ASME CODE APPENDIX I, FIGURES I.9.1 AND I.9.2. BECAUSE OF THE LIMITED AVAILABILITY OF HIGH-CYCLE FATIGUE TEST DATA (GREATER THAN 10^6 CYCLES) THE STAFF HAS NOT DETERMINED WHETHER AN ACCEPTABLE LEVEL OF SAFETY EXISTS.

STATUS:

STAFF'S POSITION IS THAT 50% OF ALTERNATING STRESS INTENSITY, S_a, AT 10^6 CYCLES SHOULD BE USED FOR 10^8-10^9 CYCLES. STAFF DOES NOT BELIEVE EXISTING DATA ALLOWS ONE TO GO TO 80% AT 10^8-10^9 CYCLES AND STILL BE CONSERVATIVE.

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7. ENVIRONMENTAL AND SEISMIC QUALIFICATION TEST PROGRAMS (3.9.3.2, 3.10, 3.11, 3.4.6)

DESCRIPTION:

THE APPLICANT HAS NOT PROVIDED THE REQUIRED INFORMATION ASSOCIATED WITH:

(1) PUMP AND VALVE OPERABILITY ASSURANCE (3.9.3.2)

(2) SEISMIC AND DYNAMIC QUALIFICATION OF SEISMIC CATEGORY I MECHANICAL AND ELECTRICAL EQUIPMENT (3.10)

(3) ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED ELECTRICAL EQUIPMENT (3.11) FOR CLASS 1E ELECTRICAL EQUIPMENT THAT GETS SUBMERGED THE QUALIFICATION PROGRAM WILL INCLUDE THIS IN CHAPTER 3. (8.4.6)

STATUS:

THE APPLICANT PROVIDED INFORMATION IN FSAR AMENDMENT 12 DATED JANUARY 1982 ON ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED ELECTRICAL EQUIPMENT. THIS INFORMATION IS BEING REVIEWED BY THE STAFF.

AFTER THE REQUIRED INFORMATION IS RECEIVED AND EVALUATED, SITE AUDITS WILL BE MADE. BASED UPON THE STATUS OF CONSTRUCTION THE AUDITS WOULD BE CONDUCTED IN JULY 1982 OR LATER.

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8. PRESERVICE AND INSERVICE INSPECTION PROGRAMS (3.9.6, 6.6, 5.2.4)

DESCRIPTION:

THE APPLICANT HAS NOT SUBMITTED HIS PROGRAM FOR:

(1) PRESERVICE AND INSERVICE TESTING OF PUMPS AND VALVES (3.9.6)

(2) PRESERVICE AND INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS (6.6)

(3) PRESERVICE AND INSERVICE INSPECTION PROGRAMS OF THE REACTOR COOLANT PRESSURE BOUNDARY (5.2.4)

STATUS:

THE APPLICANT WILL SUBMIT A PRESERVICE INSPECTION PROGRAM BY APRIL 1, 1982 OR BEFORE THE START OF PRESERVICE INSPECTION. THE PRESERVICE INSPECTION PROGRAM FOR THE RCPB WILL INCLUDE IDENTIFICATION OF AREAS WHERE ASME CODE SECTION XI REQUIREMENTS CANNOT BE MET, AND SUPPORTING TECHNICAL JUSTIFICATION.

THE INSERVICE INSPECTION PROGRAM WILL BE EVALUATED AFTER THE APPLICABLE ASME CODE EDITION AND ADDENDA CAN BE DETERMINED BASED ON 10 CFR 50.55 A (B), BUT BEFORE THE FIRST REFUELING OUTAGE WHEN INSERVICE INSPECTION COMMENCES.

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9. POOL DYNAMIC LOADS (6.2.1.8)

DESCRIPTION:

SEVERAL PHENOMENA HAVE BEEN IDENTIFIED IN THE REVIEW OF THE MARK III CONTAINMENT THAT COULD RESULT IN DYNAMIC LOADING OF STRUCTURES LOCATED IN AND ABOVE THE SUPPRESSION POOL.

STATUS: THE LOCA POOL DYNAMIC LOADS ARE BEING REVIEWED UNDER TASK ACTION PLANT (TAP) B-10, "BEHAVIOR OF BWR MARK III CONTAINMENT" AND TAP A-39, "DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT". THE END PRODUCT OF THESE TWO GENERIC PROGRAMS WILL BE APPLICABLE TO CLINTON.

A COPY OF THE DRAFT "NRC ACCEPTANCE CRITERIA FOR LOCA RELATED MARK III CONTAINMENT POOL DYNAMIC LOADS" WAS PROVIDED TO THE APPLICANT IN EARLY FEBRUARY 1982.

A MEETING WAS HELD TO DISCUSS THE CRITERIA AND POOL DYNAMIC LOADS ON FEBRUARY 17, 1982. A SCHEDULE FOR SUPPLYING THE ADDITIONAL ANALYSES WILL BE AVAILABLE IN MARCH AND ALL THE INFORMATION WILL BE SUPPLIED TO THE STAFF BY SEPTEMBER 1982. BASED UPON THE DISCUSSIONS, CLINTON DOES NOT APPEAR TO HAVE ANY SPECIAL PROBLEMS.

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10. CONTAINMENT PURGE, ISOLATION, ~~BYPASS LEAKAGE~~, AND LEAKAGE TESTING (6.2.4, 6.2.6, 6.4, 15.3.1, 6.2.2)

DESCRIPTION AND STATUS:

1. THE STAFF REQUIRES THE APPLICANT TO COMMIT TO LEAKAGE TESTING OF THE SECONDARY CONTAINMENT VOLUMES TO VERIFY THE 194 SECOND BLOWDOWN TIME TO RE-ESTABLISH A -0.25 INCH OF WATER GUAGE PRESSURE (6.2.2). THE APPLICANT HAS AGREED TO THIS TESTING. THE ITEM IS CONFIRMATORY UNTIL THIS COMMITTMENT IS DOCUMENTED.

2. THERE ARE ISOLATION PROVISIONS OF PENETRATIONS THAT DO NOT HAVE TO SATISFY THE EXPLICIT REQUIREMENTS OF GDS (55 AND 56), BUT CAN BE ACCEPTABLE ON SOME OTHER DEFINED BASIS. HOWEVER THE APPLICANT HAS NOT JUSTIFIED THE DESIGN DEVIATION FROM THE EXPLICIT REQUIREMENTS OF THE GDC. (6.2.4) THE STAFF IS AWAITING ADDITIONAL INFORMATION.

3. THE INFORMATION PROVIDED BY THE APPLICANT TO DEMONSTATE COMPLIANCE WITH CONTAINMENT LEAK TESTING REQUIREMENTS OF APPENDIX J (10 CFR 50) WAS NOT ADEQUATE. THE STAFF IS AWAITING ADDITIONAL INFORMATION ON LEAK TESTING OF VENT AND DRAIN LINES.

4. THE CONTAINMENT PURGE SYSTEM (CONTINUOUS USE OF THE 36" LINE) DOES NOT MEET BTP 6-4. THE STAFF BELIEVES THAT PURGING SHOULD BE MINIMIZED DURING NORMAL OPERATION AND SHOULD NOT BE RELIED ON FOR TEMPERATURE AND HUMIDITY CONTROL (6.2.4.1)

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THE STAFF REQUIRES THE APPLICANT TO PROVIDE A REALISTIC ESTIMATE OF THE NUMBER OF HOURS PER YEAR THAT PURGING IS EXPECTED THROUGH EACH PURGE VALVE, AND A JUSTIFICATION FOR THIS USE.

JWilliams
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11. CONTROL ROOM HABITABILITY (6.4)

DESCRIPTION:

THE THYROID DOSE IN THE CONTROL ROOM FOLLOWING A LOCA FROM RADIOACTIVE IODINE EXCEEDS THE DOSE GUIDELINES. THE APPLICANT WILL BE REQUIRED EITHER TO MODIFY HABITABILITY SYSTEMS OR REDUCE CONTAINMENT BYPASS LEAKAGE.

STATUS:

THE SER HAS BEEN WRITTEN WITH A REQUIRED TECHNICAL SPECIFICATION LIMIT ON BYPASS LEAKAGE FRACTION OF NO MORE THAN 4% OF CONTAINMENT LEAKAGE (15.3.1)

JWilliams
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12. ENGINEERED SAFETY FEATURE RESET CONTROLS (IE BULLETIN 80-06) (7.3.3.7)

DESCRIPTION:

THE STAFF REQUESTED THE APPLICANT REVIEW ALL SAFETY EQUIPMENT TO DETERMINE WHICH, IF ANY, SAFETY FUNCTIONS MIGHT BE UNAVAILABLE AFTER RESET AS WELL AS WHAT CHANGES WOULD BE IMPLEMENTED TO CORRECT ANY PROBLEMS. THE APPLICANT HAS NOT COMPLETED HIS REVIEW.

STATUS:

THE APPLICANT PROVIDED A PARTIAL RESPONSE IN A LETTER DATED DECEMBER 1, 1981 BUT DID NOT INCLUDE BOP SYSTEMS, CERTAIN NSSS SYSTEMS, OR JUSTIFICATION FOR THAT EQUIPMENT LISTED AS CHANGING MODES UPON AN ESF RESET.

THE APPLICANT IS PLANNING TO SUBMIT THE RESULTS OF HIS STUDY AT LEAST FOUR MONTHS PRIOR TO FUEL LOAD. (SEPTEMBER 1982)

JWilliams
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13. REMOTE SHUTDOWN SYSTEM (7.4.3.1, 9.5.5)

DESCRIPTION:

THE STAFF REQUIRES REDUNDANT SAFETY GRADE CAPABILITY TO ACHIEVE HOT SHUTDOWN AND SUBSEQUENT COLD SHUTDOWN FROM A LOCATION OR LOCATIONS REMOTE FROM THE CONTROL ROOM. THE STAFF IS CONCERNED THAT A FAILURE OF THE DIVISION 1 POWER SOURCE WILL PRECLUDE THIS CAPABILITY. (7.4.3.1)

THE STAFF REVIEW OF THE SHUTDOWN CAPABILITY AS A RESULT OF FIRE HAS NOT BEEN COMPLETED. (9.5.5)

STATUS:

THE APPLICANT PROVIDED HIS POSITION ON REMOTE SHUTDOWN IN A LETTER DATED DECEMBER 2, 1981. THE APPLICANT BELIEVES THE CURRENT REMOTE SHUTDOWN SYSTEM DESIGN IS ADEQUATE.

THE STAFF HAS PROVIDED COMMENTS AND HAD PLANNED TO MEET ON FEBRUARY 18, 1982 WITH THE APPLICANT TO DISCUSS FIRE PROTECTION OF THE SAFE SHUTDOWN CAPABILITY. THE MEETING WAS CANCELLED BECAUSE OF THE WEATHER AND WILL BE HELD IN MARCH.

JWilliams
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14. CAPABILITY FOR SAFE SHUTDOWN FOLLOWING LOSS OF BUS SUPPLYING POWER TO INSTRUMENTS AND CONTROLS (IE BULLETIN 79-27) (7.4.3.2)

DESCRIPTION:

THE STAFF REQUESTED THE APPLICANT TO REVIEW THE ADEQUACY OF EMERGENCY OPERATING PROCEDURES TO BE USED TO OBTAIN SAFE SHUTDOWN UPON LOSS OF ANY CLASS 1E OR NON-CLASS 1E BUS SUPPLYING POWER TO SAFETY-OR NONSAFETY-RELATED INSTRUMENTS AND CONTROLS.

STATUS:

BY LETTER DATED DECEMBER 1, 1981 THE APPLICANT HAS STATED THAT DETAILED REVIEWS HAVE BEEN INITIATED AND THE RESULTS WILL BE PROVIDED FOUR MONTHS PRIOR TO FUEL LOAD (SEPTEMBER 1982)

JWilliams
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15. CONTROL SYSTEM FAILURES RESULTING FROM HIGH-ENERGY LINE BREAKS OR COMMON POWER SOURCE OR SENSOR MALFUNCTIONS (7.7.3.1)

DESCRIPTION:

THE STAFF REQUESTED THE APPLICANT IDENTIFY ANY POWER SOURCES, SENSORS, OR SENSOR IMPULSE LINES WHICH PROVIDE POWER AND SIGNALS TO TWO OR MORE CONTROL SYSTEMS AND DEMONSTRATE THAT FAILURES OF THESE POWER SOURCES, SENSORS, OR SENSOR IMPULSE LINES WILL NOT RESULT IN CONSEQUENCES OUTSIDE THE BOUNDS OF SECTION 15 ANALYSES OR BEYOND THE CAPABILITY OF OPERATORS OR SAFETY SYSTEMS.

STATUS:

BY LETTER DATED DECEMBER 1, 1981, THE APPLICANT HAS STATED THAT DETAILED REVIEWS HAVE BEEN INITIATED AND THE RESULTS WILL BE PROVIDED FOUR MONTHS PRIOR TO FUEL LOAD. (SEPTEMBER 1982)

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16. SEPARATION OF RPS AND MSIV SOLENOID CIRCUITS AND PGCC CIRCUITS (8.4.7)

DESCRIPTION:

THE APPLICANT MUST DEMONSTRATE THE ADEQUACY OF SEPARATION BETWEEN CLASS 1E CIRCUITS AND THE SOLENOID CIRCUITS TO INSURE THAT BOTH CAN PERFORM THEIR SAFETY FUNCTION.

STATUS:

THE STAFF MET WITH THE APPLICANT, GE, AND S&L ON FEBRUARY 9, 1982. INFORMATION WAS PRESENTED TO SHOW THE ADEQUACY OF SEPARATION OF THE CIRCUITS.

THIS ITEM IS CLOSED.

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17. ORGANIZATION AND STAFFING (13.1.2.2)

DESCRIPTION:

THE ORGANIZATION TO CONDUCT OPERATION IS NOT COMPLETELY IN PLACE AND FUNCTIONING. THE STAFFING LEVELS AT THE TIME OF THE REVIEW WERE ABOUT 78% OF PROJECTIONS AT THE CORPORATE OFFICE AND 65% AT THE PLANT. THEREFORE, A SIGNIFICANT PORTION OF THE PEOPLE WHO WILL MAKE UP THE ORGANIZATION WERE NOT AVAILABLE FOR THE STAFF TO MEET TO DETERMINE QUALITY, EXPERIENCE LEVEL, AND ATTITUDES. MOREOVER, THE STAFF BELIEVES THAT AS THE APPLICANT APPROACHES LICENSING AND ATTEMPTS TO ESTABLISH A FUNCTIONAL OPERATING ORGANIZATION FURTHER INCREASES IN THE SIZE OF THE ORGANIZATION WILL BE NECESSARY.

STATUS:

THE STAFF WILL REVIEW THE ORGANIZATION AGAIN AT A TIME CLOSER TO FUEL LOAD WHEN THE OPERATING ORGANIZATION HAS FILLED OUT. THIS REVIEW IS ESTIMATED TO BE CONDUCTED 6-9 MONTHS FROM FUEL LOAD.

JWilliams
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18. EMERGENCY PLAN (13.3)

DESCRIPTION:

A REVISED PLAN WAS SUBMITTED WITH AMENDMENT 7 DATED SEPTEMBER 1981. THE PLAN IS CURRENTLY UNDER REVIEW. THE STAFF WILL DETERMINE THE ADEQUACY OF THE APPLICANT'S EMERGENCY RESPONSE PLAN WITH RESPECT TO APPENDIX E AND THE GUIDANCE OF NUREG-0654.

STATUS:

THE STAFF FORWARDED QUESTIONS ON THE EMERGENCY PLAN IN A LETTER DATED JANUARY 7, 1982. WE ARE AWAITING RESPONSES.

JWilliams
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19. SECURITY (13.7)

DESCRIPTION:

AS A RESULT OF THE STAFF EVALUATION, CERTAIN PORTIONS OF THE "CLINTON POWER STATION PHYSICAL SECURITY PLAN" WERE IDENTIFIED AS REQUIRING ADDITIONAL INFORMATION OR UPGRADING TO SATISFY THE REQUIREMENT OF 10 CFR 73.55.

THERE ARE THREE SECURITY PLANS:

(1) PHYSICAL SECURITY PLAN
(2) GUARD FORCE TRAINING AND QUALIFICATION PLAN
(3) SAFEGUARDS CONTINGENCY PLAN

STATUS:

THE STAFF HAS INDICATED IN A LETTER DATED JANUARY 8, 1982 THE ACCEPTANCE OF THE CONTINGENCY PLAN AND TRAINING AND QUALIFICATION PLAN.

IP SUBMITTED REVISION 2 OF PHYSICAL SECURITY PLAN AND SAFEGUARD CONTINGENCY PLAN WITH LETTER DATED NOVEMBER 12, 1981.

NRC FORWARDED QUESTIONS/COMMENTS TO OP WITH LETTER DATED FEBRUARY 10, 1982.

JWilliams
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20. QUALITY ASSURANCE (17)

DESCRIPTION:

THE STAFF REQUIRES ADDITIONAL INFORMATION AND CLARIFICATION IN REGARD TO THE OVERALL AUTHORITY AND RESPONSIBILITY OF THE QUALITY ASSURANCE ORGANIZATION AND THE ONSITE COMPLIANCE DEPARTMENT DURING THE OPERATION PHASE AND IN REGARD TO THE LIST OF STRUCTURES, SYSTEMS, AND COMPONENTS THAT FALL UNDER THE CONTROL OF THE QA PROGRAM.

STATUS:

THE APPLICANT HAS BEEN PROVIDED THE LIST OF QUESTIONS AS A RESULT OF THE REVIEW. A MEETING WAS HELD ON JANUARY 29, 1982 DEALING WITH BOTH CONSTRUCTION AND OPERATIONS QA. THE APPLICANT HAS INDICATED CHANGES WILL BE MADE IN THE ORGANIZATION. THE STAFF WILL DISCUSS ITS QUESTIONS AND CONCERNS AFTER THE CHANGES IN THE ORGANIZATION ARE MADE.

JWilliams
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TMI ITEMS STILL UNDER REVIEW

ITEM	SHORTENED TITLE	COMMENTS
I.C.1	SHORT-TERM ACCIDENT & PROCEDURE REVIEW	STAFF HAS INFORMATION FROM OWNERS GROUP UNDER REVIEW AND ANTICIPATES GUIDELINES FOR CLINTON WILL BE ISSUED SOON.
I.C.7	NSSS VENDOR REVIEW OF PROCEDURES	IP HAS COMMITTED TO ENSURING REVIEWS COMPLETED PRIOR TO FUEL LOAD. STAFF WILL CONFIRM.
I.C.8	PITOT MONITORING OF SELECTED EMERGENCY PROCEDURES	IP WILL USE STAFF GUIDELINES FOR LONG TERM UPGRADING OF EMERGENCY OPERATING PROCEDURES (NUREG-0799) RATHER THAN I.C.8
I.D.1	CONTROL-ROOM DESIGN REVIEWS	PRELIMINARY ASSESSMENT COMPLETE. CORRECTIONS BEING MADE.
I.D.2	PLANT-SAFETY PARAMETERS DISPLAY CONSOLE	STAFF REQUIREMENTS UNDER DEVELOPMENT. CLINTON DESIGN SATISFACTORY AT THIS TIME.
I.G.1	TRAINING DURING LOW-POWER TESTING	IP WILL REVIEW RESULTS OF SIMULATED LOSS AT ALL AC POWER TESTS FROM OTHER BWR's AND DETERMINE IF TESTS SHOULD BE DONE AT CLINTON. STAFF WILL DETERMINE IF TESTING NEEDED.

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ITEM	SHORTENED TITLE	COMMENTS
B.2	PLANT SHIELDING	IP TO SUBMIT STUDY RESULTS SHOWING DOSES TO WORKERS IN VITAL AREAS AND PROPOSED MODIFICATIONS.
II.B.3	POST ACCIDENT SAMPLING	IP HAS COMMITTED TO MEET REQUIREMENTS. ALL DETAILS NOT AVAILABLE FOR REVIEW AT THIS TIME.
II.B.7	ANALYSIS OF HYDROGEN CONTROL	
II.B.8	RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS	REVIEW OF HYDROGEN IGINITION SYSTEM UNDERWAY
II.E.4.2	CONTAINMENT ISOLATION DEPENDABILITY	AWAITING INFORMATION FROM IP.
II.F.1 (A-F)	ACCIDENT MONITORING INSTRUMENTATION	IP COMMITTED TO COMPLIANCE. DESIGN DETAILS TO BE DEVELOPED AND REVIEWED.
II.F.2	INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE-COOLING	OWNERS GROUP TO SUBMIT REPORT TO STAFF ON ICC IN JULY 1982. EVALUATION WILL BE APPLIED TO CLINTON.
II.K.3.13	HPCI AND RCIC INITIATION LEVELS	BWR OWNERS GROUP POSITION. AUTO RESTART OF RWCU ON LOW WATER LEVEL IS BEING MADE.

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ITEM	SHORTENED TITLE	COMMENTS
.K.3.15	ISOLATION OF HPCI AND RCIC	CONCEPTUAL DESIGN SATISFACTORY. MOD BY 1st OUTAGE IF NOT BEFORE.
II.K.3.18	ADS ACTUATION	IP EVALUATING 2 OPTIONS, WILL REPORT ON AT LEAST 4 MONTHS PRIOR TO FUEL LOAD
II.K.3.21	RESTART OF LPCS AND LPCI	IP HAS COMMITTED TO AUTO RESTART OF HPCS ON LOW-REACTOR WATER LEVEL. STAFF TO CONFIRM DESIGN ADEQUACY AT LEAST 4 MONTHS BEFORE FUEL LOAD.
II.K.3.27	COMMON REFERENCE LEVEL	IP HAS COMMITTED TO REQUIREMENTS. NEEDS CONFIRMATION
II.K.3.28	QUALIFICATION OF ADS ACCUMUATORS	OWNERS GROUP EVALUATION TO BE SUBMITTED.
II.K.3.31	PLANT SPECIFIC CALCULATION TO SHOW COMPLIANCE WITH 10 CFR 50.46	IP HAS COMMITTED TO PROVIDING ANALYSIS.
III.A.1.2	UPGRADE EMERGENCY SUPPORT FACILITIES	STAFF REVIEW.
III.A.2	EMERGENCY PREPARENESS	STAFF REVIEW
III.D.3.4	CONTROL ROOM HABITABILITY	IP DOES NOT MEET REQUIREMENTS.

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ACRS GENERIC BWR QUESTIONS

JANUARY 7, 1982 GESSAR MEETING

CLINTON STATION

QUESTION: Given a LOCA in which the accompanying effects destroy one train of the water level instrumentation, can the design then accomodate a single active failure?

RESPONSE: The single failure criterion is applied to safety-related systems. Non-safety-related systems are assumed to fail in the manner most adverse to the event. The single failure criterion is not usually applied as described in the question, because the water level instrumentation is safety grade and is, therefore, protected from pipe whip and jet impingement effects. Although the event described involves an additional active failure, we have evaluated the potential consequences for the Clinton design.

Considering the location of the assumed additional active failure, it is only pertinent in this evaluation to consider that the additional failure occurs in the remaining undamaged signal initiation and control systems for the RPS and ESF systems. Assuming the additional failure occurs elsewhere, such as the operational capability of these systems, results in less severe situations that have been analyzed in the past. For example, if we were to assume the most severe single active failures reported in the Clinton FSAR (failure of the diesel generator supplying the LPCI or failure of the HPCS), the initiation systems would respond normally to low vessel water level conditions (even with the assumed failure of a division of vessel water level function), and the overall plant LOCA response would be the same as that presented in the FSAR.

Reactor vessel water level signals are initiated from level (differential pressure) transmitters of the cold reference leg design. There are essentially fourteen level transmitters of interest to this evaluation, mounted on eight separate sensing lines. The sensing lines terminate outside the drywell and inside containment and tap off the reactor vessel at widely separated locations (140 degrees apart). Physical and electrical separation is provided by arranging the sensors in four independent instrumentation divisions, each electrical instrumentation division having a separate independent power supply with the exception of a loss of offsite power condition when the three diesel generators are shared by the four instrument divisions. Mechanical Division 1 corresponds to electrical Division 1, etc. Diversity of RPS and ESF initiation is provided by reactor vessel low water level signals and high drywell pressure signals. The RPS and ESF functions are also arranged into four separate divisions, A through D. From the logic information, it can be shown that, in general, two independent signals are required to initiate an RPS or ESF function. Scram is initiated on any two out of four level 3 low water level signals. Each of the four sensors is associated with a separate division with the result that failing any division in combination with a failure of any other sensor/signal, or even another entire division, in this case, will still allow automatic scram initiation.

In summary, the following combinations of safety systems having the most impact on plant response to LOCA would still be initiated automatically (In each case, the worst single failure is assumed in combination with a failure of one entire instrumentation division).:

1. Failure of Division 1 - still get reactor scram, HPCS, and RHR (LPCI).
2. Failure of Division 2 - still get reactor scram, HPCS, LPCS and LPCI (if the LOCA occurs inside containment resulting in a high drywell pressure signal). If the LOCA occurs outside of containment and the additional failure is one of the two vessel water levels signals in Division one, reactor operator action would be required to manually initiate ADS (if required to depressurize) and LPCI. If operator action is necessary, there appears to be adequate plant display instrumentation and emergency operating procedures to ensure timely initiation measures.
3. Failure of Division 3 - still get reactor scram, RCIC, HPCS (if LOCA is inside containment), ADS, LPCS and LPCA.
4. Failure of Division 4 - same as failure of Division 3.

From the above, we conclude that the Clinton design can accomodate an additional single failure in a sensor/signal with the condition that some reasonable operator action may be required for certain failure combinations as described above. The failure of Division 2 is the only case for which operator action is required. For all of the other cases there would be sufficient automatic response to mitigate the consequences of the event. In addition to accomplishing the successful initiation of the RPS and ESF systems, sufficient level instrumentation would be available to the operators to confirm core coverage and aid in bringing the plant to a cold shutdown. This would not necessarily be true if the additional single failure is assumed to fail the power supply to a second instrumentation division, thus resulting in the loss of two entire divisions.

QUESTION: Why is the scram discharge volume secured prior to full insertion of the control rods?

RESPONSE: The scram discharge volume vent and drain line valves are closed at the time of scram to (1) limit uncontrolled loss of reactor coolant and (2) minimize excessive hydro dynamic forces.

The scram discharge volume vent and drain lines are dedicated lines that discharge into the Radwaste System. The closure of the redundant vent and drain valves in these lines prior to a full scram assures that no single active failure can result in an uncontrolled loss of reactor coolant. On a scram, the scram inlet and outlet valves open providing a direct path of reactor coolant into the scram discharge volume. Failure to close the vent and drain valves would provide a direct path for the reactor coolant to travel to outside primary containment (Radwaste System). Closing of the vent and drain valves after verifying that a full scram has been achieved, would increase the amount of reactor coolant leakage. Additionally to assure that sufficient discharge volume is available for scram, diverse and redundant level sensing instrumentation on the Scram Discharge Instrument Volume is provided for automatic scram.

Damaged floats have been observed when there has been a delay in closure of the vent or drain valves. Two possible hydro-dynamic forces may have producted the observed damage.

a. Large flow rates through the float chamber that drive the float to the limits imposed by mechanical stops on the float stem, and can cause local deformation of the float wall; and
b. Water Hammer; large flow rates can be expected if the

lines from the float chamber are connected to points with large differences in static pressure. High velocities through the relatively small vent and drain lines compared to the velocity in the instrument volume on scram or reset, could result in such a condition. This condition is only possible when the system is essentially water solid which is normally the case on reset but not on scram. Normally on scram there is sufficient air in the ADV on closure of the vent and drain valves, such that the dynamic head in the vent or drain lines does not result in large static pressure differences, and the process is a simple, constant volume compression. However, given a delayed closure of a valve or a system that is already partially full, a water-solid system could result and, in the manner described above, cause damagingly high flow rates through the float chamber.

Water hammer could result on closure of the valves when water is passing through the vent or drain lines at high flow rates or possibly from steam bubble collapse. For the classical water hammer on valve closure, the causes of passing water at high flow rates through the vent and drain lines are similar to those discussed for large flows through the float chamber. For the steam bubble collapse, a delayed closure of a valve could allow hot water into the SDV which could flash, form steam and cause two-phase flow through the system. On closure of the valves, system pressurization could cause a collapse of the bubble and possibly water hammer. At one plant, damage was not limited to the floats, since the vent line was found to be pulled from its restraints. This emphasizes the importance of timely closing of the vent and drain before the ADV becomes water-solid and high velocity flow results.

QUESTION: What are the preservice and inservice inspection requirements for the CRD penetrations in the lower reactor vessel heat?

RESPONSE: Code Requirement (ASME Section XI, 1977 Edition):

Peripheral Control Rod Drive (CRD) housings welds are required to be surface examined (Dye Penetrant) once as a Preservice. Welds located in 10% of the peripheral housings require surface examination during each tem (10) year inservice inspection interval in accordance with ASME Section XI, IWB-2500-1, examination category B-O.

However, the weld areas are not accessible for surface examination without extentive dissassembly of the CRD housings, which would result in significant radiation exposure, possible contamination of personnel or equipment, and possible damage to the control rod drive mechanism. Therefore, the staff has concluded that the inspection requirement may be waived for the following reasons:

1. Welds are subject to hydrostatic testing upon completion of each outage.
2. If a weld should fail while in operation, the CRD housing supports would prevent ejection of the housing and the maximum leakage rate, by calculation, would be below the normal makeup capability.
3. Leakage detection is provided by the Leakage Detection System, with continuous monitoring in the Control Room.

QUESTION: What would be the consequences of using an alternate decay heat removal scheme, assuming a loss os pool cooling, by venting the containment?

RESPONSE: The staff has estimated that containment venting would be required about two or three hours following shutdown with a coincident loss of pool cooling. There is still a question regarding the capability of the containment structure to withstand the SRV discharge condensation loads when the suppression pool is at saturation temperature. Nevertheless, at the time of containment venting virtually all of the primary coolant activity would be steaming at a rate of approximately 35 to 50 lb/sec from the surface of the pool.

For the range of exlusion area boundary X/Q's and other factors, thyroid dose rates of the order of a few to 10 millirem per hour, principally due to ^{131}I would be calculated.

Although coolant iodine is predominantly long-lived ^{131}I, due to decay of shorter-lived iodines during diffusive escape from the fuel, of the order of a few tens of curies of ^{133}I and ^{135}I could also be present. These isotopes yield radioactive xenon daughters, which would be released to the environment during venting. Assuming ten curies of each iodine in these mass chains, about 1.7 Ci of ^{135}Xe and 0.15 Ci of ^{133}Xe would be in the primary containment at the initiation of venting, and would result in whole body doses of about 0.1 millrem at the exclusion area boundary.

The above estimates assume primary coolant iodine activity is at the standard technical specification limit of 0.2 µCi/gram.

QUESTION: What improvements have been made to the Clinton plant with regard to the "Levy List" for Allens Creek?

RESPONSE: The study by S. Levy, Inc. for Allens Creek (Rev. 1 dated February 1981) was intended to investigate potential design changes to prevent or mitigate degraded core accidents. These concepts involved major design changes and did not include a cost-benefit assessment for plants already constructed and, therefore, have not been incorporated into the Clinton design. However, certain aspects of the hydrogen control recommendations may eventually be incorporated into the Clinton design (45 FR 65466 proposed rule).

CLINTON ACRS QUESTION/RESPONSE

In the staff's SER for Clinton, the following statements appear on page 5-8: "Based on the above dicussion, the staff conclude that no changes to the current mode of depressurization are necessary for Clinton at this time. The staff notes, however, that the applicant is subject to the results of his generic review of this item."

In contrast to the above, on page 7-19 of the SER it is stated that in compliance with Item II.K.3.18 of NUREG-0737, the ADS actuation logic will be modified to ensure adequate core cooling. It is also stated that the applicant has committed to implement one of two alternatives both of which have been found to be acceptable to the staff.

The reason for the above inconsistency is that the SER section appearing on page 5-8 was written prior to receiving the information described on page 7-19 reflecting the status of the BWR owners group generic review current at the time of SER publication. This inconsistency was overlooked in the final SER editing. Furthermore, since the issuance of the SER, the staff has received additional analyses from the owners group indicating that the proposed ADS modifications may impede operator actions under consideration for ATWS conditions. The owners group has requested that an extension to the schedule for NUREG-0737 Item II.K.3.18 be granted to allow them to complete their studies and to propose alternate plant modifications. The NUREG-0737 schedule requires that proposed modifications to ADS actuation be submitted for staff review four months prior to expected issuance of an operating license, and the applicant had previously committed to that schedule prior to the recent information regarding the impact of ATWS. NURG-0737 further requires that the plant modifications to ADS actuation be implemented at the first refueling 6 months after staff approval. The staff agrees that an extension should be granted to the NUREG-0737 schedule to allow the applicant and the staff to resolve this issue. We are continuing our discussions with the applicant and will provide an appropriate update on this subject in the Clinton SSER.

CLINTON - ACRS QUESTIONS

SER: PAGE 9-18, II.B.3.(6) "THE APPLICANT STATES THAT THE POST ACCIDENT SAMPLING SYSTEM TOGETHER WITH THE SAMPLING AND ANALYSIS PROCEDURES ARE DESIGNED TO LIMIT THE RADIATION EXPOSURE TO THE OPERATING PERSONNEL BELOW THE LEVELS SPECIFIED IN GDC-19. HOWEVER, PERSON-MOTION STUDIES SHOULD BE PERFORMED TO DEMONSTRATE COMPLIANCE WITH GDC-19 FOR SAMPLING, TRANSPORT, AND ANALYSIS OF LIQUID AND GASEOUS SAMPLES, ITEM ITEM REMAINS OPEN."

QUESTION: EXPLAIN RELATION TO GDC-19 FOR MOTION STUDY.

GDC-19 LIMITS RADIATION EXPOSURES TO ANY INDIVIDUAL TO 5 REM WHOLE BODY FOR SAMPLING, TRANSPORT AND ANALYSIS OF LIQUID AND GASEOUS SAMPLES (NUREG-0737). THE APPLICANT STATES THAT POST ACCIDENT SAMPLING MEETS THE REQUIREMENTS OF GDC-19. HOWEVER, THE STAFF HAS ASKED FOR A PERSON-MOTION STUDY DURING A POST ACCIDENT SAMPLING EXERCISE WITH ESTIMATED POST LOCA RADIATION LEVELS TO VERYIFY COMPLIANCE WITH GDC-19.

JWILLIAMS
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CLINTON - ACRS QUESTION

SER Page 9-19: II.B.3 (11) "Sufficient shielding is provided to make it possible for an operator to obtain and analyze a sample with radiation exposures meeting the requirements of GDC 19, assuming source terms of Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological consequence of a loss of Coolant Accident for Pressurized Water Reactors."

QUESTION: Explain why R. G. 1.4 was used for shielding.

RESPONSE: Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant accident for Boiling Water Reactors," should be used for the shielding calculations.

CLINTON - ACRS

II.E.4.2 Containment Isolation Dependability

QUESTION: Are the valves qualified to close under adverse conditions?

RESPONSE: Valves (36", 24", 10" Butterfly valves) are manufactured by Posi-Seal.

Conversations were held with the utility on 10-29-81 an on 11-30-81. During those discussions the capability of the valve to close under high pressure loads from a LOCA was discussed. The utility indicated they believed the valves were capable of closing under these loads but were not able to supply qualification documentation in the form of test reports/analysis at that time. During the second conversation the utility indicated they were in the process of attaining qualification documentation. That information will be reviewed when submitted.

QUESTION: How are the low voltage CMDS devices used in the Clinton NSPS protected from voltage spikes, electrical noise, and other types of interferrence?

RESPONSE: All inputs and outputs to the NSPS cabinets are buffered and isolated and internal wiring is routed to prevent "cross talk" or radiated electromagnetic interferrence (EMI). For power lines to the NSPS (12 VDC), EMI is prevented by the use of switching power supplies (max noise spike 62 mV). It would take an input of greater than 4V to switch the output of a CMOS device on a low to high transition and less than 8V to switch on a high to low transition. As a comparison, TTL logic which is operated at 5V has a low to high minimum threshold of approximately .7 volt. Inputs to the analog trip modules are current loops an therefore, not vulnerable to EMI. In addition, the power input to each logic card is filtered. The staff looked at the susseptibility of the NSPS circuitry to electrical noise an voltage transients as part of the review process. The plant computers (part of the non-safety PMS/DCS) are powered from the plant UPS and charger and are battey backed. All safety related inputs are isolated.

QUESTION: Explain the staff's conclusion regarding the Clinton design of the Safety Parameter Display System (SPDS).

RESPONSE: NUREG-0696, "Functional Criteria for Emergency Response Facilities," provides general guidelines for the SPDS design. The Committee for the Review of Generic Requirements (CRGR) is currently working with the staff in the development of acceptance criteria to implement the NUREG-0696 guidelines. We expect these criteria will be complete in a couple of months.

QUESTION: Verify the containment volume.

RESPONSE: The containment free volume of 1.46×10^6 ft^3 listed in the Clinton FSAR is based on the preliminary design. All of the FSAR has not yet been updated to reflect the most recent containment volume of 1.55×10^6 ft^3 which is based on the most recent design information.

QUESTION: Why does the SER description of the bases for the plant Technical Specifications refer only to 10 CFR 20 and not 10 CFR 50, Appendix I?

RESPONSE: The SER describes a generalized statement of intent. In fact, the Technical Specifications that will be developed for the Clinton plant will be based on maintainin normal operation exposures within the limits of 10 CFR 20 and the numerical guidelines of 10 CFR 50, Appendix I, and accident exposures within the guidelines of 10 CFR 100.

Human Engineering Review of Clinton Power Station Remote Shutdown Panel

QUESTION: Was a review performed?

RESPONSE: Yes, the applicant's consultant performed a preliminary design assessment (PDA) which was submitted for NRC review October 5, 1981. The PDA reported 16 human engineering discrepancies (HED's) associated with the remote shutdown panel. Subsequently, the human factors engineering branch (HFEB) audited the PDA and performed its own human engineering review. Ten additional HED's were recorded during the HFEB review. Some human engineering aspects of he remote shutdown panel have not been reviewed due to the plart's construction status. Those aspects were communications, normal and emergency lighting, and the accessibility, organization, and storage of procedures and other documents.

QUESTION: What were the findings, plans, and schedule for any correction?

