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Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the local public document room located at the Multnomah County Library, Social Science and Science Department, 801 SW 10th Avenue, Portland, Oregon 97205. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 18th day of June 1982.

For the Nuclear Regulatory Commission.

Charles M. Trammell,

Acting Chief, Operating Reactors Branch No. 3, Division of Licensing.

[FR Doc. 82-17398 Filed 6-25-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-537]

Tennessee Valley Authority and Project Management Corp.; Availability of Site Suitability Report for Clinch River Breeder Reactor Plant

Notice is hereby given that the Office of Nuclear Reactor Regulation has published its revised Site Suitability Report for the Clinch River Breeder Reactor Plant, to be located on the Clinch River in the town of Oak Ridge, Roane County, Tennessee. Notice of receipt of Tennessee Valley Authority and Project Management Corporation [1] application to construct and operate the Clinch River Breeder Reactor Plant was published in the Federal Register on June 12, 1975 (40 FR 25110).

The report is being referred to the Advisory Committee on Reactor Safeguards and is being made available at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. 20555; at the Oak Ridge Public Library, Civic Center, Oak Ridge, Tennessee 37830; and at the Lawson McGhee Public Library, 500 West Church Street, Knoxville, Tennessee 37902, for inspection and copying. Copies of the Site Suitability Report (NUREG-0786) may be purchased, at current rates, from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161.

Dated at Bethesda, Maryland this 22 day of June 1982.

[1] The Energy Research and Development Administration (ERDA) became an applicant on May 1, 1976; subsequently, ERDA became the Department of Energy on October 1, 1977.

For The Nuclear Regulatory Commission.

Paul S. Check,

Director, CRBR Program Office, Office of Nuclear Reactor Regulation.

[FR Doc. 82-17399 Filed 6-25-82; 8:45 am]

BILLING CODE 7590-01-M

Draft Regulatory Guide; Issuance and Availability

The Nuclear Regulatory Commission has issued for public comment a draft of a proposed revision to a guide in its Regulatory Guide Series together with a draft of the associated value/impact statement. This series has been developed to describe and make available to the public method acceptable to the NRC staff of implementing specific parts of the Commission's regulations and, in some cases, to delineate techniques used by the staff in evaluating specific problems or postulated accidents and to provide guidance to applicants concerning certain of the information needed by the staff in its review of applications for permits and licenses.

The draft, temporarily identified by its task number, SG 049-4 (which should be mentioned in all correspondence concerning this draft guide), is proposed Revision 1 to Regulatory Guide 5.53 and is entitled "Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay." This guide describes methods and procedures acceptable to the NRC staff for meeting the provisions of the Commission's regulations as they relate to the use of nondestructive assay as used in material control and accounting systems to detect unaccounted-for loss or diversion of special nuclear material to unauthorized uses. This guide endorses and supplements ANSI N15.20-1975, "Guide to Calibrating Nondestructive Assay Systems."

This draft guide and the associated value/impact statement are being issued to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the draft value/impact statement. Comments on the draft value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by August 20, 1982.

Although a time limit is given for comments on these drafts, comments and suggestions in connection with (1) items for inclusion in guides currently being developed or (2) improvements in all published guides are encouraged at any time.

Regulatory guides are available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C. Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control. Telephone requests cannot be accommodated. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 22nd day of June 1982.

For the Nuclear Regulatory Commission.

Karl R. Goller,

Director, Division of Facility Operations, Office of Nuclear Regulatory Research.

[FR Doc. 82-17400 Filed 6-25-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards; Meeting

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b.), the Advisory Committee on Reactor Safeguards will hold a meeting on July 8-10, 1982, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published in the Federal Register on June 16, 1982.

The agenda for the subject meeting will be as follows:

Thursday, July 8, 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.-12:45 P.M.: Perry Nuclear Power Plant, Units 1 and 2 (Open)—The Committee will hear the report of its Subcommittee and consultants who are present regarding the request for an Operating License for the Perry Nuclear Power Plant, Units 1 and 2.

The Committee will hear and discuss reports from members 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.

45 P.M.–3:45 P.M.: NRC Safety arch Program (Open)—The members will hear and discuss the report of its Subcommittee Chairman and designated members regarding the proposed ACRS report to NRC on the proposed NRC Safety Research Program and Budget for FY 1984–85.

Members of the NRC Staff will participate as appropriate.

Portions of this session will be closed as required to discuss detailed contractual negotiation information the premature release of which would be likely to significantly frustrate the performance of the Committee's statutory function.

3:45 P.M.–6:45 P.M.: Robert E. Ginna Nuclear Power Plant (Open)—The members of the Committee will hear the reports of its Subcommittee and consultants who are present regarding the SEP review of this power plant. Members of the NRC Staff and representatives of the Applicant will also make related presentations and respond to questions by the Committee members.

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

day, July 9, 1982

8:30 A.M.–11:30 A.M.: Clinch River Breeder Reactor (Open)—The Committee will hear the reports of its Subcommittee and consultants who may be present regarding the adequacy of the site proposed for this facility.

Representatives of the NRC Staff and the "applicant" will also make related presentations and respond to questions by the Committee members.

11:30 P.M.–12:30 P.M.: Disposal of High-Level Radioactive Wastes (Open)—The members will hear the report of its Subcommittee and consultants who are present regarding the proposed NRC regulation (10 CFR Part 60), Criteria for High Level Waste Disposal.

Representatives of the NRC Staff will make presentations and respond to questions as appropriate.

12:30 P.M.–1:00 P.M.: Future Committee Activities (Open)—The members will discuss anticipated subcommittee activities and items proposed for consideration by the full Committee.

2:00 P.M.–3:00 P.M.: Decay Heat Removal Systems (Open)—The embers will hear the report of the CRS Subcommittee and consultants who are present regarding the NRC Task Action Plan (A–45), Evaluation of Alternate Decay Heat Removal Systems.

Representatives of the NRC Staff will also make presentations and respond to questions by the ACRS members.

Representatives of the nuclear industry will participate as appropriate.

3:00 P.M.–5:00 P.M.: NRC Reactor Safety Research Program (Open)—The members will continue discussion of the proposed ACRS report to NRC regarding the proposed NRC Safety Research Program and Budget for FY 1984–85.

Portions of this session will be closed as required to discuss detailed contractual negotiation information the premature release of which would be likely to significantly frustrate the performance of the Committee's statutory function.

5:00 P.M.–6:30 P.M.: ACRS Subcommittee Activity (Open)—The members will hear and discuss the reports of ACRS Subcommittee Chairmen regarding safety related matters including resolution of steam generator tube integrity problems; proposed DOE program for siting and assessment of high level radioactive waste repositories; proposed changes in 10 CFR Part 20, Standards for Protection Against Radiation and use of radioiodine blocking agents.

Saturday, July 10, 1982

8:30 A.M.–12:30 P.M. and 1:30 P.M.–3:30 P.M.—Preparation of ACRS Reports (Open/Closed)—The members will discuss proposed reports to NRC regarding items considered during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information related to matters being discussed, information involved in an adjudicatory proceeding, and information the premature release of which would be likely to seriously inhibit the Committee in the performance of its statutory function.

3:30 P.M.–4:00 P.M.: New ACRS Members (Closed)—The Members will discuss the qualifications of candidates proposed for nominations as ACRS members.

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

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

I have determined in accordance with Subsection 10(d) Pub. L. 92–463 that it is necessary to close portions of this meeting as noted above to discuss Proprietary Information [5 U.S.C. 552b(c)(4)] applicable to the matters being discussed, preliminary information the release of which would be likely to significantly frustrate performance of the Committee's statutory function [5 U.S.C. 552b(c)(9)(B)], and information the release of which would represent a clearly unwarranted invasion of personal privacy [5 U.S.C. 552b(c)(6)].

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

Dated: June 22, 1982.

John C. Hoyle,

Advisory Committee Management.

[FR Doc. 82–17401 Filed 6-25-82; 8:45 am]

BILLING CODE 7590-01-M

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

Radio Technical Commission for Aeronautics (RTCA) Executive Committee; Meeting

Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92–463; 5 U.S.C. App. I) notice is

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

June 23, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
267TH ACRS MEETING
July 8-10, 1982
WASHINGTON, DC

Thursday, July 8, 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 Remarks 1.2) Items of interest regarding ACRS activities
2) 8:45 A.M. - 12:45 P.M.	Perry Nuclear Power Plant Units 1 and 2 (Open) 2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee and consultants regarding the operating license for this plant (JJR/AJC/GRQ) 2.2) 9:15 A.M.-12:45 P.M.: Reports by and discussions with representatives of the NRC Staff and the Applicant (Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter.)
12:45 P.M. - 1:45 P.M.	LUNCH
3) 1:45 P.M. - 3:45 P.M.	NRC Safety Research Program (Open) 3.1) Proposed ACRS Report to NRC regarding FY 1984-85 Safety Research Program and Budget (CPS, et al./SD, et al.) (Portions of this session will be closed as necessary to discuss information the release of which would be likely to frustrate performance of the Committee's statutory function.)

4) 3:45 P.M. - 6:45 P.M.

Robert E. Ginna Nuclear Plant (Open)

4.1) 3:45 P.M.-4:15 P.M.: Report of ACRS Subcommittee regarding SEP review (CPS/RKM)

4.2) 4:15 P.M.-6:45 P.M.: Meeting with NRC Staff and Applicant

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

Friday, July 9, 1982, Room 1046, 1717 H Street, NW, Washington, DC

5) 8:30 A.M. - 11:30 P.M.	Clinch River Breeder Reactor (Open) 5.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee regarding the proposed site for the CRBR (MWC/PAB) 5.2) 9:00 A.M.-11:30 A.M.: Meeting with the NRC Staff and the "applicant"
6) 11:30 A.M. - 12:30 P.M.	Disposal of High Level Radioactive Waste (Open) 6.1) Report of ACRS Subcommittee regarding proposed NRC Rule (10 CFR Part 60) Criteria for High Level Waste Disposal (RCA/RCT)
7) 12:30 P.M. - 1:00 P.M.	Future Committee Activities (Open) 7.1) Discuss anticipated Subcommittee activities (MWL) 7.2) Discuss items proposed for ACRS review (RFF)
1:00 PM. - 2:00 P.M.	LUNCH
8) 2:00 P.M. - 3:00 P.M.	Decay Heat Removal Systems (Open) 8.1) 2:00 P.M.-2:20 P.M.: Report of ACRS Subcommittee regarding NRC Task Action Plan A-45, Evaluation of Alternate Decay Heat Removal Systems (DAW/RS) 8.2) 2:20 P.M.-3:00 P.M.: Meeting with NRC Staff
9) 3:00 P.M. - 5:00 P.M.	NRC Reactor Safety Research Program and Budget (Open) 9.1) Discuss proposed ACRS report to NRC regarding the proposed NRC Safety Research Program and Budget for FY 1984-85 (CPS et al./SD et al.) (Portions of this session will be closed as necessary to discuss information the release of which would be likely to frustrate performance of the Committee's statutory function.)

10) 5:00 P.M. - 6:00 P.M.

<u>ACRS Subcommittee Activity</u> (Open)

10.1) Report of Metal Components Subcommittee Chairman regarding Steam Generator Tube Integrity (PGS/EI)

10.2) Report of Acting Subcommittee Chairman regarding proposed DOE Program for Siting and Assessment of High Level Waste Repositories (RCA/RCT)

10.3) Report of Acting Subcommittee Chairman regarding proposed changes in NRC Rule (10 CFR Part 20), Standards for Protection Against Radiation (RCA/RCT)

Saturday, July 10, 1982, Room 1046, 1717 H Street, NW, Washington, DC

11) 8:30 A.M. - 12:30 P.M. — Proposed ACRS Reports to NRC (Open/Closed)
- 11.1) 8:30 A.M.-9:30 A.M.: ACRS report on Perry Nuclear Plant - OL (JJR/AJC) (Closed)
- 11.2) 9:30 A.M.-10:30 A.M.: ACRS report on CRBR site suitability (MWC/PAB)(Closed)
- 11.3) 10:30 A.M.-12:30 P.M.: ACRS report on proposed NRC Safety Research Program and Budget (CPS et al./SD et al.) (Open)

12:30 P.M. - 1:30 P.M. — LUNCH

12) 1:30 P.M. - 3:15 P.M. — Proposed ACRS Reports to NRC (Open/Closed)
- 12.1) 1:30 P.M.-2:15 P.M.: ACRS report on R.E.Ginna-SEP (CPS/RKM) (Closed)
- 12.2) 2:15 P.M.-2:45 P.M.: ACRS report/comments regarding 10 CFR 60, Criteria for High Level Waste Disposal (RCA/RCT) (Open)
- 12.3) 2:45 P.M.-3:15 P.M.: ACRS report/comments regarding Task Action Plan A-45 (Open) (DAW/RS) (Open)

13) 3:15 P.M. - 3:45 P.M. — New ACRS Members (Closed)
- 13.1) Discuss candidates proposed for appointment to ACRS (JJR/MCG)

14) 3:45 P.M. - 4:30 P.M. — Concluding Session (Open/Closed)
- 14.1) Complete preparation of ACRS reports regarding matters considered during this meeting
- 14.2) Proposed memo from R. Fraley to the EDO regarding the Torsional Ultrasonic Water Level Detector (JJR/RS) (Open)
- 14.3) J.Carson Mark remarks regarding Quantitative Safety Goals (JCM/RFF) (Open)

Issue Date:

December 13, 1982

MINUTES OF THE
267TH ACRS MEETING
JULY 8-10, 1982
WASHINGTON, DC

CERTIFIED

The 267th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Chairman P. Shewmon at 8:30 a.m., Thursday, July 8, 1982.

[Note: For a list of attendees, see Appendix I. W. Kerr, H. W. Lewis and H. Etherington were not present during the Meeting. M. Bender was not present on Friday or Saturday; D. W. Moeller was not present on Thursday or Friday; and M. S. Plesset was not present on Saturday.]

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

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

I. Chairman's Report (Open to Public)

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

Chairman Shewmon referred the Committee to a letter to Chairman Palladino by J. Carson Mark regarding supplementary comments on the subject of quantitative safety goals (see Appendix IV). C. Mark suggested that, barring any concerns from Committee Members, these comments might be appended as supplementary remarks to the Committee's letter from the 266th Meeting regarding quantitative safety goals.

II. Perry Nuclear Power Plant Units 1 and 2 Operating License Review (Open to Public)

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

OFFICIAL USE ONLY

A. Report of the ACRS Subcommittee

J. J. Ray called the Committee's attention to a memorandum regarding the Perry construction status (see Appendix V). He indicated that the consensus of the Subcommittee after the site tour on June 28th was favorable with respect to the physical status of the plant. They found good equipment spacing throughout the plant.

J. J. Ray pointed out an interesting feature of the Cleveland Electric mode of operation with respect to blackouts. He explained that the East Lake fossil plant is switched off and isolated from the system during blackout conditions by means of underfrequency relays so that it is available for black start to bring the system back. There were nine requests for time to speak from the public during the Subcommittee meeting. Their main concerns were emergency planning (particularly with respect to the practicality of alerting and evacuating the public during emergencies) and quality assurance as experienced at the plant. J. J. Ray alerted the Committee that Subcommittee Members were particularly concerned with the lack of commercial nuclear operating experience of plant personnel.

J. J. Ray pointed out several items to which the Committee should pay particular attention:

- PGCC system of controls and physical configuration of the interconnections between control systems and the control room accessories

- Differing positions of the Staff and Applicant regarding CO_2 fire suppression

- Remote shutdown panel arrangement which can disable the control room panels when a fire occurs at the remote shutdown panel.

M. S. Plesset questioned CEI as to their participation and knowledge concerning the Mark III dynamic load question. He asked whether the Applicant was aware of tests of SRVs performed in Taiwan in Mark III containments. These questions were deferred for discussion by the Applicant later in the session.

B. Introductory Statement by Cleveland Electric Illuminating Company (CEI)

D. R. Davidson, Vice-President of the System Engineering and Construction Group of CEI, described the Perry Plant site and the qualifications of certain key management people at CEI. P. G. Shewmon inquired about the engineering capabilities of the Perry staff, and M. S. Plesset requested a discussion about quality assurance and quality control. M. Bender expressed interest in the role of the constructor of the nuclear steam supply and the architect/engineer in the quality control program at Perry. M. R. Edelman, Division Manager of Nuclear Engineering Construction for CEI, indicated that the Perry project organization which manages the total construction is an integrated organization with a separate quality assurance department. He mentioned that a construction section at CEI provides second level surveillance on top of the contractor's own QA/QC program. The QA organization at CEI provides an audit function that audits contractors programs onsite. M. W. Carbon asked D. R. Davidson about the commercial BWR experience of the CEI management people.

C. Report by the NRC Staff

A. Schwencer, NRC Staff, requested that the ACRS consider writing a report addressing operation of Units 1 and 2 at full design power subject to resolution of remaining open items on the BWR/6 Mark III containment. He mentioned concerns regarding the containment from a former General Electric employee named Humphrey which would be referred to the ACRS Fluid Dynamics Subcommittee at the end of July. The issues of LOCA loads and hydrogen control will be addressed in Supplement 3 to the SER on the Grand Gulf nuclear plant. J. Ebersole expressed personal objection to the request for a full power operating license which he considered premature because of the state of construction and the number of open issues on Perry. M. Bender asked how the Humphrey questions would impact the Perry license review. J. Kudrick, NRC Staff, indicated that there are no fundamental safety concerns still outstanding that have been raised by Humphrey, although detailed analyses are necessary for final resolution of some of the Mark III containment questions.

J. Stefano, NRC Staff, presented a comparison of the Perry Plant with Clinton and Grand Gulf (see Appendix VII). He pointed out that the Perry Plant is a free standing steel vessel supported by a steel-lined reinforced concrete foundation mat while Clinton and Grand Gulf are steel-lined, concrete reinforced structures. The Committee discussed a comparison of containment static pressure safety margins for the Perry and Clinton stations. J. P. Knight, NRC Staff, pointed out that the full question of containment capability must also address the question of containment leakage. M. S. Plesset asked regarding the Perry containment response to impulse loads which occur within a very short period. B. Jeng, NRC Staff, indicated that static pressure is controlling as the basis of Staff analyses of ultimate capacity.

W. V. Johnston, NRC Staff, explained the unresolved issues regarding fire protection in the control room through the use of a CO_2 fire suppression system. One issue involved the determination of permissible levels of CO_2 in the control room. He expressed the Staff's concern that levels of CO_2 to extinguish a fire could exceed the levels that are permissible for occupation of the control room. H. Krug, NRC Accident Evaluation Branch, indicated that the Accident Evaluation Branch did not feel that the operator would have a problem with CO_2 since there would be sufficient time for the operators to utilize respiratory equipment. J. Ebersole expressed concern that since the CO_2 fire suppression systems are not seismically competent, they might, under a moderate earthquake condition, spray uncontrolled CO_2 into the control room as well as possibly the diesel generator rooms, causing the diesel generators to become inoperative. J. Stang, NRC Staff, indicated that it was his understanding that this system automatically shuts itself off after a predetermined time.

D. Okrent discussed the turbine missile issue which is based upon an unfavorable orientation of the turbines at Perry. He expressed concern regarding the adequacy of shutdown heat removal, taking account of the possibility of an earthquake induced small LOCA. N. E. Fioravante, NRC Staff, answered an additional question by D. Okrent by noting that the Staff does a compartmental internal flooding review in its review of high energy and monitored energy line pipes.

D. Okrent asked whether the Nuclear Review Safety Committee at Perry had outside employees of the operating company in membership.

D. Okrent brought up the Quad Cities event involving the loss of all power when the Licensee inadvertently deenergized the startup transformer. E. Goodwin indicated that the running unit was left without diesels for an hour and a half, and the use of some instrumentation in the control room was also lost on the operating unit. J. J. Ray suggested that the Applicant explain how they are kept aware of LERs at operating plants.

W. E. Coleman, CEI, indicated that CEI and the Staff are far apart on the issue of the CO_2 system in the control room. With regard to a fire in the remote shutdown panel, he indicated that there are two divisions in the control room and a fire in the remote shutdown panel could be isolated in only one division panel such that the other division would be still operative in the control room. He also indicated that CEI is involved in the Owners Group discussion of the Mark III containment load problem.

D. Presentation by the Applicant

M. Edelman, CEI, described the present site organization and discussed staffing levels (see Appendix VIII). He elaborated on the Nuclear Engineering Department in detail, describing in particular the Nuclear Analysis Section which will have responsibility for licensing backfits, corporate health physics, the ALARA program, PRA analysis, human factors, as well as staffing of the independent Safety Engineering Group. He discussed the Licensing and Fuel sections at Perry indicating that CEI is developing its own capability inhouse to do thermal analysis for replacement of the core on Perry as well as the ability to do core simulation.

M. J. Titas, CEI, described the Nuclear Project Training Section within the Perry Project Services Department responsible to organize and centralize construction and support training activities for the project (see Appendix IX). He mentioned an internal evaluation of the onsite training program which indicated that the training and support functions needed coordination and better organization. Since it was found that operator training at the Perry Plant unit was progressing well, operator training activities were separated from support function training with the objective of eventual recombination of these activities at the time of fuel load. He mentioned CEI's attempts at establishment of an associated degree program in quality assurance and indicated that they were in the process of setting up special training courses in transient analysis, inservice inspection and nondestructive examination, reactor physics, and PRA. P. G. Shewmon inquired regarding the procedure for selecting a qualified instrumentation control technician to handle a system problem that occurs after hours or on a weekend. J. J. Waldron, CEI, indicated that a computer printout was in preparation which would contain a list of qualified engineers with instructions that the shift operator call a qualified person to handle an after hours problem. He added that there were onshift instrumentation technicians who would be qualified to handle most routine work that would be expected.

J. J. Waldron described the responsibilities of the Nuclear Services Section (see Appendix X). He indicated that CEI had committed to the NRC Staff to have an individual with commercial BWR nuclear power plant experience on shift at least one year prior to fuel load and that a current CEI employee working on the project could fill this position. J. J. Waldron explained the CEI practice of hiring Nuclear Navy people and farming them out to operating BWR plants on temporary assignments with other utilities to gain operating experience. He indicated that these temporary assignments last six weeks to six months.

W. E. Coleman presented an overview of systems interactions at the Perry Plant (see Appendix XI). He mentioned a management action plan recently completed by Cygna Energy Services which relates PRA and systems interactions to various needs of the Perry organization. J. Ebersole inquired regarding the RPS logic of the usual 1 out of 2 logic taken twice which is used in most GE designs. He pointed out that this logic provided coincident and not redundant system design, invalidating the single failure criterion. The Committee discussed the subject with the Applicant and the NRC Staff. As a result of this discussion, J. Ebersole requested that the NRC Staff respond to this matter in generic and specific detail and that the Applicant also respond to the problem at a later date.

J. Ebersole suggested to W. E. Coleman that under even a modest-sized seismic event, since the fire protection systems are not seismically designed, CO_2 would be injected into the diesel generator rooms. W. E. Coleman indicated that this question had been presented before and CEI is studying the ability of the diesel generators to operate without air cooling in a sealed room with CO_2. J. Ebersole then requested that the NRC Staff investigate this common mode failure of the fire protection system where a seismic event could cause the fire protection system to activate and flood the diesel generator rooms with CO_2.

D. Okrent asked W. E. Coleman to list the general categories of systems interactions to be treated in CEI's proposed analysis. W. E. Coleman indicated that CEI will address three NRC control system failure questions. The purpose of the study would be to examine adverse systems interactions between nonclass 1E and class 1E systems which could affect the ability of the plant to achieve and maintain cold shutdown. He defined an adverse systems interaction as the occurrence of a set of dependent failures that defeats or jeopardizes the performance of a safety function. He explained that CEI plans to study systems interactions that affect the reactor. He also indicated that CEI would treat the probability of occurrence of the loss of all a.c. power and its survivability.

R. A. Pender, CEI, discussed the Perry evaluation of safety relief valve hydrodynamic loads. He indicated that the complete dynamic analysis included all major structures within the reactor building and as expected, showed very little effect of hydrogen loads on the auxiliary building (see Appendix XII). He indicated that pool swell continues to have a major impact on the Perry design. He then described Perry's program for resolution of the Humphrey issues suggesting that they could be resolved prior to fuel load.

J. Ebersole expressed concern regarding the extent of the buckling load which might be placed on the drywell shell as a result of hydrogen combustion. He inquired as to the margin the Applicant had in its stress analysis. E. M. Buzzelli, CEI, indicated that the design capability of the drywell structure from a hydrogen burn analysis and use of the CLASIX code was a maximum pressure differential of 16 to 18 psig. J. Ebersole asked about the ability of the SRVs to survive this impact and successfully reseat many times into the containment. E. M. Buzzelli indicated that this situation was under study and being pursued generically. Some Members expressed distrust of the use of the CLASIX code and suggested that there was another way to determine the pressure differential values.

D. Okrent asked whether CEI had inquired into the feasibility of supplying additional instrumentation beyond that normally supplied for inadequate core cooling. This instrumentation would either provide additional information or coordination of information for the operators or be available to provide data which is absent because of instrumentation which has failed in an accident beyond the design basis. P. A. Nevins, CEI, indicated that the Perry organization is relying on its active participation in the Regulatory Guide 1.97 Owners Group to provide a detailed analysis of inadequate core cooling to the Staff by the end of July 1982. P. A. Nevins described an emergency response information system, a data acquisition and displacement system to be installed at Perry which is expected to provide information to the operator under both normal and abnormal operating modes.

J. J. Ray indicated that equipment qualification was in process at Perry and that he knew of no particular problems with the process at this time. J. Ebersole asked whether CEI had an evaluation program to ascertain the appropriateness of the original determination of the location of electrical instrumentation and control components, and whether the ultimately extremely hostile environments for this instrumentation is appropriate. The Applicant indicated that no such program existed.

S. Kensicki explained the development of the Emergency Plan for Perry (see Appendix XIII). He described the Perry emergency organization, defining the responsibilities of subordinate managers to the Emergency Director. D. Okrent requested that S. Kensicki indicate an individual or individuals who have knowledge of the kinds of accident sequences, consequences, and phenomena involved as could be read in documents such as WASH-1400. W. E. Coleman pointed to the general supervisor in licensing or the division manager of nuclear engineering as having knowledge of the results of the Perry RSSMAP mini-PRA. D. Okrent explained that he was more interested in the person who had experience and would understand various postulated accident sequences which might lead to core damage or core melt, or sequences which might have an effect on hydrogen generation and pressure generation. S. Kensicki suggested that the shift supervisors would be individuals who would understand the various scenarios of possible accidents.

J. J. Ray mentioned a communication to the Subcommittee concerning the possible storage of liquid propane gas in former salt mines under the proposed location for the Lakeland County's Emergency Operations Center. He inquired whether there were voids under the site from former mining into which such propane gas might leak and therefore constitute a hazard. L. Beck, CEI, indicated that all mineral rights around the site had been acquired by CEI to a distance great enough that a propane detonation would not overpressure any of the buildings on site. He added that CEI was not aware that any mining had taken place under the site and indicated that CEI had salt rights to preclude anyone from doing solution or salt mining. R. Axtmann asked questions about the criteria for location of emergency operations facilities as noted in NUREG-0696. S. Kensicki described in detail Perry plans for the main emergency operations facility and the backup emergency facility.

Chairman Shewmon asked the Applicant what he was doing to control stress corrosion cracking of major piping and piping welds. W. E. Coleman indicated that because of problems of cracking of the reactor vessel nozzles, Perry completely replaced all of those reactor vessel nozzles to eliminate the stresses seen in some of the older BWRs. Chairman Shewmon asked about stress corrosion cracking in other primary system piping. W. E. Coleman indicated a complete NRC Staff review had been done of plant piping for intergranular stress corrosion cracking susceptibility.

Chairman Shewmon asked about the Perry Plant's capability for deaeration on startup. S. Kensicki indicated that CEI had incorporated a d.c. heater into this deaeration design to control oxygen levels during plant operation on the feedwater side. D. Kensicki indicated that the Perry Plant does not have capability for deaeration on startup. The Committee discussed the subject of oxygen content on startup including GE's recommendations for oxygen levels.

D. Okrent pointed out that an overall a.c. system reliability study done for CEI showed a lack of sophistication in the choice of data used for the diesel's reliability.

M. Bender asked whether the Staff intended to prepare a comprehensive construction report on plant quality for the Perry plant. R. L. Tedesco, NRC Staff, indicated that a comprehensive quality report is not a requirement and is only done on request. Chairman Shewmon suggested that the Staff submit to M. Bender as an example, a copy of a comprehensive construction quality report on a plant such as LaSalle or Byron. M. Bender indicated his interest in a sample report.