RESPONSE: The findings of the PDA and HFEB review can be found in the attached document. The attached document also contains the applicant's response to each HED. Twenty two of the HED's will be evaluated as part of the applicants Detailed Control Room Design Review (per NUREG-0700).

Clinton Nuclear Power Plant
Unit No. 1

Remote Shutdown Panel
Human Engineering Discrepancies

A-193

Preliminary Design Analysis
Human Engineering Discrepancies
on the
Remote Shutdown Panel

A-1. CONTROL ROOM WORKSPACE

FINDING

2.* Access to the Remote Shutdown Panel is not currently limited. (10.1)

Response
Control card access should be implemented.

4.* Meters on C61-P001 are too high to be read by the average operator. They are about 6' from the floor. (3.46)

Response
Investigate impact on the operation of the panel and consider alternatives.

5.* Some labels on C61-P001 are too high or too low to be read. (3.49)

Response
Investigate criticality when the associated controls/displays are reviewed for their placement.

A-4. CONTROLS

FINDING

1.* The use of the RHR B Shutdown Cooling Suction Valve 1E11-F006B was not apparent to operators. Remote Shutdown Panel. (4.24)

Response
Determine if this switch is extraneous, inform or train operators accordingly, and eventually remove if determined extraneous.

2.* There is little use of knob-coding on the remote shutdown panel. (4.25)

Response
Shape code throttleable controls and others per Appendix A.

A-5. DISPLAYS

FINDING

9.* Suppression pool level meter on C61-P001 has a scale ranging from -24 inches to +6 inches. This scale is not consistent with similar control room indication, and reference point is unclear. (3.52)

Response
Modify scale accordingly to conform to appropriate control room indication.

A-6. LABELS AND LOCATION AIDS

FINDING

2.* Review of a new label list for Remote Shutdown found significant improvements.(5.15)

Response
Implement these changes.

28.* Labels placed below recorders on C61-P001 are sometimes eclipsed by the recorder itself. (5.17)

Response
Labels should be placed on the doors of recorders to avoid this.

31.* Division numbers on labels for C61-P001 are very small and difficult to read. (5.16)

Response
If this information is required on labels, some alternate form of coding might be used, for example colored symbols.

34.* Some abbreviations used on labels on C61-P001 are inconsistent. For example, shutdown service is abbreviated SX and SSW on remote shutdown and control room panels. (5.18)

Response
Review labels and implement consistent abbreviations.

36.* Labels on C61-P001 are black on yellow and white on blue. (3.48)

Response
Relabel using black lettering on white labels.

FINDING

37.* Labels on the remote shutdown panel are black on yellow and other colors. Labels are also temporary. (5.14)

Response
When this panel is re-labeled, it is recommended that black letters on a white background be used.

38.* Font width on labels varies significantly from label to label on C61-P001. Some are very thick and cannot be read easily. (3.50)

Response
Recut the illegible labels.

A-8. PANEL LAYOUT

FINDING

2.* Indication of shutdown cooling, such as reactor water temperature and drywell pressure, should be considered for inclusion on the remote shutdown panel. (4.23)

Response
Evaluate the need to include additional instrumentation for effective operations.

9.* Division 1 Cross-Tie Valve switch on C61-P001 has indicator lights stacked above it. Not all lights are associated with the switch. (4.26)

Response
Move unrelated indicator lights or separate them in some fashion.

15.* There is a group of 8 meters on C61-P001 grouped together. (3.47)

Response
Accentuation though color padding, group labeling, or demarcation lines should be employed to divide large groupings into functional sub-groupings of 5 or less meters each.

Control Room Design Review/Audit Team
Human Engineering Discrepancies
on the
Remote Shutdown Panel

C-1. CONTROL ROOM WORKSPACE

FINDING

1. There is inadequate separation between the front surface of the Remote Shutdown Panel and the backpanel surface of the adjacent electrical panel. The distance between the opposing surfaces is approximately 32 inches. This is an inadequate amount of workspace for easy operation of the Remote Shutdown Panel by one or more operators. The minimum recommended separation distance between a single row equipment panel and a wall or other opposing surface is 50 inches.

   Response

   Evaluate the Remote Shutdown Panel placement readability, availability, and use time. Evaluate and assess the need for and the space allowed for protective equipment. As a result of this evaluation personnel assignments should be made so that the entire panel can be maintained under the surveillance of qualified personnel and the need for "cross-over" of one person in front of another is minimized. (1)

2. The J-handle Transfer Switches on the Remote Shutdown Panel are mounted too low. They are located between approximately 20" and 33" above the floor. This low control height makes them inconvenient to operate.

   Response

   See item C-1, 1. (1)

4. There are no provisions for storage of procedures at the Remote Shutdown Panel.

   Response

   Provide necessary procedures in an accessible locations. (1)

C-3. ANNUNCIATORS

FINDING

17. There are no annunicators at the Remote Shutdown Panel for conditions that are annunciated in the Control Room.
Examples:
a.)Suppression Pool Temperature
b.)Reactor Vessel Water Level

Response

See C-1.1 (1)

C-4. CONTROLS

FINDING

4. Transfer switch J-handles on the Remote Shutdown Panel are vulnerable to inadvertent activation. The J-handles project approximately 4 inches into a workspace that is only 32 inches deep and they are located approximately at knee level.

Response

See C-1.1 (1)

C-5. DISPLAYS

FINDING

15. A vertical series of 8 pairs of red and green inidicator lights on the Remote Shutdown Panel alternate colors left-right down the two columns. In this arrangement, color and position do not provide any type of redundant information and do not conform to the Close/Green/Left - Open/Red/Right convention used in the control room.

Response

Adopt the convention in the main control room. (1)

18. Some of the yellow alarm indicator lights on the Remote Shutdown Panel do not conform to color code convention.
Examples:
a.)RCIC TURBINE TRIP indicator light
b.)RCIC TURBINE OIL TEMP HIGH indicator light
c.)RCIC TURBINE BEARING OIL LOW PRESS indicator light

Response

See C-1.1. Adopt the color convention used in the control room. (1)

C-6. LABELS AND LOCATION AIDS

3. Some permanent labels for the Transfer switches on the Remote Shutdown Panel do not have switch numbers on them. Some switches are labeled with numbers using temporary embossed tape labels. Those labels with switch numbers appear to have numbers which are different from the switch numbers cited in the procedures.

Response

See C-3.8. (1)

16. There is temporary mimic on the Remote Shutdown Panel.

Response

Temporary mimics will be made permanent. (1)

C-9. CONTROL/DISPLAY INTEGRATION

3. The meters in the BOP section at the right end of the Remote Shutdown Panel cannot be read from the locations of associated NSS Division 1 controls in the middle and left sections of the panel.

Response

See C-1.1. (1)

C-10. FEATURES THAT COULD NOT BE EVALUATED

2. Communications facilities in the Control Room and at the Remote Shutdown Panel could not be evaluated.

4. Control Room and Remote Shutdown Panel normal and emergency lighting could not be evaluated.

6. Document organization and storage provisions in the Control Room and at the Remote Shutdown Panel could not be evaluated.

7. Accessibility and use of procedures and other reference documents in the Control Room and at the Remote Shutdown Panel could not be evaluated.

Organization and Management

L. J. Koch

March 5, 1982

2-22-82

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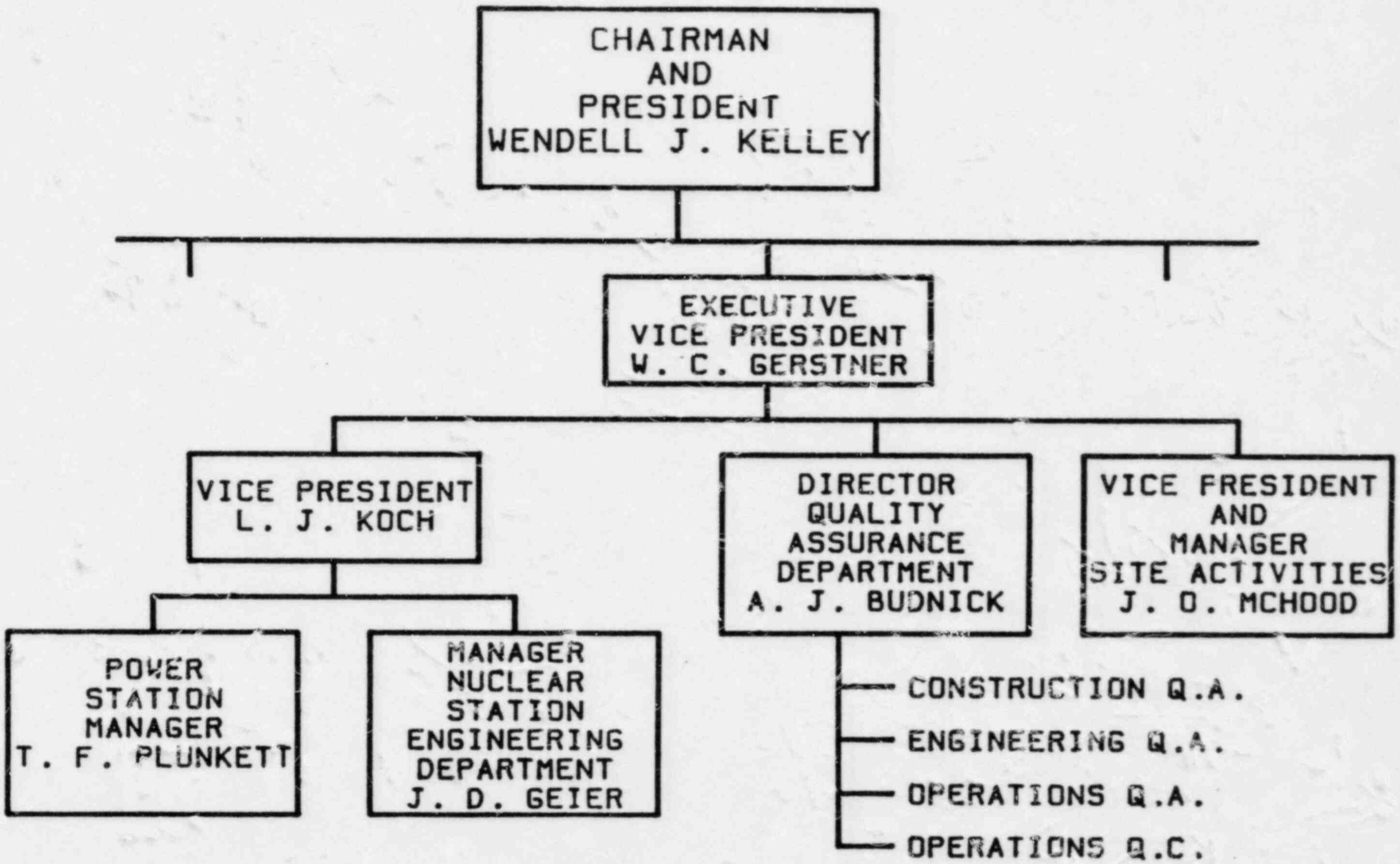
IP (PARTIAL) CORPORATE STRUCTURE

RELATIVE TO CLINTON POWER STATION



TD12

IP STAFF PARTICIPATION - C.P.S.

A-204

	PROFESSIONAL STAFF	TOTAL PERSONNEL
SITE ACTIVITIES	80	105
NSED	60	73
QUALITY ASSURANCE	23	25
OPERATIONS	91	204
TOTAL DIRECT IP	254	407

FULL TIME IP EMPLOYEES - DOES NOT INCLUDE SUPPORTING SERVICES.

ACRS Meeting March 5, 1982
Illinois Power Company Participants

JULIUS D. GEIER
Manager of Nuclear Station Engineering
BS Mechanical Engineering
10 years on CPS project
30 years total experience (20 years w/Argonne & Savannah River)

THOMAS F. PLUNKETT
Manager Clinton Power Station
BS Mechanical Engineering, MS Nuclear Engineering
5 years on CPS project
20 years total experience (7 years McDonnell Douglas - Nuclear space power; 8 years - Donald C. Cook Plant)

CHARLES C. WHEELER
Supervisor - Mechanical Engineering
BS Chemical Engineering
6 years on CPS project
30 years total experience (21 years w/General Atomic, TVA, GE, US Army)

ERICH W. KANT
Director - Nuclear Safety and Engineering Analysis
BS Electrical Engineering & MS Industrial Engineering
1 year on CPS project
10 years total experience (9 years at Donald C. Cook Plant)

ALAN L. RUWE
Supervisor - Electrical & Startup Engineering
BS Electrical Engineering & MS Nuclear Engineering
8 years on CPS project
12 years total experience (all IP)

LARRY S. BRODSKY
Assistant Manager Clinton Power Station
BS Chemistry
7 years on CPS project
13 years total experience (6 years Navy Nuclear Program)

JOHN P. O'BRIEN
Supervisor - Instrumentation & Controls Engineering
BS Electrical Engineering & MS Nuclear Engineering
6 years on CPS project
22 years total experience (all IP)

JOHN G. COOK
Technical Supervisor - Clinton Power Station
BS Engineering Physics, MS Nuclear Engineering, MBA
7 years on CPS project
12 years total experience (5 years Nuclear Navy Program)

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Organization & Experience

J. D. Geier

March 5, 1982

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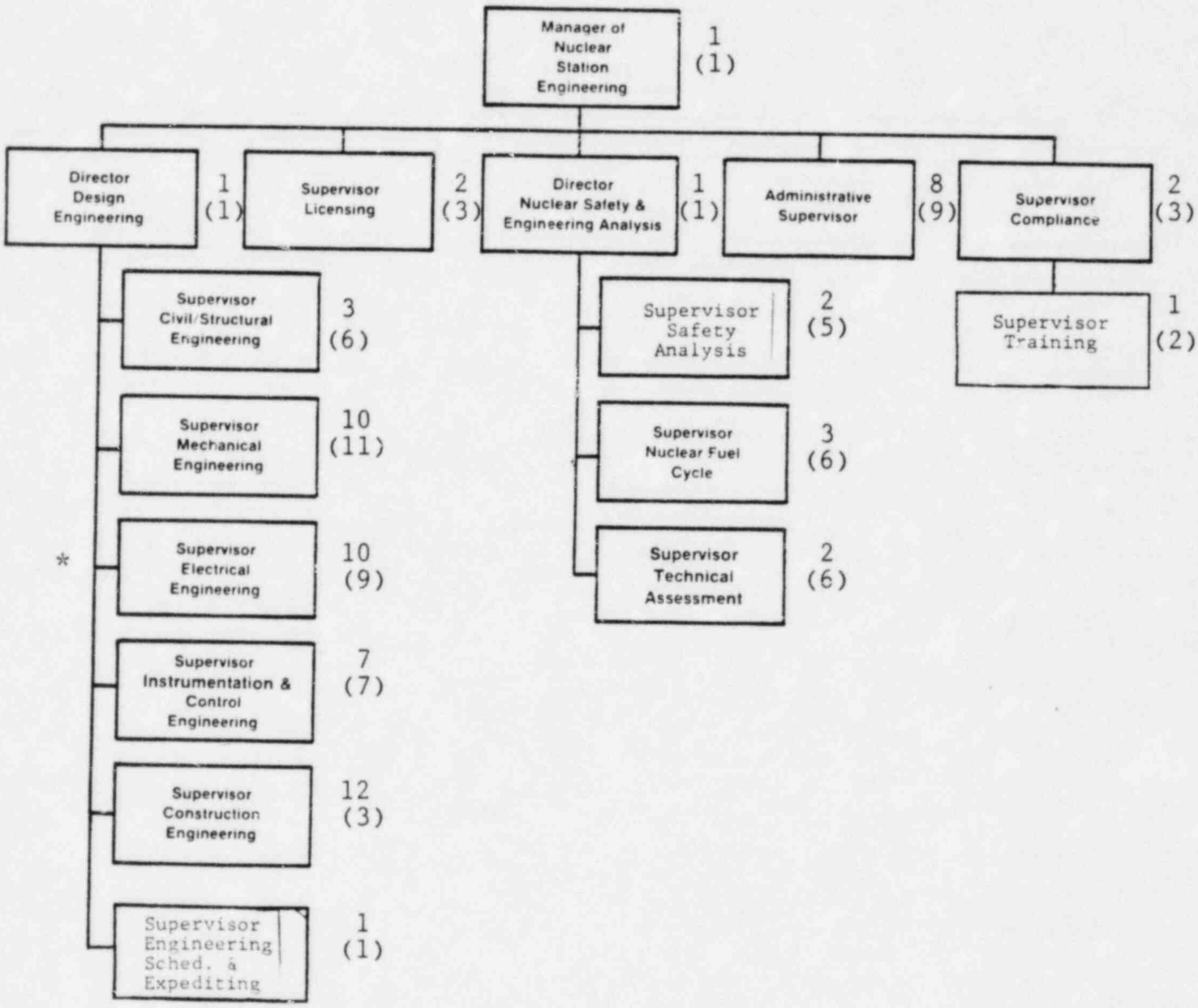

* Includes NSED Startup Coordination

CLINTON POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE 13.1-2

IP NUCLEAR STATION
ENGINEERING DEPT. STRUCTURE
RELATIVE TO CLINTON POWER STATION

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Experience Summary

J. D. Geier

March 5, 1982

A-208

February 15, 1982

NUCLEAR STATION ENGINEERING
Technical Personnel Experience Summary

	Man Years of Experience: Present	Man Years of Experience: Expected at Commercial Operation (1)
I. Department Staff		
CPS Experience	77.5	124.0
Other Nuclear Experience	21.0	21.0
Other Engineering Experience	36.5	36.5
TOTAL	135.0	181.5
II. Department Mgt. & Supv.		
CPS Experience	98.0	119.0
Other Nuclear Experience	107.5	107.5
Other Engineering Experience	64.0	64.0
TOTAL	269.5	290.5
III. Total Department		
CPS Experience	175.5	243.0
Other Nuclear Experience	128.5	128.5
Other Engineering Experience	100.5	100.5
TOTAL	404.5	472.0

(1) About September 1, 1983

February 15, 1982

NSED STAFF EXPERIENCE

Position	Degree	Engineering Experience (Years) CPS	Other Nuclear	Other	Total
Supervising Engineer	BSEE, 1973	3½	-	5	8½
Supervising Engineer	BSEE, 1969	5½	-	7	12½
Supervising Engineer	BS Math & Physics, 1966	5½	10	-	15½
Construction Engineer	BSEE, 1967	1½	-	13	14½
Project Engineer	BSNE, 1975	4½	1	-	5½
Project Engineer	BSIE, 1967	8½	6	-	14½
Project Engineer	BSEE, 1973	4½	-	5	9½
Staff Engineer	BSEE, 1977	4½	2	-	6½
Staff Engineer	BSEE, 1975	4½	-	2	6½
Staff Engineer	BSMET,1978	3½	-	-	3½
Staff Engineer	BSEE, 1978	3½	-	-	3½
Staff Engineer	BSME, 1977	4½	-	-	4½
Staff Engineer	BSEE, 1976	1½	-	4	5½
Staff Engineer	BSME, 1977	4½	-	-	4½
Staff Specialist	BS ,1977	-	2	-	2
Engineer	BSME, 1981	1	-	-	1

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NSED STAFF EXPERIENCE

Position	Degree	Engineering Experience (Years) CPS	Other Nuclear	Other	Total
Engineer	BSME, 1981	1	-	-	1
Engineer	BSME, 1981	1	-	-	1
Engineer	BSME, 1977 MSNE, 1980	1	-	-	1
Engineer	BSEE, 1981	-	-	-	-
Engineer	BSEE, 1981	-	-	½	½
Engineer	BSME, 1981	1	-	-	1
Engineer	BSNE, 1977 MS, 1978 MBA, 1981	1½	-	-	1½
Engineer	BSME, 1980	1½	-	-	1½
Engineer	BSEE, 1979	2½	-	-	2½
Engineer	BSME, 1981	1	-	-	1
Engineer	BSEE, 1979	1½	-	-	1½
Engineer	BSCE, 1979	2½	-	-	2½
Engineer	BSEE, 1981	1	-	-	1
Engineer	AAS BSEE, 1981	1	-	-	1
Engineer	BSNE, 1981	-	-	-	-

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NSED MANAGEMENT EXPERIENCE

February 15 , 1982

			Engineering Experience (Years)			
	Degree	Registration	CPS	Other Nuclear	Other	Total
Manager, NSED	BSME, 1952	-	10.0	20.0	-	30.0
Director-Design Engineering	BSEE, 1964 MSNE, 1977	IL	10.0	-	8.0	18.0
Dir-Nuclear Safety & Eng. Analysis	BSEE, 1971 MSIE, 1980	MI, WI	1.5	8.5	-	10.0
Supervisor-Nuclear Fuel Cycle	BSME, 1960	IA	8.5	14.0	-	22.5
Supervisor-Compliance	BSPhys, 1965 BSEE, 1977	-	5.0	-	15.0	20.0
Supervisor-Instrumen. & Controls Eng.	BSEE, 1960 MSNE, 1977	-	5.0	-	16.0	21.0
Supervisor-Civil/Structural Engineering	BSCE, 1949	IL, IN	7.0	15.0	10.0	32.0
Supervisor-Elec. & Startup Engineering	BSEE, 1970 MSNE, 1977	IL	8.0	-	4.0	12.0
Supervisor-Safety Analysis	BSME, 1958	-	5.0	8.0	11.0	24.0
Supervisor-Mechanical Engineering	BSChE, 1952	CA	6.0	15.0	-	21.0
Supervisor-Licensing	BSCh, 1952	-	10.0	20.5	-	30.5
Supervisor-Technical Assessment	BSPhys, 1970 MSNE, 1972	-	7.5	3.0	-	10.5
Supervisor-Construction Engineering	BSME, 1971	-	5.5	3.5	-	9.0
Supervisor-Training	BSPhys, 1971 MSNE, 1974	-	9.0	-	-	9.0

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PLANT STAFF

1. CPS STAFF ORGANIZATION
2. STAFFING SUMMARY
3. PLANT STAFF EXPERIENCE SUMMARY
4. PROCEDURE DEVELOPMENT
5. PLANT STAFF PROJECT INVOLVEMENT
6. STARTUP FUNCTIONS/ACTIVITIES

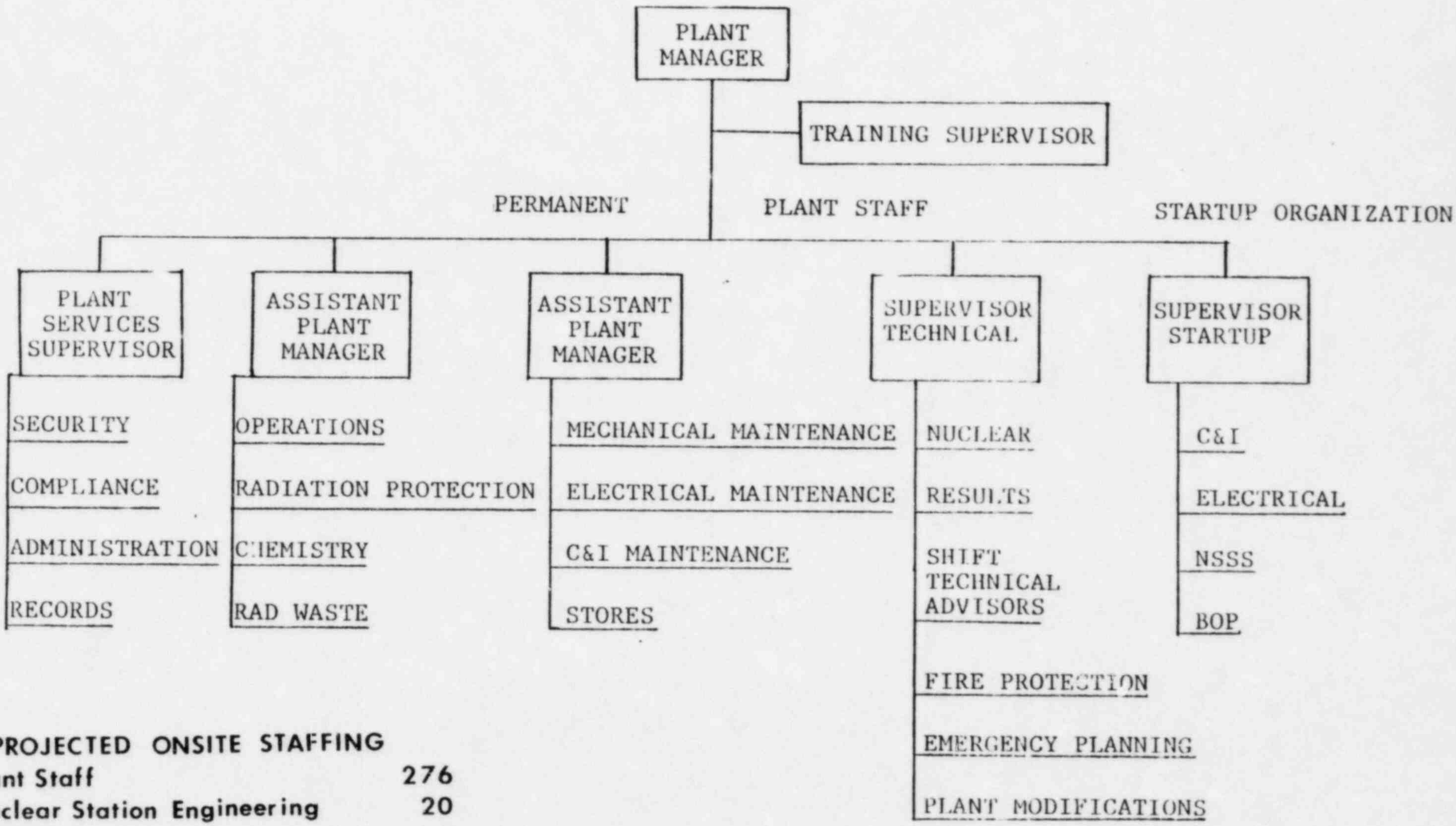
PLANT STAFF

1. CPS STAFF ORGANIZATION
2. STAFFING SUMMARY
3. PLANT STAFF EXPERIENCE SUMMARY
4. PROCEDURE DEVELOPMENT
5. PLANT STAFF PROJECT INVOLVEMENT

STARTUP

6. STARTUP FUNCTIONS/ACTIVITIES
7. STARTUP STAFF EXPERIENCE SUMMARY

CPS STAFF ORGANIZATION



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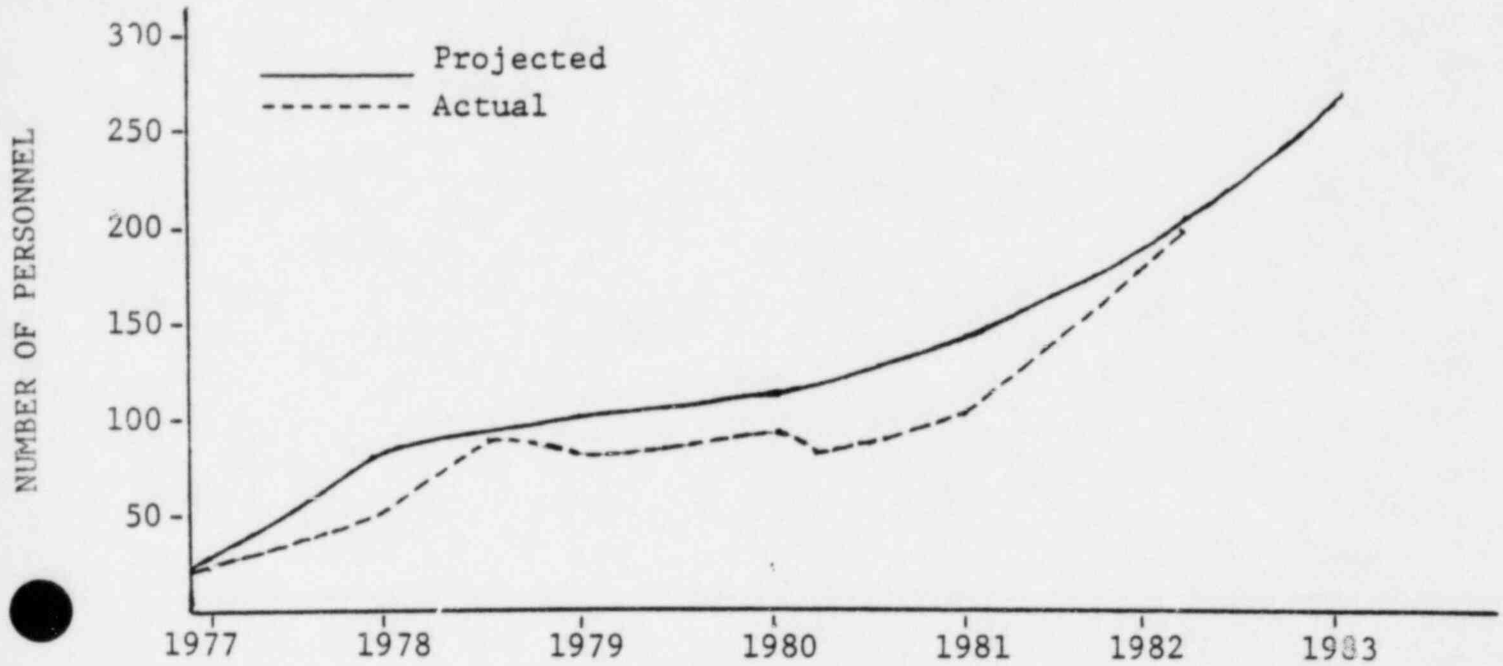
PROJECTED ONSITE STAFFING

Plant Staff	276
Nuclear Station Engineering	20
QA/QC	30
Total	326

STAFFING SUMMARY



	To Date	Projected
MAINTENANCE	63	85
OPERATIONS	49	58
RADCHEM	27	44
TECHNICAL	30	36
OTHER (MANAGEMENT, ADMIN-STRATIVE, TRAINING, COMPLIANCE, SECURITY)	35	53
	204	276

Attrition rate last 12 months 9%
Attrition rate in Operations 2%

PLANT STAFF EXPERIENCE SUMMARY

- PREVIOUSLY LICENSED 10 (6 BWR)
- 458 MAN YEARS COMMERCIAL NUCLEAR EXPERIENCE
- 762 MAN YEARS NAVY NUCLEAR EXPERIENCE
  114/204 HAVE NAVY NUCLEAR BACKGROUNDS
- 12 PERSONNEL WITH BWR PLANT EXPERIENCE
- 34 BWR SIMULATOR CERTIFICATION (ALL SRO CERTIFIED)
- 70 REFUELING OUTAGES
- 40 COLLEGE DEGREES (31 TECHNICAL)
  11 ADVANCED DEGREES
- 18 ASSOCIATE DEGREES (10 TECHNICAL)

PROCEDURE DEVELOPMENT

OBJECTIVE IS TO HAVE PROCEDURES DEVELOPED EARLY BY THE USERS

BENEFITS

* RETAIN EXPERIENCE OF DEVELOPING PROCEDURE IN HOUSE
* ALLOWS CHECKOUT AND USE OF PROCEDURES PRIOR TO CORE LOAD

STATUS

	IDENTIFIED	WRITTEN
PLANT STAFF PROCEDURES	1266	1048
ANNUNCIATOR PROCEDURES	2377	2370
STARTUP PROCEDURES	283	221

PLANT STAFF PROJECT INVOLVEMENT

* ALL PLANT TRAINING
* PREPARATION OF PROCEDURES
* SPARE PARTS SELECTION AND PROCUREMENT
* PREVENTIVE AND CORRECTIVE MAINTENANCE
* PERSONNEL SUPPORT FOR STARTUP TESTING
* REVIEW OF PLANT DESIGN AND CONSTRUCTION
* ISI PROGRAM MANAGEMENT
* PERSONNEL SELECTION AND STAFFING
* PROCUREMENT OF SHOP EQUIPMENT
* IMPLEMENTATION OF ADMINISTRATIVE PROGRAMS
* RECEIPT, STORAGE, AND ISSUE OF PARTS
* SYSTEM DESCRIPTIONS
* RAD. ENVIRONMENTAL MONITORING
* REVIEW SAFETY RELATED PROCUREMENT
* METEOROLOGICAL MONITORING
* VENDOR SURVEILLANCE
* DAM MONITORING & RELIEF WELL INSTALLATION
* FIRE PROTECTION PLAN
* REVIEW THERMAL DISCHARGE CALULATIONS
* TMI RESPONSES
* FIRE PUMP SURVEILLANCE
* CONTROL ROOM LAYOUT
* DEVELOPED CPS UNIQUE DCS DISPLAYS
* EMERGENCY PLAN PREPARATION AND IMPLEMENTATION
* PRELIMINARY CONTROL ROOM SURVE
* ERF SITING AND FACILITIES
* RECORDS SYSTEMS DEVELOPMENT
* EMERGENCY WARNING SYSTEM DESIG
* DEVELOPING TEST COUPON PROGRAM
* ULTIMATE HEAT SINK MONITORING
* BWR TMI OWNER'S GROUP
* RAD PROTECTION REVIEW OF PLANT LAYOUT
* DEVELOPMENT OF PROCESS RAD MONITOR SYSTEM
* DEVELOPMENT OF RADWASTE OPERATIONS CENTER
* DEVELOPMENT OF WATER MANAGEMENT PROGRAM
* COMMUNICATIONS SYSTEM
* STATION KEYING PLAN
* DEVELOPED CPS TECH. SPECS.
* LESSON PLANS FOR BOP SYSTEM AND SYSTEM TRACING GUIDES
* REVIEW OF PIPING SYSTEM ISO DRAWINGS
* CONTROL ROOM CHECKOUT AND TESTING

STARTUP FUNCTIONS/ACTIVITIES

* COMPONENT TESTS
  - INSPECTION AND CHECKOUT OF:

    MOTORS

    PUMPS

    FANS

    BREAKERS

    REMOTE OPERATED VALVES

* CALIBRATION OF INSTRUMENTATION

* PIPE FLUSHING
  - VELOCITY FLUSHING OF PIPING SYSTEMS
  - PLANT STAFF OPERATORS UTILIZED FOR OPERATION OF PLANT EQUIPMENT

* HYDROTEST
  - PRESSURIZATION AND INSPECTION OF PIPING SYSTEM

* PREOPERATIONAL TESTS
  - VERIFIES OPERATION OF ENTIRE SYSTEM
  - VERIFIES OPERATING PROCEDURES WHERE POSSIBLE
  - PLANT STAFF OPERATORS UTILIZED FOR OPERATION OF PLANT EQUIPMENT

* STARTUP TESTS
  - SYSTEM TESTS PERFORMED FROM FUEL LOAD TO COMMERCIAL OPERATIONS

    CORE PERFORMANCE TESTS

    ROD CONTROL TESTS

    TURBINE GENERATOR TESTS

    PIPING THERMAL EXPANSION TESTS

    IN CORE INSTRUMENTATION TEST

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TRAINING

1. TRAINING FOR PERSONNEL WITH
   NO PRIOR COMMERCIAL NUCLEAR EXPERIENCE

2. PHILOSOPHY FOR PLANT STAFF TRAINING

3. TRAINING CONTROLS

4. LICENSED OPERATOR TRAINING
   NON-LICENSED OPERATOR TRAINING

5. BALANCE OF STAFF TRAINING (CURRENT TRAINING)
   BALANCE OF STAFF TRAINING (FUTURE TRAINING)

6. STA TRAINING

7. TRAINING INVOLVEMENT

TRAINING FOR PERSONNEL WITH NO PRIOR COMMERCIAL NUCLEAR EXPERIENCE

* EMPHASIZED HIRING NAVY NUCLEAR & PROVIDING EXPERIENCE

* COMMERCIAL FOSSIL EXPERIENCE
  - ° 36 MAN-YEARS AT IPC FOSSIL UNITS
  - ° 7 MAN-YEARS STARTUP TESTING

* COMMERCIAL NUCLEAR EXPERIENCE
  - ° 14.5 MAN-MONTHS REFUELING OUTAGE EXPERIENCE FOR 7 RADCHEM PERSONNEL
  - ° 3.3 MAN-MONTHS OPERATIONAL EXPERIENCE FOR CHEMISTRY SUPERVISOR
  - ° 3.5 MAN-MONTHS REFUELING OUTAGE EXPERIENCE FOR PLANT OPERATORS
  - ° 31 MAN-MONTHS BWR OBSERVATION TRAINING

* COMMERCIAL NUCLEAR EXPERIENCE - GRADUATE ENGINEERS
  - ° 9 ENGINEERS SRO TRAINED OF WHICH
    8 ENGINEERS SRO CERTIFIED AT SIMULATOR
  - ° 18 ENGINEERS RECEIVED SYSTEM TRAINING
  - ° 3 ENGINEERS SCHEDULED FOR GRAND GULF STARTUP TESTING
  - ° 2 ENGINEERS SCHEDULED FOR LASALLE STARTUP TESTING

PHILOSOPHY FOR PLANT STAFF TRAINING

CORPORATE & PLANT MANAGEMENT INVOLVEMENT

CENTRAL PLANT TRAINING STAFF

- ° LICENSED OPERATOR TRAINING
- ° GENERAL EMPLOYEE TRAINING
- ° FIRE BRIGADE TRAINING
- ° REQUALIFICATION TRAINING
- ° STA TRAINING (PLANT SPECIFIC)
- ° NON LICENSED OPERATOR TRAINING (CLASSROOM PORTION)

DEPARTMENT TRAINING

- ° OPS
- ° RADCHEM
- ° MAINTENANCE
- ° TECHNICAL
- ° PLANT SERVICES

- ° CONTROL LOOP CALIBRATION
- ° RADIOCHEMISTRY ANALYTICAL PROCEDUR
- ° NDE QUALIFICATIONS
- ° RADWASTE OPERATOR TRAINING
- ° EQUIPMENT REPAIR & MAINT. TRAINING
- ° NUCLEAR ENGINEERING/CORE MANAGEMEN
- ° COMPUTER SOFTWARE/HARDWARE TRAININ
- ° ENVIRONMENTAL MONITORING TRAINING
- ° WELDER TRAINING
- ° ISI TRAINING
- ° SECURITY GUARD WEAPONS TRAINING
- ° NON LICENSED OPERATOR - TRAINING (OPERATIONAL PORTION)

TRAINING CONTROLS

* TRAINING SUPERVISOR SOLEY RESPONSIBLE TO PLANT MANAGER FOR QUALITY OF ALL TRAINING
* ALL DEPARTMENT ANNUAL & QUARTERLY TRAINING SCHEDULES APPROVED BY TRAINING SUPERVISOR
* ALL CHANGES TO DEPARTMENT TRAINING MUST BE APPROVED BY TRAINING SUPERVISOR
* TRAINING DEPARTMENT STAFF MUST EVALUATE OTHER DEPARTMENT TRAINING EVOLUTIONS QUARTERLY
* TRAINING SUPERVISOR FILES AN ANNUAL TRAINING STATUS REPORT WITH VICE PRESIDENT
* NRAG OVERVIEW (CORPORATE & CONSULTANTS)
* QUALITY ASSURANCE DEPARTMENT AUDITS TRAINING DEPARTMENT ANNUALLY

LICENSED OPERATOR TRAINING

REACTOR FUNDAMENTALS	5-16 WEEKS
UNIVERSITY OF ILLINOIS REACTOR STARTUPS	1 WEEK
NSSS SYSTEMS	5 WEEKS
BALANCE OF PLANT SYSTEMS	5 WEEKS
TURBINE GENERATOR OPERATIONS	3 WEEKS
PRE SIMULATOR	2 WEEKS
BWR TECHNOLOGY	4 WEEKS
OBSERVATION TRAINING	4 WEEKS
SIMULATOR	9 WEEKS

34/35 COMPLETED TRAINING AND SRO SIMULATOR CERTIFICATION
17 LICENSED OPERATOR CANDIDATES IN TRAINING
11 NON LICENSED OPERATOR CANDIDATES IN TRAINING

NON-LICENSED OPERATOR TRAINING

OPERATOR FUNDAMENTALS	6 WEEKS
NSSS SYSTEMS	4 WEEKS
BALANCE OF PLANT SYSTEMS	4 WEEKS
QUALIFICATION WORK ON SHIFT	6 MONTHS

BALANCE OF STAFF TRAINING
(CURRENT TRAINING)

MAINTENANCE	RADIATION/CHEMISTRY	TECHNICAL
1. Reactor Fundamentals	1. Reactor Fundamentals	1. Reactor Fundamentals
2. BWR Technology	2. BWR Technology	2. BWR Technology
3. Balance of Plant	3. Balance of Plant	3. Balance of Plant
4. BWR Operator Training	4. BWR Operator Training	4. BWR Operator Training
5. Observation Training	5. Observation Training	5. Observation Training
6. Reactor Startup	6. Reactor Startup	6. Reactor Startup
7. License Review	7. License Review	7. License Review
8. Large Steam Turbine Generator	8. Core Analysis	8. Core Analysis
9. Limitorque Valves	9. BWR Chemistry	
10. Bearings	10. Plant Outages	
11. Process Instrumentation	11. Rad Chem Technician - RP Qualification Program	
12. Nuclear Instrumentation	12. Rad Chem Technician - CH Qualification Program	
13. Analog and Digital Electronics Correspondence	13. Radwaste Operator Qualification Program	
14. Alterrex	14. Nuclear Chemist Qualification Program	
15. Rod Control and Information		
16. Mark II EHC Introduction		
17. Honeywell 4500 Maintenance		
18. Honeywell USER Programming		

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BALANCE OF STAFF TRAINING
(FUTURE TRAINING)

MAINTENANCE	RADIATION/CHEMISTRY	TECHNICAL
1. Control Rod Drive Maintenance	1. Eberline Instrument Operation and Maintenance	1. Station Nuclear Engineering
2. Main Steam Isolation Valve Maintenance	2. Short Course on Radiation Protection	2. Station Nuclear Engineering Refresher
3. BWR Refueling Floor Activity		3. PCIOMR Orientation
4. Mark II EHC Technician		4. Core Management Engineering
		5. Eberline Instrument Operation and Maintenance

3/82

STA TRAINING

* GOAL
  - TRAIN DEGREED ENGINEERS FOR SHORT TERM ROLE AS STA'S
    + ACADEMIC COURSES - SELECTED TOPICS
    + PLANT SPECIFIC TOPICS
    + SUPERVISOR DEVELOPMENT TRAINING
  - UPGRADE OPERATIONS SHIFT SUPERVISION TO ELIMINATE THE NEED FOR STA'S
    + ACADEMIC COURSES (49 SEMESTER HOURS)
    + PLANT SPECIFIC TOPICS
    + SUPERVISOR DEVELOPMENT TRAINING

* STATUS
  - 21 ACTIVE TRAINEES
  - 49 SEMESTER HOURS TOTAL
    + 21 SEMESTER HOURS COMPLETE
    + 9 SEMESTER HOURS IN PROGRESS

* PHILOSOPHY
  - PROVIDE UNDERGRADUATE LEVEL ENGINEERING TRAINING TO OPERATIONS SUPERVISORS
    + IMPROVE ANALYTICAL ABILITIES OF OPERATIONS SUPERVISION
    + LONGTERM UNIFICATION OF RESPONSIBILITY AND AUTHORITY

TRAINING INVOLVEMENT

* INPO - PLANT MANAGER - EMERGENCY PLAN TRAINING
  RADCHEM SUPERVISOR - GENERIC RADIATION PROTECTION TRAINING
  MAINTENANCE SUPERVISION - MECHANICAL, ELECTRICAL & C&I TRAINI

* RICHLAND COMMUNITY COLLEGE
    ° IPC DEVELOPED TRAINING PROGRAMS IN OPERATIONS, RADIATION PROTECTION, ELECTRICAL AND C&I
    ° PROVIDE INSTRUCTORS (7), TUITION SCHOLARSHIPS, CURRICULUM DEVELOPMENT AND STUDENT COUNSELING

* UNIVERSITY OF ILLINOIS
    ° INSTALLED ELECTRONIC BLACKBOARD
        + STA TRAINING
        + POST GRADUATE/UNDERGRADUATE COURSES
    ° PREVIOUS MSNE PROGRAM

* MIDWEST NUCLEAR TRAINING ASSOCIATION
    ° CHARTER MEMBER
    ° NGET AGREEMENT

* CPS UNIQUE SIMULATOR
    ° SPECIFICATION DEVELOPMENT
    ° DESIGN AND PROJECT EXECUTION

Mark III Containment Issues

C. C. Wheeler

March 5, 1982

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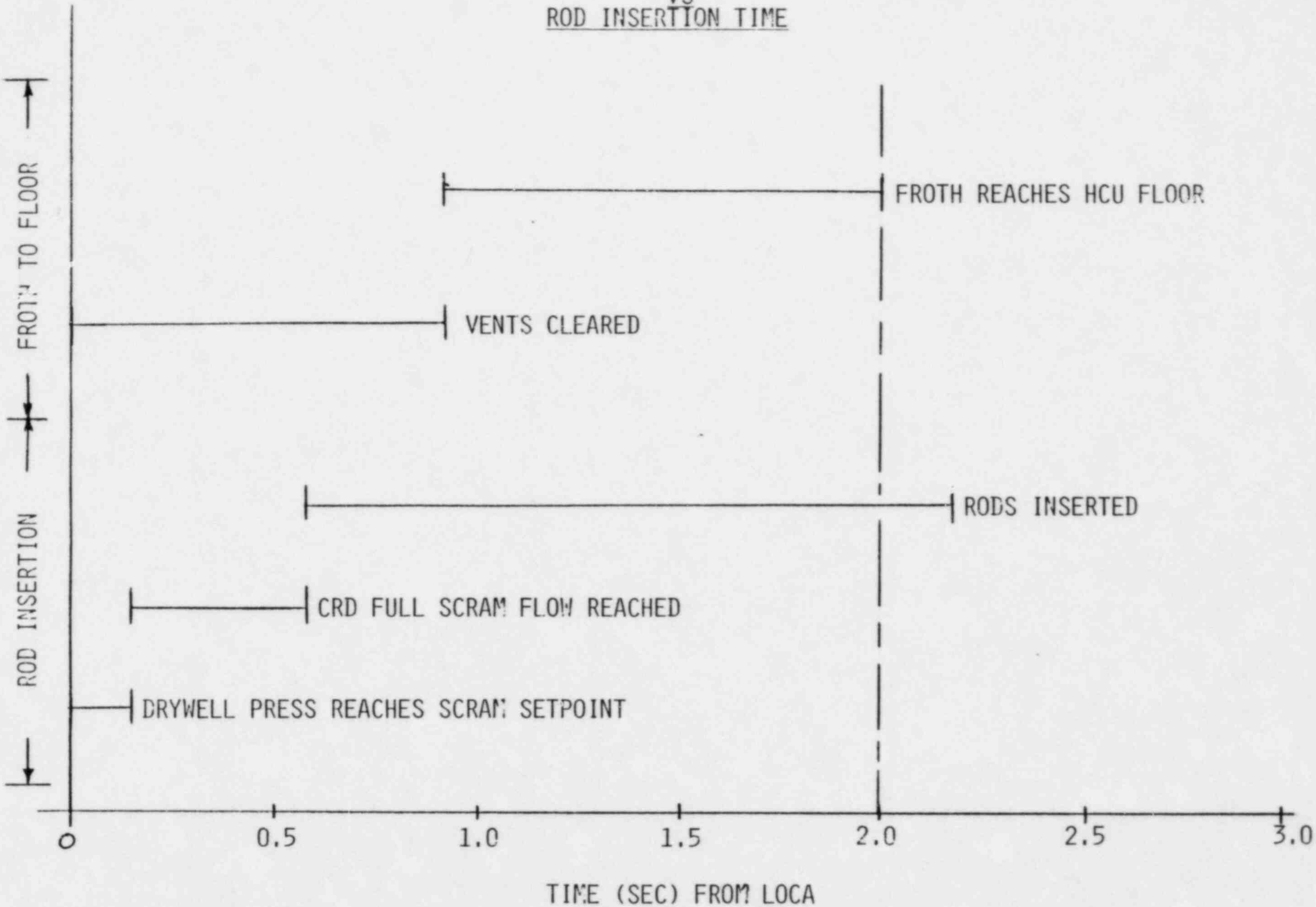
POOL SWELL VS ROD INSERTION TIME



A-2-30

CCW
2-25-82
N83-12

A-231

CPS DESIGN

VS

NRC DYNAMIC LOADS CRITERIA

	DESIGN	EVALUATION	ADEQUATE (X) STRUCTURES	EQUIP'T	PIPING
SRV LOADS	RAMSHEADS CHANGED TO GE "X" QUENCHER	MONTE-CARLO & 35% LOAD RED'N	X	X	X
LOCA LOADS					
POOL SWELL IMPACT	GESSAR II	NRC - SAME	X	X	X
POOL SWELL DRAG	GESSAR II 40'/SEC	NRC - 50'/SEC	X	X	X
FROTH SWELL IMPACT	GESSAR II	FREQUENCY HZ	X	(1)	(1)
FROTH SWELL DRAG	GESSAR II 11 PSID	GESSAR II 11 PSID &	X	(1)	(1)
		$(V/40)^2$(11PSID)	HCU FLOOR (1)	(1)	(1)

(1) ASSESSMENT NOT COMPLETE - NO PROBLEMS TO DATE

CCW
2/23/82
N83-2

CONTAINMENT COMPARISON

	CPS (218)	GESSAR (238)	GG (251)
THERMAL POWER (MWT)	2,894	3,579	3,800
CNTMT DIA (FT)	124	120	124
CNTMT HT (FT)	212	183'7"	208
CNTMT NET VOL (FT^3)	1,457,500	1,100,000	1,400,252
FT^3/MWT	*500	300	370
DRYWELL I.D. (FT)	69	73	73
DRYWELL NET VOL. (FT^3)	241,500	256,000	270,124
POOL VOL. (GAL)	1,123,720	974,345	1,038,065
GAL/MWT	**388	272	273
POOL (DW) AREA (FT^2)	455	533	553

* CPS HAS 64% MORE CNTMT SPACE/MWT THAN GE STD. PLANT.

** CPS HAS 43% MORE GALS OF SUPPRESSION POOL H_2O/MWT THAN GE STD. PLANT

CCW
2/24/82
N83-1D

HUMAN FACTORS ENGINEERING

J. G. Cook

March 5, 1982

ILLINOIS POWER COMPANY'S INVOLVEMENT IN IMPROVING THE MAN-MACHINE INTERFACE

* Based on review by IP operating personnel, IPC committed to the Nuclenet Control Room in 1973.

* Key IPC operations personnel were hired early in the project.

* IPC operations personnel took the lead in organizing and directing the BWR-6 Control Room Owner's Group in 1975. By unifying the utilities, IP provided an operator input which had significant impact on design.

* IPC built and used a plywood mockup of the proposed Nuclenet Control Room so that in reviewing the design the operators and engineers had a clear physical representation of the Control Room. Placement of switches were reviewed by physically walking through operating sequences.

* IPC coordinated the design so that the conventions adopted for the Main Control Room were used throughout the plant.

* Between 1978 and 1980 IPC operation staff engineers and technicians directed the factory checkout and testing of the Nuclenet Control Room in San Jose. This involved 7 people full time or 14 man years of detailed experience with the Control Room.

* Since 1980, the IPC personnel who were in San Jose have lead a larger team of operators and technicians in performing the in-plant checkout and testing of the Control Room.

* Beginning in 1981, IPC started a formal human factors review of the Control Room. Although an independent human factors consultant was used; operators walked through operating procedures including emergency procedures and they conducted a panel by panel review of the completed Control Room.

* The displays have been developed by IPC operations personnel. The SPDS has been integrated into the display system by making revisions to the previously provided plant summary display. Normally this display is available on CRT #5.

CONTROL ROOM DESIGN OBJECTIVES

* Enhance the interface so as to improve the operator's ability to handle normal operation and transients.

  . Decrease time necessary for operator to determine the existence of a problem.

  . Decrease the time necessary to analyze the problem.

  . Decrease response time.

* In this manner, minimize operator errors.

* Implemented in the following ways:

  . Integrating Process Displays based on functional usage

  . CRT displays were chosen because of the clarity with which they can provide real time information to the operator.

  . Arranging displays to present the operator interface in an organized manner.

FEATURES OF THE DISPLAY CONTROL SYSTEM

* Short response panel; 10 CRTs

* Long response panel; 1 CRT

* Supervisor's console; 2 CRTs

* The DCS is capable of 90 formats; approximately 60 have been prepared. Any of the formats can be called up at any screen.

* Normally the operator uses the system in the master mode where depressing one pushbutton will bring up predetermined formats at each of the 9 system related CRTs.

* There is a menu for each system related group of formats. By depressing the menu pushbutton the operator can determine which format displays a certain piece of information.

FURTHER DEVELOPMENTS TO SUPPORT
HUMAN FACTORS

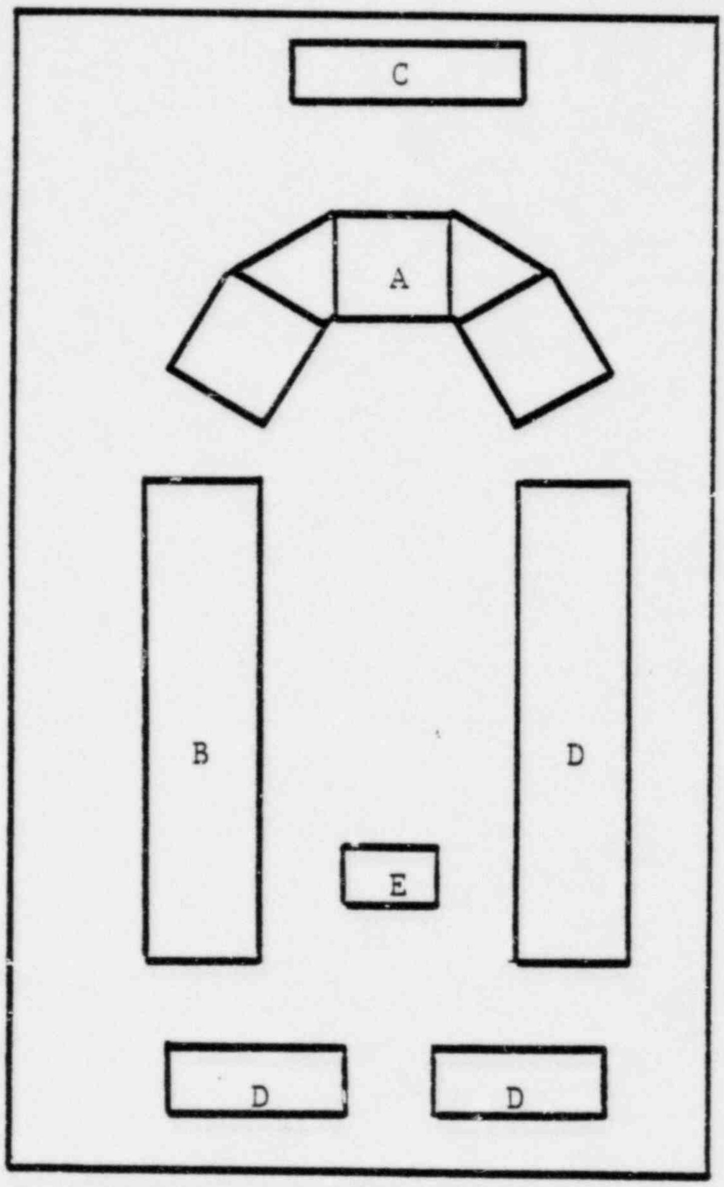
* Procurement of a Plant Specific Simulator

* IPC has developed formats which integrate the SPDS (Safety Parameter Display System) with the display control system.

* Development of the Radwaste Operations Center using human factors conventions from the Main Control Room.

* Provide the displays on CRTs or printers in the TSC and EOF.

* Development of a process radiation monitoring system using the principles of the display control system.



Key:

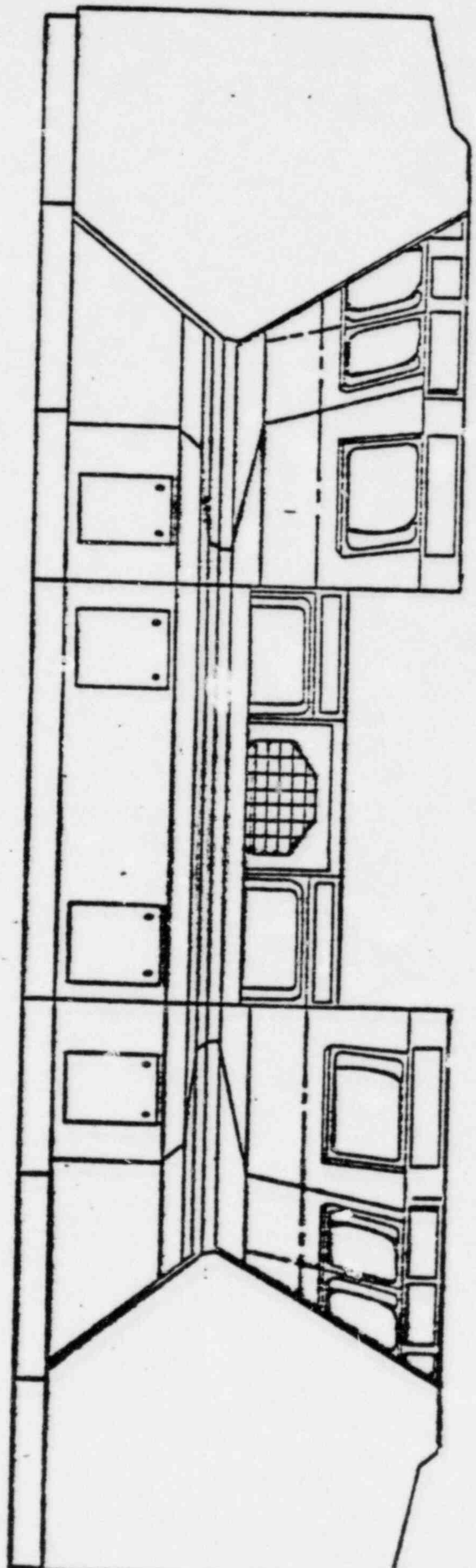
A. Short Response Panel

B. Reactor Core Cooling Benchboard

C. Standby Information Panel and
Process Radiation Monitoring Panel

D. Long Response Panel

E. Supervisory Panel



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REMOTE SHUTDOWN PANEL

J. G. Cook

March 5, 1982

A-240

Remote Shutdown Panel

* Design Bases

  - The Remote Shutdown Panel is redundant to and entirely separate from the Main Control Room.

  - The use of transfer switches allows the Remote Shutdown Panel to operate regardless of the situation in the Main Control Room.

  - Cold Shutdown can be reached through use of the panel.

* Human Factors Considerations

  - The necessity of allowing the operators quick access to other electrical equipment requires that the panel be placed in the Aux. Bldg. at the 781' elevation.

  - The panel contains relatively few switches and controls relatively few systems.

  - Specific procedures have been prepared for operation of this panel. These procedures have been walked through on the panel by operators. The panel is capable of performing its function.

  - The CPS simulator will include the Remote Shutdown Panel; its arrangement will be similar to the plant. Operators will be accustomed to operating the panel under crowded conditions.

  - The panel has been reviewed. Changes under way at present are; permanent mimics for the panel, revised labels and indicator lights and the assignment of one RO and one SRO to the panel. The evaluation of other improvements is continuing.

  - During the ACRS subcommittee tour of the panel there were a number of items which may have prejudiced individual attitudes toward the panel. First 14 people, rather than the normal 2 operators, tried to crowd into the admittedly confined space. Temporary construction lighting was used rather than permanent lighting. Construction noise levels were high and cleanliness reflected the construction activity rather than the operating state.

  - The panel will be used to cool down the plant during startup testing. These tests will conclusively demonstrate that the panel can perform its function.

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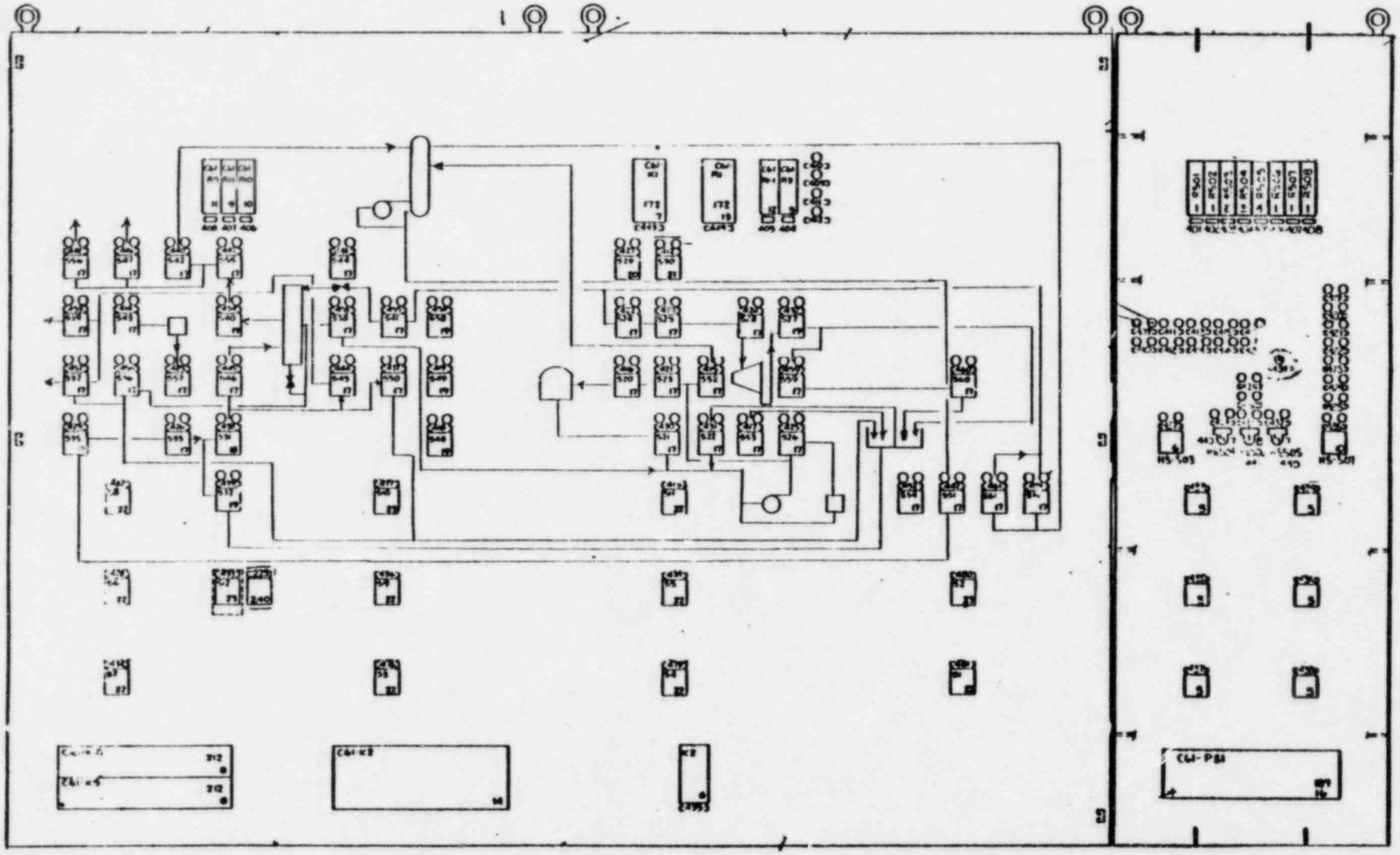


REMOTE SHUTDOWN PANE

Station Blackout

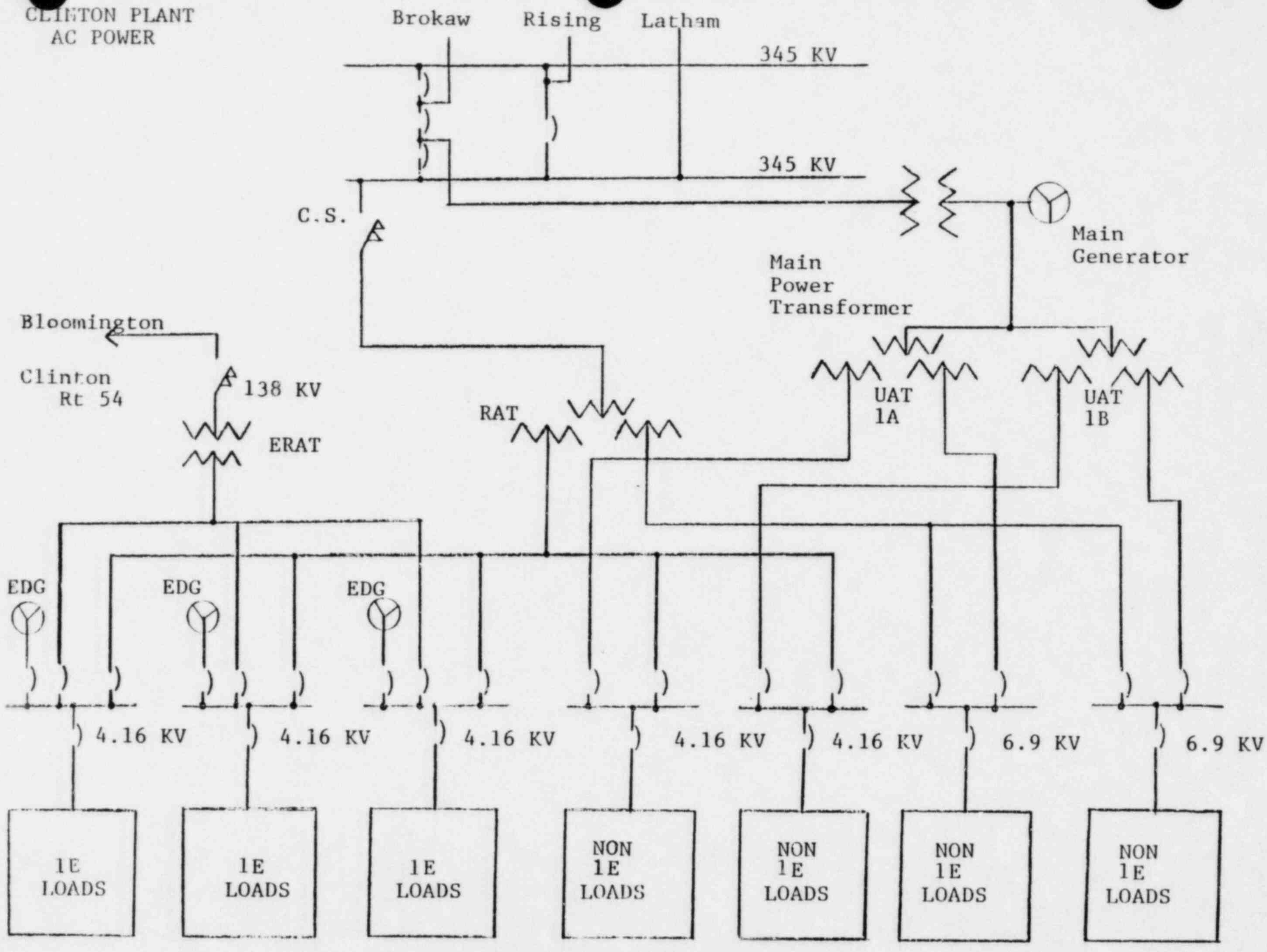
Offsite and Onsite Power Systems

A. L. Ruwe

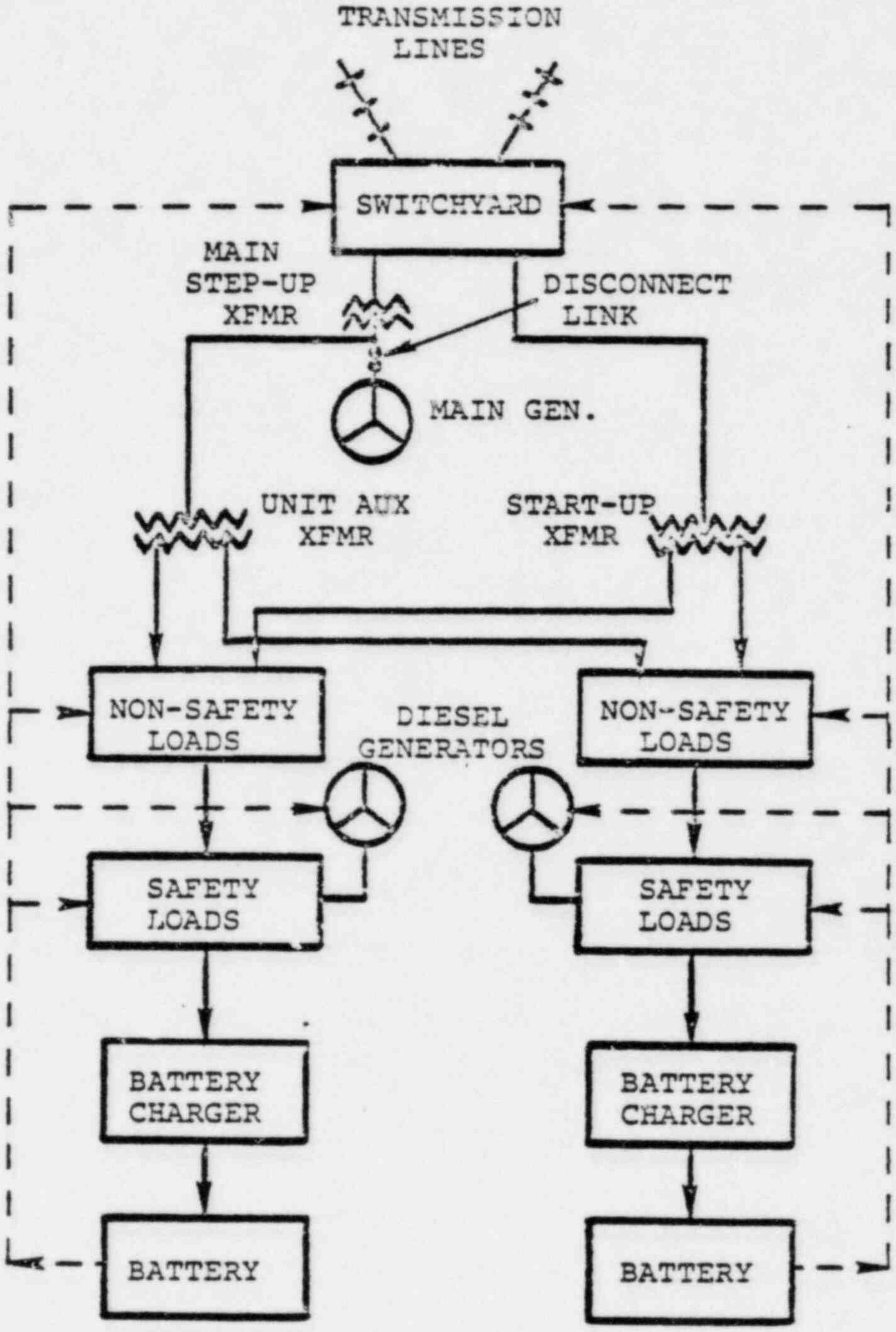
March 5, 1982

A-283

CLINTON PLANT
AC POWER



A-294



SAFETY DIVISION A SAFETY DIVISION B

NOTE: EITHER DIVISION CAN ALONE PROVIDE ALL SAFETY FUNCTIONS

FIGURE 1. TYPICAL TWO DIVISION AC/DC ELECTRICAL DISTRIBUTION SYSTEM

SYMPTOMS ADDRESSED BY PROCEDURES

* Reactor Vessel Water Level Low
* Isolation with Scram
* Drywell Temperature/Pressure High
* Containment Temperature High/Suppression Pool Temperature High
* Suppression Pool Water Level High/Low
* Failure to Shutdown (SCRAM)

SYMPTOM ORIENTED OFF-NORMAL PROCEDURES

* Level Control - Emergency
* Containment Control - Emergency
* Cooldown - Emergency
* Reactivity Control - Emergency

REACTOR VESSEL INJECTION

* Condensate/Feedwater
* RCIC
* HPCS
* CRD
* LPCS
* RHR
* Water Leg Pumps
* SLC
* Fuel Pool Cooling and Cleanup
* Shutdown Service Water
* Fire Protection

DECAY HEAT REMOVAL

As Addressed in Cooldown Procedure

MAIN CONDENSER AVAILABLE

* Bypass Valves
* RCIC
* RHR - Steam Condensing
* Main Steam Line Drains
* RWCU
* RHR - Shutdown Cooling

MAIN CONDENSER NOT AVAILABLE

* RCIC
* RHR - Steam Condensing
* HPCS
* SRV's
* RWCU
* RHR - Shutdown Cooling

Seismic Reevaluation

J. D. Geier

March 5, 1982

A-250

I. CLINTON SSE DESIGN BASIS

A. ASSUME OCCURRENCE OF A MAXIMUM REGIONAL EARTHQUAKE OF INTENSITY VIII AT THE SITE

- 0.25G FREE FIELD ZERO PERIOD FOUNDATION ACCELERATION
- RG 1.60 SPECTRA SHAPE AT SURFACE

B. ASSUME RECURRENCE OF 1811/1812 NEW MADRID EARTHQUAKE AT VINCENNES, IN

C. USE FINITE ELEMENT METHOD OF SSI

II. DESIGN PROCEDURE

A. DECONVOLVE SURFACE MOTION TO FOUNDATION LEVEL

B. USE ASSUMED AVERAGE SOIL PROPERTIES

C. USE SPECIFIED MINIMUM MATERIALS PROPERTIES

D. CONSERVATIVE DESIGN PROCEDURES FOR STRUCTURES, SYSTEMS, AND EQUIPMENT

III. NRC STAFF SER REVIEW (AUG., 1981)

A. FINITE ELEMENT METHOD OF SSI ALONE NO LONGER ACCEPTABLE
SOIL SPRING METHOD OF SSI SHOULD ALSO BE USED

B. DECONVOLUTION OF SURFACE MOTION TO FOUNDATION LEVEL NOT ACCEPTABLE

C. SOIL PROPERTY VARIATION SHOULD BE CONSIDERED

D. THE TWO ACCEPTABLE REEVALUATION OPTIONS DESCRIBED IN SER:

OPTION 1: DEVELOP 5.8 MAGNITUDE SITE SPECIFIC SPECTRA AND EVALUATE IMPACT ON DESIGN

OPTION 2: DEVELOP A 5.3 MAGNITUDE SITE SPECIFIC SPECTRA AND PERFORM A SITE HAZARDS ANALYSIS TO JUSTIFY ITS USE

IV. STATUS

A. REEVALUATION OF PLANT STRUCTURES, PIPING, AND EQUIPMENT COMPLETED BASED ON OPTION 1 (5.8 MAGNITUDE EARTHQUAKE)

B. RESULTS

1. PLANT CAPABILITY EXCEEDS 5.8 MAGNITUDE EARTHQUAKE
2. INCREASE IS GENERALLY LESS THAN 40% WITH AN AVERAGE INCREASE OF ABOUT 15%
3. CLINTON SIGNIFICANTLY LESS AFFECTED BY NEW SEISMIC CRITERIA THAN OTHER PLANTS DUE TO EXTRA CONSERVATISM IN ITS DESIGN BASIS ESTABLISHED FOR CONSTRUCTION PERMIT

Seismic Reevaluation

A. K. Singh

March 5, 1982

A-255

We have performed a state-of-the-art seismic re-evaluation of the plant to show that the Clinton design meets the present NRC requirements. This re-evaluation (based on SER Option 1) included:

A. Development of a site-specific spectrum for an $m_B = 5.8$ earthquake using the methodology used at the Sequoyah and Midland plants. The site-specific spectrum has been submitted to the staff for review. The staff has found it to be generally acceptable.

B. A new soil spring soil-structure interaction analysis including soil property variation and no deconvolution. The staff has reviewed the SSI methodology and the responses and has found them to be acceptable.

C. Evaluation of the plant structures, piping, and equipment for the effects of the new seismic analysis to show that the Clinton design is adequate to resist the new loads. The evaluation results have been presented to the staff for review.

Specific information on stresses and margins in design were presented at the ACRS Subcommittee held in Decatur February 25 and 26, 1982.

We believe we are close to resolving the seismic issue with the staff. This presentation summarizes the evaluation we have performed to resolve the seismic issue.