M. Bender asked whether the NRC Staff had developed a position or judgment on the adequacy of the training program at Perry. M. L. Gildner, NRC Staff, indicated that the nonoperator licensing training as indicated in the SER has been fragmented and has had some problems. He indicated that it was premature to evaluate the new centralized training facility organized over the past four months at Perry. M. Bender then asked the Staff about their judgment with respect to the overall organizational capability at Perry. M. L. Gildner responded generally favorably to CEI's record on retention of experienced personnel, indicating that experiences gained are likely to be retained inhouse. J. J. Ray questioned if the Staff had rated Perry's quality assurance organization. C. Williams, NRC Region III, indicated that it was his personal opinion that the Perry organization should be considered above average in relationship to its responsiveness to problems and issues and in response to the generally displayed construction of the plant.

J. J. Ray questioned what studies CEI has participated in or applied from the viewpoint of human factors in the control room. A. G. Migas, CEI, mentioned a detailed control room checklist to review control rooms developed by a subcommittee of the BWR Owners Group in which CEI is a participant (see Appendix XIV). He indicated that an Owners Group survey had been conducted of the Perry control room in September of 1981. This survey included operator interviews, reviewing panel layouts for conformance to human factors criteria, review of control room environment, and task analysis. He indicated that the task analysis was based on emergency guidelines developed by the BWR Owners Group.

J. Ebersole asked whether CEI contemplated control room exercises and tests to deliberately synthesize the disablement of nonqualified apparatus which is not class 1E, such as enunciators and indicators, to observe operator response to a degraded state of instrumentation.

A. G. Migas indicated that this testing is taking place on a simulator involving accident scenarios such as the loss of a.c. power.

III. <u>Site Suitability Review for the Clinch River Breeder Reactor (CRBR)</u> (Open to Public)

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

A. <u>Report of CRBR Subcommittee</u>

M. W. Carbon explained that the purpose of this meeting was to review NUREG-0786, the site suitability report in the matter of the Clinch River Breeder Reactor Plant, and to consider the suitability of the CRBR site for such a plant. He pointed out an important distinction made by the Staff in asking the ACRS to review the proposed site as "a suitable location for a reactor of the general size and type as the CRBR." He listed some major topics of import for the current review.

- The Staff's understanding of the general hazards of a plant of this size and type
- Comparison of the safety of an LMFBR with that of an LWR
- Basis for selecting the site suitability source term
- Appropriate level of safe shutdown earthquake
- Letter from T. Cochran of the National Resources Defense Council Inc. (see Appendix XV) which alleges that the Staff and Applicant have made strikingly different presentations on identical topics to the CRBR Subcommittee and the Licensing Board.

B. <u>Overview of CRBR Site Suitability Report</u>

P. Check, NRC Staff, stressed the distinction that the ACRS review is to consider the suitability of the Clinch River site, not the acceptability of the Clinch River reactor. He indicated that the reason for the review is the Applicant's request for a limited work authorization to begin certain site preparation activities, not to include any safety related work prior to receipt of the construction permit.

C. Thomas, Section Leader at NRC (CRBR Program Office) described the purpose of the Site Suitability Report (SSR) in terms of the definition of an LWA-1, proposed site preparation activities, the NRC's approach to a site suitability review, and the Site Suitability Report itself (see Appendix XVI). After presenting background material on an LWA-1, C. Thomas detailed site preparation activities into four general categories:

. General site clearing and grading

. Excavation

. Installation of temporary construction facilities

. Other miscellaneous activities allowable under 10 CFR 50.10(E).

C. Thomas defined the approach to the site suitability review as consisting of defining characteristics of the facility of the general size and type proposed relative to site suitability, determining characteristics of the proposed site, and assessing capability of the site and facility characteristics. He indicated that once the facility is defined, an early site review would be conducted, at which time a conclusion could be made as to the suitability of the facility of the general size and type at the proposed site. C. Mark asked whether a 1300 megawatt LWR electric plant could be built at the particular site being evaluated. C. Thomas suggested that he knew of no overriding consideration that would preclude licensing a larger plant at that site.

C. Thomas referred to a listing of contentions on the site suitability portion as well as the construction permit part of the ASLB hearing (see Appendix XVII). He indicated that only contentions 1A, 2, 3A - D, 5A and B, and 11D1 were relevant to site suitability and were being considered in the site suitability portion of the review. The ASLB has limited consideration of contentions 1, 2, and 3, which deal with inclusion of the core disruptive accident (CDA), the design basis accident spectrum, and the feasibility of designing the plant in such a way that the probability of CDAs could be made so low that they could be excluded from the design basis accident spectrum. He indicated that contention 5A dealt with site meteorology and population density, contention 5B dealt with long-term evacuation, and contention 11B1 dealt with radiological organ dose and equivalent limits. D. Okrent expressed interest in the NRC response to contention 5B. C. Thomas indicated that NRC was not offering an opinion since the NRC does not normally look at the effects of evacuation of nearby industrial facilities during the course of its review. C. Thomas added that the Department of Energy is ultimately responsible for making a decision to site a nuclear plant near facilities such as K 25. D. Okrent requested an explanation of contentions 3B - 3D. C. Thomas indicated that it is the NRC's position that core disruptive accidents are sufficiently improbable or could be made sufficiently improbable that they do not have to be considered in the spectrum of design basis accidents since the site suitability source term bounds the sources of that accident spectrum.

B. Morris explained the Staff's basis for the belief that the CRBR risk will be comparable to a current generation LWR in licensing today (see Appendix XVIII). He indicated that the site suitability source term for Clinch River is a nonmechanistic, postulated release of radionuclides into the containment but with no containment failure. That source includes some contribution from core melting that could only realistically be considered associated with some core disruption. He pointed out, however, that the source term is not based on an attempt to bound all postulated CDAs that one might consider. The NRC has required (in previous communication with the Applicant) that certain design measures be imposed on the Clinch River design to assure that severe accidents such as CDAs will be improbable and hence beyond the design basis spectrum.

Chairman Shewmon asked the meaning of a core disruption that involves an accident where radioactivity might get out of the pressure vessel. B. Morris indicated that once core disruption is postulated, you have to consider the possibility that there will be some mechanical or thermal damage done to the primary system. Chairman Shewmon asked whether the probability of this accident included other considerations beyond the initial core disruption.

B. Morris explained that the probability includes the initiation and the core disruption but not the subsequent failures of the primary system and/or containment. D. Okrent referred to the unresolved question of severe accident rulemaking with respect to LWRs, and questioned how the Staff and Applicant would accommodate severe accident rulemaking for the CRBRP. B. Morris indicated that measures should be taken to assure that the accidents are very improbable, and that the design should be capable of accommodating a severe accident with such design measures as to assure that the containment will survive for a long enough time that the consequences will be acceptably low.

B. Morris indicated that there are a number of deterministic criteria, that when applied will insure the NRC that the risk will be acceptably low from the CRBRP. The current plant design allows possible venting of the material inside containment subsequent to a core melt accident. Under the new set of criteria which supersede the 24 hour criteria presented in May, 1976 NRC indicated that such venting should not result in consequences greater than 10 CFR part 100. B. Morris pointed out, in addition, that these guidelines are designed for the purpose of assuring that venting will not be a severe health hazard compared to the subsequent failure. He also noted that the Staff would like for the containment to be capable of retaining radionuclides for a sufficiently long period of time subsequent to a core melt that the risk would be acceptably low to demonstrate comparability with light water reactors. Chairman Shewmon requested that the NRC Staff provide D. Okrent with a written explanation of the Staff's 24 hour containment design criteria covering core melt accidents as they now apply.

E. Rumble, SAI, described a risk comparability, quick scoping analysis, comparing the CRBRP design to LWRs (see Appendix XIX). He indicated that assumptions regarding human interactions, human factor items that would be compared with LWR procedures, were not available and were excluded from the probabilistic risk assessment. He discussed certain basic considerations for evaluating core disruptive accidents which point to similar initiator sources and causes for the CRBRP and LWRs (see the first page of Appendix XIX). He mentioned three types of accidents which could occur at the facility and indicated that internal plant failure, the first of these, was the only one considered in the analysis. External forces and sabotage, the other two categories, were excluded. He indicated that there were three phases to the analysis: an initiation phase, an investigation of the challenges to the primary system, and a look at challenges to the containment. He described several internal plant failures, the primary coolant system response to a core disruption, and the expected containment response.

D. Okrent asked what the weak points in the analysis were. E. Rumble indicated that the analysis was not a complete one, not a full blown PRA, and it did not include external events, sabotage, or human interactions. J. Ebersole questioned what the reliability was to estimate closure of the containment in a mechanical context involving the large purge valves hypothesized to close under pressure pulses and release rates one might see during an accident.

C. Site Suitability Source Term

J. Hulman, NRC, Chief of the Accident Analysis Branch of NRR, presented four interrelated subjects

- Risk of a beyond design basis accident
- Site suitability source term
- Dose guidelines for site acceptability
- Design basis accident enveloping event used for site suitability

He indicated that NRC has concluded that the risks of severe accidents beyond design basis events are generally comparable not only to a light water reactor of similar size to the CRBRP but also to a contemporary reactor of 1000 or 1200 Mw electrical (see Appendix XX).

J. Hulman indicated that the Staff concludes that they can use a nonmechanistic event analogous to that used for a light water reactor and postulate the source term for a limiting type of accident for site suitability. He added that the only difference between the two source terms would be the addition of plutonium as a significant potential dose contributor in the case of an LMFBR. J. Hulman indicated that, because of the possibility of releasing different radionuclides for the CRBRP, long bone marrow and liver organ dose equivalents have been added to the light water reactor dose guidelines in Part 100. In the case of the enveloping event, J. Hulman indicated that the Staff has basically tried to insure that the risk from the breeder for design basis accidents would not exceed the risk from a light water reactor. He indicated that criteria were developed on that basis and resulting doses were found to be a small fraction of that guideline.

Chairman Shewmon questioned the applicability of the source term. J. Hulman indicated that it is the judgment of the Staff that the source term postulated in terms of its contribution to potential doses is representative of some kinds of beyond design basis accidents and is not as conservative for other events. It is not a bounding source term for all possible breeder events. C. Mark asked whether plutonium is the dominant contributor to dose of all the heavy elements in the source term. J. Hulman volunteered to provide C. Mark with a written discussion of the matter of comparison of the dose contribution of plutonium to the dose contribution from the element curium which C. Mark suggested had 10 times as much activity as plutonium. This matter was resolved later in the meeting.

D. Okrent asked the Staff how they would treat the term sufficiently or probabilistically with regard to CDAs before the ASLB. B. Morris indicated that a numerical value as a discriminator would not be provided. He indicated that the Staff would base its answer on the deterministic criteria and the feasibility of achieving a high reliability for those systems that are supposed to prevent severe accidents.

R. Axtmann expressed concern that translating the LWR source term over to the CRBRP might violate assumptions of the original source term because of the difference in the water chemistry such as metal water reactions in an LWR. J. Hulman explained the original source term in Part 100 by referring to a TID document TID-14844. He expressed his opinion that the source term for the light water reactors was probably conservative and this conservatism would translate into the CRBRP source term. D. Okrent noted that personally he was more interested in the question of the containment capability for a spectrum of accidents rather that the source term for the DBA to meet Part 100.

G. Clare, Westinghouse, presented a table comparing the CRBRP site suitability source term to that used for siting foreign LMFBRs (see Appendix XXI). He indicated that he was unaware of an equivalent to the U.S. site suitability source term in France or West Germany.

G. Clare addressed the subject of nonradiological effects of sodium reaction product aerosols. He explained that it was prudent to provide aerosol mitigation features on the plant. He applied this specifically to the steam generator building where significant quantities of sodium in the intermediate heat transfer system piping might leak from that pipe in a fire or subsequent sodium fire and release significant amounts of aerosols to the environment. He presented an analysis and evaluation of the steam generator design basis leak (see Appendix XXI). G. Clare indicated that it is Westinghouse's conclusion that the offsite concentration of sodium reaction product aerosols from this accident would be low. D. Okrent asked whether the scenario of severe injury or death from sodium aerosols was studied. G. Clare indicated that Westinghouse had not studied that type of accident specifically.

R. Stark presented a handout summarizing the findings of the Staff on population and site location (see Appendix XXII). H. Piper, NRC Staff, pointed out in answer to a question by Chairman Shewmon that the Oak Ridge Laboratories are 4 1/2 miles from the proposed site.

D. Okrent expressed concern that the NRC Staff could demonstrate an adequate degree of protection for this plant taking account of the combination of the shutdown earthquake design and design basis in the actual design and the qualifications of various aspects of the plant. He added that the Staff could not draw fully on LWR experience on margins. D. Okrent indicated that he was concerned that the margin with regard to structural capability looked sufficiently large such that the plant could withstand an earthquake of lower probability but greater intensity than the design basis earthquake.

C. P. Siess asked whether a PRA for the CRBRP existed which included seismic effects. B. Morris indicated that a PRA will be performed which will include seismic margins and evaluations.

D. Hydrology

R. Lee, TVA, briefly described the determination of the design basis flood level at the CRBRP (see Appendix XXIV.) He indicated that two types of events were considered: a rainfall flood, and seismically caused floods. He added that in the seismically caused floods two types of conditions were considered:

- Operational basis earthquake (OBE) coincident with a 1/2 probable maximum flood (PMF)

- Safe shutdown earthquake, SSE, coincident with a 25 year flood (see Appendix XXIII)

He indicated that the controlling situation in both cases was failure of Norris Dam and the controlling elevation, the OBE failure of Norris. The Committee discussed the definition as per the TVA study of the standard project flood and the controlling elevation. Chairman Shewmon asked whether TVA had seismic criteria for the design of their dams. T. J. Abraham, TVA, indicated that the Norris Dam was analyzed for acceleration equal to 0.10g. He indicated that TVA had done a conservative analysis which indicated that the Norris Dam would not fail under the maximum earthquake that could be expected in the region. He added, however, that in the interest of conservatism in the siting of this nuclear plant, TVA took the most conservative position postulating failure of the dam.

Committee Members expressed interest in whether the Norris Dam was seismically analyzed to take account of the postulated SSE for the CRBRP of 0.125g. T. J. Abraham indicated that since the SSE was a more severe earthquake than that assumed in the analysis of the dam which was done previously on a judgmental basis, the breech of the dam was expanded on each side as a best estimate of what would happen. This increased the horizontal breech from 665 ft. to 835 ft. wide which TVA considered a very conservative analysis (see Appendix XXIV).

T. J. Abraham described the physical features of Norris Dam and indicated that in dam safety analysis, TVA designed the Norris Dam for a maximum credible earthquake of 0.15g. He mentioned certain conservatisms built into the seismic analysis:

- Pseudo static method of analysis where acceleration is maximized and these forces applied as a static stability

- Simplified dynamic analysis to arrive at the moments and sheers in the amplification of the rock acceleration load

- Concrete with no tensile ability

- Conservative judgment in assessing stress analysis after the final stability analysis has been made.

C. P. Siess discussed with T. J. Abraham the details of the hydrodynamic forces involved in the seismic analysis. He asked if the Norris Dam failed, what would be the consequences other than possible damage to the CRBRP such as possible damage to the Watts Bar Nuclear Plant or the towns of Chattanooga, Oak Ridge, and Knoxville. R. Lee indicated that TVA had not carried the failure of the Norris Dam down that far. He did note that Knoxville is on high ground and would not have the same flood problem as Chattanooga. T. J. Abraham stressed the TVA dam safety program which paid particular attention to seismic analysis. The Norris Dam is considered to be safe against normally expected maximum credible earthquakes.

IV. SEP Review of the Robert E. Ginna Nuclear Plant (Open to Public)

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

A. Report of the ACRS Subcommittee

C. P. Siess summarized the detailed review that took place at the subcommittee meeting on June 30, 1982 (see Appendix XXV). He directed the Committee's attention to a probabilistic risk assessment done by Sandia Laboratories based partly on the WASH-1400 three-loop Westinghouse Plant and the Crystal River-3 IREP study for a two loop CE plant with a containment similar to the two-loop Westinghouse Ginna Plant.

B. Introduction by Licensee

J. Larizza, Ginna Station Electrical Inspection Engineer, presented a milestone historical summary of the Ginna Station including the site and a description of the pressurized water reactor and systems (see Appendix XXVI).

C. Presentation by the NRC Staff

A. Wang, NRC Integrated Assessment Project Manager for Ginna, described the process of topic selection and resolution for Ginna (see Appendix XXVII). He indicated that reactor vessel integrity of the Ginna Plant design has been handled as a generic issue. C. P. Siess indicated that the subcommittee looked into the turbine missile issue. In a discussion of the seismology, the subcommittee found that in terms of the Regulatory Guide 1.60 spectrum, the Ginna plant has need for structural strengthening of the turbine building and auxiliary building.

A. Wang discussed a table of topics for which the Ginna plant met current criteria or equivalent, based on modifications implemented or committed to by the Licensee. D. A. Ward asked what the schedule was for the longest lead item on that list. A. Wang indicated that it is probably the pipe break outside containment topic (service water), which has three or four common mode failures and is to be integrated with the structural upgrade program. When a list of the 27 topics considered for backfit was shown, C. P. Siess inquired as to why the seismic design considerations were on the list as not requiring backfit. A. Wang indicated that that topic referred just to the turbine building and the Applicant had performed further analyses to show that the bracing was adequate.

A. Wang presented a list of topics with procedural backfits.

J. Ebersole, with regard to overpressurization protection of the shutdown cooling system, asked whether the administrative controls were sufficiently reliable to prevent repressurization of the RHR system.

He expressed his concern that the NRC has sought to rely on administrative procedures to protect the plant against an extremely important type of interaction. A. Wang acknowledged that that is to be a Technical Specification change and is already in the Ginna procedures. W. T. Russell explained that the low temperature overpressure protection system involves the dual setpoint relief concept where the Staff is relying upon the lowered relief setpoint on the PORV to provide additional relief protection for the RHR system. A. Wang individually explained the topics with hardware backfits

R. Mecredy indicated that installation of the final modifications for the structural upgrade will take at least several years and will probably not be completed in 1984. W. T. Russell explained that this was a rather major program, the first portion of which was involved in deciding the new design basis for these structures. He noted that this was an example of an integration of several topics into one program involving estimates of cost to the applicant for the upgrade of between $20 and $40 million.

D. Okrent inquired regarding the effectiveness of additional d.c. monitoring and a d.c. trouble alarm that the NRC is requiring Rochester Gas and Electic to install to monitor d.c. battery current. W. T. Russell indicated that this instrumentation to alert for battery failures is a recommendation for improving the reliability of d.c. systems. The proposed change for Ginna should make the d.c. system more effective.

The Committee discussed topics with analysis of potential hardware backfits. A. Wang indicated that the first topic where differences exist between Rochester Gas and Electric and the NRC Staff involves the flood level of Deer Creek based upon the probable maximum flood.

C. P. Siess explained that when the plant was originally licensed, considerable attention was given to possible flooding from Lake Ontario that lies immediately north of the plant. At that time no consideration was given to Deer Creek, a small steam that runs just south of the plant and could locally flood.

R. Mecredy indicated that both RG&E and the Staff have been studying the matter of local flooding and the primary problem for RG&E involves a potentially significant unknown cost in rebuilding the screenhouse to protect against the standard flood plus 1 ft. W. T. Russell indicated that the Staff position is actually in two parts, the first of which would have the Licensee provide protection for the standard project flood plus 1 ft., and the second position to demonstrate that providing protection for the probable maximum flood is too expensive or could not be done. The Committee discussed with the NRC Staff the details of potential flooding in the area of Ginna. After considerable Committee questioning, R. Mecredy informed C. P. Siess that RG&E would be prepared to discuss the subject of flooding in detail when they appear before the ACRS's Systematic Evaluation Programs Subcommittee regarding application for an FTOL.

A. Wang introduced the topic involving effects of high water level on structures, and effects of groundwater level. The Applicant explained its reasoning for concluding that the groundwater level is lower than grade, a contention the NRC disputes because of insufficient data presented by RG&E (see Appendix XXVIII). In answer to a question by C. P. Siess, T. Weis, RG&E, indicated that the soil in question, granular backfill soil around the walls of the auxiliary building, is saturated. C. P. Siess suggested that since the water is already there, it would take only natural conditions to bring the free water level up to grade, and since it has not happened in 11 years of operation, it is unlikely to happen in the next 27 years. He explained that the problem, therefore, involves the free water level which is now 20 ft. below grade gradually coming up to grade and exerting pressures on structures. W. T. Russell indicated that the Staff had calculated that Ginna has margins for the static effects of groundwater in the cases of control building walls and walls in the diesel building. He indicated that the item remained open because the Staff did not have sufficient engineering drawings or details to make determinations with respect to floor slabs and other walls in the structure.

M. Fliegel, NRC Staff, indicated that if RG&E had information that the groundwater level was below the plant grade for a five year period of extreme precipitation, the Staff might be able to lower the level that it requires be used in calculating postulated effects for loading conditions.

A. Wang explained RG&E's contention that they do not require contain- - isolation valves on some portions of the containment isolation system. He indicated that the Staff's basis for its position for closed systems is that General Design Criteria (GDC-57) requires one containment isolation valve for each penetration. Other reasons for the Staff's position include

. Service water lines are large (8 inch)

. System pressure and outlet lower than accident pressure

. Access to the manual valve now present may be limited due to radiation

. Time to close these manual valves may be significant

. The fan coolers have recently had a minor leak.

G. Wrobel, RG&E, presented a diagram of the penetrations involved in the service water system (see Appendix XXIX). He indicated that RG&E does not consider these as containment isolation barriers since, following a safety injection signal or when ECCS service water pumps are actuated supplying water to the fan coolers and compartment coolers, the pressure at the inlet of the containment is greater than the overall containment pressure. He contended that except for the first two or three hours of the design basis event, the pressure at the discharge of these valves would always be higher than the containment pressure. Therefore, leakage from containment into the service water would not be expected. G. Wrobel did point out that the valves are accessible just outside the containment. The radiation source field for a TMI-2 type source term was about 3 R per hour, and for a design basis loss of coolant accident, about an order of magnitude lower. W. T. Russell indicated that the Staff might reassess its position on the basis of the lower radiation doses revealed. G. Wrobel pointed out the significant cost of replacing those manual valves with remote manual valves as the Staff requires. W. T. Russell indicated that the 3 R per hour radiation level may not be sufficiently high to require RG&E to go to a remote manual valve. It may be sufficient to have a manual valve to isolate the containment for a leak check. W. T. Russell added that if RG&E was able to demonstrate that the reactor operator could isolate, that they had adequate procedures, and that the valves were not in the high radiation area, then the Staff would be inclined to agree with a manual isolation valve with operator action to close it locally rather than requiring the valve to be closed from the control room. Therefore, this issue may be resolved with this new information.

D. Comments by Rochester Gas and Electric on the SEP Program

R. Mecredy reviewed the SEP objectives identified by the NRC in 1977 (see Appendix XXX) and noted that the documentation base was not satisfactory regarding many of the topic assessments. He suggested that it is not yet possible to perform an integrated assessment within the SEP program. There did appear to be a basis for future integration given sufficient time. He pointed out that the lack of program definition, the lack of personnel within the NRC assigned to SEP, and the high turnover among NRC reviewers early in the program, plus the impact of TMI-2, resulted in several years of lost effort. He indicated that overall, despite the relatively slow start, the SEP has been relatively efficient both in manpower and money. He pointed to the elusive final objective of obtaining a full term license conversion since its application for a full term license in 1972. He contended that the final SEP must provide the basis for the license conversion. He indicated that the SEP has identified several areas where modifications to the Ginna plant procedures can be made to improve safety margins. Through the increase of documentation, RG&E has been able to confirm the adequacy of a number of design aspects of the plant. He noted that for a particular plant to accomplish the program in a reasonable time period, the review must be narrowly focused on a limited number of topics, with the number of topics clearly spelled out before beginning the program.

V. Task Action Plan A-45, "Evaluation of Alternative Decay Heat Removal Systems" (Open to Public)

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

A. Report of the ACRS Subcommittee

D. A. Ward indicated that the purpose of the meeting was to hear a report from the Staff on their revision of the officially approved Task Action Plan A-45 regarding the requirements to improve the reliability of decay heat removal systems. He noted that there was not a particular requirement for the Committee to write a letter endorsing the plan.

B. Report of the NRC Staff

A. R. Marchese, NRC Staff, presented a discussion outline indicating that the overall purpose of Task Action Plan A-45 is to evaluate the adequacy of current licensing design requirements to insure that nuclear power plants do not pose unacceptable risks due to failure to remove shutdown decay heat (see Appendix XXXI). He defined the objectives of the programs to develop a comprehensive and consistent set of decay heat removal (DHR) system requirements for existing and future LWRs and to evaluate alternate means of decay heat removal. He noted that the schedule for the program had been reduced from 48 months to 30 months by deleting most of the work on future plants, although acceptance criteria for decay heat removal systems for future plants will be developed. He indicated that quantitative criteria now will be based on frequency of core melt due to decay heat removal failures, rather than overall risk, as was originally planned. In answer to a question by D. Okrent, A. R. Marchese indicated that the Staff intends to avoid including containment heat removal systems unless there is an interaction. He indicated that the plant systems the Staff will be studying will include those which deal with frequent event transients and a small break LOCA spectrum. A. R. Marchese indicated that it was likely that the Staff would become involved with the feed and bleed concept and would consider interaction with the containment systems. In answer to a question by J. Ebersole, A. R. Marchese noted that the BWR studies would include consideration of the BWR's suppression pool cooling. J. Ebersole expressed concern that the Staff might not go far enough into extrapolating present designs and future ideas, or include new concepts. A. R. Marchese indicated that the Staff will study new concepts if they look attractive for existing plants. They will be assessed and ranked based on value impact evaluations.

D. Okrent suggested that the Staff consider earthquakes more severe than the Safe Shutdown Earthquake in the evaluation of the current capability of decay heat removal systems to deal with small break LOCAs. After some discussion on the matter, A. R. Marchese indicated that the Staff would reflect on the comments by D. Okrent regarding seismic capabilities of decay heat removal systems by checking on the present status of current industry and NRC programs that deal with the seismic issue.

A. R. Marchese indicated that as one Subtask in the program, the Staff will concentrate on the phenomenological aspects, including all of the thermal hydraulic tests and information available from the LOFT and Semiscale programs, in the area of modes of heat transfer that involve natural convection and reflux cooling. He indicated that the

Staff is to review to the extent that this information can be extrapolated to full size systems for a range of plant configurations. J. Ebersole suggested that the Staff should seriously consider feed and bleed at reduced pressures in the context of pressurized thermal shock problems. D. Okrent asked if the program had a task or subtask which involved a critical evaluation of the decay heat removal requirements for other countries that are using LWRs. A. R. Marchese indicated that the NRC Staff intends to elicit information from a number of countries, and become familiar with the basis behind design decisions.

In answer to a question by D. Okrent concerning the Staff's approach to dealing with the uncertainties of existing PRAs, D. Berry, Sandia, described his approach. He explained that the plan was to use completed studies to define criteria for systems in the power plant for different accident events, and use that existing work to define the components that are called upon to meet the accident situations. He indicated Sandia's intent to quantify these sequences for evaluation and determine weaker plants from the review of DHRs. D. Okrent suggested that this does not solve the question of dealing with the uncertainties. NRC must begin to address the uncertainties for decision making and go beyond sensitivity studies to alleviate uncertainties.

K. Kniel indicated that one way of alleviating the uncertainty was to formulate criteria for core meltdown or severe core damage and to say in those criteria that there are substantial margins to cover the uncertainties.

D. A. Ward questioned how the Staff planned to involve industry to TAP A-45 program. D. Berry indicated that the plan was to establish consultant agreements from support agreements with people and industry to establish a peer review relationship for reports at different milestones within the program. The Committee decided not to write a report regarding this matter at this time.

VI. Disposal of High-Level Radioactive Waste (Open to Public)

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

R. C. Axtmann indicated that the ACRS has written three letters on the subject of high-level waste disposal in geological repositories. He indicated that one of these letters, dated September 16, 1981, contained 13 specific ACRS suggestions or comments on the Staff's draft of 10 CFR 60, The Technical Criteria for Disposal of High-Level Waste. Among the Committee's suggestions were the following:

- Inclusion of the retrievability as a part of the rule rather than as background material

- Elimination of design and construction material from the rule

- Permitting the licensee to meet an overall safety goal without requiring separate subsystem goals (i.e., the package, the backfill, etc.).

- Beginning early work on the evaluation and comparison of computer models for the reservoir

- Relegating the regulation of transuranic waste to a separate document.

He indicated that the Staff has considered in the recent past the choice of a contractor to verify the longevity of the 1000 year waste containment package, domestic and foreign approaches to the overall problem, and the latest draft of 10 CFR 60. The Committee went into executive session for a first reading of a letter drafted by D. W. Moeller.