NSSS EVALUATION

WE HAVE EVALUATED ALL ESSENTIAL EQUIPMENT FOR THE INCREASED SEISMIC RESPONSE, INCLUDING:

- RPV AND INTERNALS
- GE-SUPPLIED REACTOR VESSEL EQUIPMENT
- MAIN STEAM AND RECIRCULATION PIPING
- FLOOR-MOUNTED EQUIPMENT, INCLUDING PUMPS, MOTORS, HEAT EXCHANGERS, REFUELING EQUIPMENT, AND TANKS.

BASED ON THIS EVALUATION, WE CONCLUDE THAT ALL NSSS EQUIPMENT IS ADEQUATE FOR THE NEW SEISMIC LOADS.

EVALUATION OF STRUCTURES

WE HAVE EVALUATED THE MAIN BUILDING COLUMNS, FLOOR SLABS, AND SHEAR WALLS, THE CONTAINMENT AND INTERNAL STRUCTURES AND THE EARTH STRUCTURES FOR THE NEW SEISMIC LOADS TO CONCLUDE THAT CLINTON PLANT STRUCTURES CAN WITHSTAND THE EFFECTS OF THE REVISED SEISMIC LOADS.

THE STAFF HAS REVIEWED OUR EVALUATION AND HAS FOUND IT TO BE ACCEPTABLE.

PIPING EVALUATION

WE HAVE EVALUATED THE RESIDUAL HEAT REMOVAL AND THE DECAY HEAT REMOVAL SYSTEM TO CONCLUDE THAT THE DESIGN OF THESE SYSTEMS ARE NOT AFFECTED BY THE NEW SEISMIC INPUT.

IN ADDITION, WE HAVE REANALYZED THE CRITICAL PORTIONS OF THE FOLLOWING PIPING SYSTEMS TO CONCLUDE THAT CLINTON PIPING DESIGN IS CONSERVATIVE AND IS ADEQUATE TO WITHSTAND THE EFFECTS OF THE NEW SEISMIC INPUT.

- REACTOR WATER CLEAN-UP SYSTEM (RT01 AND RT02)
- RESIDUAL HEAT REMOVAL SYSTEM (RH08 AND RH34)
- SHUTDOWN SERVICE WATER SYSTEM (SX16)
- LOW PRESSURE CORE SUBSYSTEM (LP04)
- COMPONENT COOLING SYSTEMS (CC06 AND CC07)
- REACTOR CORE ISOLATION COOLING SYSTEM (RI11)
- STAND-BY GAS TREATMENT SYSTEM (VG13)

THESE SUBSYSTEMS WERE CHOSEN BECAUSE THEY WERE JUDGED MOST AFFECTED BY THE CHANGE IN SEISMIC INPUT BASED ON SUBSYSTEM FREQUENCY AND THEIR LOCATION IN THE PLANT OR THEY WERE JUDGED TO BE MOST HIGHLY STRESSED SUBSYSTEMS.

BALANCE OF PLANT (BOP) EQUIPMENT

COMPARISON OF THE NEW COMBINED RESPONSE SPECTRA TO THOSE IN THE PROCUREMENT SPECIFICAITON SHOWS THAT THE PROCUREMENT SPECTRA BOUND THE NEW SPECTRA FOR ALL VERTICAL COMPONENTS AND MANY HORIZONTAL COMPONENTS IN THE EQUIPMENT FREQUENCY RANGES. FOR LOCATIONS WHERE THE SPECIFICATION SPECTRA DO NOT BOUND THE NEW SPECTRA, THE AVERAGE INCREASE IS 5% AND THE MAXIMUM INCREASE IS 15% IN THE RANGE OF CRITICAL EQUIPMENT FREQUENCIES.

WE HAVE REVIEWED THE DESIGN CALCULATIONS OF SAFETY RELATED BOP MECHANICAL EQUIPMENT AND THE ACTUAL TEST SPECTRA OF 1E BOP ELECTRICAL EQUIPMENT TO CONCLUDE THAT THE INTEGRITY AND FUNCTIONABILITY OF BOP ELECTRICAL AND MECHANICAL EQUIPMENT IS NOT AFFECTED BY THE SMALL INCREASE IN THE SEISMIC INPUT.

SUMMARY

1. WE HAVE RE-EVALUATED THE PLANT STRUCTURES, PIPING AND EQUIPMENT TO THE CURRENT NRC STAFF POSITIONS AND HAVE DEMONSTRATED THAT THE CLINTON DESIGN MEETS THE PRESENT NRC CRITERIA.

2. BASED ON THE CLINTON SPECIFIC EVALUATION AND OUR EXPEFIENCE WITH SIMILAR EVALUATIONS FOR OTHER PLANTS, WHERE INCREASES IN DESIGN LOADS DUE TO SEISMIC RE-EVALUATION ARE GREATER THAN CLINTON, WE CONCLUDE THAT THE CLINTON PLANT STRUCTURES PIPING AND EQUIPMENT DESIGN IS CONSERVATIVE AND IS CAPABLE OF WITHSTANDING THE NEW SEISMIC LOADS WITHOUT PLANT MODIFICATIONS. THUS ADDITIONAL ANALYSIS SHOULD NOT BE REQUIRED.

3. WE HAVE PRESENTED THE RE-EVALUATION RESULTS TO THE NRC STAFF AND WE BELIEVE WE ARE CLOSE TO RESOLVING THIS ISSUE WITH THE STAFF.

EMERGENCY PLANNING

J. G. Cook

March 5, 1982

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DEVELOPMENT OF THE EMERGENCY PLAN

* Strong IP involvement

  - Preparation of the CPS Emergency Plan
  - Preparation of the Evacuation Time Study
  - Preparation of Implementing Procedures
  - Coordination Activities with other Organizations
  - Training

* State Agencies

  - Illinois Emergency Services and Disaster Agency
  - Illinois Department of Nuclear Safety
  - Comprehensive & Coordinated response between all political units
  - IPRA covers all nuclear plants in Illinois (5 full scale tests)
  - IPRA is an extension of other emergency activities
  - Demonstrated ability in emergency operations
  - Facilities demonstrated in other drills

* Local Agencies

  - DeWitt County ESDA - (Excellent job in developing and demonstrating plans for all risks)
  - DeWitt County Sheriff
  - Hospital
  - Ambulance
  - Fire Department
  - Doctors, etc.

SITE AND FACILITIES

* Site

  . Regular Topography
  . No Obstructions to Traffic
  . Low Population Density
  . Favorable Meteorology

* Facilities

  . TSC adjacent to Main Control Room
  . EOF on site
  . Press Center near site
  . Backup EOF in Decatur
  . Emergency Warning System

* Communications

  . Nuclear Accident Reporting System
  . Emergency Notification System
  . Health Physics Network
  . Emergency Automatic Ringdown
  . CPS Switchboard
  . Decatur Switchboard
  . Remote Telephone Central Offices
  . Facsimile Transmission
  . ESDA Radio
  . Microwave
  . IPC Radio

* Assessment Facilities

  . Extensive Plant Data Availability through DCS/PMS
  . Real Time Release and Dose Projections available through the PRM System and Report Generator
  . Hard Copy Plant Data
  . Procedures

OPERATION IN EMERGENCY

* IPC

  - Control the Plant and Plant Personnel
  - Classify Accident and take prescribed Actions
  - Notification
  - Radiation Surveys and Assessment
  - Recommendations for Off-site Activities

* ESDA

  - Operational Control of Emergency Activities
    State & County EOC
    State Command Post (inside EPZ)
  - Coordinate the Activities of Police, Ill. Dept. of Transportation, Public Health Dept., Dept. of Conservation, Ill. Commerce Commission, etc.
  - Traffic Control, Access Control
  - Evacuation (as ordered by Governor)
  - Operation of Relocation Centers
  - Social Services

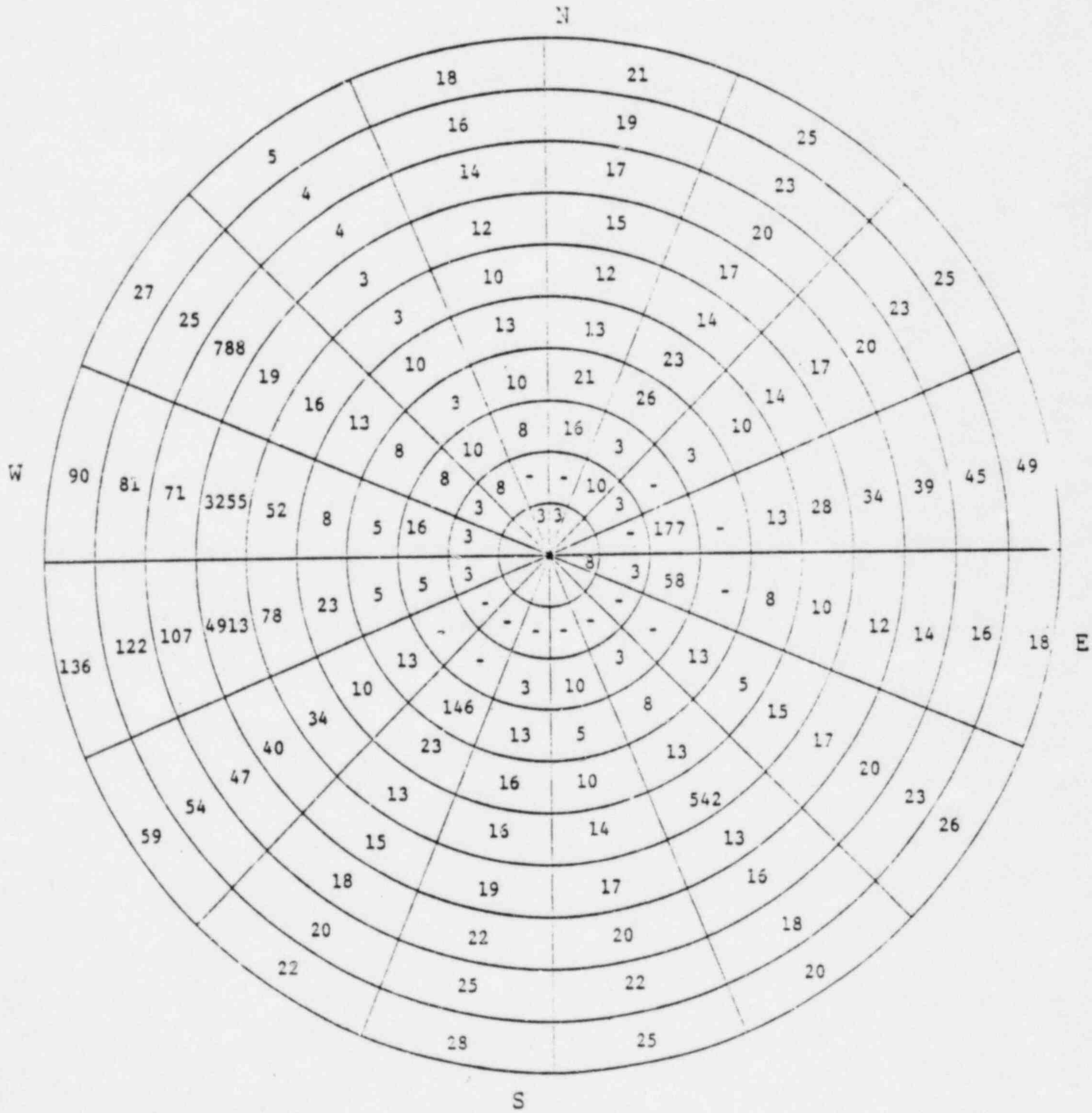
* Ill. Dept. of Nuclear Safety

  - Assessment (Rad. Emergency Assessment Center)
  - Food, Water, Milk Control
  - Exposure Control

POPULATIONS
0-10 Miles

Total Population

1981 - 12,698
1990 - 13,405
2000 - 13,860
2010 - 14,254
2020 - 14,644



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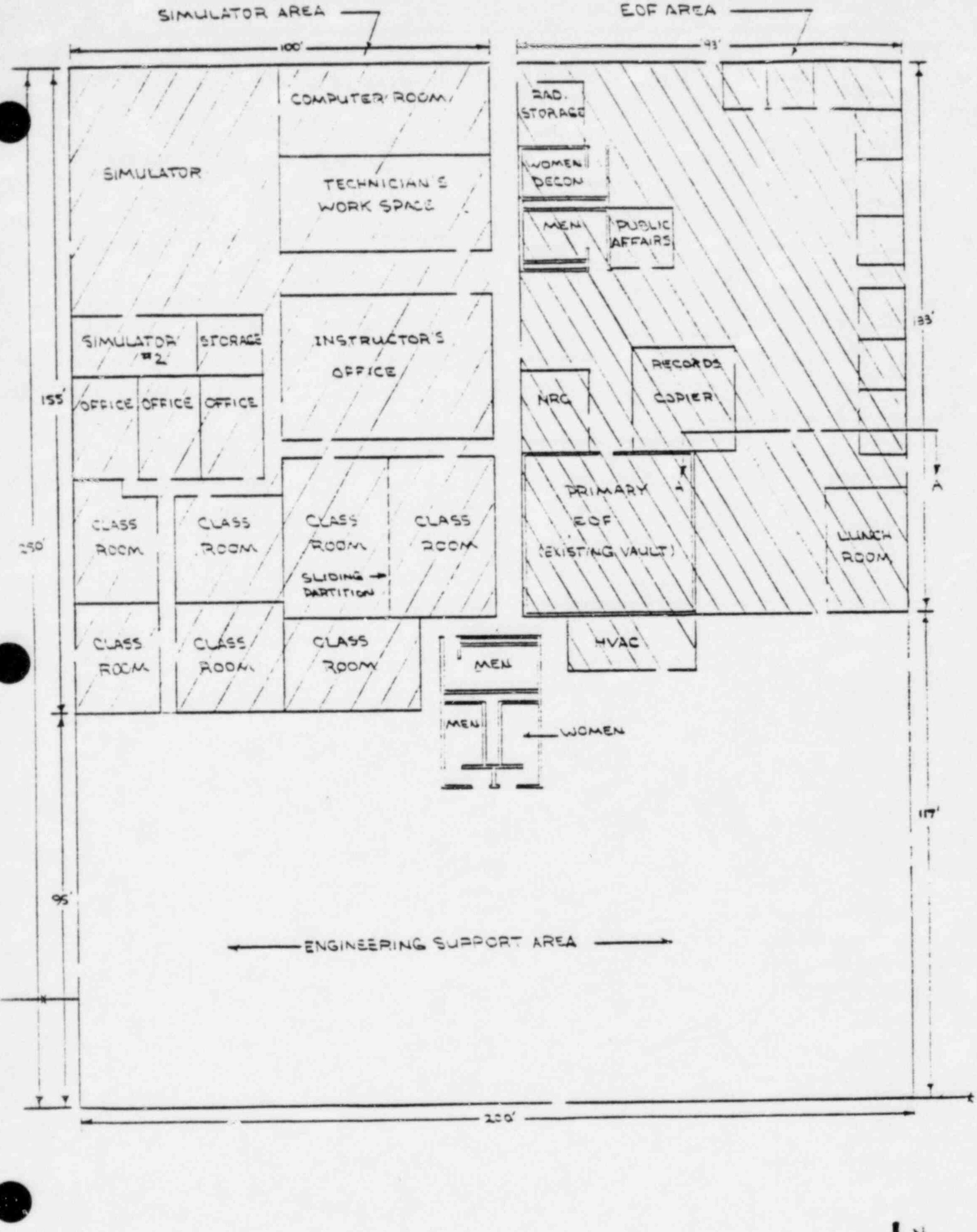

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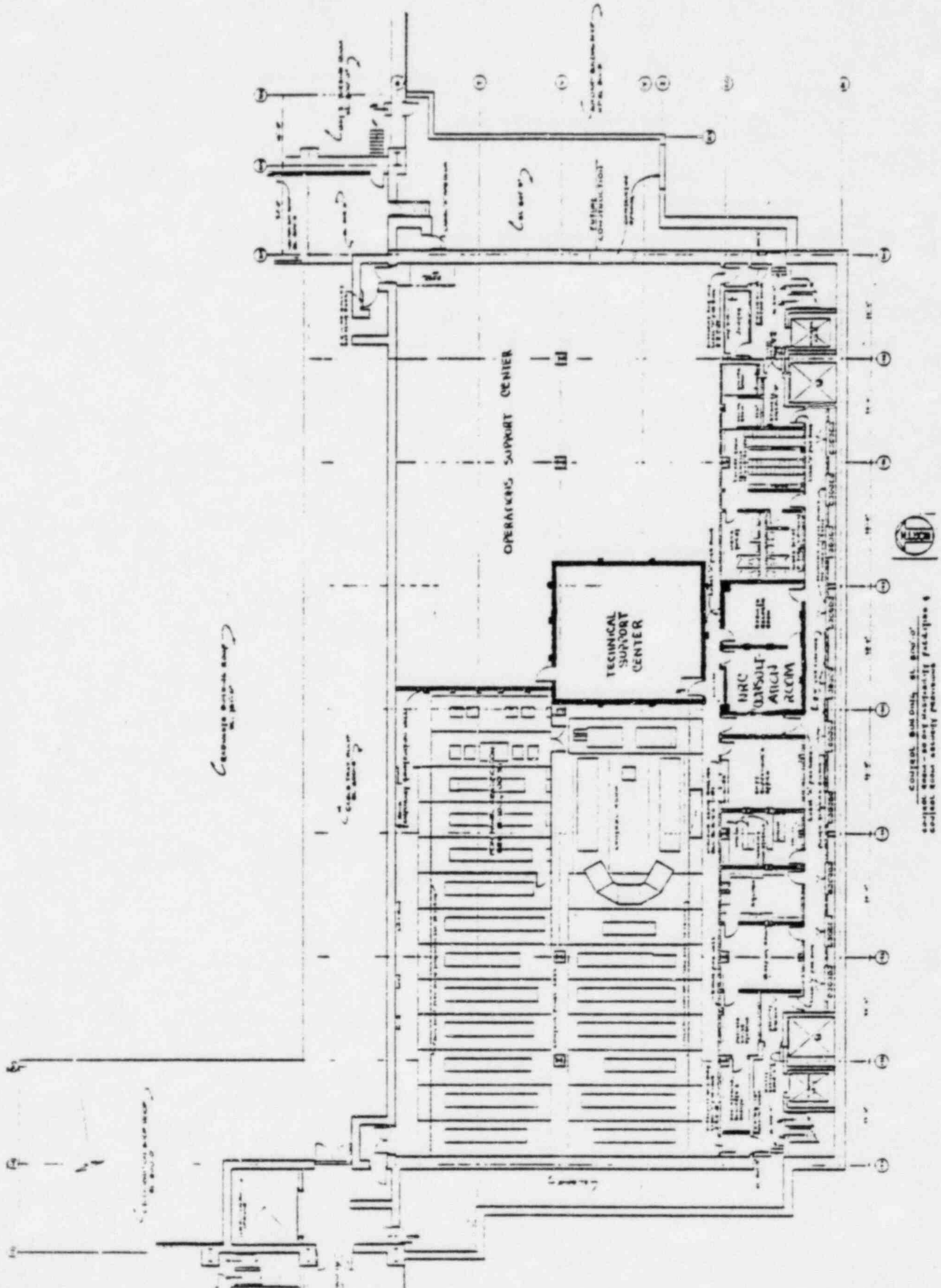
CONSTRUCTION OFFICE BUILDING


OPERATIONS SUPPORT CENTER
TECHNICAL SUPPORT CENTER
NRC CONSULTATION ROOM

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INITIATIVES TO IMPROVE QA AT CONSTRUCTION SITES

I. INVOLVE INDUSTRY CHIEF EXECUTIVE OFFICERS IN QA

II. NRC CONSOLIDATION OF STAFF RESOURCES

III. THIRD PARTY AUDITS

IV. ENFORCEMENT SANCTIONS

V. TRAINING, QUALIFICATION AND CERTIFICATION OF QA PERSONNEL

I. INVOLVE INDUSTRY CHIEF EXECUTIVE OFFICERS IN QA

RECOMMEND A MEETING BETWEEN CHIEF EXECUTIVE OFFICERS AND THE COMMISSION AND SENIOR STAFF. MEETING OBJECTIVES WOULD BE:

A. DELIVER A STRONG MESSAGE THAT IN TOO MANY INSTANCES INDUSTRY HAS NOT MET ITS RESPONSIBILITY FOR QA AT NUCLEAR POWER PLANTS UNDER CONSTRUCTION.

B. OBTAIN FEEDBACK ON CORRECTIVE ACTIONS PLANNED BY INDUSTRY.

C. OBTAIN A COMMITMENT FOR DEVELOPMENT OF INDUSTRY PLAN OF ACTION.

1. SHORT RANGE PLAN WOULD ADDRESS THOSE PLANTS EXPECTING AN OPERATING LICENSE IN FY 82.

2. LONG RANGE PLAN WOULD ADDRESS THOSE PLANTS TO BE LICENSED AFTER FY 82, AND WOULD ADDRESS THE EVALUATION OF DESIGN AND CONSTRUCTION ACTIVITIES AT THOSE PLANTS.

D. EMPHASIZE NRC's INTENTION TO USE PROMPT AND STRINGENT ENFORCEMENT SANCTIONS FOR NRC IDENTIFIED QA PROGRAM BREAKDOWNS.

E. INFORM UTILITY CHIEF EXECUTIVE OFFICERS THEY WILL BE REQUIRED TO CERTIFY THAT THE PLANT HAS BEEN DESIGNED, CONSTRUCTED AND TESTED IN ACCORDANCE TO THE FSAR PRIOR TO THE ISSUANCE OF AN OPERATING LICENSE.

III. THIRD PARTY AUDITS

A. NRC WILL CONTINUE TO ENCOURAGE INDUSTRY, THROUGH INPO, ASME, AND COMPARABLE INDUSTRY ORGANIZATIONS, TO PURSUE VIGOROUS USE OF CONSTRUCTION QUALITY ASSURANCE STANDARDS.

B. FOR PLANTS TO BE LICENSED THIS YEAR, THE UTILITY WILL BE REQUIRED TO PERFORM A RIGOROUS SELF APPRAISAL OF ITS QUALITY ASSURANCE PROGRAM. THIS INFORMATION WILL BE PROVIDED TO THE STAFF FOR AN INDEPENDENT ASSESSMENT OF THE ACCEPTABILITY OF DESIGN AND CONSTRUCTION FOR SELECTED AREAS.

C. INDUSTRY HAS REQUESTED INPO TO DEVELOP AND MANAGE A SPECIAL AUDIT PROGRAM FOR DESIGN AND CONSTRUCTION.

IV. ENFORCEMENT SANCTIONS

A. EMPHASIZE NRC'S INTENTION TO EXERCISE PROMPT AND STRINGENT ENFORCEMENT SANCTIONS FOR THOSE PROBLEM AREAS IDENTIFIED BY NRC.

B. NRC WILL CONSIDER MODIFYING ENFORCEMENT SANCTIONS FOR UTILITY-IDENTIFIED VIOLATIONS

V. TRAINING, QUALIFICATION AND CERTIFICATION OF QA PERSONNEL

NRC STAFF IS GIVING SERIOUS CONSIDERATION TO FORMAL CERTIFICATION OF VARIOUS LEVELS OF QUALITY ASSURANCE PERSONNEL. THE FOLLOWING ALTERNATIVES WILL BE CONSIDERED:

A. THIRD PARTY CERTIFICATION OF PERSONNEL ENGAGED IN QUALITY ASSURANCE/QUALITY CONTROL INSPECTION FUNCTIONS.

B. NRC LICENSING OF PERSONNEL ENGAGED IN QUALITY ASSURANCE/QUALITY CONTROL FUNCTIONS.

C. CERTIFICATION OF SELECTED UTILITY, ARCHITECT-ENGINEER AND MAJOR VENDOR EMPLOYEES AS DESIGNATED NRC INSPECTORS (ANALOGOUS TO THE FAA DESIGNATED ENGINEER).

ADDITIONAL APPROACHES UNDER CONSIDERATION

I. QUALITY VERIFICATION PROGRAM FOR PROBLEM FACILITIES

II. DESIGN MANAGEMENT

III. REVISED INSPECTION PROGRAM

IV. APPROVED BIDDER'S LIST

I. QUALITY VERIFICATION PROGRAM FOR PROBLEM FACILITIES

THE STAFF IS CONSIDERING CRITERIA FOR SELECTIVELY APPLYING INSPECTION PROGRAMS FOR PROBLEM FACILITIES.

THE NATURE OF THESE PROGRAMS WOULD DEPEND ON THE EXTENT AND NATURE OF THE BREAKDOWN IN THE UTILITY'S QUALITY ASSURANCE PROGRAM. ALTERNATIVES TO BE CONSIDERED WILL INCLUDE:

A. INTENSIVE REINSPECTION AND/OR REANALYSIS OF QUESTIONABLE AREAS BY THE UTILITY.

B. INTENSIVE INSPECTION AND/OR ANALYSIS OF QUESTIONABLE AREAS BY NRC USING, AS APPROPRIATE, INTEROFFICE AUDIT TEAMS AND/OR CONTRACTOR ASSISTANCE TO PROVIDE THE REQUIRED SPECIFIC TECHNICAL EXPERTISE.

C. USE OF THE NRC MOBILE LAB WITH INDEPENDENT TECHNICAL INSPECTION EXPERTISE CONTRACTED FOR BY THE NRC.

II. DESIGN MANAGEMENT

THE STAFF IS CONSIDERING INITIATIVES TO VERIFY THE PROPER MANAGEMENT AND IMPLEMENTATION OF THE DESIGN FOR NUCLEAR POWER PLANTS. THE ALTERNATIVES UNDER CONSIDERATION INCLUDE:

A. THE DEVELOPMENT OF CRITERIA TO EMPLOY CONTRACTOR DESIGN AUDITS AT SELECTED SITES IN THE CONSTRUCTION PHASE. THE AUDITS WOULD PROVIDE ADDITIONAL CONFIDENCE IN THE PROPER IMPLEMENTATION OF THE DESIGN IN QUESTIONABLE AREAS.

B. REVISING THE VENDOR AND CONSTRUCTION INSPECTION PROGRAMS TO INCLUDE INSPECTION ELEMENTS TO BE USED TO VERIFY THE PROPER MANAGEMENT AND IMPLEMENTATION OF THE DESIGN.

C. ASSIGNMENT OF RESIDENT INSPECTORS AT SELECTED MAJOR VENDORS AND ARCHITECT-ENGINEERS.

D. SOME TYPE OF ACCREDITATION OF DESIGN ORGANIZATIONS, INCLUDING THOSE OF NSS'S AND AE'S.

III. REVISED INSPECTION PROGRAM

NEW INITIATIVES ARE BEING CONSIDERED FOR INCLUSION IN THE PRESENT CONSTRUCTION INSPECTION PROGRAM. THESE ARE:

A. DEVELOPMENT OF INCREASED NRC CAPABILITY FOR INDEPENDENT MEASUREMENTS UTILIZING CONTRACTOR SUPPORT.

B. INCREASE INSPECTION PROGRAM EMPHASIS ON ENSURING THE EFFECTIVENESS OF THE UTILITY'S MANAGEMENT CONTROL (QUALITY ASSURANCE) SYSTEMS.

C. CONTINUE DIRECT VERIFICATION INSPECTIONS ACROSS THE TECHNICAL DISCIPLINES.

D. INCREASE INSPECTION (NRC) EMPHASIS ON INSPECTION OF DESIGN AND DESIGN INTERFACES.

E. PERFORM A PERIODIC OVERALL ASSESSMENT OF THE ACCEPTABILITY OF CONSTRUCTION WORK CONSIDERING ALL INSPECTION RESULTS.

IV. APPROVED BIDDER'S LIST

PRESENTLY, THREE MEASURES EXIST WHICH HAVE SOME FEATURES OF AN APPROVED BIDDER'S LIST APPROACH.

A. MAJOR VENDOR TOPICAL QA REPORTS.

B. COORDINATED AGENCY SUPPLIER EVALUATION (CASE)

C. ASME CERTIFICATION OF AUTHORIZATION (N-STAMP) TO INCLUDE ACCREDITATION FOR 10 CFR 50 APPENDIX B PURPOSES.

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

March 4, 1982

APPENDIX XIV
PROPOSED SUMMARY OF ACRS SUBCMTE MEETING
OF REACTOR OPERATIONS - LER RULE,
3/3/82

MEMORANDUM FOR: W. Mathis, Chairman
ACRS Subcommittee on Reactor Operations

FROM: Richard Major, Richard Major
Senior Staff Engineer

SUBJECT: PROPOSED SUMMARY OF ACRS SUBCOMMITTEE MEETING OF REACTOR OPERATIONS - LER RULE, MARCH 3, 1982 . WASHINGTON, D.C.

A summary of the subject meeting is attached for your review. Copies are being distributed to the other ACRS members for their information and comment. Corrections and additions will be included in the minutes of the meeting.

Attachment:
1. Meeting Summary
2. Mr. Ahearne's comments on SECY-82-3
3. Meeting Slides from AEOD

cc: ACRS Members
ACRS Technical Staff, w/o attach.
C. Ryder, Fellow, w/o
C. Sun, Fellow, w/o
E. Case, NRR, w/o
E. Goodwin, NRR, w/o
C. Michelson, AEOD, w/o
J. Haltemes, AEOD, w/o
F. Hebdon, AEOD, w/o

PROPOSED SUMMARY OF THE MARCH 3, 1982 MEETING
OF THE ACRS SUBCOMMITTEE ON REACTOR OPERATIONA
LER RULE

Purpose:

The purpose of the meeting was to continue discussions with NRC's Office of Analysis and Evaluation of Operational Data and other interested parties on the proposed Licensee Event Report (LER) Rule outlined in SECY 82-3, "Proposed Addition of 10 CFR 50.73 Establishing the Licensee Event Report (LER) System." Since the last subcommittee meeting on this issue, additional questions had arisen on the proposed LER rule from the Committee and the Commission. The meeting was designed as a round table discussion to address the concerns.

Attendees:

ACRS

W. Mathis
D. Moeller
H. Etherington
J. Ebersole
D. Ward
M. Bender
R. Major, D.F.E.
C. Ryder, Fellow
C. Sun, Fellow

NRC

R. Colmar, DST
J. Pittman, RES
K. Bissell, OCM
F. Manning, RES
B. Buchbinder, RES
R. Robinson, RES
C. Willis, NRR
J. Roe, OCM
T. Marsh, OCM
E. Abbott, OCM
W. Booth, OPE
L. Ong, OPE
A. Bates, SECY

NRC/AEOD

C. Michelson
J. Heltemes
F. Hebdon
R. Dennig
E. Boyle

NRC/IE

E. Jordon
B. Mills
A. Patton

Others

E. Howard, KMC
R. Leyse, NSAC
B. Cohn, INPO
J. Wagoner, US DOE

A-280

Background:

This was the third subcommittee meeting with AEOD to discuss the proposed LER system. Previous meetings were held on September 9, 1981, and December 8, 1981.

Following the September meeting an Advanced Notice of Rulemaking was published to inform the public of the following: that a proposed rule to modify and codify existing LERs would be developed; that the Staff endorses INPO to fund and manage NPRDS; and encourage INPO to assure that NPRDS is set up to meet industry and NRC needs to support probabilistic risk and reliability assessment programs. Members of the Subcommittee expressed agreement with the proposals discussed in the proposed advance notice of rulemaking.

In December, a second meeting was held with AEOD to discuss the details of the LER rule prior to asking for Commission approval to release the draft rule for public comment. As a result of Subcommittee and full Committee discussions, the Committee decided it would have no objection to the proposed LER rule being issued for public comment.

The current status of the rule has remained unchanged since December. The rule has not been released by the Commission for public comments.

Discussion:

1. The LER rule was developed to address problems with reporting requirements; the Sequence Coding and Search System (SCSS) was developed to address problems with data storage and retrival capabilities. Although both systems have been coordinated to ensure the maximum use of information provided one does not depend on the existance of the other.

2. The SCSS will be backfitted to include some of the past LERs. All LERs from 1981 are being coded. It has not been decided how far the backfitting should be extended. INPO will update the current LER data base until SCSS is fully operational. AEOD believes no existing data will be discarded and no search capability lost.

3. The LER rule will require reporting of events not now reported (reactor trips, inadvertent actuation of safety systems). It will also require more detailed report narratives. However, AEOD cannot increase the reporting burden on licensees. To accomplish this, AEOD will give up some reports currently received. Implementation will involve relying on NPRDS for reports of single, random failures and no longer requiring reports on certain types of events such as operation in a degraded mode, and minor technical specification violations. This information would be published in monthly operating reports rather than LERs.

A-281

4. AEOD believes that the new LER system and an effective and efficient NPRD System will increase the NRC's and the nuclear industry's ability to analyze most operational experience data.

5. The proposed rule will require that licensees report any instance of personnel error, equipment failure, procedure violation, or discovery of design, analysis, fabrication, construction, or procedural inadequacies that alone could prevent the fulfillment of the safety function of structures or systems that are needed to shutdown the reactor, remove residual heat; or control the release of radioactive material. Many events reportable under other criteria will include, "Human errors."

6. There was discussion of Commissioner Ahearne's comments (attached) which could be interpreted as follows:

   . The proposed rule which uses a narrative to describe an event, does not provide a data base for a system which would provide a general data matrix for detecting generic problems. It also lacks detailed retreival and analysis capability. There is no provision for selecting a list of important plant data to file with the LER.

   . The proposed LER rule does not provide a base for future expansion into the above mentioned matrix.

   . The rule should not be approved until it can provide a logical step towards a more meaningful analysis tool for the future.

7. In the discussion of above points, a clear definition of what would be desirable for the immediate and long range data base to satisfy Commissioner Ahearne's comments was not found. However, it was suggested that an example would be helpful in understanding what would satisfy the proposed non-narrative concept of data collection.

   The subcommittee was in general agreement that such a proposal (non-narrative data collection) should be considered, but that considerable time (years) would be required to study such a system. It was suggested that this could be a topic for a possible research project.

Conclusion:

The proposed LER Rule represents a substantial improvement over the present LER system and should be approved for release for public comment. The ACRS would still expect to review the final version of the rule after public comments have been received and considered. The Subcommittee recognized that after experience with the proposed LER system and as the state of the art in information collection and analysis progresses, it is possible that this LER system would be revised or replaced. It was felt that the proposed LER system represents the next logical evolution in incident reporting and should not be delayed.

Future Meetings:

The Subcommittee would expect to meet again on this rule following consideration of public comments.

The following pages A-284 thru A-286 has been deleted as 1.

DELETION

4-287

Licensee Event Report Rulemaking

ACRS Reactor Operations Subcommittee Meeting

March 3, 1982

Prepared By: Frederick J. Hebdon

QUESTION A. EXPLANATION OF RELATIONSHIP BETWEEN THE LER RULE, SEQUENCE CODING, AND ANALYSIS OF THE DATA COLLECTED.

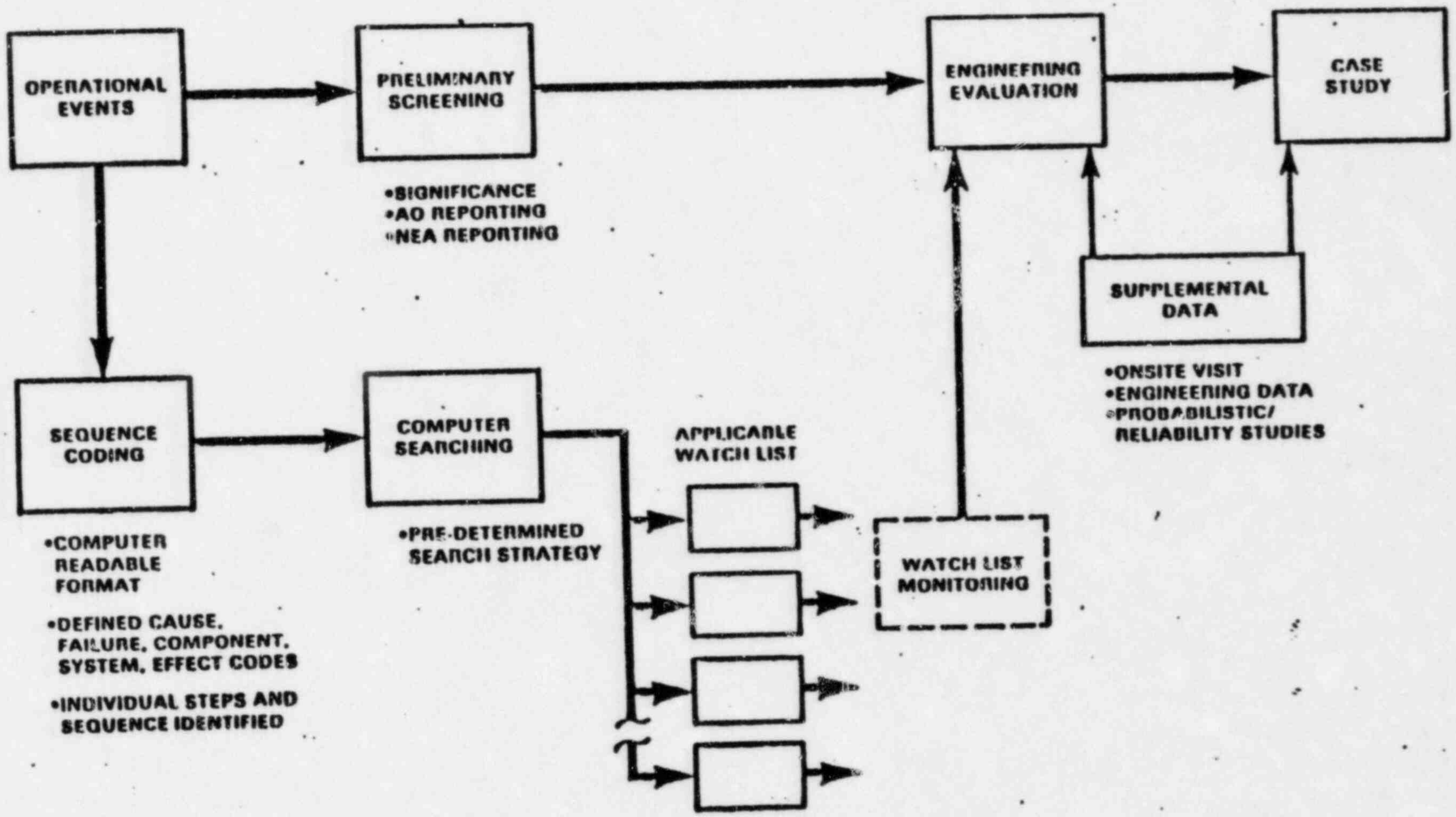
ANSWER.