VII. Report of the Metal Component Subcommittee Regarding Steam Generator Tube Integrity (Open to Public)

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

Chairman Shewmon presented a summary of the June 7, 1982 meeting of the ACRS Subcommittee on Metal Components at the Conference Center at the Electric Power Research Institute (EPRI), Palo Alto, California (see Appendix XXXII). He indicated that the Steam Generator Owners Group (SGOG-1) has made technical progress regarding recognition that mistreatment of steam generators is a major cause of downtime, showing that recommended modifications of plant and operating procedures can stop denting and help in reducing the average plugging rate of steam generator tubes to a tolerable level. He mentioned information presented about TMI-1 steam generators involving considerable stress corrosion cracking in the top of their steam generators, probably caused by thiosulphate contamination of the primary coolant. The source of the thiosulphate was apparently the containment spray additive system. Plants that use this technique now use sodium hydroxide. He indicated that the Ginna event will engender an unresolved safety issue which will include steam generator corrosion, emergency core cooling system, pressurized thermal shock, and severe transients as components of the issue. C. Mark questioned how an applicant could avoid steam generator damage. Chairman Shewmon indicated keeping oxygen out of the primary coolant and the use of titanium alloy condenser tubes are two ways.

VIII. ACRS Subcommittee Report Regarding Proposed DOE Program for Siting and Assessment of High-Level Waste Repositories (Open to Public)

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

R. C. Axtmann explained that the draft DOE national plan for siting high-level radioactive waste repositories and environmental assessment describes a step wise plan to identify sites for the first repositories using three major candidate media: basalt, tuff, and salt. The report focused on salt and pointed out that neither the national plan nor 10 CFR 60 considers population density. He also indicated that the national plan does not mention population density near the site nor the candidate medium shale.

R. Axtmann mentioned a FEMA presentation regarding the distribution of potassium iodide tablets in the early stages of a power plant accident to mitigate thyroid exposures from iodine release. He indicated that a medical doctor explained that radioactive iodide could cause benign thyroid nodules as well as malignancies. It was noted that while potassium iodide tablets had not been stockpiled before the TMI accident, a pharmaceutical firm manufactured thousands of potassium iodide tablets during the accident.

In answer to a question, R. Axtmann reported that there was no discussion of the iodine source term from the repositories since the radioiodine would have decayed before burial. J. Ebersole asked what the reduction in uptake of radioiodine was when potassium iodide tablets were used. R. Axtmann said that he thought it was in the neighborhood of a factor of 20 to 50, once the thyroid gland was flooded with nonradioactive iodine.

IX. Report of Subcommittee Regarding Proposed Changes in NRC Rule 10 CFR 20 (Open to Public)

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

R. Axtmann briefly discussed the subcommittee meeting on Reactor Radiological Effects held on June 23, 1982. The status of NRC's proposed revision to 10 CFR 20 was reviewed. He indicated that the Staff was presently in the process of revising the rule but it was in a preliminary stage. Comments by K. Z. Morgan, ACRS Consultant, were distributed at the meeting (see Appendix XXXIII).

X. ACRS Report to the Commissioners on the Safety Research Program and Budget for FY 1984-85) (Open to Public)

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

C. P. Siess indicated that there are discussions under way regarding the formation of a consortium that would obtain considerably increased financial support from foreign countries for continuing the test program in LOFT.

C. P. Siess suggested that there is a need for an integral facility that would simulate the Babcock and Wilcox or Combustion Engineering pressurized water reactor designs. He indicated that the ACRS strongly supports use of the proposed Semiscale MOD V to reproduce the characteristics of the B&W reactor and suggested that the NRC seek significant financial support from industry for this effort.

D. W. Moeller expressed his concern for reduction in funding for research programs involved in the low-level radioactive waste and high-level radioactive waste area. He pointed to several programs which he felt should be retained and prioritized the low-level waste programs in descending order as follows:

1. Low-Level Waste Program on Engineered Disposal and Alternates to Shallow Land Burial

2. Low-Level Waste Program on Characterization of the Chemical Components of Low-Level Waste

3. Low-Level Waste Program on Nondestructive Test Methods for Waste Packages

4. Low-Level Waste Program on the Scope and Pace of Work on Source Term of Radioactive Isotopes in Shallow-Land Burial

It was also suggested that the high-level waste program involving fracturing and geomechanics of jointed rocks also be retained if possible.

The Committee recommended no change in the total budget for FY 1984. For FY 1985, the Committee proposed increases in funding for some Decision Units, corresponding generally but not in all cases to those recommended for FY 1984.

XI. Executive Sessions (Open to Public)

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

A. ACRS Reports, Letters, and Memoranda

1. ACRS Report on the Perry Nuclear Power Plant, Unit 1

The Committee prepared a report to the Commissioners of its review of the Perry Nuclear Power Plant, Unit 1 regarding the request for an operating license. The Committee concluded that, if due consideration is given to the recommendations in the body of the report, and subject to satisfactory completion of construction, staffing, and preoperational testing, operation at full power is acceptable.

2. ACRS Report on the Suitability of the Clinch River Breeder Reactor Plant Site

The Committee prepared a report to the Commissioners of its review of NUREG-0786, Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant (CRBRP) and considered the suitability of the proposed site for such a plant. The NRC Staff has concluded that the CRBRP can be designed and constructed in such a manner that it will present no greater risk to the health and safety of the public than an LWR plant meeting current safety criteria. The ACRS believes the proposed site is suitable for such a plant.

3. Comments on the NRC Safety Research Program and Budget for Fiscal Years 1984 and 1985

The Committee prepared a report to the Commissioners transmitting its comments on the Office of Nuclear Regulatory Research Budget proposed for FY 1984 and 1985. Only that portion of the budget relating to Program Support was considered. Sections of the report relate generally to programs for which the ACRS thinks greater effort or emphasis is needed, specific comments on the proposed programs in each Decision Unit, and specific recommendations regarding the Research Program Support Budget.

4. RES Sponsored Research on Torsional Ultrasonic Instrumentation

The Committee approved a memorandum from the ACRS Executive Director to the EDO providing ACRS comments on the need for NRC-sponsored research aimed at further development of a torsional ultrasonic system to measure liquid level in reactor vessels. The ACRS found no reason at this time for the NRC to sponsor such research.

B. Generic Safety Items

1. Additional Remarks Appended to ACRS Comments on Proposed Policy Statement on Safety Goals for Nuclear Power Plants, (NUREG-0880, "A Discussion Paper")

   The Committee heard additional remarks submitted by J. C. Mark and M. S. Plesset relative to the ACRS report to the Commissioners issued June 9, 1982, regarding the adoption of a policy statement on quantitative safety goals.

2. General Electric Instrumentation Logic

   J. Ebersole expressed his concern regarding control instrumentation system logic in General Electric nuclear power plants (commonly referred to as "one-out-of-two logic taken twice") which violates the single-failure criterion. He suggested that this issue be the subject of suitable research as part of the NRC Safety Research Program. The Committee decided that the matter was to be handled as a generic item and J. Ebersole was designated to draft a letter for consideration at the 268th ACRS Meeting in August.

3. Proposed Regulations on High-Level Waste Disposal

   The Committee discussed the status of the draft regulation, Disposal of High-Level Radioactive Wastes in Geologic Repositories, 10 CFR 60. Several Members expressed concern regarding the proposed change in the definition of the "accessible environment" as it relates to the potential impact of radioactive wastes in a repository. Therefore, the Committee's report to the Commissioners was deferred to the 268th ACRS Meeting so that the implications of this definition change could be more closely studied.

4. Discussion of Task Action Plan A-45

   The Committee discussed the status of approved Task Action Plan (TAP) A-45, Shutdown Decay Heat Removal Requirement, with members of the NRC Staff but decided not to write a report regarding this matter. ACRS Members did make individual recommendations/comments regarding implementation of TAP A-45:

   - Inclusion of containment heat removal systems in the criteria for decay heat removal systems

   - Use of plant-specific evaluations of alternate decay heat removal systems (DHRS) as part of the program analyses

- Consideration of lower probability earthquakes than the SSE in evaluation of the current capability of decay heat removal systems to deal with small break LOCAs

- Reconsideration of the adequacy of the treatment of shutdown decay heat removal systems in the current plan

- Consideration of the pressurized thermal shock issue in system constraints

C. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 268th ACRS Meeting, August 12-14, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

D. Systematic Evaluation Program Review of R. E. Ginna Nuclear Power Plant

The Committee reviewed the results of the Systematic Evaluation Program, Phase II, as it has been applied to the R. E. Ginna Nuclear Power Plant but was unable to complete its report to the Commissioners regarding this matter. Completion of the review was deferred to the 268th ACRS Meeting during August 1982.

E. American Nuclear Society Ad Hoc Committee

W. Kerr has been asked to serve as a member of an American Nuclear Society ad hoc committee to study and prepare comments on the Source Term question. The ACRS shall discuss approval of the request to serve on the Committee during the 268th ACRS Meeting in August.

F. FRG/RSK Meeting on October 5-6, 1982

The FRG/RSK confirmed their plans to meet with the ACRS on October 5-6, 1982, consistent with previous negotiation to discuss the following:

- Use of PRA and Quantitative Safety Goals in the regulation of nuclear power plants (Lead ACRS member - D. Okrent)

- Recent or proposed changes in safety related policy including consideration of Class 9 accidents (Lead ACRS member - W. Kerr.)

. Recent or proposed changes in safety related technology such as use of the DEPB as the basis for design of limited plant features, prevention/mitigation of reactor pressure vessel thermal shock, etc. (M. Bender and P. Shewmon share the ACRS Lead regarding this area.)

The Committee endorsed an extra agenda item requested by the FRG/RSK concerning the status of activities regarding radwaste management and disposal for which D. W. Moeller will be the Lead member.

G. <u>Contentions on Clinch River Breeder Reactor from the Natural Resources Defense Council, Inc.</u>

The Committee endorsed a letter response to T. B. Cochran who verbally discussed the Natural Resources Defense Council, Inc. intervention and related concerns on the Clinch River Breeder Reactor project.

H. <u>New ACRS Members</u>

The Committee discussed candidates for appointment to the ACRS membership which will be made vacant when W. M. Mathis completes his present term. The ACRS selected three individuals to recommend to the Commissioners.

The 267th ACRS Meeting was adjourned at 2:45 p.m., Saturday, July 10, 1982.

ACRS-2011

APPENDIXES
TO
MINUTES OF THE 267TH ACRS MEETING
JULY 8-10, 1982

ATTENDEES
267TH ACRS MEETING
JULY 8-10, 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
Carson Mark
Dade W. Moeller
David Okrent
Milton S. Plesset
Chester P. Siess
David A. Ward

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
M. Norman Schwartz, Technical Secretary
Herman Alderman
William M. Baldewicz
Stuart K. Beal
Alden Bice
William M. Bock
Paul A. Boehnert
Don Bucci
Anthony J. Cappucci
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
R. C. Tang

NRC ATTENDEES

267TH ACRS MEETING

JULY 8, 1982

NUCLEAR REACTOR REGULATION

J. P. Knight, DE
P. Shemanski, DE
J. Stefano, DL
J. Stancy, DE
A. Singh, SKI
T. Collins, DSI
G. Thomas, DSI
R. L. Tedesco, DL
C. Grimes, DL
E. Goodwin
G. Lainas, DL
M. Fliegel
D. Persinko, DL
D. Terao, DE
D. Jeng
D. Terao
L. Yang
G. Bagchi, EQB
J. Manck, DSI
O. D. Parrk DSI
N. E. Fioravante, DSI
D. I. Scrig, DHFS
W. V. Johnston, DE
J. W. Clifford, DHFS
S. Salah, DHFS
A. Schwencer
A. Wang
C. Williams
W. T. Russell

INSPECTION AND ENFORCEMENT

R. DeFayette
J. Mathis

HFEB
D. Tondi

Region III
M. L. Gildner, Perry Res. Inspector
C. C. Watter
L. G. McGregor
W. D. Shafer

NRC ATTENDEES

267TH ACRS MEETING

July 9, 1982

NUCLEAR REACTOR REGULATION

W. Pasuah, RAB
R. Stark
J. Swift
T. L. King
R. A. Becker
L. W. Bell
R. B. McMullen
C. Quay
R. B. Cedell
R. L. Roth

R. Roberts, NRC
E. Goodwin
C. Thomas
M. E. Wangler
M. Thadani
P. S. Check
B. Morris
J. P. Knight
D. B. West
A. Spano
H. Holz
M. Fliegel
C. P. Tan
A. R. Marchese

NUCLEAR REGULATORY RESEARCH

C. Kato
P. M. Wood
L. Russell

CNSNS-GSB/MEX

V. Robio

NUCLEAR MATERIAL SAFETY & SAFEGUARDS

M. Bell

ATTENDEES - APPLICANT

267TH ACRS MEETING

July 8, 1982

CLEVELAND ELECTRIC ILLUMINATING CO.

M. J. Titas
B. D. Walratm
P. G. Klann
M. Edelman
R. Farrell
L. Beck
R. J. Tadyett
S. Kensicki
P. A. Nevins
K. A. Matheny
A. G. Migas
D. Davidson
R. A. Stratman
C. M. Shuster
W. E. Coleman
E. M. Buzzellt
R. Pender
J. J. Waldron
A. F. Silakoski
T. E. Mahon

GILBERT ASSOCIATES, INC.

C. A. Vath
L. E.Wise
J. S. Holton
R. W. Alley
M. G. Capiotis
W. E. Meek
P. B. Gudikurst
M. Plica
M. Waselus
S. M. Gresdo
J. F. Hilbish
J. D. Grier

GENERAL ELECTRIC

R. C. Mitchell
D. T. Shew
W. M. Davis

A-V

ATTENDEES - APPLICANT

267TH ACRS MEETING

July 9, 1982

WESTINGHOUSE POWER CORP.

Goeser
Clare
Dickson
P. J. Docherty
E. T. Murphy
P. J Docherty

DEPARTMENT OF ENERGY

N. Kaushal, CRBRP-PO
P. Gross, CRBRP-PO
J. Dornhoff
T. Schleiter
H. W. Hibbitts

TENNESSEE VALLEY AUTHORITY

J. A. Domer
H. Piper
J. Hunt
R. Lee
T. J. Abraham

PUBLIC ATTENDEES

267TH ACRS MTG.

July 8, 1982

T. E. Tipton, Atomic Industrial Forum
L. D. Schultz, Weston Geophysical
G. Leblanc, Weston Geophysical Corp.
J. E. Booker, GSU
Robin Wilson, Ohio/Wash. News Service
J. A. Kirkebo, Stone & Webster Engr.
W. J. Reed, Gulf States Utilities Co.
L. A. England, Gulf States Utilities
P. W. Merloe, NRC Calendar
D. L. Holtzscher, Illinois Power Company
M. A. Lechowilt, Bechtel
R. S. Boyd, KMC, Inc.
F. C. Fogarty, MSE
D. Silverman, LNR&A
R. Shaffer, AP
G. Wrobel, Rochester Gas & Electric
A. Larizza, Rochester Gas & Electric
R. E. Smith, Rochester Gas & Electric
R. McCredy, Rochester Gas & Electric
T. Weis, Rochester Gas & Electric
A. Toblin, NUS Corporation
R. Spulak, Sandia National Laboratories
L. S. Lang, Rochester Gas & Electric
Leyse, NSAC
C. Taylor, WPPSS
K. Connor, Doc-Search Associates
J. E. McEwen, TSI
J. Ferraro, Rochester Gas & Electric
J. Berga, Electric Poer Research Inst.
J. F. O'Brien, NUS
E. Murphy, Westinghouse
H. H. Vargt, LeBoeuf, Lamb

PUBLIC ATTENDEES

267TH ACRS MTG.

<u>July 9, 1982</u>

Leyse, NSAC
Johnson, Science Applications
N. Strand, Self
L. Connor, The NRC Calendar
B. White, Law Engineering
S. M. Kowkabany, Burns & Roe
A. D. Burkhart, Burns & Roe
R. E. Palm, Burns & Roe
Laraam, Oak Ridger
G. Elyn, PML/MLB
M. Vandenboch, Friends of the Earth
M. Barrett, AP
R. Liner, SAI
E. Rumble, SAI
M. White, Doub & Muntzing
E. Shilter, Bechtel Power Corp.
M. Yawery, AP
Leyse, NSAC
J. Potter, Nuke Waste News
J. Berga, EPRI
T. Huston, CP Co.
D. L. Berry, Sandia
L. Lave, UCLA
J. Nelson, Quadrex

APPENDIX II
FUTURE AGENDA

AUGUST

Grand Gulf Station Unit 1--outstanding OL Items	
Watts Bar Plant Units 1 and 2--OL	4 hrs
Discuss Implementation Plan Regarding Proposed NRC Quantitative Safety Goals	2 1/2 hrs
ACRS Comments on Control Room Habitability	
Meeting with Adm. Kinnard R. McKee regarding Naval Reactors Program Policies and Practices.	
High Level Waste Disposal--complete discussion of 10 CFR 60	

Subcommittee Reports

Subcommittee on Transportation of Radioactive Material regarding the proposed revision of 10 CFR Part 71 (CPS/SD) (see Attachment 1).	Deferred to Sept.
Subcommittee on Fluid Dynamics regarding potential safety issues raised by J. Humphrey regarding Grand Gulf/BWR Mark III containment design (MSP/PAB)	Tentative
Subcommittee on Reactor Radiological Effects regarding occupational radiation exposure at nuclear power plants (DWM/RCT)	
Subcommittee on Qualification Program for Safety Related Equipment concerning proposed NRC rule regarding Accreditation of Qualification Testing Organizations (JJR/AJC)	Deferred

Subcommittee on Reactor Radiological Effects on NRC Policy regarding consideration of seismic events in emergency planning (DWM/RCT) — Tentative

Subcommittee on Safety Research Program regarding the Long-Range Research Plan (CPS/SD)

NRC Staff Reports Regarding the Status of:

- Proposed NRC Generic Review Plan for Systems Interactions Studies per Attachment 2, Memorandum from E. F. Goodwin to R. F. Fraley dated April 16, 1982 — Tentative
- Proposed NRC Plan for resolution of ATWS rule per Attachment 3, memorandum from P. A. Boehnert to W. Kerr dated June 29, 1982 — Tentative

7/10/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JULY

20	Reactor Radiological Effects (Tang/McKinley) - Moeller, Ebersole, Ray, Okrent (tent.). Purpose: (1) To review PWR Occupational Radiation Exposure histories and recent experiences in exposure reduction at several plants, (2) To discuss the status of NUREG-0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees."
29 & 30	Fluid Dynamics (San Jose, CA) (Boehnert) - Plesset, Ebersole, Ray, Ward. Purpose: To discuss potential safety concerns raised by Mr. J. Humphrey on the BWR Mark III containment design.

AUGUST

3	Reliability & Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Kerr, Siess, Mark, Lewis, Bender, Ebersole. To review the Staff action plan for implementation of a safety goal.
10	Watts Bar (Beal/Quittschreiber) - Ebersole, Bender, Ward. Purpose: To complete the review of the application for an operating license.
10 CANCELLED	Regulatory Activities (Duraiswamy) - Siess, Carbon, Ray, Kerr. Purpose: To review proposed Regulatory Guides and Regulations.
11 (a.m.)	Safety Research Program (Duraiswamy) - Siess, Okrent, Plesset, Ward, Shewmon, Bender, Kerr, Moeller, Mark, Carbon (tent.). Purpose: To provide early input to the RES Staff for their preparation of the Long-Range Research Plan for FY 85-89.
11 (1:00 p.m.)	Grand Gulf (Alderman) - Okrent, Siess, Ebersole, Mark, Bender, Plesset (tent.). Purpose: To continue the review of Grand Gulf for an operating license.
12-14	268th ACRS Meeting
18 & 19	CRBR Working Group on Structures and Materials (Cappucci/Quittschreiber) - Shewmon, Axtmann, Etherington, Siess (tent.) Carbon (tent.). Purpose: To discuss "leak before break" criteria for CRBR, overall leakages, and leak detection; inservice inspection plan and the structural integrity of critical transition joints; and elevated temperature design, including supports.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

AUGUST

24 — Transportation of Radioactive Materials (Duraiswamy) - Siess, Bender, Mark, Moeller. Purpose: To continue the review of the adequacy of the NRC package certification procedures, and to review 10 CFR Part 71.

25 — Reactor Operations (Major) - Ebersole, Bender, Kerr, Moeller, Okrent, Ray, Ward. Purpose: (1) To discuss NRC's enforcement policy governing enforcement actions for violations of NRC regulations and license applications; (2) A discussion of regionalization effort within I&E; and (3) To discuss the current status of I&E's Performance Appraisal Team inspection program and Systematic Assessment of Licensee Performance program.

31 & SEPT. 1 — WPPSS 2 (Hanford, WA) (Griesmeyer/Quittschreiber) - Plesset, Ebersole, Mark, Mathis, Ward (tent.). Purpose: To review application for an operating license.

SEPTEMBER

8 — Regulatory Activities (Duraiswamy) - Siess, Carbon, Ward*, Bender*. Purpose: To review proposed Regulatory Guides and Regulations.

8 (tent.) — Metal Components Working Group (Igne) - Bender*, Shewmon, Axtmann, Ward*, Etherington, Okrent*, Plesset, Lewis. Purpose: To continue the review regarding pressurized thermal shock.

8 — AC/DC Power Systems Reliability (Savio) - Ray, Ebersole, Kerr, Okrent*. Purpose: (1) To review the status of NRR's actions on the implementation of the recommendations in NUREG-0666; (2) to review the status of RES's ongoing work on station blackout; and (3) to conduct a discussion on matters generally relating to the reliability of AC/DC power systems components as time permits.

9-11 — 269th ACRS Meeting

* Conflict to be resolved.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

OCTOBER

5 & 6	Meeting with RSK (Fraley) - Shewmon, Bender, Okrent, Kerr, Etherington, Lewis. Purpose: To discuss: (1) Use of PRA and quantitaive safety goals in the design and regulation of nuclear power plants; (2) Recent or proposed changes in safety-related policy, including items such as consideration of Class 9 accidents in the design of nuclear plants, reemphasis on standardization in the design and licensing of nuclear plants, etc.; and (3) Recent or proposed changes in safety-related technology, including items such as the use of the DEPB as the basis for plant design, pressurized thermal shock of reactor pressure vessels, et.
7-9	270th ACRS Meeting

DATES TO BE DETERMINED

Date to Be Determined (Late Sept.)	Reliability and Probabilistic Assessment (Griesmeyer/Quittschreiber) - Okrent, Kerr, Siess, Mark, Lewis, Bender, Ebersole. Purpose: To discuss the ACRS review of the Limerick Probabilistic Risk Assessment.
Date to Be Determined	ECCS (location to be determined) (Boennert) - Plesset, Ebersole, Ward, Okrent. Purpose: To discuss B&W Small Break LOCA model.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 20, 1982	REACTOR RADIOLOGICAL EFFECTS	(TANG/MCKINLEY) Moeller, Ebersole, Ray, Okrent (tent.)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Moeller

Purpose: (1) To review PWR Occupational Radiation Exposure histories and recent experiences in exposure reduction at several plants.
(2) To discuss the status of NUREG-0761.

Representatives from Westinghouse, Combustion Engineering, Babcock & Wilcox, VEPCO, Florida Power & Light, Toledo Edison, INPO, and DOE will make presentations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees," dated March 1981.

2. Project Status Report and Tentative Schedule dated July 8, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 29 & 30, 1982	FLUID DYNAMICS	(BOEHNERT) Plesset, Ebersole, Ray, Ward
		Cons: Bush, Garlid, Schrock, Theofanous, Zudans

LOCATION: San Jose, CA

BACKGROUND:

Who proposed action: M. Plesset

Purpose: To discuss potential safety concerns raised by Mr. J. Humphrey on the BWR Mark III containment design.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Letter, J. Humphrey to P. Boehnert, dated 6/21/82 offering comments and suggestions regarding participation in July 29&30, 1982 meeting.
2. Letter, J. Humphrey to A. Schwencer (NRC Licensing) dated 6/17/82 providing general and specific comments as follow-up on a 5/27/82 NRC/GE/MP&L meeting.
3. Letter, L. Dale, MP&L to H. Denton, NRC, dated 5/28/82 providing detailed response to J. Humphrey concerns raised at 5/27/82 meeting noted above.
4. Transcript of 5/27/82 meeting.
5. NRC List of Humphrey Original Concerns (not dated)
6. Memos, H. Alderman to D. Okrent dated 5/21 & 5/18/82 informing ACRS of J. Humphrey concerns.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 3, 1982	RELIABILITY & PROBABILISTIC ASSESSMENT	(GRIESMEYER/QUITTSCHREIBER) Okrent, Kerr, Siess, Mark, Lewis, Bender, Ebersole

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To review the Staff action plan for implemenation of a safety goal.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. NUREG-0880, "Discussion paper on Safety Goals."
2. Staff action plan to implement a safety goal (not yet received).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
~~AUGUST 10, 1982~~ CANCELLED	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Carbon, Ray, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review proposed Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 10, 1982	WATTS BAR	(BEAL/QUITTSCHREIBER) Ebersole, Bender, Ward Cons: Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To complete the review of the application for an OL.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER was released on July 8, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 11, 1982 (a.m.)	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Okrent, Plesset, Ward, Shewmon, Bender, Kerr, Moeller, Mark, Carbon (tent.)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action. RES Staff

Purpose: To provide early input to the RES Staff for the preparation of the Long-Range Research Plan for FY 1985-1989.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 11, 1982 (1:00 p.m.)	GRAND GULF	(ALDERMAN) Okrent, Siess, Ebersole, Bender Plesset (tent.), Mark Cons: G. Schott

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To continue the review of Grand Gulf for an operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SSER 2 received 6/21/82.
2. SSER 3 expected to be received 7/15/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 18 & 19	CRBR WORKING GROUP ON STRUCTURES AND MATERIALS	(CAPPUCCI/QUITTSCHREIBER) Shewmon, Axtmann, Etherington, Siess (tent.), Carbon (tent.)
		Cons. Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: P. Shewmon

Purpose: To discuss "leak before break" criteria for CRBR, overall leakages, and leak detection. Inservice inspection plan and the structural integrity of critical transition joints will also be discussed. Elevated temperature design, including supports will also be discussed.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

W-ARD-0185
Memo, Cappucci to Shewmon, "Proposed Review Plan for CRBR Working Group on Structures and Materials," dated 4/22/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 24, 1982	TRANSPORTATION OF RADIOACTIVE MATERIALS	(DURAISWAMY) Siess, Bender, Mark, Moeller Cons: Langhaar, Shappert, Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: To continue review of the adequacy of the NRC package certification procedures, and to review 10 CFR Part 71.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 25, 1982	REACTOR OPERATIONS	(MAJOR) Ebersole, Bender, Kerr, Moeller, Okrent, Ray, Ward Cons.: Mathis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: J. Ebersole

Purpose: (a) To discuss NRC's enforcement policy governing enforcement actions for violations of NRC regulations and license applications. Including a discussion of the types of enforcement actions available to the NRC and the circumstances under which they will be used.

(b) A discussion of the regionalization effort within NRC's Office of Inspection and Enforcement, which is beginning to place the responsibility for technical reviews with the regional field offices. Current progress, future aims, relationship between regional offices, headquarters could be topics for discussions.

(c) Current status of IE's Performance Appraisal Team (PAT) inspection program and the Systematic Assessment of Licensee Performance (SALP) program. How IE perceives these programs' interface with the INPO evaluation programs would also be of interest.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Nuclear Regulatory Commission, 10 CFR 2, "General Statement of Policy and Procedure for Enforcement Actions." Revised general statement of policy. Effective Date: March 9, 1982

2. Memorandum for: W. Kerr From: M. Libarkin, Subject: NRC Enforcement Policy, dated May 12, 1982.

3. Nuclear Regulatory Commission, SECY-82-150, Subject: "The Performance Appraisal Team (PAT) Inspection Program," dated April 8, 1982.

4. Memorandum for: Mr. Ward and Mr. Bender, From: Dr. Kenneth D. Kirby, Subject: The IE Performance Appraisal Team Inspection Program, dated May 7, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
AUGUST 31 & SEPT. 1, 1982	WPPSS-2	(GRIESMEYER/QUITTSCHREIBER) Plesset, Ebersole, Mark, Ward (tent.) Cons: Mathis

LOCATION: Hanford, WA

BACKGROUND:

Who proposed action: NRR

Purpose: To review application for operating license.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SER, April 12, 1982 without seismic evaluation.
SSER with seismic evaluation due June 4, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPT. 8, 1982	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Carbon, Ward, Bender

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review proposed Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPT. 8, 1982 (Tentative)	Metal Components	(IGNE) Bender, Shewmon, Ward, Axtmann, Etherington, Okrent, Plesset, Lewis Consultants: Kouts, Theofanous, Catton, Zudans, Irwin, Abbott, Binford, Gall, Weschler Fellow: Bock

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: S. Hanauer/Bender, Shewmon

Purpose: To continue the review regarding pressurized thermal shock.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Interim Staff position on PTS and basis. Available some time in August.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
SEPTEMBER 8, 1982	AC/DC POWER SYSTEMS RELIABILITY	(SAVIO) Ray, Ebersole, Kerr, Okrent

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: J. Ray

Purpose: (1) To review the status of NRR's actions on the implementation of the recommendations in NUREG-0666.
(2) To review the status of RES's ongoing work on station blackout.
(3) To conduct a discussion on matters generally relating to the reliability of AC/DC power systems components as time permits.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
OCTOBER 5 & 6, 1982	MEETING WITH RSK	(FRALEY) Shewmon, Bender, Okrent, Kerr, Etherington, Lewis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action:

Purpose: (1) To discuss use of PRA and quantitative safety goals in the design and regulation of nuclear power plants.