- THE LER RULE WAS DEVELOPED TO ADDRESS PROBLEMS WITH REPORTING REQUIREMENTS
- THE SEQUENCE CODING AND SEARCH SYSTEM (SCSS) WAS DEVELOPED TO ADDRESS PROBLEMS WITH DATA STORAGE AND RETRIEVAL
- NEITHER SYSTEM DEPENDS UPON THE EXISTANCE OF THE OTHER.
- THE PROPOSED LER RULE WILL PERMIT AN IMPROVED UNDERSTANDING OF THE NATURE OF REPORTED EVENTS
- THE SCSS WILL PERMIT GREATER USE OF REPORTED DATA

4288

QUESTION A. (CONTINUED) AS DATA ON EVENTS IS GATHERED BY LERS HOW WILL IT BE FORMULATED, AND INTERPRETED?

AEOD Analysis and Evaluation Program



A-287

QUESTION B. HOW WILL THE PROPOSED RULEMAKING ON IMMEDIATE REPORTING AND THE PROPOSED RULEMAKING ON LERs BLEND TOGETHER?

ANSWER.

- THEY WILL FORM AN INTEGRATED AND COORDINATED SET OF REPORTING REQUIREMENTS

- THE CRITERIA CONTAINED IN 50.73 ARE IDENTICAL IN MANY RESPECTS TO THE CRITERIA IN 50.72, "NOTIFICATION OF SIGNIFICANT EVENTS"

- WE ARE COMMITTED TO INSURE THAT THE TWO RULES ARE CONSISTENT

- THESE EFFORTS MAY RESULT IN COMBINING 50.72 AND THE PROPOSED 50.73 INTO A SINGLE RULE

A-290

QUESTION C. How will the new LER rule interface with the current LER system?

(i) Will the data base accumulated over the years by the present system be incorporated into the new system?

ANSWER.

- The SCSS will be backfitted to include some past LERs

- The existing NSIC data base of LERs will continue to be updated

- The abstracts from the NSIC data base will be available in SCSS as an output option

- The LER data base on the NIH computer will no longer be maintained by AEOD

- No existing data will be discarded

- No search capability will be lost

A-391

QUESTION C(11). WHAT, IF ANY, DATA IS LOST BETWEEN THE TWO LER SYSTEMS, WHAT INFORMATION COULD BE LOST AND WHAT IS ITS SIGNIFICANCE?

ANSWER.

- ASSUMPTIONS:
  - -- THERE ARE EVENTS THAT ARE NOT NOW REPORTABLE THAT SHOULD BE REPORTED
  - -- THE CONTENT OF THE EXISTING LER IS NOT, IN MANY CASES, ADEQUATE
  - -- WE CANNOT REALISTICALLY EXPECT TO INCREASE THE REPORTING BURDEN.
- CONCLUSION:
  - -- IF WE WANT MORE DETAILED REPORTS AND WE WANT REPORTS ON EVENTS THAT ARE NOT CURRENTLY REPORTABLE, WE MUST GIVE UP SOME REPORTS THAT ARE CURRENTLY RECEIVED.
- IMPLEMENTATION:
  - -- RELY ON NPRDS FOR REPORTS OF SINGLE, RANDOM FAILURES
  - -- NO LONGER REQUIRE ORTS ON CERTAIN TYPES OF EVENTS
    - --- OPERA A DEGRADED MODE
    - --- MINOR ICAL SPECIFICATION VIOLATIONS

4-292

QUESTION C(III). WOULD THE NEW LER SYSTEM PRECLUDE ANY STUDIES THAT HAD BEEN DONE IN THE PAST USING THE CURRENT LER DATA BASE?

ANSWER.

- WE BELIEVE THAT THE NEW LER SYSTEM AND AN EFFECTIVE AND EFFICIENT NPRD SYSTEM WILL INCREASE THE NRC'S AND THE NUCLEAR INDUSTRY'S ABILITY TO ANALYZE MOST OPERATIONAL EXPERIENCE DATA.

QUESTION C(III) CONTINUED. HOW COULD STUDIES BE PERFORMED UNDER THE NEW LER SYSTEM ON THESE EXAMPLES FROM NUREG-0572, "REVIEW OF LICENSEE EVENT REPORTS (1976-1978)- ADVISORY COMMITTEE ON REACTOR SAFEGUARDS.

ANSWER.

(A) UNAVAILABILITY OF VITAL SERVICES (D-II).

- EVENTS WHERE THE INITIATING OCCURRENCE IS CAUSALLY LINKED TO THE FAILURE OF ONE DIVISION (E.G., TRAIN) OF A SAFETY SYSTEM REQUIRED TO MITIGATE THAT INITIATING OCCURRENCE ARE SPECIFICALLY IDENTIFIED AS REPORTABLE NONCONSERVATIVE INTERDEPENDENCIES.

A-293

(B) UNAUTHORIZED BYPASSING OF INTERLOCKS (D-XI).

- UNAUTHORIZED BYPASSING OF CONTAINMENT BUILDING ACCESS DOOR INTERLOCKS WOULD BE REPORTABLE IF SUCH A PERSONNEL ERROR OR EQUIPMENT FAILURE ALONE COULD PREVENT THE FULFILLMENT OF THE SAFETY FUNCTION OF A STRUCTURE (I.E., THE CONTAINMENT) NEEDED TO CONTROL THE RELEASE OF RADIOACTIVE MATERIAL.

P. 294

(c) FAILURES IN AIR-MONITORING, AIR-CLEANING, AND VENTILATING SYSTEMS (D-XV).

- THE PROPOSED RULE REQUIRES THE REPORTING OF ANY EVENT THAT ALONE COULD PREVENT THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO SHUTDOWN THE REACTOR, REMOVE RESIDUAL HEAT, OR CONTROL THE RELEASE OF RADIOACTIVE MATERIAL

- TO THE EXTENT THAT THESE FAILURES COULD COMPROMISE THE INTEGRITY OF THE CONTAINMENT, COMPROMISE THE CONTROL OF THE AIRBORNE RADIOACTIVE RELEASES, AND COMPROMISE THE OPERATION OF OTHER KEY SAFETY EQUIPMENT; THEY WOULD BE REPORTABLE

A-295

QUESTION D. How will NPRDS component failure rates be factored into the LER data base? What will be the interface between the two systems? Will component failure rates be solely the role of NPRDS or will LERs also generate some of this information?

ANSWER.

- The NPRD and the LER systems are complementary and together represent an overall system for reporting operational experience (IOERS)
- NPRDS component failure rates and the NPRDS engineering data base may be used to support studies that also use the SCSS
- SCSS will not be used to calculate failure rates
- The NPRD reports for each component failure that contributed to the event described in an LER will be referenced in the LER
- A compatible set of component function identifiers and system descriptors will be used

QUESTION E. HOW ARE THE RESULTS OF THE AEOD STUDIES DISTRIBUTED TO THE INDUSTRY?

ANSWER.

By a variety of methods -

1. Some studies are sent to all affected licensees by NRR generic letters

2. All studies are sent to industry organizations (i.e., INPO and NSAC) who make further distribution

3. AEOD provides reports to the directly involved utility and NSSS vendor

4. All reports are placed in PDR which is closely monitored by industry

5. Some reports have been distributed by IE information notice

6. Summaries of AEOD cases appear in Power Reactor Events

A-297

QUESTION F. METHODS FOR CULLING OUT "HUMAN ERROR" LERS. CAN THEY BE BROKEN INTO SUBCATEGORIES (E.G., PROCEDURE ERROR, DESIGN ERROR, OPERATOR ERROR, MAINTENANCE ERROR)? HAS THE EXISTING SYSTEM BEEN MODIFIED UNDER THE PROPOSED RULE TO BETTER RECORD HUMAN ERROR; HOW? DURING THE DECEMBER 8, 1981 LER MEETING THERE WAS AN ALLUSION TO INPO STARTING TO DEVELOP AN "NPRDS-LIKE" SYSTEM FOR GATHERING EXPERIENCE ON HUMAN FAILURES: IS THERE ANY PROGRESS BY AEOD AND INPO ON THIS?

ANSWER.

- THE PROPOSED RULE REQUIRES THE LICENSEE TO REPORT ANY INSTANCES OF PERSONNEL ERROR, EQUIPMENT FAILURE, PROCEDURE VIOLATION, OR DISCOVERY OF DESIGN, ANALYSIS, FABRICATION, CONSTRUCTION, OR PROCEDURAL INADEQUACIES THAT ALONE COULD PREVENT THE FULFILLMENT OF A SAFETY FUNCTION

- THE PROPOSED RULE REQUIRES THAT THE NARRATIVE REPORT:

  -- INCLUDE OPERATOR ACTIONS AND PROCEDURAL DEFICIENCIES THAT AFFECT THE COURSE OF THE EVENT

  -- DESCRIBE THE EVENT FROM THE PRESPECTIVE OF THE OPERATOR

- SCSS WILL CAPTURE THIS INFORMATION USING SPECIAL CODES

A-298

QUESTION G. WILL EITHER THE LER SYSTEM OR THE NPRDS BE ABLE TO REFLECT ERRORS IN SOFTWARE? (A SEPARATE ISSUE FROM HUMAN FAILURES CAUSED BY OPERATORS OR MECHANICS)--SHOULD SUCH A SYSTEM BE ESTABLISHED IF NOT ALREADY IN PLACE?

ANSWER.

- LERs FOR REPORTABLE EVENTS ARE TO DESCRIBE ANY ERRORS IN DESIGN ANALYSIS, INCLUDING SOFTWARE, THAT CONTRIBUTED TO THE EVENT

QUESTION H. WILL ERRORS IN QUALITY CONTROL AND QUALITY ASSURANCE BECOME EVIDENT IN THE PROPOSED LER SYSTEM? WHAT MECHANISM WILL BE USED TO DISCOVER SUCH ERRORS?

ANSWER.

- LERs FOR REPORTABLE EVENTS ARE TO DESCRIBE ANY PERSONNEL ERROR OR PROCEDURAL INADEQUACIES, INCLUDING QUALITY CONTROL OR QUALITY ASSURANCE DEFICIENCIES, THAT CONTRIBUTED TO THE EVENT

- SCSS CAN IDENTIFY EVENTS WITH SPECIFIC CHARACTERISTICS THAT OCCUR OVER A PERIOD OF TIME

- THESE TRENDS OR PATTERNS OF RECURRING EVENTS ARE FREQUENTLY INDICATIVE OF QUALITY CONTROL OR QUALITY ASSURANCE ERRORS

A-300

QUESTION I. Other feasible means of gathering and recording data in LERs besides the use of the narrative.

(i) Could actual plant data from plant recorders be placed into the LER data pool?

ANSWER.

- The proposed LER rule does not require that the licensee provide actual plant data from the plant recorders

- The licensee may include such data if it contributes to the description of the event

- The staff may require that the licensee submit specific additional information including data from relevant plant recorders

- Such data would be noted in the COMMENTS field of the SCSS

- The actual data would be available from the NRC Document Control System

A-301

QUESTION I.(II) COULD SEQUENCE CODING BE DONE AT THE PLANT FROM RAW DATA, RATHER THAN PRODUCED FROM THE LER NARRATIVE?

ANSWER.

- CONSISTENT CODING OF INFORMATION IS ESSENTIAL
- IT WOULD NOT BE FEASIBLE TO MAINTAIN THE REQUIRED LEVEL OF CONSISTENCY IF THE CODING WERE DONE AT THE MORE THAN 70 OPERATING PLANTS

A-302

QUESTION I.(III) POTENTIAL ADVANTAGES AND DISADVANTAGES OF SYSTEMS USING RECORDS OTHER THAN THE NARRATIVE ACCOUNT OF AN EVENT.

A. BETTER ABLE TO RECONSTRUCT A PARTICULAR EVENT AND NOT LOSE SUBTLETIES OF THE EVENT WHICH MAY BE OF IMPORTANCE.

B. DIFFICULTIES INTRODUCED BY PLANT SPECIFIC VARIANCES.

ANSWER.

- THE PROPOSED RULE REQUIRES A CLEAR, SPECIFIC, NARRATIVE DESCRIPTION OF WHAT OCCURRED SO THAT KNOWLEDGEABLE READERS CONVERSANT WITH THE DESIGN OF COMMERCIAL NUCLEAR POWER PLANTS BUT NOT FAMILIAR WITH THE DETAILS OF A PARTICULAR PLANT CAN UNDERSTAND THE COMPLETE EVENT

- THE PROPOSED RULE ALSO INCLUDES A LIST OF SPECIFIC INFORMATION THAT MUST BE INCLUDED, AS A MINIMUM, IN THIS NARRATIVE DESCRIPTION

- THIS APPROACH WILL PROVIDE THE MOST COMPLETE AND ACCURATE DESCRIPTION OF THE EVENT

A-303

QUESTION J. METHODS OF COMPARING EVENT SEQUENCES AMONG THEMSELVES. CAN A TECHNIQUE BE DEVELOPED THAT WOULD SEARCH FOR HIDDEN CORRELATIONS AMONG EVENTS?

ANSWER.

- ONE OF THE MAJOR PURPOSES OF THE SCSS IS THE COMPARISON OF EVENT SEQUENCES
- STEPS IN A SEQUENCE ARE CLEARLY IDENTIFIED AS BELONGING TO THE SAME SEQUENCE AND THEIR LOCATION WITHIN THE SEQUENCE IS INDICATED
- ANY OF THE FIELDS OF ONE SEQUENCE CAN BE COMPARED WITH ANY OF THE FIELDS OF ANY OTHER SEQUENCE

QUESTION J. (CONTINUED) WILL THE SEQUENCE CODING SCHEME USE THE DATA BASE PREVIOUSLY COMPILED UNDER THE PRESENT LER SYSTEM?

ANSWER.

- THE EXISTING NSIC DATA BASE OF LERs WILL CONTINUE TO BE UPDATED AND WILL BE AVAILABLE USING THE DOE/RECON NETWORK
- THE ABSTRACTS OF LERs PREPARED BY NSIC WILL BE INCLUDED IN THE SCSS DATA BASE AND WILL BE AVAILABLE AS AN OUTPUT OPTION

A-305

<u>QUESTION K</u>. CAN SEQUENCE CODING TAKE OTHER FORMS BESIDES A LINEAR RELATIONSHIP AMONG EVENTS IN AN INCIDENT? COULD MULTIPLE CAUSATIONS OF AN EVENT BE IDENTIFIED?

<u>ANSWER</u>.

- WE ASSUME THAT "LINEAR RELATIONSHIPS" MEANS

  1 → 2 → 3 → ETC.

- THE SCSS CAN, AND FREQUENTLY DOES, CODE SEQUENCE FORMS OTHER THAN LINEAR RELATIONSHIPS AMONG OCCURRENCES

- IT IS ALSO POSSIBLE TO CODE A MULTIPLE CAUSATION FOR A SINGLE OCCURRENCE

QUESTION K. (CONTINUED) ARE INTERACTIONS BETWEEN SYSTEMS IDENTIFIABLE?

ANSWER.

- INTERACTIONS, INCLUDING LEAKS, BETWEEN SYSTEMS ARE IDENTIFIABLE
- THE COMPUTER COULD BE ASKED TO FIND ALL SEQUENCES THAT INCLUDE ANY TWO OR MORE SYSTEMS SPECIFIED BY THE USER
- SCSS INCLUDES A FIELD THAT INDICATES THE INTERFACING SYSTEM

QUESTION L. GIVEN THE PROPOSED LER SYSTEM PROCEEDS FORWARD, HOW WILL ADJUSTMENTS TO THE SYSTEM BE MADE IF THE NEED IS INDICATED BY EXPERIENCE? WILL IT TAKE FURTHER RULEMAKING?

ANSWER.

- THERE ARE SEVERAL WAYS THAT CHANGE IN THE LER SYSTEM COULD BE MADE
  - THE EDO MAY GRANT EXEMPTIONS WITHOUT RULEMAKING
  - THE REGULATORY GUIDE CAN BE CHANGED WITHOUT RULEMAKING
  - THE LER RULE CAN BE CHANGED THROUGH RULEMAKING

A-308

QUESTION L. (CONTINUED) HOW ABOUT ADJUSTMENTS TO THE SEQUENCE CODING SYSTEM, WILL IT TAKE RULEMAKING?

ANSWER.

- CHANGES TO THE SCSS CAN BE MADE WHENEVER THE PROGRAM MANAGER (I.E., AEOD) DECIDES THAT A CHANGE IS NEEDED

- CHANGES TO THE SCSS WILL NOT REQUIRE RULEMAKING

Response To ACRS Questions On Equipment Qualification

Presented At The March 4, 1982 Meeting Of The ACRS

Paul Shemanski
Goutam Bagchi
Equipment Qualification Branch, NRR

The Following Questions Were Raised During The Committee's Review Of The Comanche Peak Plant. Comanche Peak Was Required To Comply With The Qualification Testing Of IEEE-323-1974. It Is Important To Determine If The 323-1974 Testing Requirements Do Achieve The High Reliability Desired In Nuclear Plant Safety Related Electrical Components. To This End:

Question 1

Does Environmental Qualification Really Demonstrate An Ability To Function In The Event Of Conditions Such As Potential Flooding, Vibratory Motion From A Seismic Event, Gamma Radiation, High Temperture, High Humidity, Or Live Steam Conditions?

- SCOPE OF QUALIFICATION
  - ENVIRONMENTAL, SEISMIC & DYNAMIC EFFECTS
  - ELECTRICAL & MECHANICAL EQUIPMENT
- QUALIFICATION PARAMETERS
  - TEMPERATURE
  - PRESSURE
  - HUMIDITY
  - SUPERHEATED STEAM
  - RADIATION
  - SEISMIC & DYNAMIC
  - FLOODING
  - AGING (THERMAL, MECHANICAL, RADIATION)
  - CHEMICAL SPRAY
- APPLIES TO NORMAL OPERATION, MAINTENANCE, TEST, AND ACCIDENT ENVIRONMENTS
- IEEE 323-1974 TEST
  - ACCIDENT AT END OF QUALIFIED LIFE
  - SEQUENTIAL TEST
- EXAMPLE - TRANSMITTER
  - ACCEPTANCE TEST
  - THERMAL AGING
  - FUNCTIONAL AGING
  - RADIATION EXPOSURE (NORMAL + ACCIDENT)
  - SEISMIC AGING
  - LOCA TEST (318 °F/86.7 PSIA/8 HRS. TO 150 °F @ 14 DAYS)
  - POST ACCIDENT TEST (203 °F, 23 DAYS)
  - BASELINE TEST

Question 3

The Value Of This Massive Qualification Program Could Be Negated If A Weak Link Exists Because Some Element Had Not Been Properly Qualified. The Weak Link Question Applies Especially To Electric Circuitry, Signal Transmission Devices Or Activation Equipment. For Example, Failure Of A FireDamper Actuator Might Negate The Value Of A Qualified Fire Wall.

- Scope Of Qualification Based On GDC-4
  - Environmental, Seismic & Dynamic Effects
  - Electrical & Mechanical Equipment
- Applies To Normal Operation, Maintenance, Testing And Postulated Accidents
- Equipment/Systems Safety Design Functions
  - Emergency Reactor Shutdown
  - Containment Isolation
  - Reactor Core Cooling
  - Containment And Reactor Heat Removal
  - Prevention Of Radioactive Release To Environment
- Inside Containment
  - LOCA
  - HELB (MSLB)
- Outside Containment
  - (HELB)
- Objective Of Environmental Qualification Program
  - To Provide Reasonable Assurance That The Safety Related Equipment Required To Mitigate LOCA's/HELB's Can Perform Its Intended Function In The Most Limiting Environment In Which IT Is Expected To Function

Example Of Qualification Testing For The Containment Isolation Function

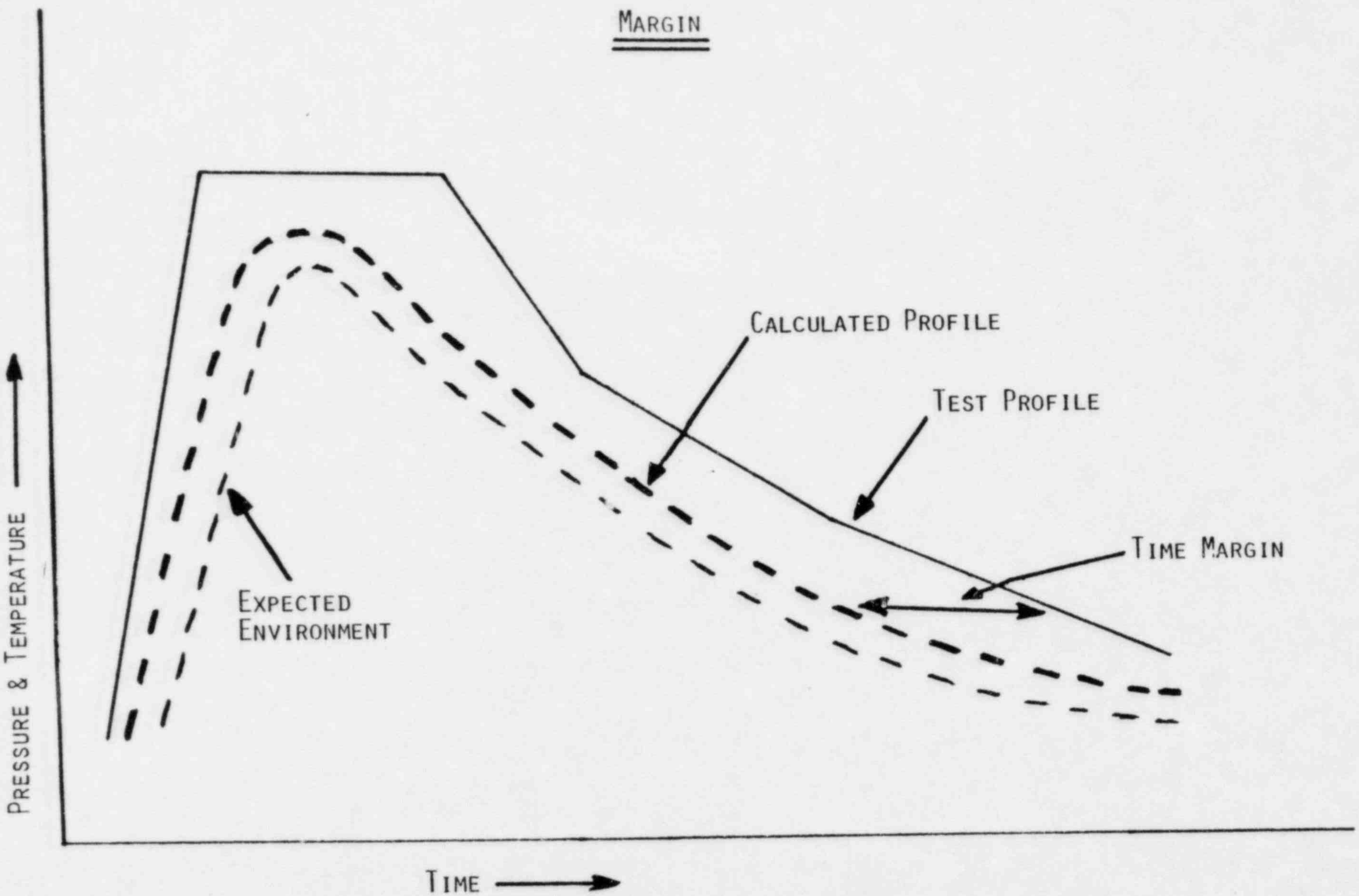
- Series Elements (Major)
  - Pressure Sensor
  - Electrical Cabling
  - Penetration Assembly
  - Pressure Transmitter
  - Valve Actuator
  - Valve Mechanism
- All Equipment That Performs A Safety System Must Be Qualified
- Components Of The Item Being Qualified Are Examined For Any <u>Weak Link</u> That Might Exist - Organic Materials Are Looked At Closely Because Of Their Susceptibility To Aging From Temperature And Radiation
- Reliance On Redundancy And Physical Separation Coupled With Qualification Testing Allow Us To Make A Judgement As To Whether Or Not Reasonable Assurance Exists That The Equipment Can Perform Its Intended Function

Question 4

Does "Type" Testing Of Selected Components And Equipment Provide A Reliable Test For Production Equipment That Will Experience A Variety Of Operating And Maintenance Transients During A 40-Year Lifetime, Including Potential Abuses From Operator/Maintenance Errors, Physical Damage, Etc.?

- TYPE TEST - DEFINITION
- MARGINS
- MAINTENANCE/SURVEILLANCE PROGRAM
  - MANUFACTURER'S RECOMMENDATIONS
  - PLANT EXPERIENCE
  - LER'S
  - TREND ANALYSIS

Margin



Calculated Profile
Test Profile
Time Margin
Expected
Environment
Pressure & Temperature
Time

QUESTION 2

DOES THE SPECIFIC LOCATION OF SUCH EQUIPMENT IN THE CONTAINMENT BUILDING INFLUENCE THE POTENTIAL FOR UNANTICIPATED OR EXTREME ENVIRONMENTAL CONDITIONS AND THE POTENTIAL FOR INTERACTION WITH SYSTEMS NOT COVERED BY THE QUALIFICATION PROGRAM SUCH AS FIRE MAINS, SERVICE AIR, COMPONENT COOLING WATER SYSTEMS, ETC.?

A320

EQUIPMENT QUALIFICATION:

- IDENTIFY EQUIPMENT TO BE QUALIFIED
  - EMERGENCY REACTOR SHUTDOWN
  - CONTAINMENT ISOLATION
  - REACTOR CORE COOLING
  - CONTAINMENT HEAT REMOVAL
  - CORE RESIDUAL HEAT REMOVAL
  - PREVENTION OF SIGNIFICANT RADIOACTIVE RELEASE
  - SUPPORTING SYSTEM
- ESTABLISH ENVIRONMENTAL PARAMETERS

  TEMPERATURE, PRESSURE, HUMIDITY, RADIATION, CHEMICAL SPRAY, SUBMERGENCE, AGING, VIBRATION, SEISMIC AND DYNAMIC
- PERFORM TYPE TESTS PER IEEE 323-74 STANDARD

  TEST PLAN, MOUNTING, CONNECTIONS, MONITORING, MARGIN, TEST SEQUENCE, AGING, RADIATION, VIBRATION, OPERATION UNDER NORMAL AND ACCIDENT CONDITIONS INCLUDING EXTREME ENVIRONMENTS
- ESTABLISH SURVEILLANCE AND MAINTENANCE PROGRAM BASED ON QUALIFIED LIFE

Influence Of Specific Location Within Containment:

Integrated Radiation Dose Rate And Level Based On

- Containment Atmosphere
- Containment Surface Dose
- Dose Near Sump Water
- Proximity To Recirculation Lines
- Component Specific Shielding Effect

Submergence Effect

- Elevation Of Equipment Within Containment

Seismic And Dynamic Load

- Elevation Of Equipment Within Containment Distance Away From Initiating Event

Unanticipated Environmental Conditions:

- Maintain Awareness Of Operational Experiences
  - LER, AEOD Reports

- Upgrade Qualification Requirements Consistent With Safety Significance Of Unanticipated Events

Interaction With Unqualified System:

Fire Mains:

Within Containment

Redundency, Separation, Qualification To Chemical Spray And Submergence

Outside Containment

Redundency, Separation, Protective Housing

Service Air:

Upgraded Criteria For Air Quality, Plant Specific Fixes For Air Binding Of Service Water, Greater Awareness Of Interaction Through Operating Experience

Component Cooling Water Systems:

Within Containment

Generally Not Safety Grade In PWR.

PWR Containment Environment Envelopes Adverse Effects Of Failure In Non-Safety Parts.

Generally Safety Grade In BWR.

Local Environment Caused By Leakage From Scram Discharge Line Should Be Used To Qualify Appropriate Equipment (NUREG-803)

Outside Containment:

Generally Safety Grade

No Adverse Interaction Anticipated.

QUESTION 5

PLEASE COMMENT ON THE SUGGESTION OF PERFORMING A SURVEY (USING NRC CONTRACTOR) OF A PLANT (SUCH AS COMANCHE PEAK) TO DETERMINE IF THE 323-1974 REQUIREMENTS WILL ACHIEVE THE DESIRED HIGH RELIABILITY NOTED ABOVE.

Equipment Reliability Achieved Through IEEE 323-1974 Requirements

Plant Survey

Staff Site Audit Of Comanche Peak

- Staff And Contractor Assistance
- Selected Electrical And Mechanical Equipment Review
- Inspection Of On-Site Installation
- Comparison Of Test Vs. Installation
- Documentation Review To Establish Design Vs. Qualification

Survey Of A Brand New Plant

- Equipment Reliability Can Only Be Determined Through Successful Performance During A Challenge
- Equipment Operability Assurance Encompasses
  - Proper Specification Of Environmental And Performance Parameters
  - Protype Testing To Proper Standards
  - Quality Of Equipment Manufacturing, Testing, And Installation
  - Well Defined Maintenance And Surveillance Schedules

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Equipment Reliability Achieved Through IEEE 323-1974 Requirements

Recommendations:

- Select Equipment Types:
  - Equipment In Harsh Environment
  - Electrical Components
    - Transmitters, Switches, Relays, Cabinets
  - Mechanical Equipment
    - Pump, Prime Mover, Accessories
    - Valve, Actuators, Accessories
- Review Manufacturer And Vendor Tests:
  - Equipment Performance During Test
  - Design Changes
  - Equipment Modifications
- Equipment Performance Data Base:
  - Performance From Experience Data
  - Correlation Between Reliability And Standards
- Qualification Beyond Type Tests:
  - Sample Testing Of Selected Production Items Such As Splices, Terminal Blocks

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

March 4, 1982

APPENDIX XVI
MEETING SUMMARY - SUBCMTE ON SAFETY
PHILOSOPHY, TECHNOLOGY & CRITERIA
2/26/82

MEMORANDUM TO: ACRS Members

FROM: J. M. Griesmeyer, Staff Engineer

SUBJECT: MEETING SUMMARY - SUBCOMMITTEE ON SAFETY PHILOSOPHY, TECHNOLOGY AND CRITERIA, FEBRUARY 26, 1982

The Safety Philosophy, Technology and Criteria Subcommittee met on Friday, February 26, 1982, in Washington, D.C. The purpose of the meeting was to review the Power Authority of the State of New York (PASNY) proposal for a Systems Interaction Study (SIS) of the Indian Point 3 (IP3) Nuclear Power plant and to review the NRC systems interaction program. The principle attendees were:

ACRS	NRC
D. Okrent	A. Thadani
J. Ebersole	F. Coffman
M. Bender	J. Conran
D. Ward	
J. M. Griesmeyer	**PASNY**
G. R. Quittschreiber	J. Lambersky
	Y. Kishinevski
	G. Wilverding
	K. Sander

The highlights of the meeting were:

1. G. Wilverding/PASNY gave a short history of the licensing activity for IP3 and the development of the SIS proposal which took 1-1/2 years and was delayed by some last minute changes.

2. Y. Kishinevski/PASNY presented the SIS proposal (a copy of his written remarks is attached). The study is to address events of unintensional adverse interactions that effect the safety of the plant by one system acting upon one or more other systems, with emphasis on non-safety - safety types of interactions. The proposal includes a systematic search for hidden interconnections or couplings that link safety and non-safety systems, and evaluation of the effects of non-safety system failure or maloperation propagated into the safety systems. The objective of the submitted proposal is to describe the methodology and evaluation criteria to be used to identify and evaluate systems interactions and to apply them to the IP3 Auxiliary Feedwater System as an example.

A strigent threshold search criterion will be used for identifying adverse systems interaction, which will then be evaluated by considering the licensing basis for IP3.

Several methods are needed to perform an adequate review of systems interactions. All involve a three step process of (1) selecting specific systems for detailed evaluation, (2) identification of dependencies or commonalities, and (3) evaluation of the systems interactions through the determination of their relative importance to safety. For the IP3 proposal, the conditions considered to be adverse and have a significant potential for feeding to core damage are: failure to achieve or maintain reactor subcriticality, failure to remove decay heat, failure of the reactor coolant pressure boundary and breach of containment integrity. The systems needed to prevent these conditions are to be addressed in the IP3 SIS.

For the connected systems or process-coupled portion of the study, a dependency analysis technique is used as the primary means of identifying systems interactions. Event tree and fault three information will also be used. The first step of the search is accomplished by developing functional shutdown logic diagrams that describe the general functions necessary to prevent core damage. The second step, identification of dependencies or commonalities, is accomplished by further developing safety system auxiliary diagrams and auxiliary safety systems commonality diagrams to provide the link between the functional systems and support systems necessary to achieve a safety function.

Evaluation of the systems interaction, through the determination of relative importance to safety, is accomplished using deterministic logic; i.e. failure modes and effects analysis.

The scope of failures or malfunctions includes consideration of the results of the following abnormal events or actions: loss of motive or control power of electrical, hydrolic or pneumatic type, chemistry, cooling, lubrication and operating vibratory motion.

The scope of failures to be excluded are: operator-induced failures, equipment unavailability due to testing or maintenance, sabotage and act of war. However, systems interactions caused by the operator acting in accordance with faulty information or procedures are included.

For the nonconnected portion of the study, the possibility of adverse interactions transported in the environment due to spacial or physical proximity during design basis events are investigated by performing a systematic plant walkdown. The first step is to classify nonconnected specially coupled systems, components and structures as either sources or targets. Equipment that requires protection from potential event induced transactions are designated as targets. An interaction is identified whenever the event induced behavior of a source could lead to a detrimental effect on a nearby target. Potential sources are evaluated as a part of the program to determined if events can credibly led to detrimental interaction with a target.

The plant walkdown is performed by two interdisciplinary teams of experienced engineers. Six categories of event-induced interactions are investigated and the information is documented.

The next refueling for IP3 is to be in March and it is tentatively planned to perform the walkthrough while the containment is accessible. For $1 million by late 1982 a draft report that identifies the interactions scheduled to be completed. The balance of the study would take about 2-1/2 years and $2-1/2 million for evaluation, corrections and resolutions. This would not include the cost of safety improvements suggested by the study.

3. The Subcommittee Members questioned the limitation of the study to non-safety system sources. M. Bender said that the relevant question should be what systems may effect the safety of the plant and the order in which they may be effected, and that the distinction between safety and non-safety systems may be a distortion of that concern. J. C. Ebersole said that the separation between functional design groups can lead to safety-systems interactions and that such interactions have not been systematically looked at before.

   Y. Kishinevski said that safety systems interactions are addressed by other regualtions that (1) make it unlikely that safety systems will become sources, and (2) require the ability to withstand single failures. He said that PASNY planned to take credit for what has already been done.

   D. Okrent said that it is not clear what the criteria should be. He said that he was not asking PASNY to include failures of equipment that is not qualified for a particular environment when it was throught to be qualified, but he thought they should look for conditions for which equipment was not qualified or intended to be.

4. M. Bender was concerned that the search criteria for identificaiton of systems interactions was too stringent and evaluation of all those identified would be impossible. J. C. Ebersole suggested that for seldom used systems the criteria may be too conservative but for systems that are often needed and used it may not be stringent enough. He said that he was concerned that the seldom used ESFs may receive attention that should be given to more often used systems.

5. J. C. Ebersole said that the homework needed before the walkthough could be prodigous and suggesed that they proceed the walkthough by a review

of the designers intensions to avoid obvious interactions. M. Bender said that the walkthrough, while important cannot identify all non-connected systems interactions because much equipment may be hidden.

6. J. Conran/NRR presented the Staff review of the IP3 Program by first identifying the acceptance standards by which they reviewed the proposal, then evaluating the proposal and finally describing the planned monitoring effort and audit method.

   The acceptance standards were based on explicit consideration of required basic safety functions:

   - Ability to achieve and maintain entire core subcritical
   - Ability to transfer decay heat to ultimate heat sink
   - Ability to maintain integrity of the reactor coolant pressure boundary
   - Ability to provide engineered safety features unimpaired

   The program should provide for systematic and comprehensive analysis of interactions that adversely effect the basic safety functions (with emphasis on non-safety system/component failure effects). It should treat various types of interactions (i.e. interconnected and nonconnected systems interactions and humanly coupled systems interactions) and it should employ suitable combinations of analysis techniques.

   In the NRR review of the original PASNY proposal for the IP3 SIS eleven questions were raised that have been negotiated to resolution in the present proposal. It is generally consistent with the Staff's acceptance standards. While the Staff would not characterize the study as full scope, they felt it was very broad in scope and it was acceptable to them.

7. J. Conran said that the emphasis on non-safety system and components as sources in the systems interaction program has several historical roots among which is the fact that non-safety systems have not been subject to extensive analysis in the past. For best return on the use of resources it was felt that concentraction on non-safety systems would be reasonable; not implying that safety systems have been given completely adequate treatement but that they normally have been analyzed extensively. D. Okrent expressed the concern that emphasis on non-safety sources may translate in practice to indefinite postponement of the treatment of safety systems as sources for systems interactions.

   M. Bender said that the demarcation between safety and non-safety systems was too vague. He recognized the need to limit the scope of the SIS and said that it might reasonable to look at systems that have not been looked at before. However, it might be possible to find better criteria that consider the classes of events that have been problematic in the past, and the frequency at which a system is needed or that concentrate on high probability events that may have been overlooked.

The Subcommittee felt that pirorization was a problem but that it was not clear that non-safety sources are more important than safety sources.

D. Okrent suggested that if safety systems have been treated to some extent in the past it may not be difficult to include them as sources in a SIS.

8. F. Coffman NRR described the proposed NRC system interaction program plan. The main objective of the pilot review projects is to evaluate methodologies for systems interaction studies. The plan calls for four different teams to study four different plants using two different methodologies for the screening and evaluation of systems interactions. The four plants that have been tentatively identified as Midland 2, Perry, San Onofre 3, and Seabrook. They are all NTOLs so there should not be much interference with operations and the radioactivity exposure will be kept low. In addition, each vendor is represented.

   The two methodologies were failure modes and effects analysis (FMEA) with fault trees, and a matrix based digraph method. The Staff preliminary evaluation of the procedures has lead the Staff to believe that the FMEA with fault trees is very much dependent upon the analyst and that the matrix based methods may not be. In addition, the fault trees became too large and inscrutiable in the Sandia SIS and were not able to identify some known interactions. The digraph matrix methods can be automated to some extent by manipolations of the matrices to maintain scrutibility while not severly limiting to scope.