(2) To discuss recent or proposed changes in safety-related policy, including items such as consideration of Class 9 accidents in the design of nuclear power plants, reemphasis on standardization in the design and licensing of nuclear plants, etc.

(3) To discuss recent or proposed changes in safety-related technology, including items such as the use of the DEPB as the basis for plant design, pressurized thermal shock of reactor pressure vessels, etc.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
TO BE DETERMINED	RELIABILITY AND PROBABILISTIC ASSESSMENT	(GRIESMEYER/QUITTSCHREIBER) Okrent, Kerr, Siess, Mark, Lewis, Bender, Ebersole

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To discuss the ACRS review of the Limerick Probabilistic Risk Assessment.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided later.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
TO BE DETERMINED	EMERGENCY CORE COOLING SYSTEMS	(BOEHNERT) Plesset, Ebersole, Ward, Okrent

LOCATION: To Be Determined

BACKGROUND:

Who proposed action: NRR/Plesset

Purpose: To discuss B&W Small Break LOCA model.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in the near future.

APPENDIX IV
C. MARK SUPPLEMENTARY REMARKS ON
QUANTITITATIVE SAFETY GOALS

J. CARSON MARK
4900 SANDIA DRIVE
LOS ALAMOS, NEW MEXICO 87544

(Not sent)
~~June 14, 1982~~

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

Dear Dr. Palladino:

In our conversation on June 4, you expressed interest in my concern over one aspect of the Commission's statement on Quantitative Safety Goals. My comment represents a purely personal view, and may very possibly be mistaken. It is not submitted on behalf of the ACRS, which may well disagree with the view expressed. Nevertheless, I should like to think that it had been noted in the course of formulation of the Commission's Safety Goal policy.

1. In my opinion, the primary (quantitative) definition of the Safety Goal objectives should not be stated in terms of health effects. It is true that these are the ultimate measure of the safety of potentially exposed persons; but there is no way -- on the basis of the terms presently proposed to define the goals -- of ascertaining if the goals have been met. This is evidently the case in the instance of an 0.1% increment in delayed cancer; and most probably also with respect to an 0.1% increase in prompt fatalities. Indeed, perhaps the only circumstance in which a clear comparison could be made between performance and the goals would be one in which something happened resulting in prompt fatalities which showed that the goals had <u>not</u> been met.

2. Since (in my opinion) conformance with the goals is not demonstrable in terms of the units used to define them, it will be necessary to have recourse to some particular dose-response relationship. At present, however, there is no fully accepted relationship of this sort. In BEIR III, for example, the majority of the Committee on Somatic Effects offers three different models, with a choice in each case between an "absolute risk" and a "relative risk" projection of consequences (which themselves differ by a factor between 3 and 4). The chairman of that Commmittee argues strongly that the majority has underestimated the true effects; and another distinguished member of the Committee feels strongly that the real effects are much smaller than suggested in the Report. There is a factor of at least 20 between the extremes of the values presented. There is the additional complication that some of the estimates relate to cancer incidence, and some to cancer mortality. To cope with this, I would suppose that the NRC would find it necessary to use some demonstrably high envelope merely to avoid, or at least to soften, possible litigative arguments; and thus embed in the documentation yet another item of unreal numerology which may be difficult or impossible to dislodge.

The above comment relates particularly to the delayed cancer effects. With respect to the prompt fatalities, things are not much better. BEIR III does not discuss this at all. WASH-1400 offers three curves: one without much medical treatment; one with some (supportive) treatment; and one with "heroic" treatment. A primary basis for the first curve is the data from the Japanese atomic bomb experience; and that is in the course of being re-assessed.

In this general connection it might be of interest to note that at its annual meeting in Washington in April 1982, the National Council for Radiation Protection (NCRP) gave favorable consideration to the notion of developing a risk system for setting limits for exposure to radiation. However, the NCRP concluded that the full development of such an approach is not possible at this time, because the necessary dose-risk data are not available.

3. If health effects are not used as the defining quantities, it would be necessary to establish the objectives in terms either of radiation exposure (both in the near field, and in an extended area) or in terms of the radioactive release to the atmosphere (possibly broken down into the two components of fission products and radioactive heavy elements). These do have the advantage of being measurable; but they have the disadvantage of seeming to be mysterious, or, at least, only indirectly related to the "real" effects. That they are indirectly related has been suggested above. In fact, the exposure estimates depend on the uncertain linkage of attempts to calculate the dispersion of the release under one or another assumption as to meterological conditions, terrain features, and so forth, as depicted by the highly stylized assumptions necessarily embedded in (for example) the CRAC code. In my view, this argues for the use of release criteria as the basis for the goal.

4. Admittedly, if one were to stipulate release criteria, these do not translate readily into "real" effects. It would be necessary, and also proper, to use some version of the CRAC (or an improved) code to interpret these into possible exposure levels; and to use some dose-effect relationship to assess the significance in terms of health effects. However, all this could be done in the presentation of "background information"; and the particular relationships used -- though admitted, and presumably stated to be open to some question and debate -- would not have to be defended in detail, and would not offer clear targets for litigation.

In this way, while using releases for criteria even though health effects were the matter of real concern, one would be following the precedent set by FDA and OSHA, for whom health effects are also the primary matter of concern but who state their criteria in terms of something measurable, such as parts per million or micrograms per cubic meter, and so forth.

5. At least part of the point to promulgating quantitative safety goals is said to be that of putting design requirements on a more rational and uniform basis. For this purpose, the more directly the criteria can be related to actual plant characteristics the better. It is true that release criteria are not very easily relatable to plant design features, but they are much more direct than radiation exposure levels away from the plant, or than health effects. The PRA buffs, who can already be seen whetting their teeth over the open field offered by the present formulation of the goals, might be disappointed by bringing the goals closer in to the plant; but they shouldn't be too upset -- there will still be more than enough for them to do.

Yours sincerely,

Carson Mark

Carson Mark

Cy: Commissioners: Ahearne
Asselstine
Gilinsky
Roberts

Executive Director ACRS: R. F. Fraley

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

July 7, 1982

MEMORANDUM FOR: J. J. Ray, Chairman
Subcommittee for the Perry Nuclear Power Plant

FROM: A. Cappucci, Staff Engineer

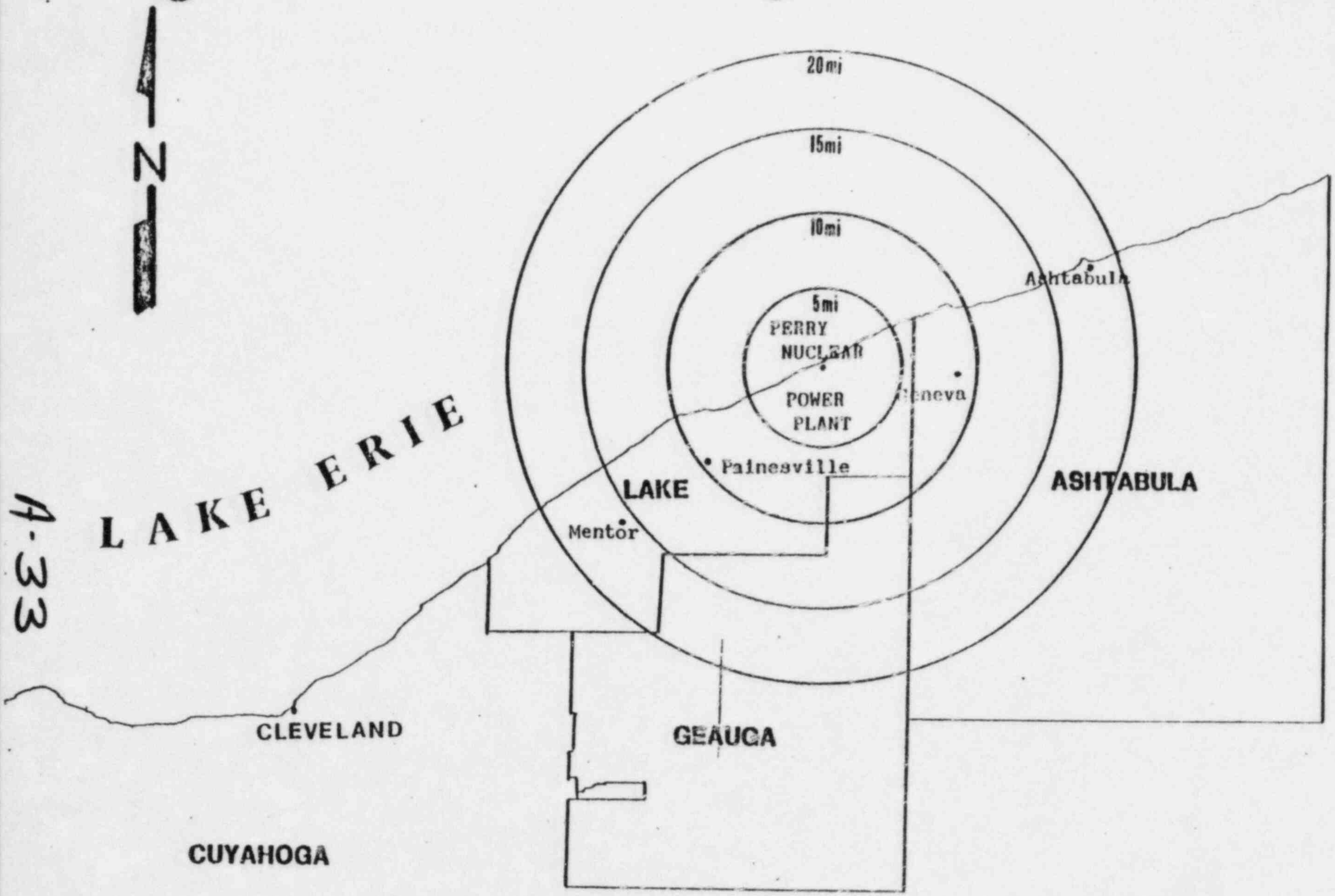
SUBJECT: PERRY CONSTRUCTION STATUS

As of May 31, 1982, the percent completion for construction is as follows:

Unit 1 - 83.4%
Unit 2 - 42.9%

Please change these numbers in the Project Status Report under Tab 2 of the 267th ACRS Meeting Books.

cc: ACRS Members
R. Fraley
M. Libarkin
G. Quittschreiber



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Distance between sectors = 5 miles radius

AREA WITHIN 20 MILES OF
PERRY NUCLEAR POWER PLANT

CAPCO NUCLEAR GENERATING UNITS

Project	CEI Share	Year of Operation	Construction and Operation Responsibility
Davis-Besse	51.38%	In Service	Toledo Edison Company
Beaver Valley			
Unit 1	-0-	In Service	Duquesne Light Company
Unit 2	24.47%	1986	
Perry			
Unit 1	31.11%	1984	Cleveland Electric
Unit 2	31.11%	1988	Illuminating Company

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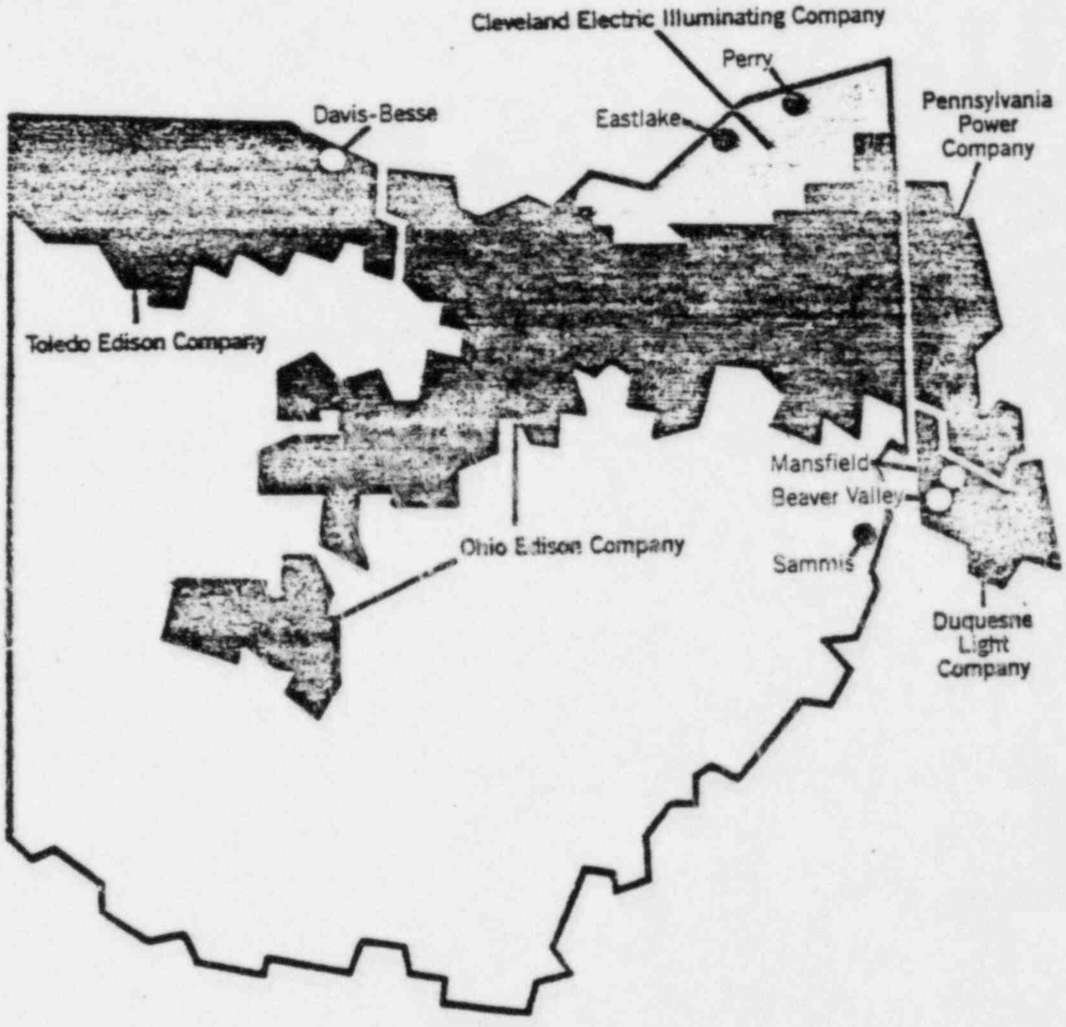