   The NRR Systems Interaction Program has identified general requirements for SIS's and is testing the two procedures to fulfill the requirements.

9. The Subcommittee questioned how good the comparison of methodologies could be if a different team is to investigate each of the four plants. The wide variations in the plants and in the application of the methods by the different teams will obsure the difference due to the choice of methodology.

10. D. Okrent noted that while the NRC after the TMI accident has said that its primary concern will be with operating plants, the pilot program plans to address only NTOL's. He asked when the operating plants would be considered. They may have more potential systems interactions that the NTOLs which were designed with more consideration of such effects.

   A Thadani said that NRR may move along with the Systems Interaction Program before the pilot studies are complete. The expedited schedule would depend upon the IP3 SIS progress and results and could rely on the methodology developed for the IP3 SIS.

11. D. Okrent asked if there were some quick investigations that the licensees could do to pick up some obvious interactions and which could complement the program. F. Coffman said that he would encourage these quick studies, but the NRC would be swamped with reviews if credit were to be given. Both D. Okrent and M. Bender suggested that the NRC might not need to evaluate and review the short studies done by licensees. The studies would be for their good and might save them considerable problems if some quick fixes are identified.

12. D. Okrent asked the Staff if there were plans to provide feedback to the licensee's if the studies that are done discover some generic problems. He also expressed the concern that the NRC program could ask for studies that are too large and too difficult to be done.

13. M. Bender expressed concern that the screening methods may not be fruitful if the evaluation of the many systems interactions becomes too difficult.

14. D. Okrent told the PASNY representatives that the full Committee would have difficulty in scheduling a complete discussion of the IP3 SIS at the March meeting, but that the full Committee may approach it in April with the NRC Systems Interaction Program. He also said that the Subcommittee felt that the IP3 SIS should include safety systems as sources.

15. In Summary, Mr. Bender said he felt that in general, PASNY had made a good effort. He felt that the following points should be addressed:

- Support systems are important - what happens when they only partially operate.
- Normally operating systems are important for systems interactions.
- Out of sequences events should be addressed.
- Some interactions of interest may be too sophisticated and complex to be treated in a broad brush approach - there is a need to be selective and to focus the effort.
- Walkthoughs should be required of all plants.
- If items are to be left out of th IP3 SIS the reasons, such as previous coverage by other requirements, should be documented.

16. J. C. Ebersole said that safety systems as sources have not been previously treated in detail. He felt that the AFWS sample study for IP3 contained too much information and that it needed to be focused. He suggested that it would be possible to do a SIS more simply by addressing the enviromentally weakest components.

17. D. Ward reiterated that he felt that PASNY needed to categorize the sources of interactions properly and that safety vesus non-safety is not appropriate. To the Staff, he said that there are some cases were "non-safety" systems should be considered as targets, that sneak circuit analysis should be adapted to help in systems interaction studies, and that computer software is becoming important in reactor operations and may be a very important factor in systems interactions.

18. PASNY expressed concern at the comments made by the Subcommittee members and reiterated the safety systems as sources are treated indirectly by current requirements that (1) largely prevent safety systems from becoming sources and (2) eliminate many interactions by the application of the single failure criterion. They questioned whether they should continue with their plant walkthrough during their refueling that starts in March. The Subcommittee indicated that they strongly felt that the walkthrough should not be delayed.

Attachment: As Stated

PASNY Presentation
Safety Philosophy, Technology, and Criteria Subcommittee meeting, February 26, 1982

Narrative Draft

Indian Point 3 Nuclear Power Plant - Systems Interaction Study

This presentation describes an effort intended to address the concerns of systems interactions. The identification, evaluation and correction or modification of adverse systems interactions will enhance the level of safety and permit a continued operation of the Indian Point 3 Nuclear Power Plant at full power. To derive a working definition of systems interaction, it is necessary to consider a number of associated concepts. In the design of a nuclear power plant, provisions are made to ensure that the release of radioactivity to the environment is extremely unlikely by providing independent ways in which the safety functions can be performed. These provisions are expressed in terms of redundancy and diversity so that multiple independent system failures would not necessarily result in a safety function failure. Systems, which support safety functions, may be designed to interact with each other. These interactions are intentional. An adverse interaction results when the conditions of one system affect or degrade the ability of another system to perform its safety function. Therefore, systems interactions are those events that affect the safety of the plant by one system acting upon one or more other systems in a manner not intended by design, with emphasis on non-safety - safety types of interactions. A systems interaction analysis involves:

1. The systematic search for hidden interconnections or couplings that link safety and nonsafety systems in the plant.
2. The evaluation of the effects of a nonsafety system failure or maloperation propagated into the safety system by such interconnections or oouplings.

The systems interaction process is an attempt to reevaluate, in a systematic fashion, those potential events which direct effect or natural cascading features could reduct plant safety margins.

The objectives of this recently submitted report are:

1. To describe the methodology and evaluation criteria to be used to identify and evaluate systems interactions.
2. To apply these methodology and evaluation criteria to a systems interaction review of the Indian Point 3 Auxiliary Feedwater System. This methodology and evaluation criteria then will be utilized to evaluate other systems that will be identified for review.

During review of the preliminary issue of the report by the systems interaction section of the RRAB, it was agreed explicitly that the threshold for identification of adverse systems interactions will be a nonsafety system or component failure that leads to the defeat of at least one train of a safety system or engineered safety feature, even if the remaining train of the affected system or ESF could perform the intended safety function.

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This is the most stringent application of the single failure criteria currently used in the licensing review processes; but, it is only to be used as a systems interaction search criterion. Systems interactions identified by applying this search criterion will be evaluated considering the licensing basis for Indian Point 3.

For the connected systems or process-coupled portion of the study, a dependency analysis technique is used as the primary means for identifying systems interactions. In addition, event-tree, fault-tree and information for individual systems will be considered when searching for and evaluating systems interactions. For the nonconnected system portion of the study, the possibility of adverse interactions transported to the AFS via spatial or physical proximity considerations during the design basis events, such as earthquake, tornado, fire, high-energy pipe rupture, internal or external flooding and internally - or externally - generated missiles were investigated. These latter events were investigated for the interactions via the plant walkthrough and by a review of reports previously prepared on these subjects.

Through an evaluation of methodology techniques prescribed by state-of-the-art reviews, it was concluded that any one method cannot perform an adequate review for determining adverse systems interactions. However, all the methods evaluated included a process of sifting out adverse systems interaction by (1) selecting specific systems for detailed evaluation, (2) the identification of dependencies or commonalities, and (3) evaluation of the systems interactions through the determination of their relative importance to safety. It is this three-step process which provided the foundation for performing a systems interaction study for the Indian Point 3 Nuclear Power Plant.

The initial task was to determine if an adverse systems interactions could occur and, if so, whether or not a significant impact on the degradation of the reactor core and the release of an unacceptable level of radioactivity to the site environs could result. Those conditions considered to be adverse and have a significart potential for leading to core damage are: failure to achieve or maintain reactor subcriticality, failure to remove decay heat, failure of the reactor coolant system pressure boundary and breach of containment integrity.

Since the Auxiliary Feedwater System is required for decay heat removal, it was, therefore, chosen as a test system to apply the proposed investigative methodology and evaluation criteria.

Interconnected systems are defined as those mechanical and electrical complexes which are process-coupled to one another physically via piping, instrumentation tubing or electrical wiring.

The first step in searching for adverse systems interactions is accomplished by developing functional shutdown logic diagrams that describe the general functions necessary to prevent core damage. The logic diagrams are based upon system descriptions, Instrumentation & Control logic diagrams, and Electrical schematic, block and wiring diagrams.

The second step, the identification of dependencies or commonalities is accomplished by further developing the functional system shutdown logic diagrams into subsystems more commonly referred to as auxiliary diagrams. The auxiliary diagrams provide the link between the functional system and support systems necessary to achieve a safety function.

The third step, evaluation of the systems interaction through the determination of their relative importance to safety, is accomplished by using deterministic logic; i.e., failure modes and effects analysis.

The shutdown logic diagrams, safety systems auxiliary diagrams and the auxiliary safety systems commonality diagrams are basically devices employed for identifying the safety and support systems, including non-safety systems that are to be analyzed for interconnections, and for correlating and combining the results of FMEAs on individual systems in order to understand and portray how interconnections, couplings and dependencies among all systems can propagate nonsafety system failures into the safety system.

For interconnected mechanical or electrical complexes, process-connected systems dependency analyses form the basis for the systems interaction analysis. The shutdown logic diagrams, safety system auxiliary diagrams and the auxiliary safety system commonality diagrams describe the combinations of components which, if the loss of redundancy is an unacceptible failure, would result in loss of any of the four basic safety functions. These documents are the vehicles for the identification and evaluation of systems interactions which could significantly compromise the safety of Indian Point 3 Nuclear Power Plant.

As a supplemental device in searching for systems interactions and as one of the principal methods for the evaluations of systems interactions identified, consideration will be given to the use of fault-trees on individual systems already available from the Zion/Indian Point 3 PSS Analysis. Not all systems interactions identified will require the use of fault-trees for evaluation. Engineering judgment, based on appropriate reflection of existing deterministic criteria, will be used in some cases.

Fault-trees are used to determine the likelihood of a failure of the various systems identified in the event-tree accident paths. A fault-tree starts with the definition of an undesired event, such as the failure of a system to operate and then determine, using engineering and mathematical logic, the ways in which the system can fail.

Having described the safety functions and the specific plant safety systems, the next step is to identify the required responses or safety actions that must be accomplished in order to achieve a safety function. The sensed variables are identified that cause or require the system response. In cases where the system does not automatically respond, operator action is required to initiate the identified system. As the safety systems and their actions are identified, they are arranged in functional order, forming success paths or operation sequences, leading to the required safety function. The arrangement of success paths becomes the shutdown logic diagram for that event.

After completion of the shutdown logic diagram for a postulated event, each safety system displayed on the diagram is analyzed to determine the specific support requirements necessary to produce the safety action. The Analyst refers to the shutdown logic diagram to determine every sequence in which a safety system is required, thereby, ensuring that all support requirements are identified. After identification of the support requirements, the plant systems that provide these support requirements are identified. These systems are the auxiliary safety systems. A safety system auxiliary diagram is then prepared on which the prime safety system and its auxiliary safety systems are displayed.

To complete the safety system auxiliary diagram, the Analyst must review the shutdown logic diagrams for all the postulated events to identify all safety sequences in which the subject auxiliary safety system appears.

After completion of the shutdown logic diagrams for each postulated event and the safety systems auxiliary diagrams, the auxiliary safety system commonality diagram for each auxiliary safety system is developed. This diagram indicates all the safety systems that a given auxiliary safety system supports. The auxiliary safety system commonality diagram allows evaluation of the overall plant response to the operations of each auxiliary safety system.

The failure mode and effects analysis provides for the evaluation of partial success or failure, or abnormal operation. From these failure mode and effects analyses, performed in conjunction with the fault-tree analysis, interconnected systems interactions can be identified and evaluated.

The scope of failures or malfunctions includes the consideration of failures caused by adverse interactions of interconnected systems and components that result as a direct consequence of abnormal events or actions. These abnormal events or actions are: loss of power (both motive and control power of the electrical, hydraulic and pneumatic type, chemistry (fluid purity), cooling (including HVAC equipment), lubrication and operating vibratory motion.

The scope of failures to be excluded are: operator-induced failures, equipment unavailability due to testing or maintenance, sabotage and act of war.

During the review of the preliminary submittal of this report, the NRC systems interaction staff was concerned about the treatment of nonsafety control system failure effects, nonsafety power system failure effects and nonsafety instrumentation failure effects. This type of systems interaction has been termed induced operator error and involves a set of circumstances in which (1) a nonsafety system failure causes the massive loss of control instrumentation display, and (2) the operator is assumed to act correctly, procedurally speaking, on the basis of incorrect readings produced by the initiating failure.

In reviewing this concern, it has been concluded that the failure of nonsafety systems is adequately considered; a comprehensive dependency analysis will uncover nonsafety control failures that may lead to operator-induced failures.

Nonconnected systems are identified as all safety and nonsafety Mechanical, Electrical and Civil Systems, which are associated with the physical arrangement or spatial coupling of each other.

The identification of nonconnected spatially coupled systems is based upon the review of Plant General Arrangements. The Plant General Arrangement and its association with spatially coupled systems is determined by performing a systematic plant walkdown.

When considering systems interactions of nonconnected systems for the design basis events previously described, the structures, systems and components, important to safety, shall not be prevented from carrying out their required safety functions because of physical, mechanical, fluid or electrical interactions caused by the event-induced failure of equipment not qualified or designed to withstand event consequences. Nor shall they lose the redundancy required to compensate for random single failures because of such interactions.

An event will include the following: earthquake, up to and including the safe shutdown earthquake, pipe failure, pipewhip, jet impingement, jet reaction, severe environment, temperature, pressure, humidity, physical impact from missiles generated internally and externally, flooding from internal failures or external effects due to rain, snow, etc., tornado depressurization/overpressurization, and fire.

For nonconnected systems interactions, the first step is to classify nonconnected spatially coupled systems, components and structures as either a source or a target.

Equipment which requires protection from potential event-induced transactions are designated as targets.

For seismically-induced events, the sources of detrimental interactions are any nonseismically-supported or qualified structures, systems or components which, due to their proximity or connection to the targets, may interact through physical, mechanical, electrical or environmental means to compromise the integrity or operability of the target.

A plant walkdown is performed by an interdisciplinary team of experienced engineers. During the inspection, possible interactions are postulated for source equipment that might affect the targets to be protected. Consideration is given to local equipment arrangement and geometry, and to possible results of these failures. Once the field system evaluation has been completed, the following information is documented:

a. Location of potential interaction

b. Components and systems involved in the potential interaction are identified on an interaction matrix form and documented on the interaction documentation forms.

c. The specific criteria used for the evaluation, which includes the type of interaction documented on the documentation forms.

d. A photographic record for each identified interaction is made. The photograph is crossreferenced with the interaction matrix form and the interaction documentation; a small arrow indicates the general location of the target.

All such interactions are listed and evaluated, using the established failure criteria previously described.

The plant walkdown by the interdisciplinary team considers the effects of intercompartmental interactions. The potential intercompartmental interactions are identified, and relevant data, such as location is documented. The walkdown team physically inspects adjacent compartments that may have interaction effects.

An interaction is identified whenever the event-induced behavior of a source could lead to a detrimental effect on a nearby target. An assessment is made of the induced behavior of the sources. An interaction is not identified by the field walkdown team, if it can be established by inspection that no credible failure mode can be induced in the sources by events of credible severity, which would violate the acceptance criteria.

Event-induced interactions identified will be in one or more of the following categories:

a. Contact between a source and a target that would compromise operability.

b. Fluid leakage from one or more sources that would degrade the environment of the target component and thereby prevent it from properly funcioning.

c. Contact between a missile generated by a source and a primary target that would compromise the pressure boundary of a secondary target component.

d. Contact between a missile generated by a source and a primary target that would compromise operability of a secondary target component.

e. Failure of nonsafety-related electrical equipment that would compromise the operability or integrity of target components.

f. Secondary effects or cascading influences caused by any of the above-mentioned interactions.

The effects of pipe failure-induced systems interactions are consistent with the guidelines provided in Standard Review Plans 3.6.1 and 3.6.2 and Regulatory Guide 1.46.

The effects of internally and externally-generated missile-induced systems interaction are consistent with the guidelines provided by Standard Review Plans 3.5.1, 3.5.2 and 3.5.3.

The effects of flooding-induced systems interactions are consistent with the guidelines provided by Standard Review Plans 3.4.1 and 3.4.2.

The effects of fire-induced systems interactions is consistent with the guidelines provided in Standard Review Plan 9.5.1 and companion Branch Technical Position APCSB-9.5-1.

The effects of severe environment-induced systems interactions are consistent with the guidelines provided by Standard Review Plans 3.3, 3.4, 3.5, 3.6 and 3.11.

As previously mentioned, the evaluation and acceptance criteria to be employed for evaluating identified systems interactions are consistent with the licensing basis for this plan; that is to say, the single failure criteria for nuclear power plants, as it was applied to the licensing of Indian Point 3 governs.

The evaluation of interconnected system interactions and their effects on plant safety will be based on satisfying the failure criteria previously described, using the techniques of failure modes and effects analysis. Also, as described, postulated system interactions induced by random failures of safety-related components will be considered acceptable if it does not compromise the functional capability of the system to perform its required safety function.

Potential sources are evaluated as a part of the program to determine if events can credibly lead to detrimental interaction with targets. They are categorized in one of the following:

a. Events will not lead to interaction because of defensible qualification of the sources by analysis, test or experience with the same or similar items.

b. Events may lead to damage or failure of the sources, but the credible failure modes are not threat to the safety function of the target.

c. Events may lead to a credible failure mode of the source which has potential to cause an adverse interaction.

In addition to the above scope, the NRC systems interaction staff, during their review, emphasized that consideration of operating experience is an important element in the systems interaction analysis. It was concluded that the operating experience at Indian Point 3 can be extrapolated from events that have actually occurred. The suitability or workability of a proposed systems interaction analysis methodology can be demonstrated if it can be shown that the application of that methodology will identify links to what has already actually occurred, or that these methods would have identified systems interactions similar to those which have occurred in the past. To this end, the scope of the study was expanded to include a review of the licensee event reports and significant occurrence reports that have been reported during 1980 and the first half of 1981 for the Indian Point 3 facility. Although this approach appeared to be a satisfactory method for determining the effectiveness of the proposed methodology, its true value can only be assessed upon completion of the study. This is due to the fact that the study presently has only included an application of the methodology and criteria to the Auxiliary Feedwater System.

Another suggestion that was made by the systems interaction staff was the investigation of using the Indian Point simulator for uncovering systems interaction dependencies for the treatment of first order types of systems interactions. The systems interaction staff believed that to the extent that such a training simulator accurately models at least direct interconnection between safety and non-safety front-line systems and their support systems, it may be possible to do more comprehensive and systematic analysis of their failure effects more easily and more efficiently by the use of a simulator. It was agreed that as part of the systems interaction effort, and investigation would be made to examine the possibility of using the training simulator. To this end, the NRC systems interaction staff was invited to observe and participate in the initial trials on September 23-24, 1981 at the Indian Point simulator facility. Subsequent to those trials, it was arranged to further investigate the use of the simulator for specific malfunctions modeled into the simulator. This activity was accomplished on October 29, 1981. In general, the results obtained during the initial trials and the malfunction tests confirmed that the use of the simulator does not effectively uncover systems interactions between safety and nonsafety systems. This was primarily due to the fact that the simulator is modeled as a training tool, consistent with the current operator training programs. Hidden dependencies between safety and nonsafety systems and components is not part of the software package of the simulator.

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

3/3/82

APPENDIX XVII
STAFF/PASNY PARTICIPATION IN SUBCMTE.
REPORT SESSION ON SYSTEMS INTERACTION
and RELATED MATTERS

NOTE TO: Mike Greismeyer, ACRS Staff

FROM: Jim Conran, Systems Interaction Staff, RRAB

SUBJECT: STAFF/PASNY PARTICIPATION IN SUBCOMMITTEE REPORT SESSION ON SYSTEMS INTERACTION....AND RELATED MATTERS

In response to Dr. Okrent's request (relayed by you in our telephone conversation this morning), the staff will attend the Subcommittee Report Session scheduled late Thursday evening prepared to discuss the results and conclusions of the SRP review we have undertaken. The questions we are examining specifically in our review is what types and to what extent safety system/safety system interactions are included under existing regulations (specifically, under the "licensing basis" for IP-3). The primary purpose, of course, is to try to better understand just how the systematic and comprehensive treatment of that aspect of systems interaction (particularly in the"spatially-coupled" systems context) would add to the scope, cost, schedule, etc. of the IP-3 program if included therein. That's a very tall order, of course; and I do not expect to complete such a review by Thursday evening. But I will be there to share with the Committee whatever I know of these matters by then. I have also encouraged PASNY to continue to work on this question, and asked if they would be prepared to provide whatever insight they can gain in this regard by the appointed hour. They have agreed to do so (with the same caveats I have expressed). I am more than happy to respond in this fashion (and any other reasonable way) to try to resolve this issue that was raised at the Subcommittee meeting, because I genuinely fear that it jeopardizes the proposed IP-3 program. From conversations with PASNY yesterday and this morning, I believe that there is no doubt left that they are seriously reconsidering their options (regarding either going ahead or withdrawing their proposal) as a result of the reaction they received from the Subcommittee on Friday.

For our part, Mike, we feel strongly that the PASNY program is an integral and important element of the overall SI program, and is an important vehicle for testing very "promising" methodology that we feel can be applied effectively and efficiently in the broad scope treatment of systems interactions in reactor plants. The Subcommittee's questions and comments regarding this (comparative methodology) aspect of our program are understandable; but ultimately such questions are not resolvable by speculation or discussion. They must be given a real trial....and that is what PASNY is offering in their proposed program. As a further important consideration, because in plant operations it is clear that the "unexpected" will continue to happen because of the "unanalyzed", we believe that the PASNY program (with all its perceived imperfections) could contribute significantly to improving safety in our operating plants generally, and in IP-3 specifically.

I surely hope that this current difficulty can be resolved and the PASNY program can go forward as planned. I believe that a formula can be found to work out differences on these important questions; and, in fact, I believe that one such formula is implicit in

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the outline below (which reflects our understanding and interpretation of what transpired at the Subcommittee meeting and regarding the comments and positions stated by the principals involved):

1. The Subcommittee has not reviewed extensively the PASNY submittal (Proposed SI Analysis Program Description); but the Systems Interaction Staff has carefully reviewed the proposal and found it acceptable (i.e., sufficiently broad scope and safety beneficial).

2. The Subcommittee appeared to agree generally with the staff's judgment regarding acceptability of the proposed program, but noted that the matter of safety system/safety system interaction (particularly in the "spatially-coupled systems" context) did not appear to be considered explicitly or comprehensively in determining the scope of the proposed program.

3. The Systems Interaction Staff acknowledged the Subcommittee's concern in the safety system/safety system interaction area and stated that that aspect of the overall SI analysis issue will be treated explicitly and comprehensively in the context of the generic "pilot program" SI studies.

   A requirement to include treatment of safety/safety interactions in the IP-3 study beyond existing regulations/guidance (e.g., SRP methods and criteria) cannot, however, be imposed legally on PASNY....and the Stello Committee is chartered specifically to prevent the staff from trying to do so "informally" without due process.

4. The Subcommittee encouraged the performance of the proposed PASNY program because of the potential safety benefits to be derived. (There is also the added benefit of full-scale trial of a "candidate" SI analysis methodology considered "promising", i.e., effective and efficient, by the SI staff). It was understood, however, that, even if PASNY completes the broad-scale program they have proposed at IP-3, if the staff's generic programs identify the need for additional safety system/safety system SI analysis requirements, those requirements would be imposed (as appropriate) on IP-3 as well as other operating plants. (In other words, no gradfathering of IP-3 in this regard).

5. PASNY indicated the need for some expression of ACRS acceptance of their proposed program as a condition for going forward at all. So, a Committee letter documenting the points outlined above (as the Committee perceives and interprets them, of course) would seem an appropriate vehicle and format for doing this.

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I hope that the time allotted during this full Committee meeting will enable full consideration of all the respective parties' positions as indicated above, and that some sort of letter (or indication of ACRS' intent to provide a letter) indicating positive ACRS response to PASNY's proposal and encouragement to get on with the program can result from the Committee's consideration of these matters at this time. If we can be of further help in bringing this about, please let me know.

Conran

Jim Conran
Reliability and Risk Assessment Branch
Division of Safety Technology

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

March 3, 1982

APPENDIX XVIII
POSSIBLE SYSTEM INTERACTION STUDY TOPICS

MEMORANDUM FOR: ACRS Members

FROM: R. Savio, Senior Staff Engineer R. Savio

SUBJECT: POSSIBLE SYSTEM INTERACTION STUDY TOPICS

I have enclosed a list of topics which might be studied in systems interaction studies of limited scope and in cases where the available resources would not support a more broad based study. Mr. Bender has asked that you look at this list and consider what topics might be added and how the listed study topics might be augmented or changed. It is my understanding that Mr. Bender will discuss this within the context of our response to PASNY on the Indian Point systems interaction study.

3-2-82

SYSTEMS INTERACTION STUDY TOPICS

1. Equipment failures resulting from failure of air supply systems, including consideration of abrupt loss of pressure, loss of air pressure, low pressure, and contamination.

2. Effects of fires in various fire zones in other areas of the plant, including compartment heatup from fires in adjacent zones and credible failures of fire zone barriers.

3. Plant walk-through to identify:

   a) physical interference and potential threats to safety equipment from safety-grade and nonsafety-grade equipment;

   b) the potential for creating adverse environments which could pose a threat to equipment important to plant safety;

   c) potential unrecognized dependence between independent systems which exist because of location, common environments, and interconnection.

   The walk-throughs would be used to discover the obvious (in the as built plant) potential for systems interactions and to identify, under a predetermined criteria, aspects of the plant design which would need to be subject to further evaluation.

4. An evaluation of equipment failures which would result in internal flooding and the effect on other equipment important to plant safety which might be affected.

5. An evaluation of LER experience for a class of plants to uncover potential systems interactions. The evaluation might be performed by a team composed of persons familiar with the detailed design of the plant and persons versed in transient and systems analysis.

6. Examination of maintenance and test procedures for potential errors involving system configuration changes which could create system dependence and the potential for system interaction. A team, such as described in Item 5, might be used. The number of "target" systems could range from the most important to all systems important to plant safety, depending on the resources available.

7. The effect of the failure of or contamination of plant service water/component cooling water on plant equipment.

8. The effect of the failure of or degradation of plant ventilation and cooling on plant equipment.

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SYSTEMS INTERACTION STUDY TOPICS (CONT'D)

9. The effect of unusually low ambient temperatures on the ability of plant equipment to deal with an anticipated transient (for example, freezing of water condensation in the diesel air systems).

10. The interaction between DC and emergency AC power systems and faulted loads and test and maintenance configurations of these power systems.

11. Application of the best state-of-the-art sneak circuit analysis to safety and control systems.

12. FMEA for control systems and effect of control system failures on safety system and safety system interactions.

13. Failure of nonsafety-grade equipment in an accident environment and the effect of these failures on plant safety and safety equipment (for example, failures which could cause secondary or primary side LOCAs, flooding, or confusing and misleading information to be transmitted via nonsafety-grade instruments).

14. Review of the dependence of systems used to remove decay heat and monitor plant status during a station blackout in AC power and the impact on plant recovery when AC power is restored (for example, turbine lube oil pumps, RCP seal cooling, or compartment temperatures).

15. Missiles created by the failure of safety and nonsafety equipment and their effect on equipment important to plant safety (for example, failures of air tanks, motors, or a dropped compressed gas storage bottle).

16. Review of plant systems for interconnections which have the potential for introducing contamination into air or fluid systems (for example, demineralizer resins into component cooling water or water into air systems).

17. Degradation of modern solid-state equipment and computers vice voltage surge, software errors/failures, hookup of diagnostic computers, and equipment which could introduce a rapid (but still within equipment, temperature limits) change in temperature.

18. Effects of waterhammer on valves and pumps and physical interactions with safety equipment.

19. Systems interactions given the unmonitored failure of electrical isolation devices.

SYSTEMS INTERACTION STUDY TOPICS (CONT'D)

20. Diablo Canyon type seismic systems interaction studies for plant in active seismic areas or older plant designed to less stringent seismic criteria.

21. Review of startup test procedures to identify "blind spots" in testing of systems and an examination of the plant to uncover construction errors which could lead to unexpected plant behavior or system interaction.

22. Failures in decay heat removal systems which could also initiate scram.

23. Failures of boundaries between high and low pressure systems.

PROPOSED SUMMARY OF ACRS SUBCOMMITTEE MEETING
ON METAL COMPONENTS/WASTE MANAGEMENT
FEBRUARY 12, 1982
WASHINGTON, D. C.

Purpose:

The joint Metal Components/Waste Management Program Subcommittee met on February 12, 1982, to:

1. Perform an independent review of the technical capabilities of the NRC's Contract Review Panel recommended contractor prior to to final selection, and
2. Review the scope of the Request for Proposal Program entitled, "Long-Term Performance of Materials Used for High-Level Waste Packaging," and determine if it satisfies the appropriate sections of 10 CFR Part 60. In addition, the scope of the recommended contractor was reviewed to determine if it meets the requirements of the proposal.

Attendees:

The ACRS Members in attendance were P. Shewmon, D. Moeller, R. Axtmann and J. Ray. The ACRS Consultants in attendance were M. Steindler, D. Orth, D. Readey and T. Kassner. The NRC Staff was represented by F. Arsenault, J. Davis, and K. Kim.

Highlights, Agreements and Requests:

1. F. Arsenault presented a brief chronology of the waste package materials performance program. He stated the program was initiated about two years ago. The recommended contractor was selected by the review panel on Janu. ry 28, 1982.

2. The Commissioners, after reviewing the RFP with the NRC Staff, requested by letter dated August 10, 1981, that the ACRS review the technical capabilities of the selected contractor before the award becomes final.

3. F. Arsenault stated that the EPA defined the general radiation standards for the waste repository and the NRC developed the guidelines to ensure containment package licenseability.

4. The containment package must be designed to 1000-year life and be retrievable in 50 years as defined in 10 CFR Part 60.

5. The Staff stated that the RFP was purposely written to be flexible so that the contract could be modified without rebidding. This feature of the RFP was questioned by the Subcommittee.

6. Contract requirement states that non-DOE contractor has preference over DOE contractors unless the DOE proposal exhibits significant technical advantage. NRC received four bids: two from the national laboratory and two from private industries. The DOE contractors' proposals were not considered because the bids did not possess significant technical advantage.

7. A major objective of this program is to develop a containment material performance predictive methodology based on an improved technical understanding of the long-term behavior of waste packages emplaced in different geological media.

8. K. Kim stated that the waste form materials to be studied are borosilicate glass, high-silicate glass and SYNROC. The containment materials to be studied are iron, nickel and titanium alloys.

9. The RFP's three tasks are (1) work plan development, (2) methodology development and (3) validation and benchmark. The Subcommittee questioned the validation technique and the data to be used for benchmarking.

10. A consultant suggested frequent (at least quarterly) progress reports and peer review of these reports.

11. K. Kim stated some of the parameters that will be evaluated are pressure, temperature, radiation, solution chemistry, cycling and rate effects.

12. The NRC Staff stated that the containment accident scenarios will be defined as the program is developed.

13. A consultant suggested that a work plan be developed by the NRC Staff to adequately monitor contractor's progress.

14. The Subcommittee has determined that sufficient information has been presented to bring this matter before the full Committee for discussion. The Subcommittee feels that the contract review panel selection is adequate subject to the following conditions: 1) close monitoring of work, 2) frequent progress reports and 3) provision for periodic peer review of contractor's proposed program and results.

15. Reports by ACRS Consultants Steindler, Orth, Readey and Kassner are attached.

Future Meetings:

The Subcommittee will present its findings to the full ACRS Committee during the March Meeting.

The following pages A-361 thru A-368 has been deleted as 1.

DELETION

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

March 9, 1982

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

SUBJECT: REPORT ON THE LONG-TERM PERFORMANCE OF MATERIALS USED FOR HIGH-LEVEL WASTE PACKAGING

Dear Dr. Palladino:

In response to your letter of August 10, 1981, the Advisory Committee on Reactor Safeguards has reviewed the NRC's Contract Review Panel recommendation for the selection of a contractor to develop a methodology for predicting Long-Term Performance of Materials Used for High-Level Waste Packaging. On the basis of this review, we offer the following comments and recommendations.

We believe that the proposed contractor is technically capable of conducting the requested research. However, we believe that both the NRC Staff and the contractor should establish plans to provide for close monitoring of the work as it progresses, that the NRC should assure that the contractor issues reports on the progress of his work at frequent intervals (at least, quarterly) and that provisions should be made for periodic peer review both of the contractor's proposed program and his results.

NRC Staff monitoring of the work and the efforts of the peer reviews should be directed toward assuring that the program is responsive to NRC's needs, and that the contractor is gaining maximum benefit from state-of-the-art knowledge and experimental techniques. In particular, detailed guides need to be developed concerning the extent and nature of the tests that must be conducted, and the accuracy levels required, to assure that the results can be extrapolated to the time spans required for a waste repository and that the associated degree of uncertainty can be defined. Specific attention will need to be directed to the determination, early in the research program, of those parameters that are important, as well as those that are not important, to the assessment of the performance of waste package materials.

The Committee would like to take this opportunity to note that this letter is narrowly responsive to the question you posed to us in your letter of August 10, 1981, and that we have some concern about the rationale for the

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extraordinarily high standards for long-term survival of these waste containers. In this matter we urge you to follow the approach used in pursuit of quantitative safety goals so that society is not penalized by the imposition of arbitrarily derived criteria for waste isolation.

Sincerely,

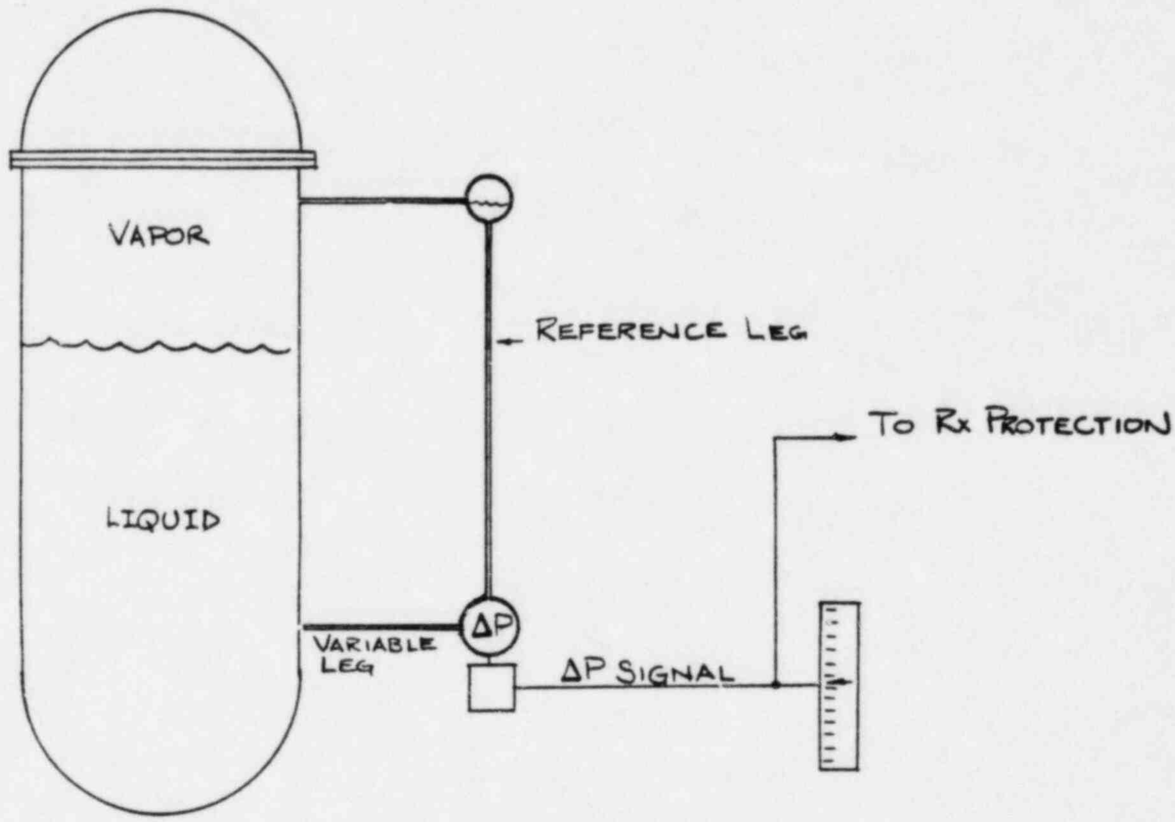
P. Shewmon

P. Shewmon
Chairman

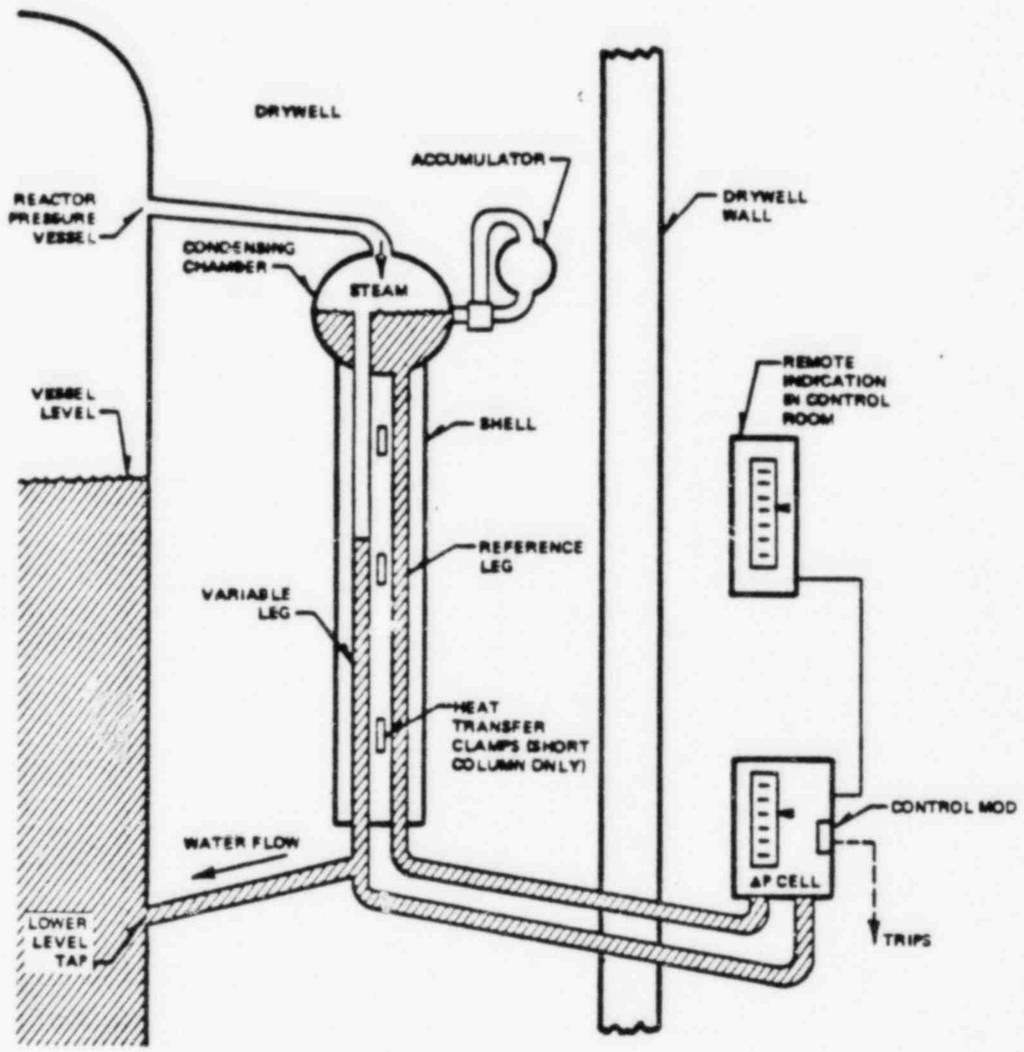
The following pages A-371 thru A-373 has been deleted as 1.

DELETION

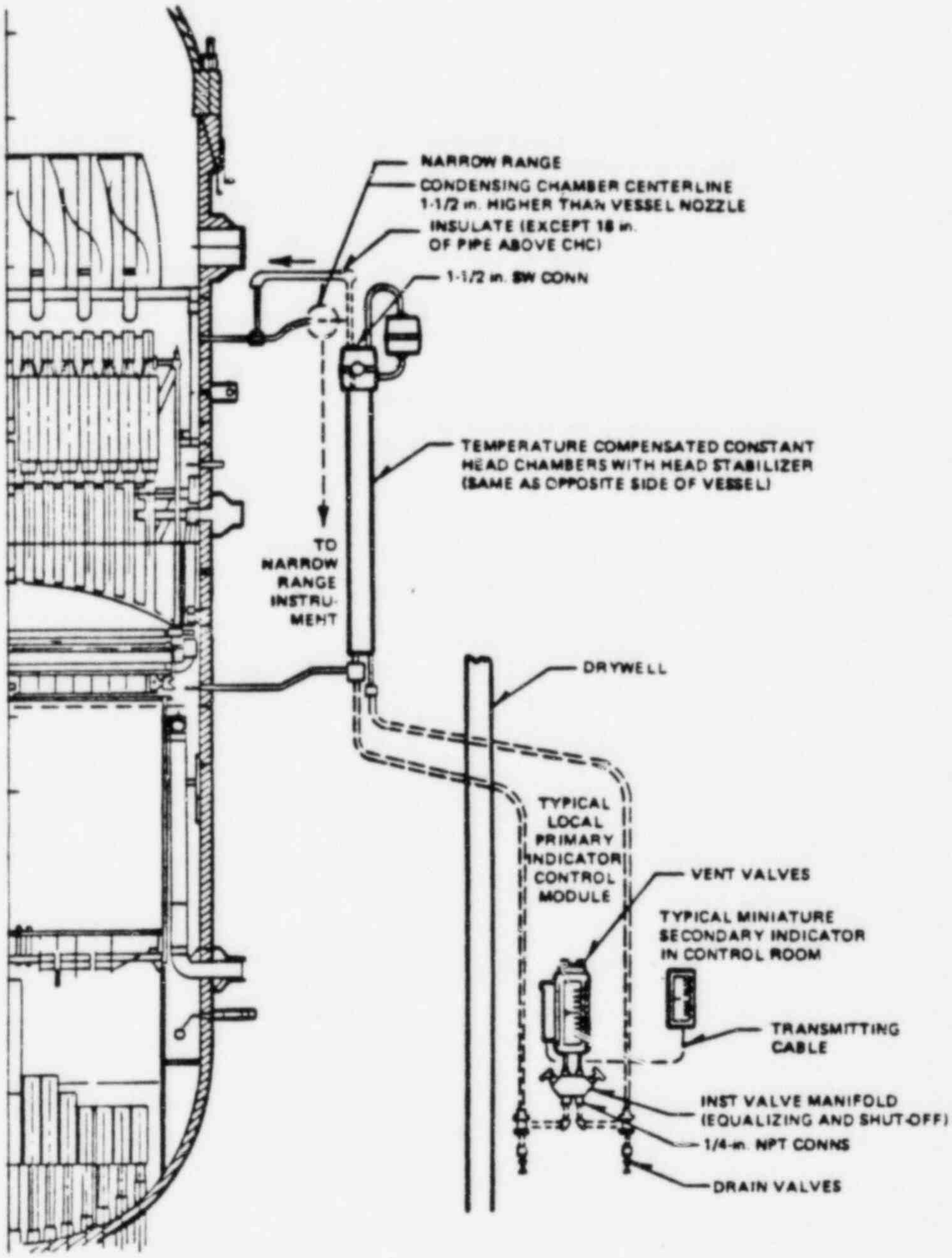
REACTOR VESSEL
DIFFERENTIAL PRESSURE
LEVEL INSTRUMENTS



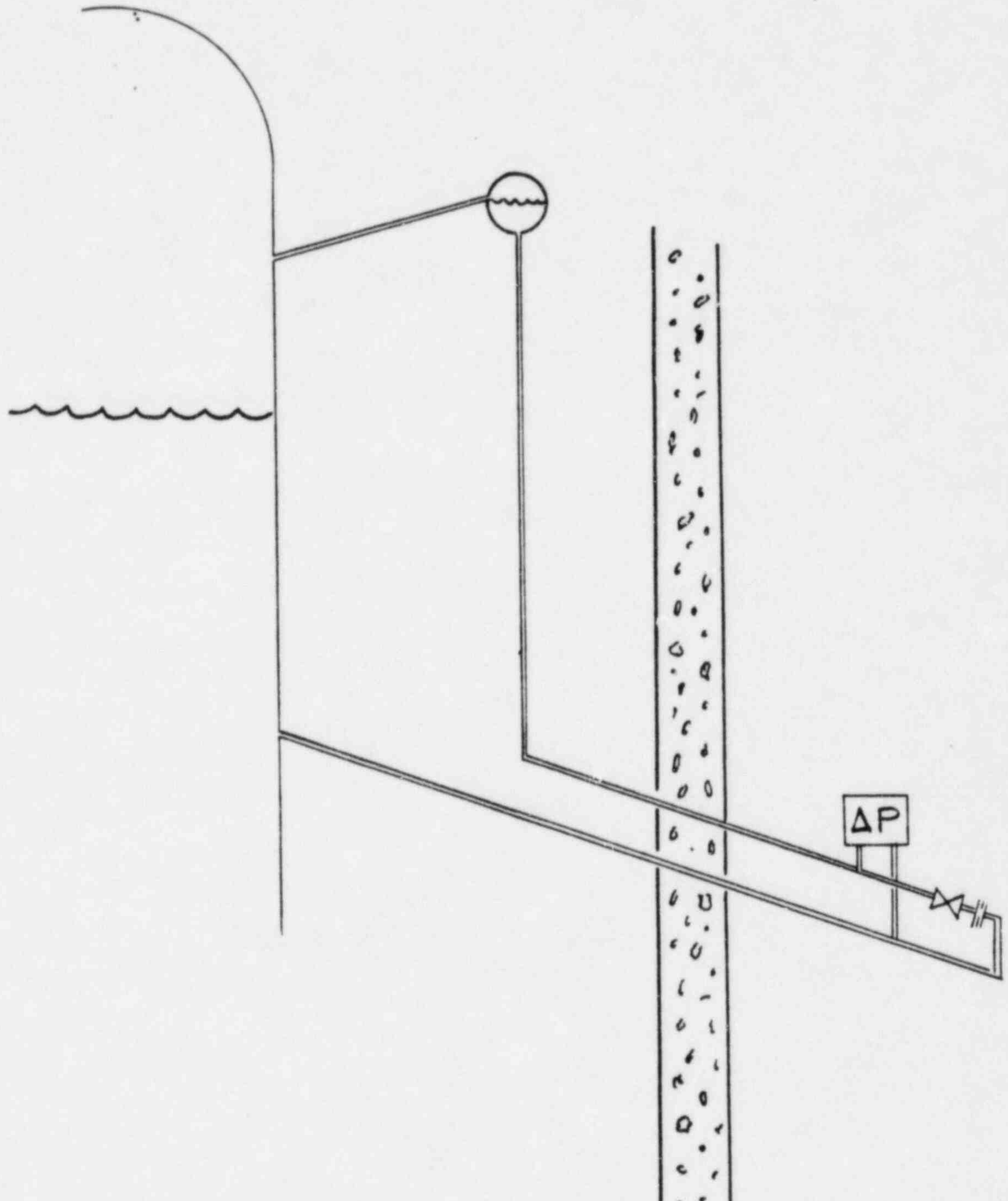
Δ P CELL PUTS OUT A SIGNAL PROPORTIONAL TO THE DIFFERENCE OF WATER WEIGHTS (PRESSURE) BETWEEN THE REFERENCE AND VARIABLE LEGS

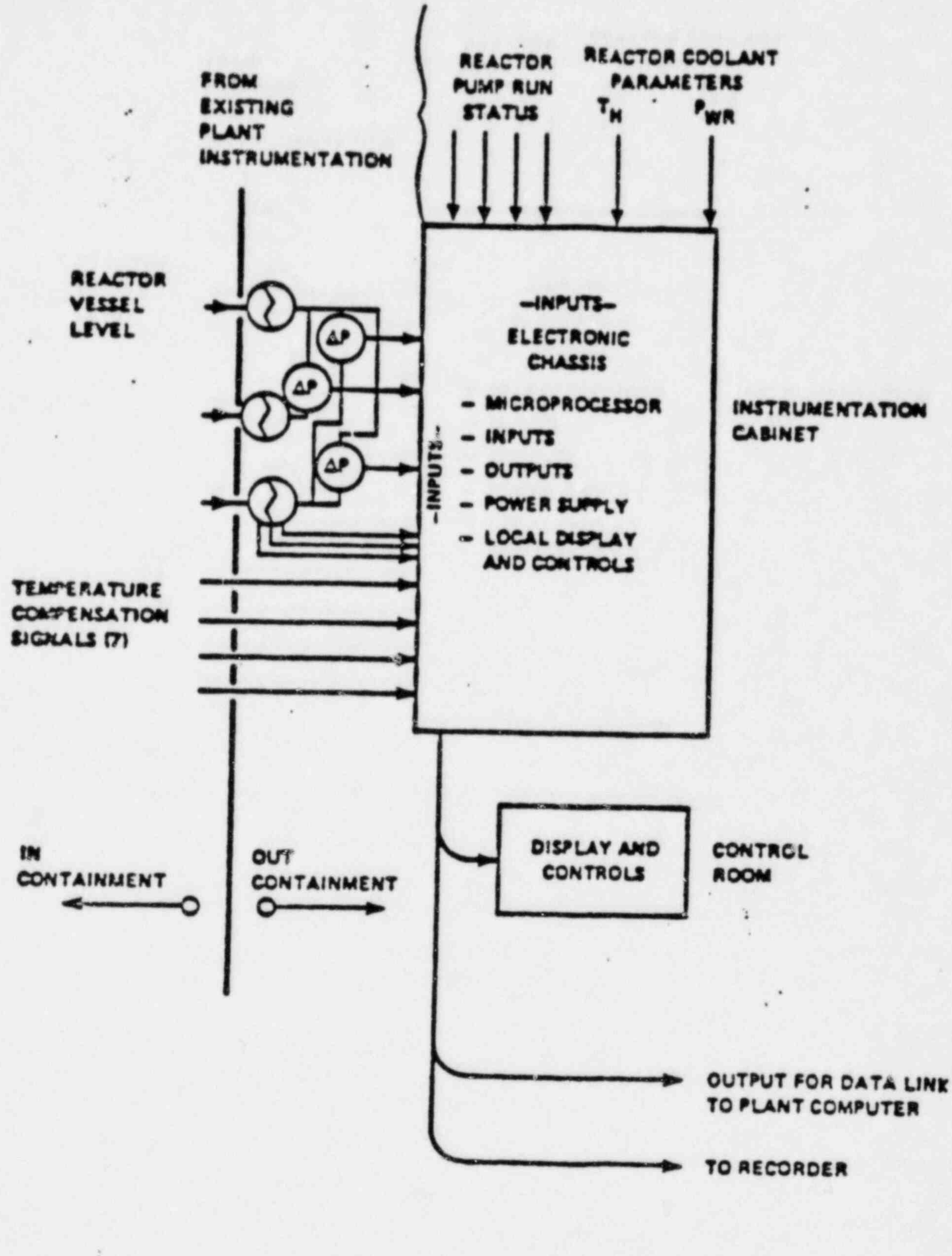


YARWAY LEVEL MEASURING INSTRUMENT SCHEMATIC



TYPICAL 210 INCH YARWAY PIPING ARRANGEMENT

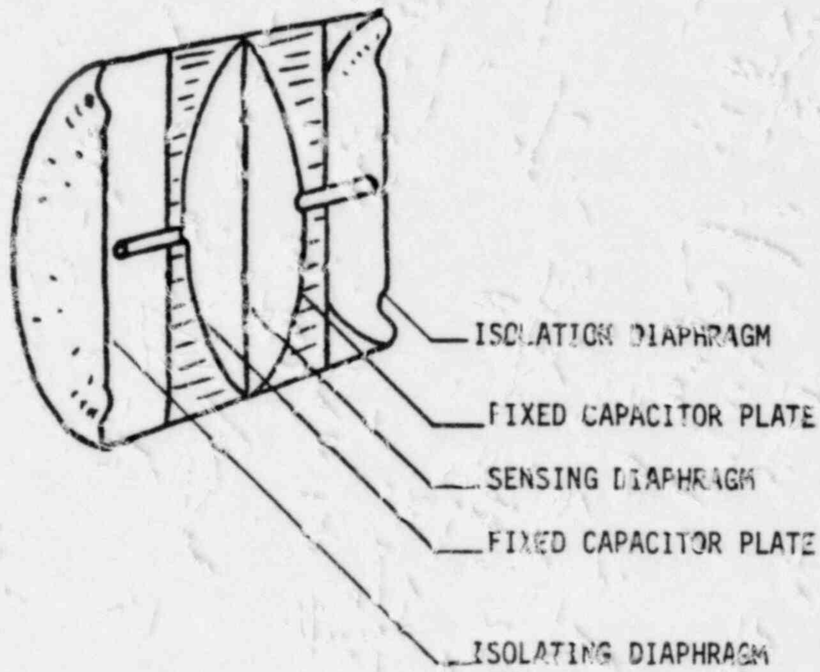

ΔP

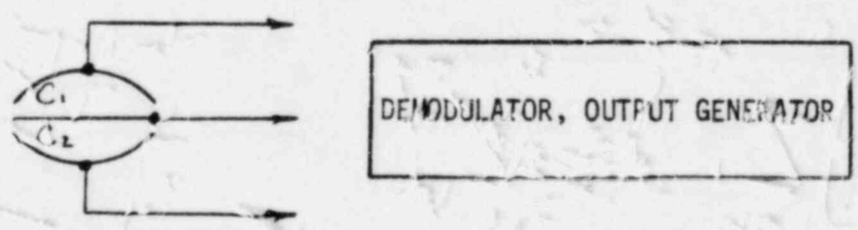


Reactor Vessel Level Instrument System Block Diagram (One Set of Two Redundant Resets Shown)

ROSEMONT 1152

" δ " CELL





$$\Delta P = K \frac{C_1 - C_2}{C_1 + C_2}$$

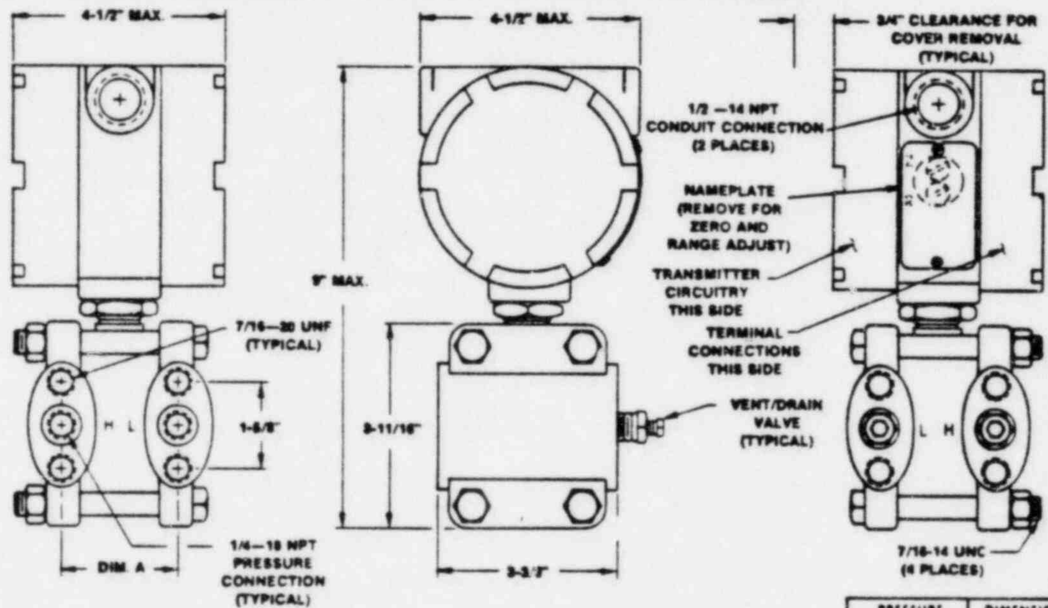
PARTS/DRAWINGS

THIS SECTION CONTAINS THE FOLLOWING DRAWINGS:

7.

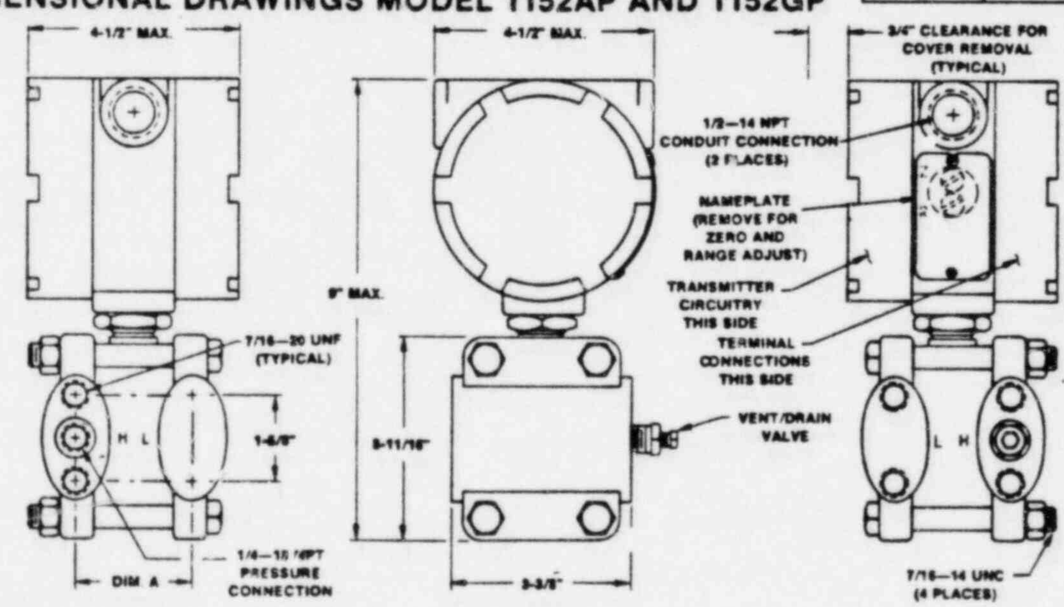
Outline Dimensions

DIMENSIONAL DRAWINGS MODEL 1151DP AND 1152HP



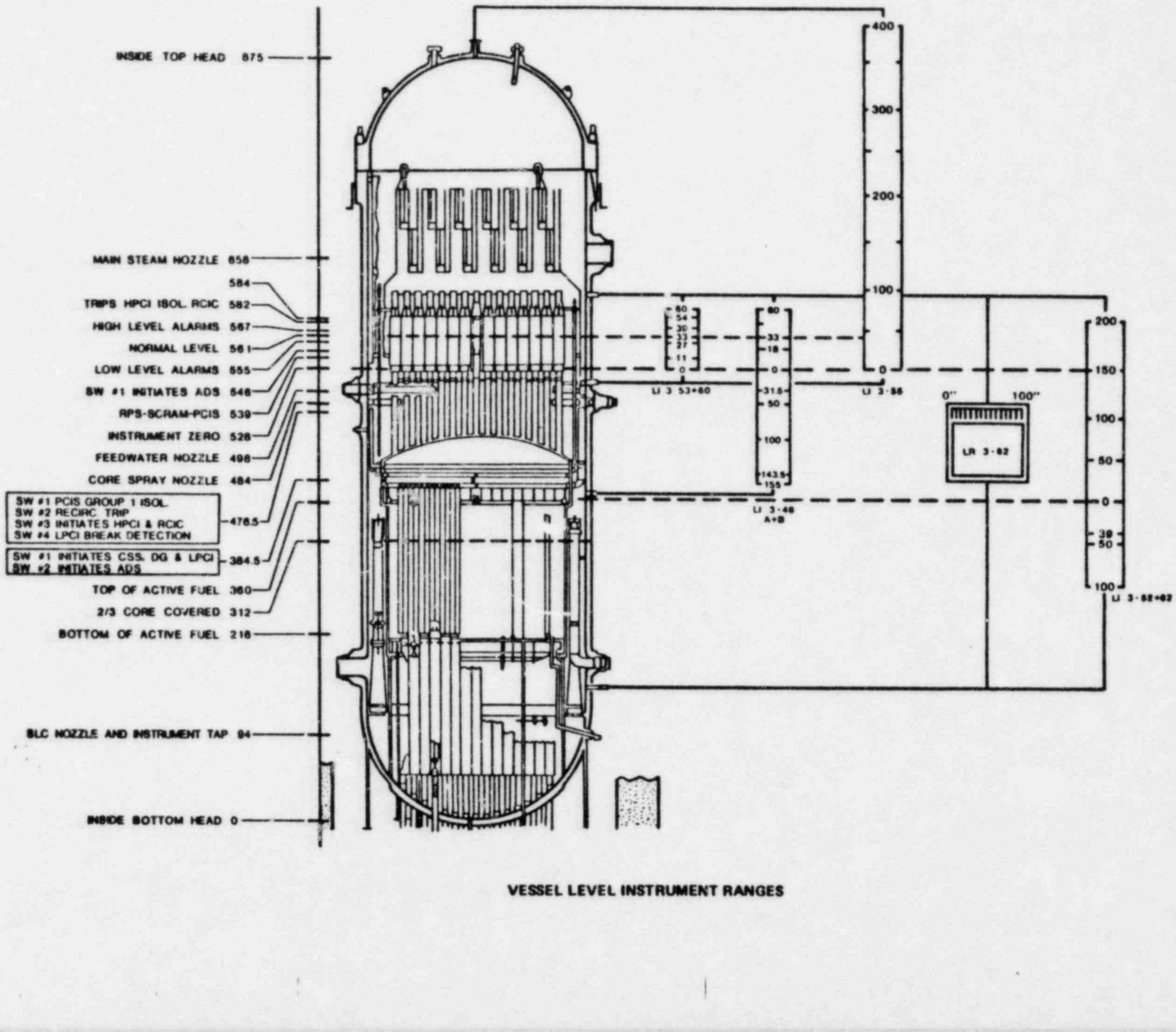
PRESSURE RANGE CODE	DIMENSION A
3,4,5	2-1/8"
6,7	2-3/16"
8	2-1/4"
9	2-9/32"

DIMENSIONAL DRAWINGS MODEL 1152AP AND 1152GP



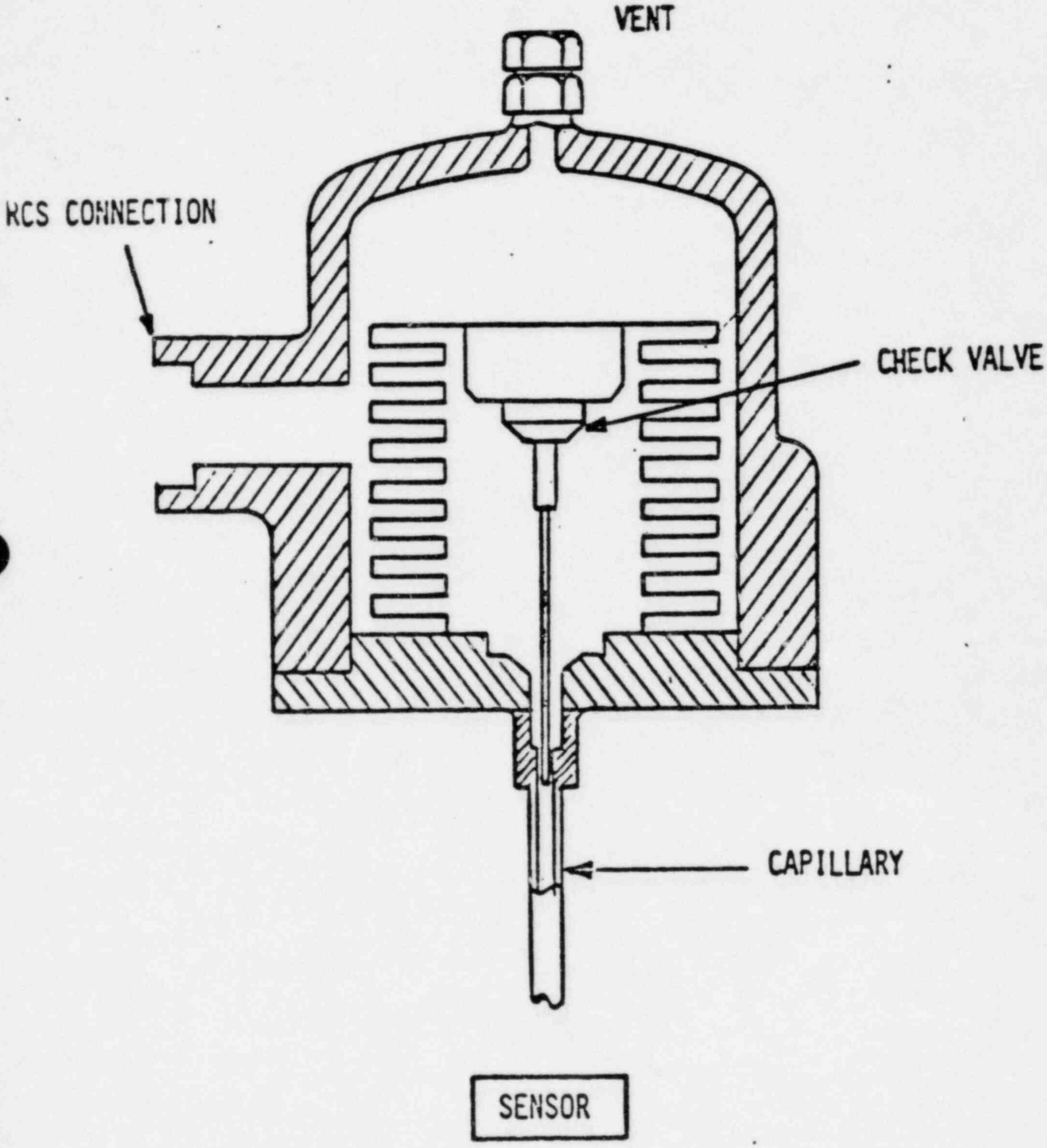
A-381

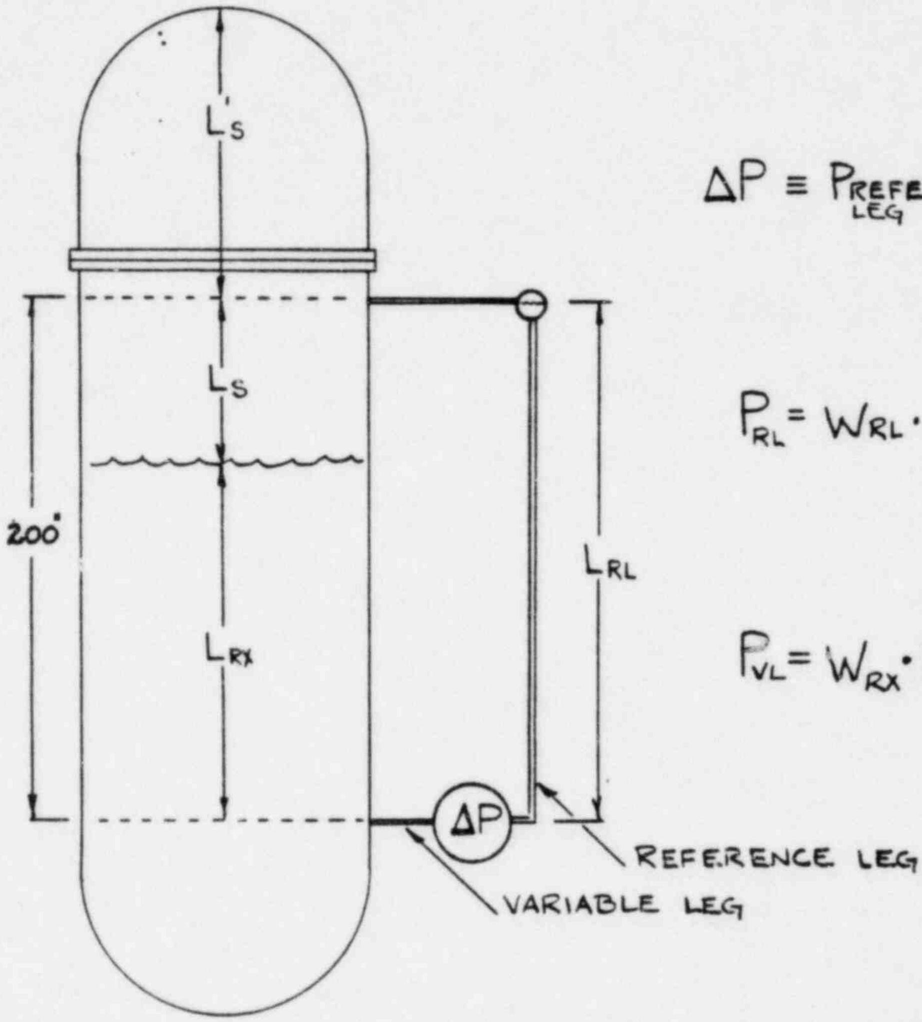
8.



VESSEL LEVEL INSTRUMENT RANGES

A-382


VENT
RCS CONNECTION
CHECK VALVE
CAPILLARY
SENSOR



$$\Delta P \equiv P_{\substack{\text{REFERENCE}\\\text{LEG}}} - P_{\substack{\text{VARIABLE}\\\text{LEG}}}$$

$$P_{RL} = W_{RL} \cdot L_{RL} + P_{RX} + W_S' L_S'$$

$$P_{VL} = W_{RX} \cdot L_{RX} + P_{RX} + W_S (L_S + L_S')$$

$$\Delta P = W_{RL} \cdot L_{RL} - W_{RX} \cdot L_{RX} - W_S (L_{RL} - L_{RX})$$

WHERE:

W_{RL} = WEIGHT OF LIQUID IN REFERENCE LEG

W_{RX} = WEIGHT OF LIQUID IN REACTOR VESSEL (ABOVE ZERO)

W_S = WEIGHT OF STEAM IN REACTOR VESSEL

L_{RL} = LEVEL OF LIQUID IN REFERENCE LEG

L_{RX} = LEVEL OF LIQUID IN REACTOR VESSEL (ABOVE ZERO)

SOURCES OF ERROR	CAUSE	MAGNITUDE
REFERENCE LEG TEMPERATURE CHANGE	1. INADEQUATE COOLING (100°F INCREASE POSSIBLE) 2. SMALL BREAK LOCA (340°F INCREASE POSSIBLE)	~5" HIGH VERY LARGE ERRORS IF REF LEG FLASHES. (118psia: UNCOMP) (425psia: COMP)
REACTOR COOLANT TEMP. CHANGE	COOLDOWN OR HEATUP	DEPENDS ON FINAL TEMP
CARRYOVER (LIQUID IN STEAM LINE) CARRYUNDER (STEAM IN SEPARATED LIQUID) ANNULUS SUBCOOLING RECIRCULATION VELOCITY HEAD & FRICTION DRYER ΔP	NORMAL OPERATION	SLIGHTLY HIGH ~2" LOW ~ 3" HIGH ON WIDE RANGE 4"-18" 7"-15"
PRESSURE WAVE TIME DELAY	ACOUSTIC TIME LAG (8-10"/sec)	0.5" High
YARWAY TEMPERATURE VARIATIONS	INADEQUATE MEASUREMENTS DURING STARTUP	DEPENDS ON TEMP ERROR
CORE ΔP	1. INADEQUATE CORE COOLING 2. FORCED CIRCULATION FLOW	VARIES WITH BLOCKAGE, PUMPS
LEVEL RATE PERCEPTION ERROR	VARYING VOLUMES OF WATER vs. LEVEL	NOT AN ACTUAL ERROR

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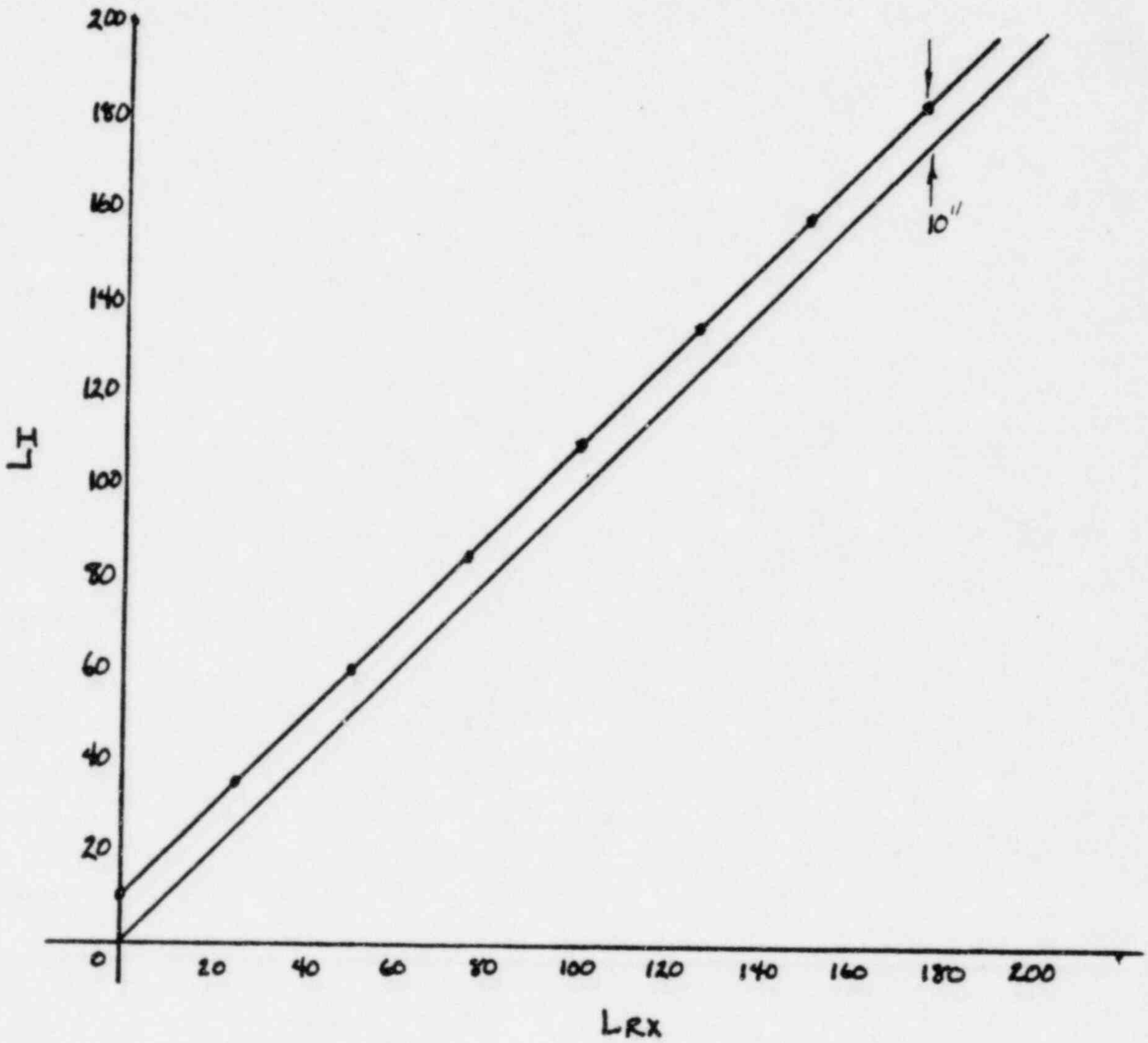
UNCOMPENSATED REFERENCE LEG

EFFECT OF RAISING $T_{CONTAINMENT}$ FROM 135°F to 235°F



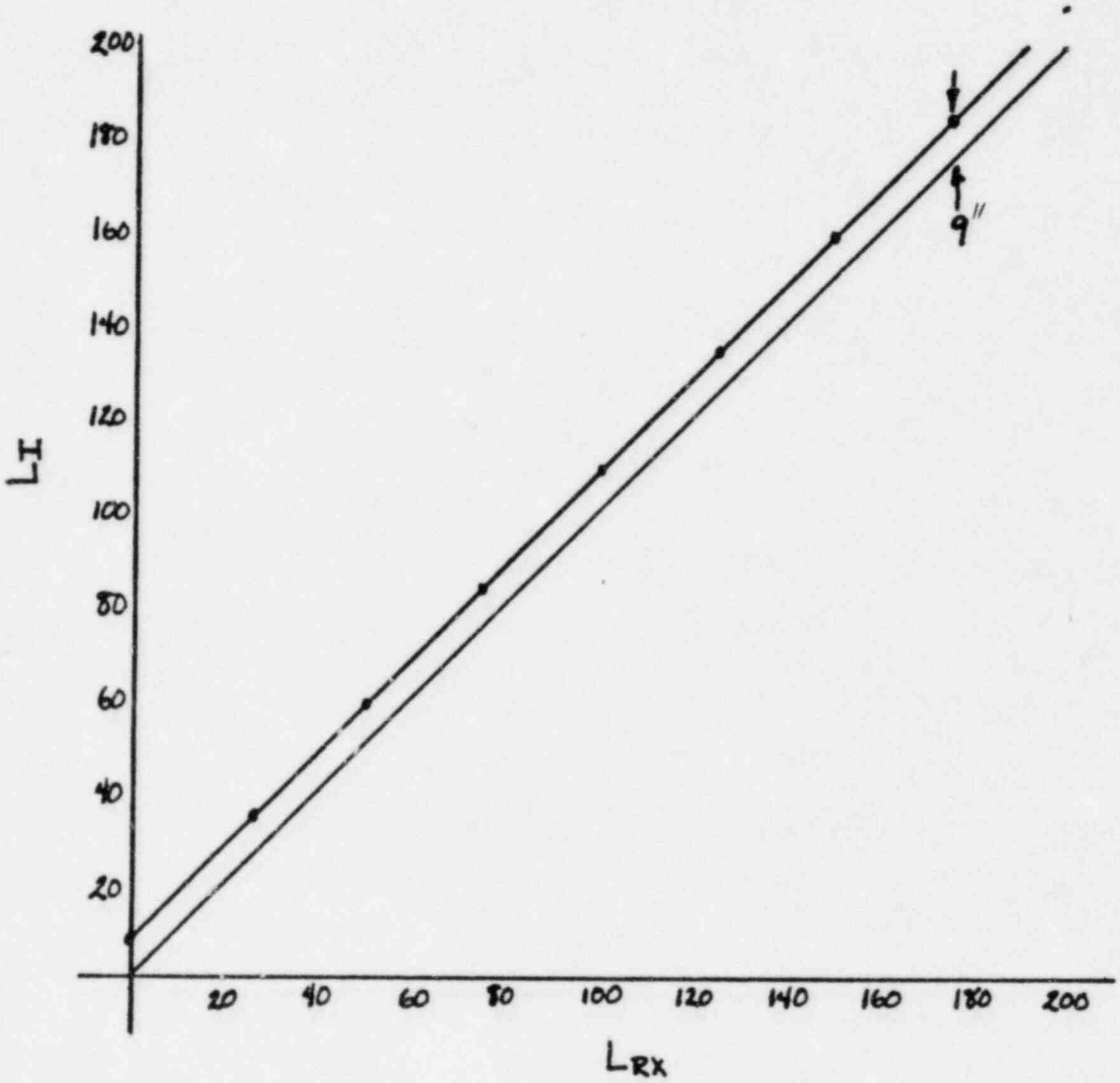
dP CELL CALIBRATED FOR P_{RX} = 1000 psi T_{RL} = 135°F

FINAL CONDITIONS P_{RX} = 1000 psi T_{RL} = 235°F

A-38C

COMPENSATED REFERENCE LEG

EFFECT OF INCREASING $T_{CONTAINMENT}$ FROM 135°F to 235°F

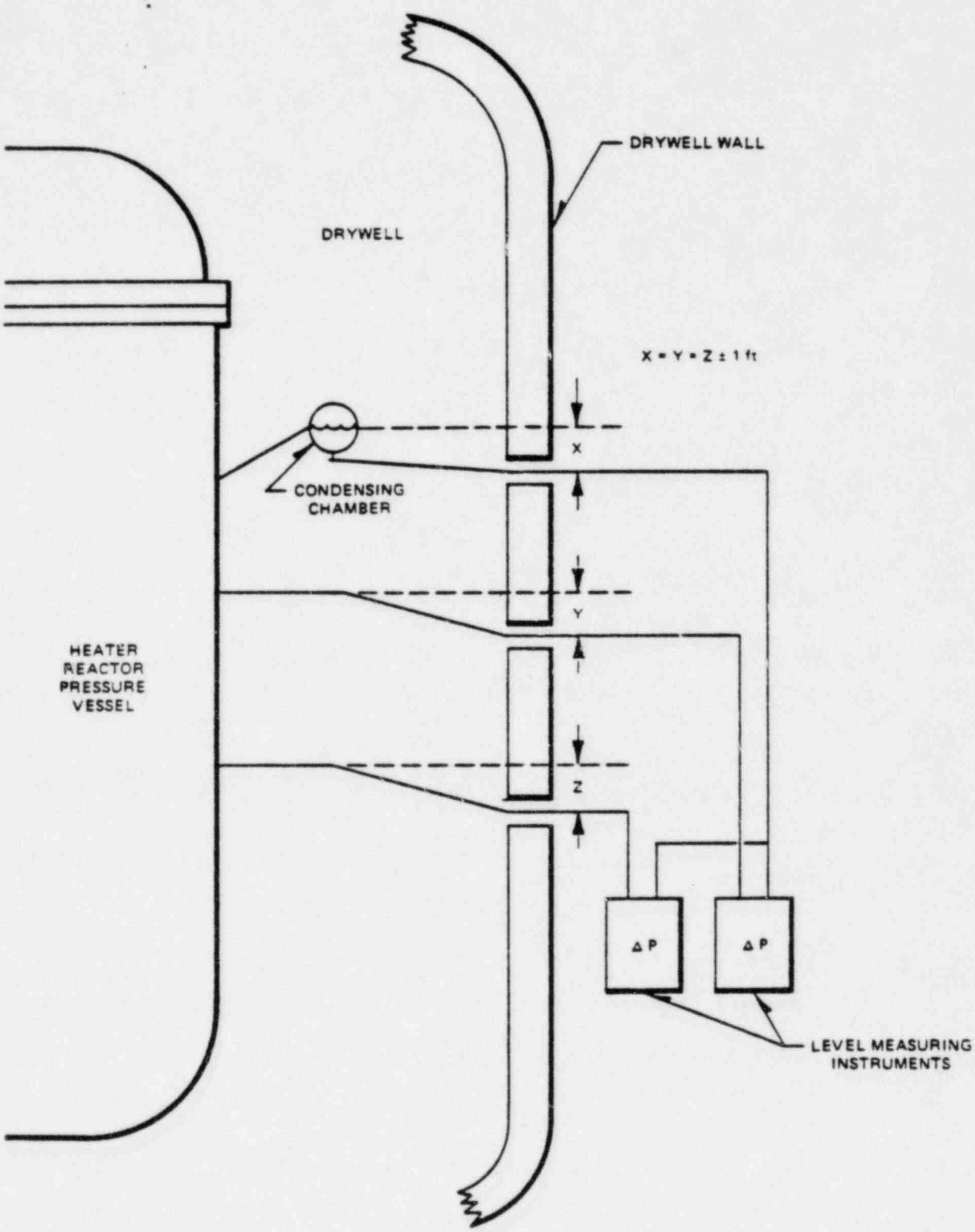


dP CELL CALIBRATED FOR P_{RX} = 1000 psi T_{RL} = 299°F

FINAL CONDITIONS P_{RX} = 1000 psi T_{RL} = 359°F

$$T_{RL} = T_{CONTAINMENT} + 0.4\ (T_{RX} - T_{CONTAINMENT})$$

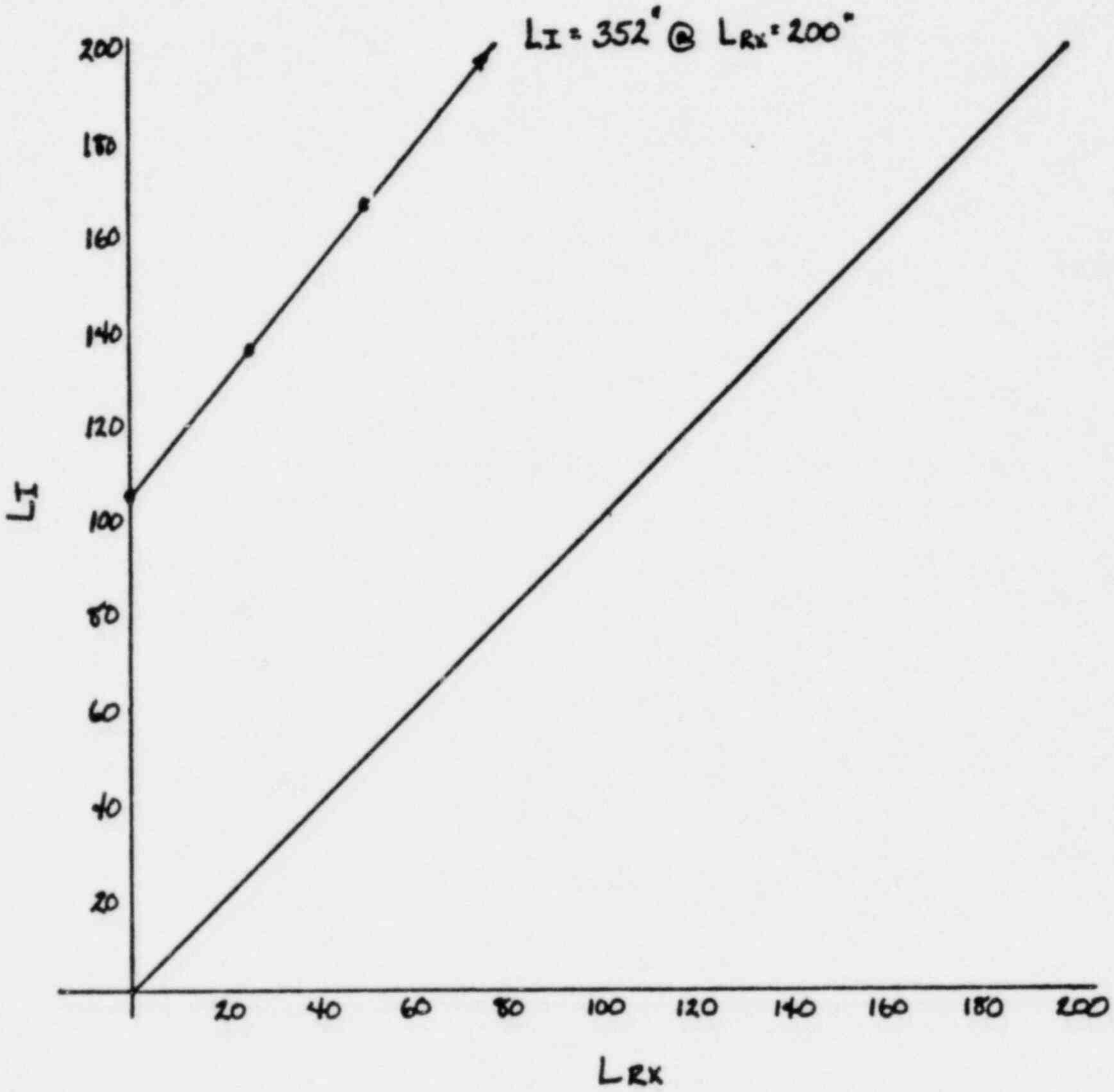
A387



Cold Reference Leg Water Level Instrument Piping Routing

A-388

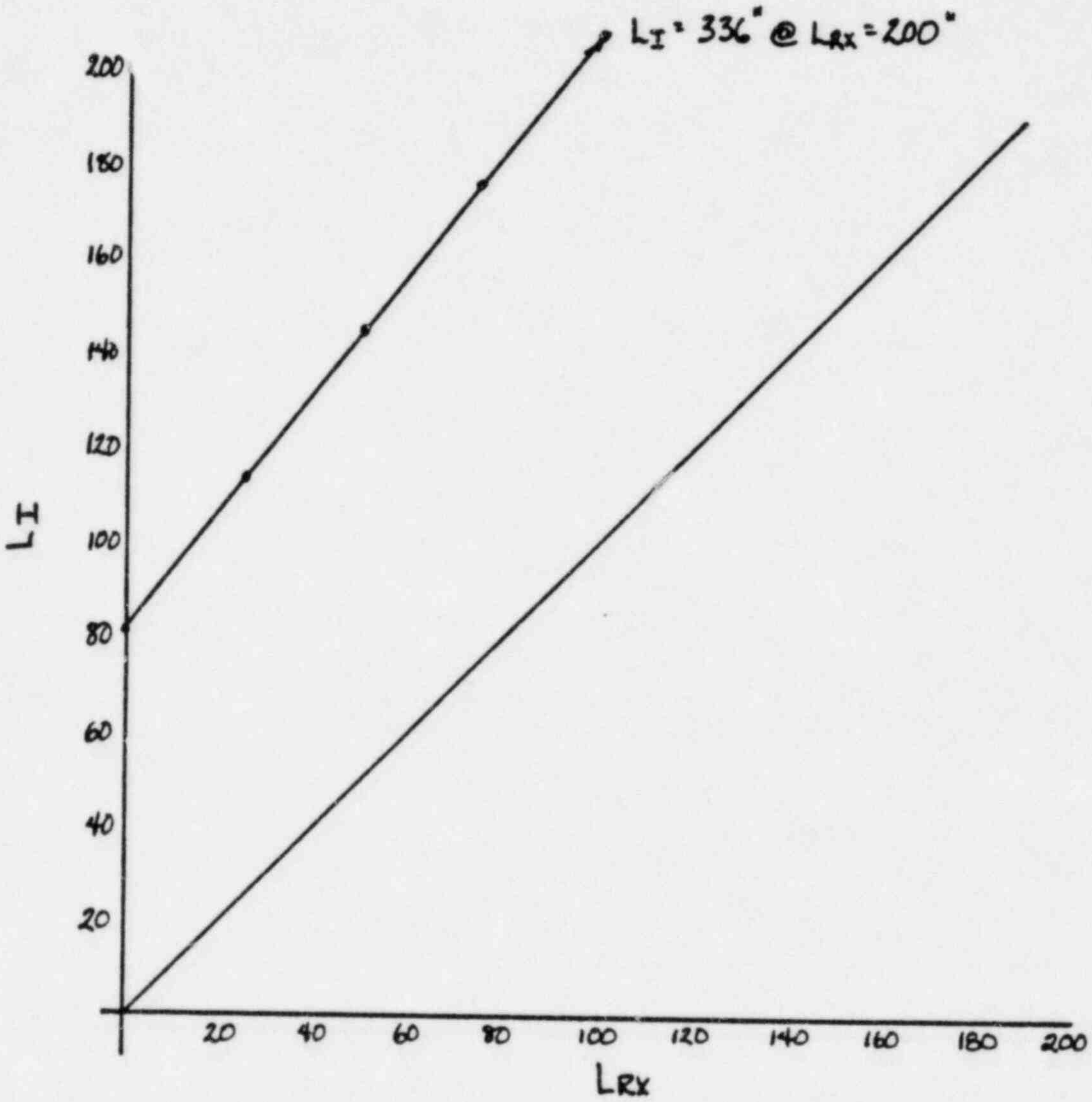
UNCOMPENSATED REFERENCE LEG
EFFECTS OF REFERENCE LEG FLASHING FROM 200" → 133"



dP CELL CALIBRATED AT	P_{RX} = 1000 psi	T_{RL} = 135°F
FINAL CONDITIONS	P_{RX} = 100 psi	T_{RL} = 340°F
		L_{RL} = 133"

COMPENSATED REFERENCE LEG

EFFECT OF REFERENCE LEG FLASHING FROM 200" TO 133"



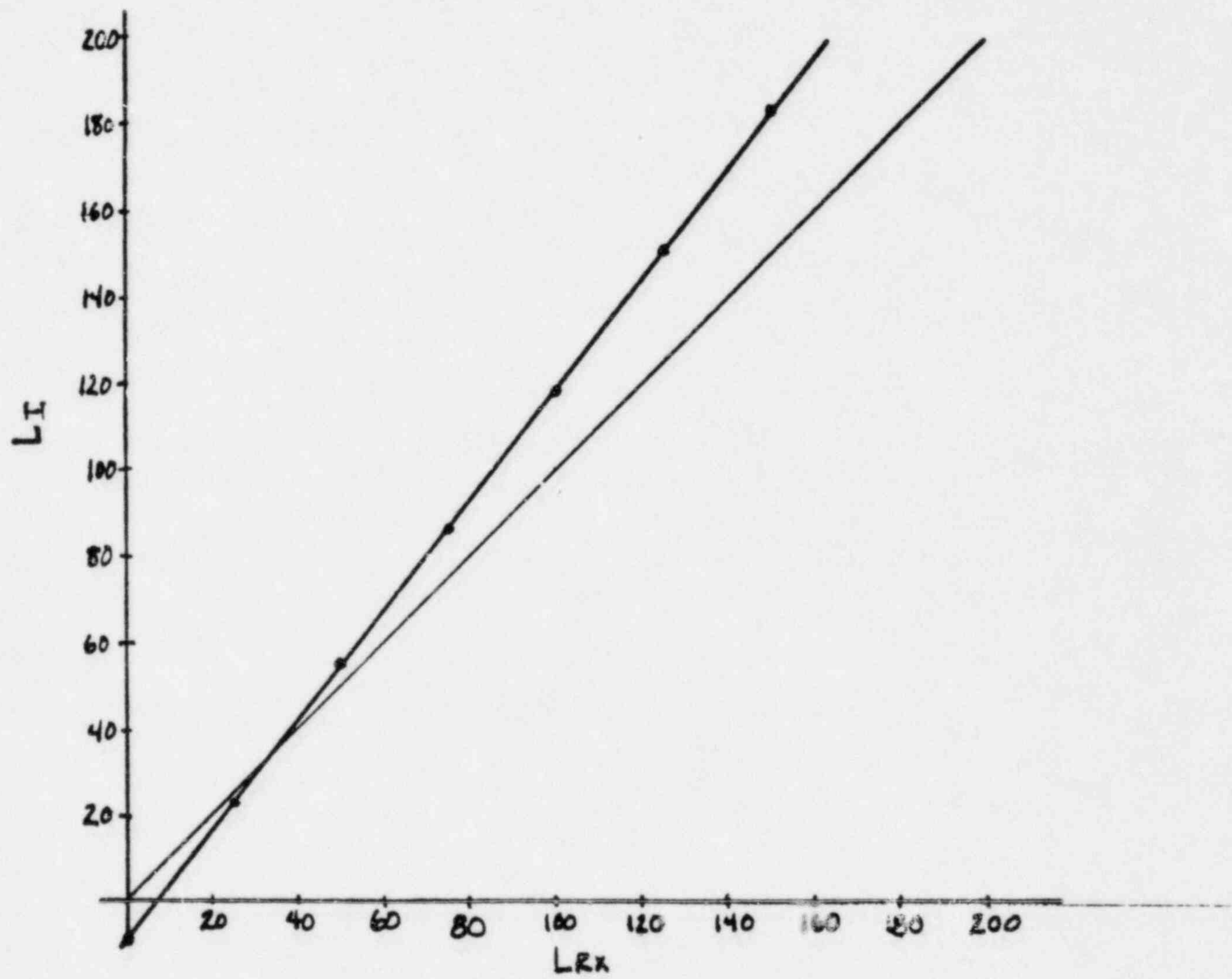
dP CELL CALIBRATED AT P_{RX} = 1000 psi T_{RL} = 299°F

FINAL CONDITIONS: P_{RX} = 100 psi L_{RL} = 133"

$$T_{RL} = T_{CONTAINMENT} + 0.4\ (T_{RX} - T_{CONTAINMENT})$$

UNCOMPENSATED REFERENCE LEG

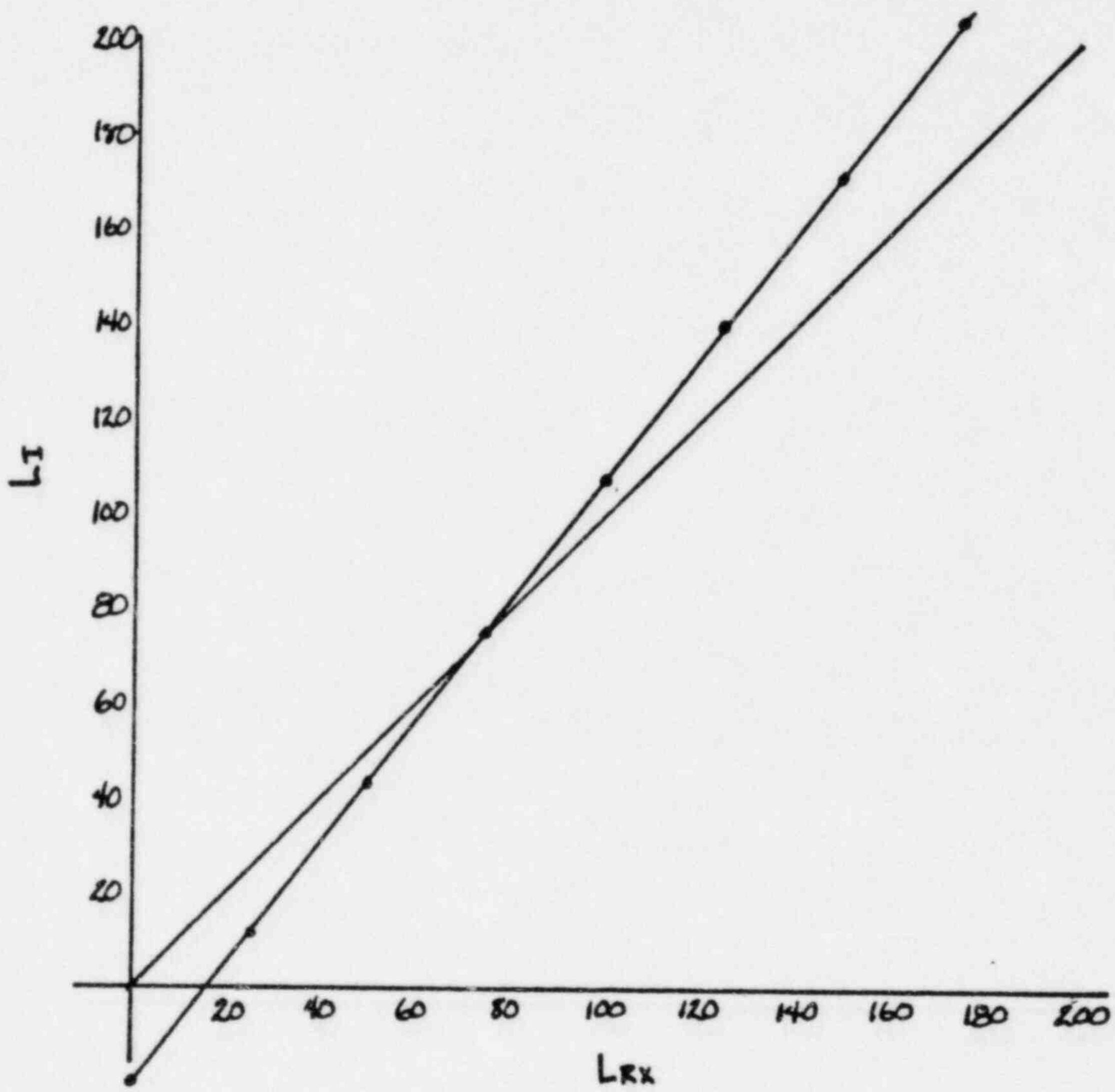
: EFFECT OF REDUCING P_{RX} FROM 1000 psi ⟶ 100 psi



dP CELL CALIBRATED FOR P_{RX} = 1000 psi T_{RL} = 135°F

FINAL CONDITIONS - P_{RX} = 100 psi T_{RL} = 135°F

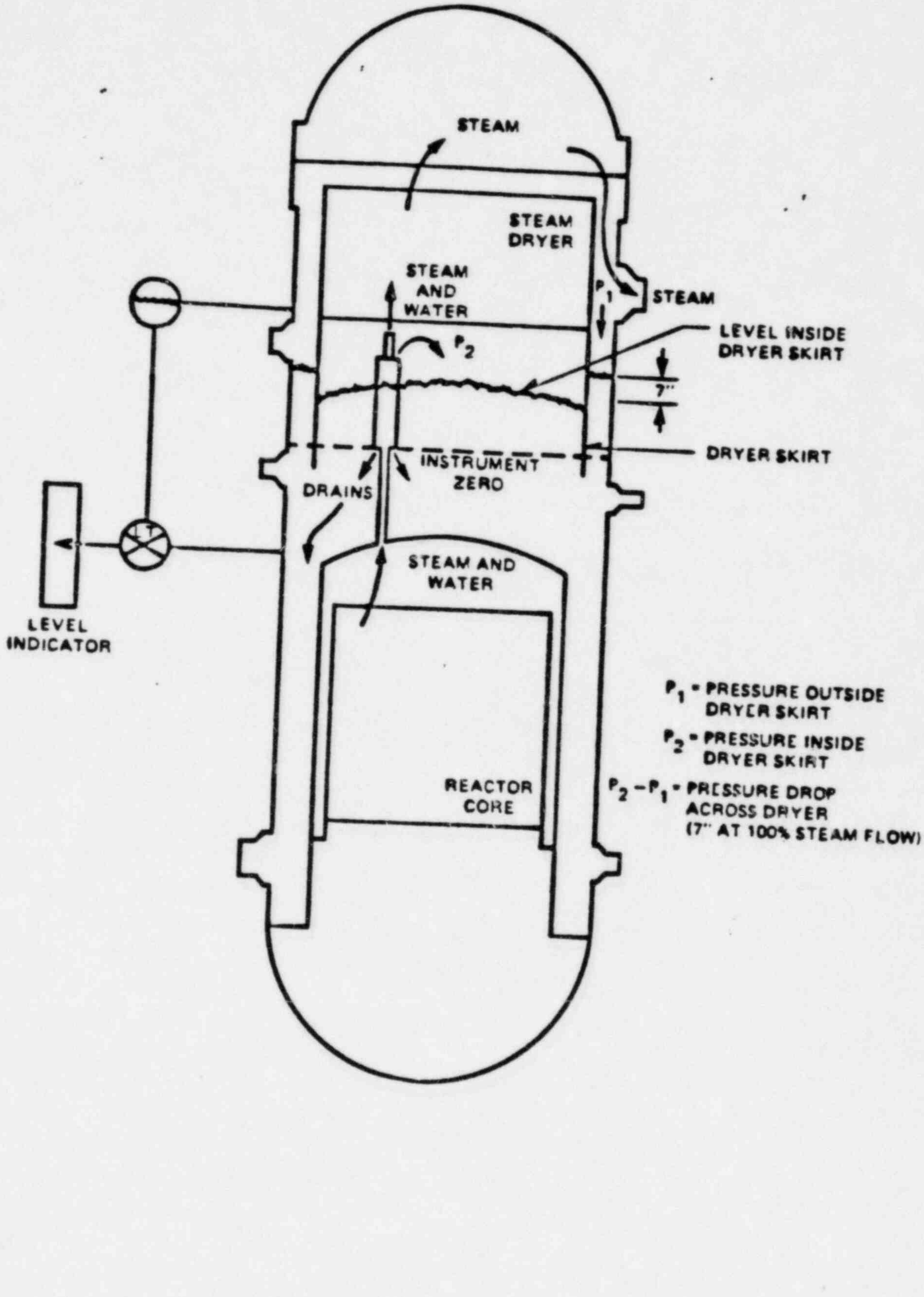
COMPENSATED REFERENCE LEG
EFFECT OF REDUCING P_{RX} FROM 1000 psi to 100 psi



dP CELL CALIBRATED FOR P_{RX} = 1000 psi T_{RL} = 299°F

FINAL CONDITIONS: P_{RX} = 100 psi T_{RL} = 212°F

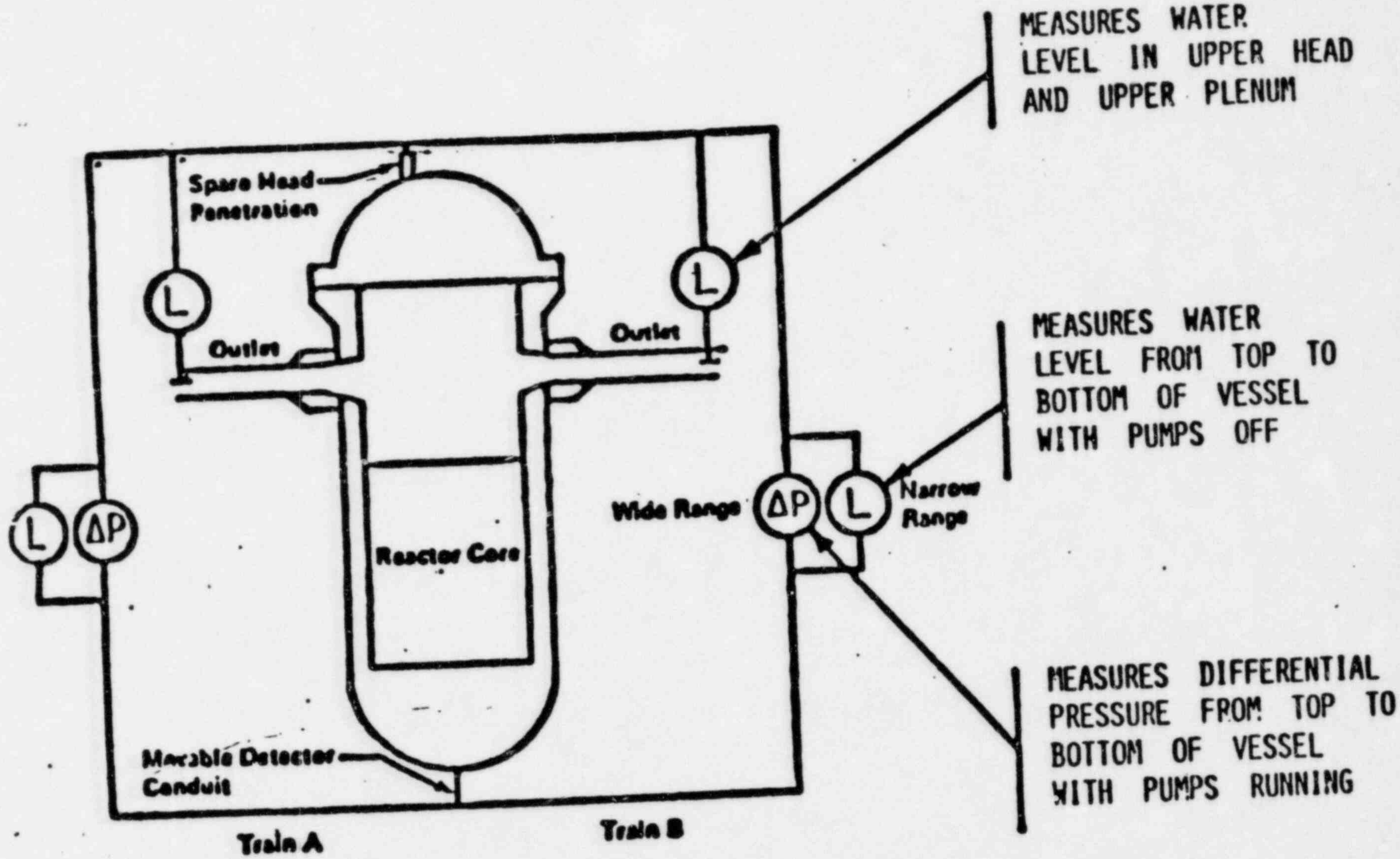
$$T_{RL} = T_{CONTAINMENT} + 0.4\ (T_{RX} - T_{CONTAINMENT})$$



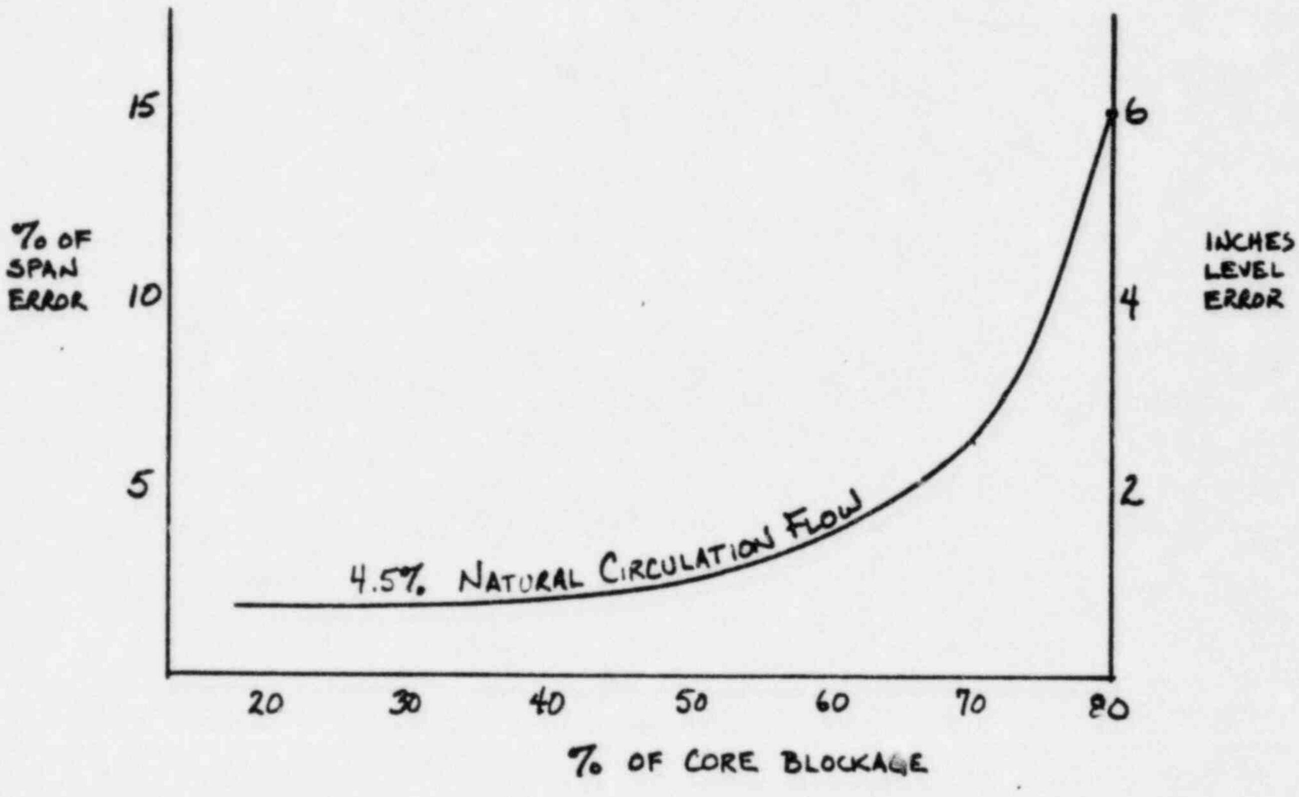
Steam Flow Effect on Vessel Level

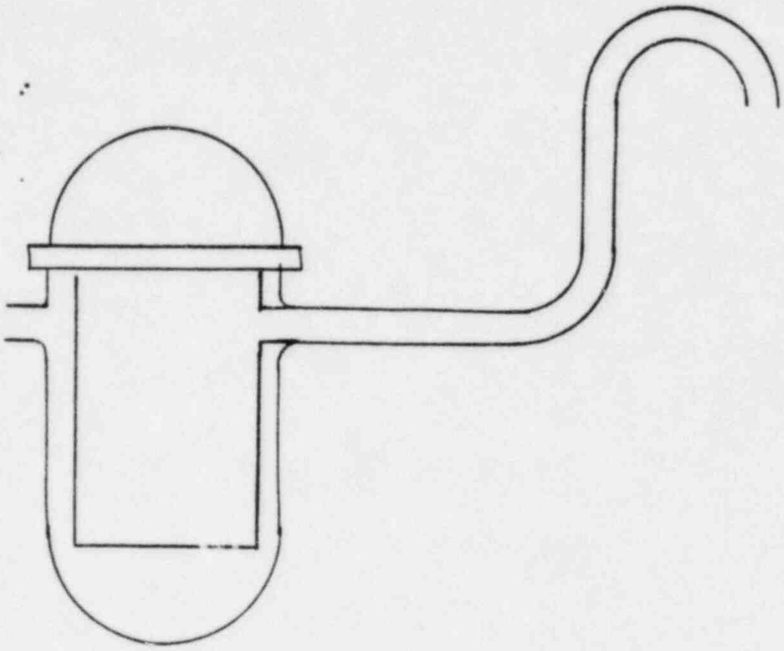
20.

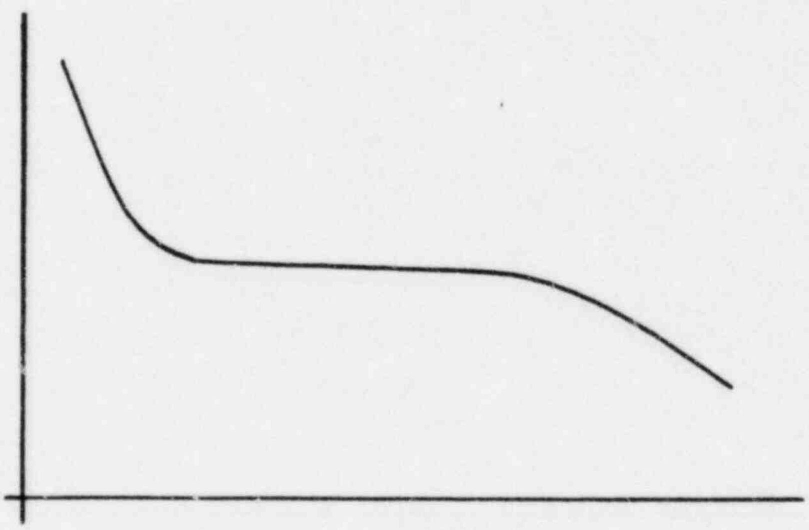
A-39Y



REACTOR VESSEL WATER LEVEL MEASUREMENT SYSTEM


15
10
5
% OF
SPAN
ERROR
6
4
2
INCHES
LEVEL
ERROR
4.5% NATURAL CIRCULATION FLOW
20
30
40
50
60
70
80
% OF CORE BLOCKAGE





L_I
t

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. A. Goldberg and M. C. Juhas "Lower-Bound K_{Iscc} Values for Bolting Materials - A Literature Study", NUREG/CR-2467 (Lawrence Livermore National Laboratory), October 1, 1981.

2. Memorandum, C. Kammerer, Director, Office of Congressional Affairs of NRC to Commissioners, Udall Committee Oversight Hearing on the NRC FY 1983 Budget, March 4, 1982.

3. Memorandum, S. Duraiswamy, ACRS Staff Engineer to ACRS Members, Material Associated with Mr. Udall's Subcommittee Hearing on the NRC FY-1983 Safety Research Budget, March 5, 1982.