CAPCO Companies
Service Areas &
Jointly Owned Plants

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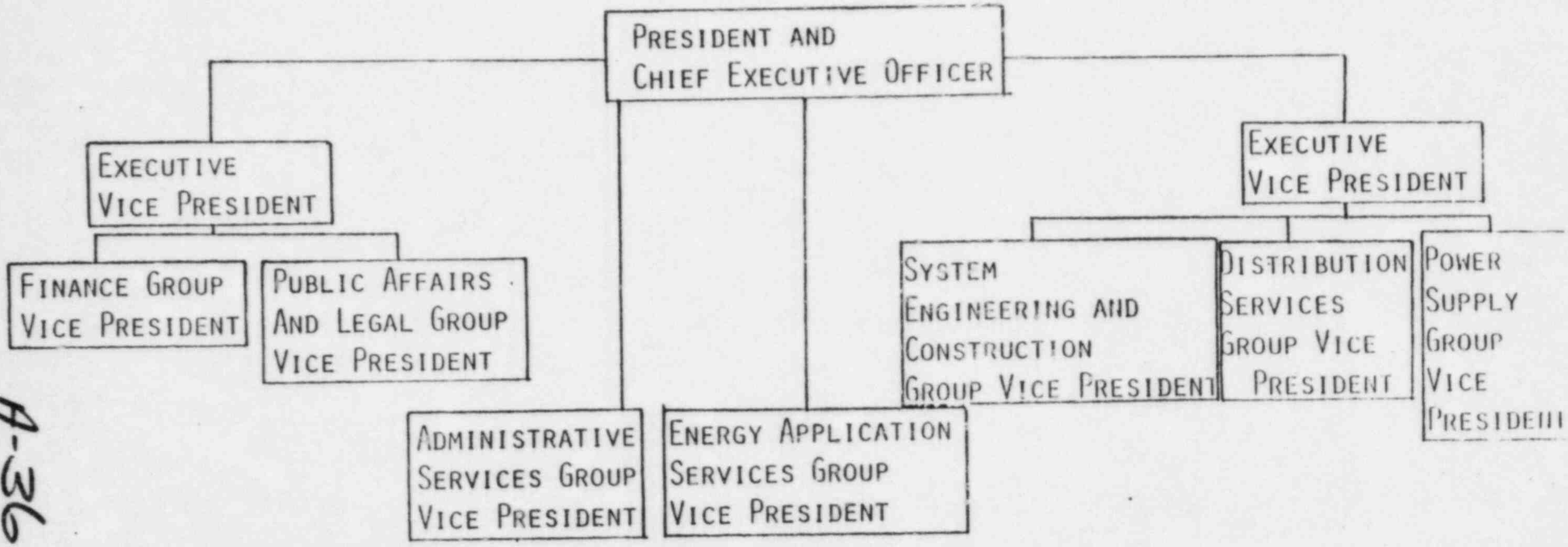
CLEVELAND ELECTRIC ILLUMINATING COMPANY
CORPORATE ORGANIZATION



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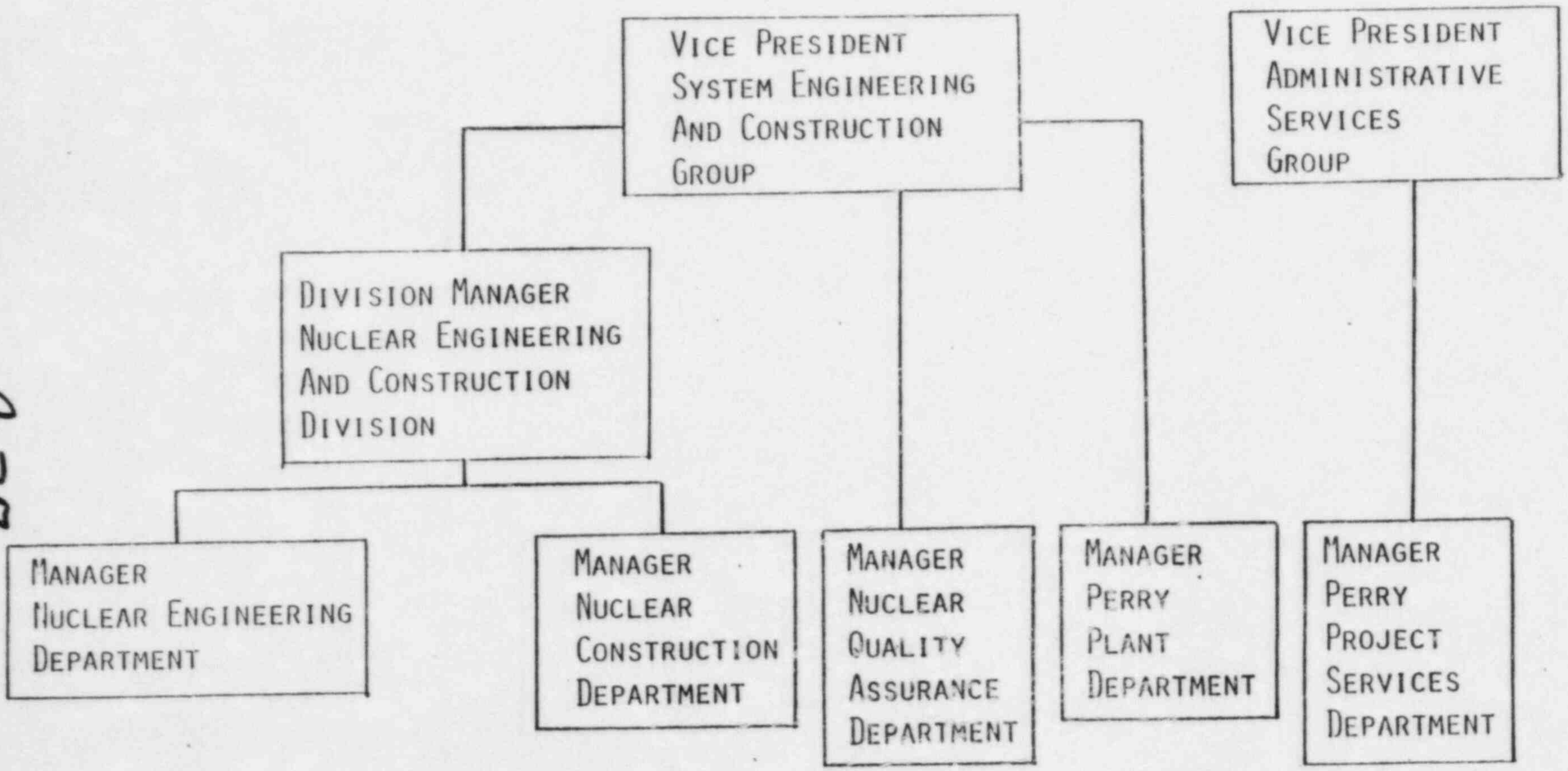
ORGANIZATIONAL CHANGES SINCE
JANUARY 1982 NRC MANAGEMENT AUDIT AT PNPP



MURRAY R. EDELMAN
Divison Manager
Nuclear Engineering and
Construction Division

FORMAL EDUCATION AND TRAINING:

B.S. Mechanical Engineering, Case Institute of Technology, 1961

Juris Doctor, Baldwin-Wallace Cleveland Marshall Law School, 1965

EXPERIENCE:

1961 - Present: The Cleveland Electric Illuminating Company

Held various engineering positions in CEI including General Supervising Engineer of the Civil and Mechanical Engineering Department and Manager of the Nuclear Quality Assurance Department before being named as Division Manager, Nuclear Engineering and Construction Divison.

MICHAEL J. TITAS

MANAGER

PERRY PROJECT SERVICES DEPARTMENT

FORMAL EDUCATION AND TRAINING:

B. S. in Electrical Engineering, Case Institute of Technology, 1964

M. S. in Electrical Engineering, Cleveland State University, 1969

EXPERIENCE:

Held various engineering positions in CEI including positions in Systems Operations and General Supervisor positions in Trouble Dispatching, and Electrical Maintenance and Construction before being named Manager of the Perry Project Services Department.

JOHN J. WALDRON
Manager
Perry Plant Department

FORMAL EDUCATION AND TRAINING:

Bachelor of Mechanical Engineering Degree, Marquette University, 1951

Eight-Day PWR Design Orientation Course (B&W), 1969

Three-Week BWR Design Orientation Course (GE), 1972

Three-Week Nuclear Technology Course for Power Plant Engineers (General Physics Corporation), 1976

Twenty-Week Academic Program for Nuclear Power Plant Personnel (General Physics Corporation), 1979

Five-Week Perry Nuclear Plant Technology (GE), 1980

Nine-Week Operator Training Course, Perry Simulator (GE), 1980

Certified SRO on Perry Simulator

EXPERIENCE:

1954 - Present: The Cleveland Electric Illuminating Company

Held various engineering positions with CEI including Results Engineer, Plant Technical Engineer, and Operations General Supervisor at CEI's Avon Lake Power Plant (fossil-fired plant) from 1958 to 1972. In 1972, transferred to Perry Project Team and in 1974, named to current position as Manager of the Perry Plant Department.

LAWRENCE O. BECK
General Supervising Engineer
Nuclear Licensing and
Fuel Management Section

FORMAL EDUCATION AND TRAINING:

B.S. Electrical Engineering, Purdue University, 1958

Master of Business Administration, Case Western Reserve University, 1967

EXPERIENCE:

1956 - Present: The Cleveland Electric Illuminating Company.

Held various engineering positions with CEI including Senior Engineer in the Civil and Mechanical Engineering Department and Senior Project Engineer responsible for preliminary engineering work and environmental studies for the Perry Plant before assuming his current position.

SUMMARY OF LICENSING STATUS

PERRY NUCLEAR POWER PLANT
UNITS 1 & 2

JUNE 23, 1973	APPLICATION TO CONSTRUCT UNITS 1 & 2
JULY 1974	CP--SER ISSUED
MAY 3, 1977	CONSTRUCTION PERMITS ISSUED (CPPR-148 AND CPPR-149)
JUNE 20, 1980	APPLICATION FOR OPERATING LICENSE TENDERED
JUNE 1981	ASLB PREHEARING
MAY 1982	OL--SER ISSUED
NOVEMBER 1982	ASLB HEARINGS SCHEDULED TO BEGIN
NOVEMBER 1983	APPLICANT'S ESTIMATED FUEL LOAD FOR UNIT 1

COMPARISON WITH OTHER PLANTS

FEATURE	PERRY	CLINTON	GRAND GULF
TYPE REACTOR	BWR/6	BWR/6	BWR/6
CONTAINMENT	MARK III 1/	MARK III	MARK III
FUEL (GE DESIGN)	8X8/748	8X8/624	8X8/800
RV DIA. (INSIDE)	238	218	251
FUEL RODS PER ASSY.	62	62	62
MOVEABLE CONTROL RODS	177	145	193
RATED THERMAL POWER (MW_T)	3579	2894	3833
SYSTEM PRESS. (NOMINAL IN STEAM DOME-PSIA)	1040	1040	1040
RV DESIGN PRESS. (PSIG)	1250	1250	1250
ELECTRICAL SYSTEM	RELAY LOGIC	SOLID STATE LOGIC	RELAY LOGIC
NSSS	GE	GE	GE

1/ FREE-STANDING STEEL VESSEL SUPPORTED BY STEEL-LINED REINFORCED CONCRETE FOUNDATION MAT. CLINTON & GRAND GULF ARE STEEL-LINED CONCRETE REINFORCED STRUCTURES.

PERRY (UNITS 1 & 2) NUCLEAR POWER PLANT

19 OUTSTANDING ISSUES WHICH HAD NOT BEEN RESOLVED WITH THE APPLICANT AT THE TIME THE SER WAS ISSUED.

49 CONFIRMATORY ITEMS - ITEMS WHICH HAVE ESSENTIALLY BEEN RESOLVED TO THE STAFF'S SATISFACTION BUT FOR WHICH CERTAIN ADDITIONAL AND CONFIRMATORY INFORMATION IS STILL REQUIRED WHICH THE APPLICANT HAS COMMITTED TO FURNISH THE STAFF IN THE RELATIVELY NEAR FUTURE.

15 LICENSING CONDITIONS - SIX WHICH MUST BE RESOLVED PRIOR TO ISSUANCE OF OPERATING LICENSE FOR UNIT 1: NINE WHICH ARE LONGER TERM RESOLUTION ISSUES WHICH MAY BE CITED IN THE OPERATING LICENSE FOR UNIT 1 TO ENSURE THAT NRC REQUIREMENTS ARE MET DURING PLANT OPERATIONS.

6 TECHNICAL SPECIFICATIONS - ISSUES THAT WILL BE ADDRESSED IN TECHNICAL SPECIFICATIONS IN THE LICENSE ISSUED DEFINING FEATURES, CONDITIONS AND CHARACTERISTICS GOVERNING PLANT OPERATIONS THAT CANNOT BE CHANGED WITHOUT PRIOR APPROVAL OF THE NRC STAFF.

PERRY NUCLEAR POWER PLANT

UNITS 1 & 2

OUTSTANDING ISSUES

1. TURBINE MISSILE PROTECTION
2. SEISMIC SYSTEM AND SUBSYSTEM ANALYSIS
3. PROTOTYPE REACTOR INTERNALS VIBRATION TEST PROGRAM
4. MECHANICAL & ELECTRICAL EQUIPMENT QUALIFICATION (ENVIRONMENTAL/SEISMIC-DYNAMIC)
5. INSERVICE TESTING OF PUMPS AND VALVES
6. TRANSIENT AND ACCIDENT ANALYSIS FOR ECCS, OVERPRESSURE PROTECTION & OPERATING MCPR
7. CONTROL ROOM DESIGN ASSESSMENT/AUDIT
8. CONTAINMENT SYSTEM (RECENT MARK III CONTAINMENT ISSUES)
9. POOL DYNAMIC LOADS
10. CONTAINMENT PURGE
11. PERIODIC TESTING OF ADS ACTUATION SYSTEM DURING PLANT OPERATION
12. MANUAL INITIATION/TERMINATION OF ESF SYSTEMS
13. IE BULLETIN 79-27
14. CONTROL SYSTEM FAILURE
15. FIRE PROTECTION--SAFE SHUTDOWN
16. PGCC SYSTEM FIRE PROTECTION IN CONTROL ROOM
17. HPCS ENGINE SKID PIPING CLASSIFICATION TO REG. GUIDE 1.26
18. INTERIM SHIFT STAFFING
19. EMERGENCY PREPAREDNESS PLANS

CONFIRMATORY ITEMS

STATUS

06-28-82

A. LONG TERM - REMAIN OPEN FOR 08/82 SSER.

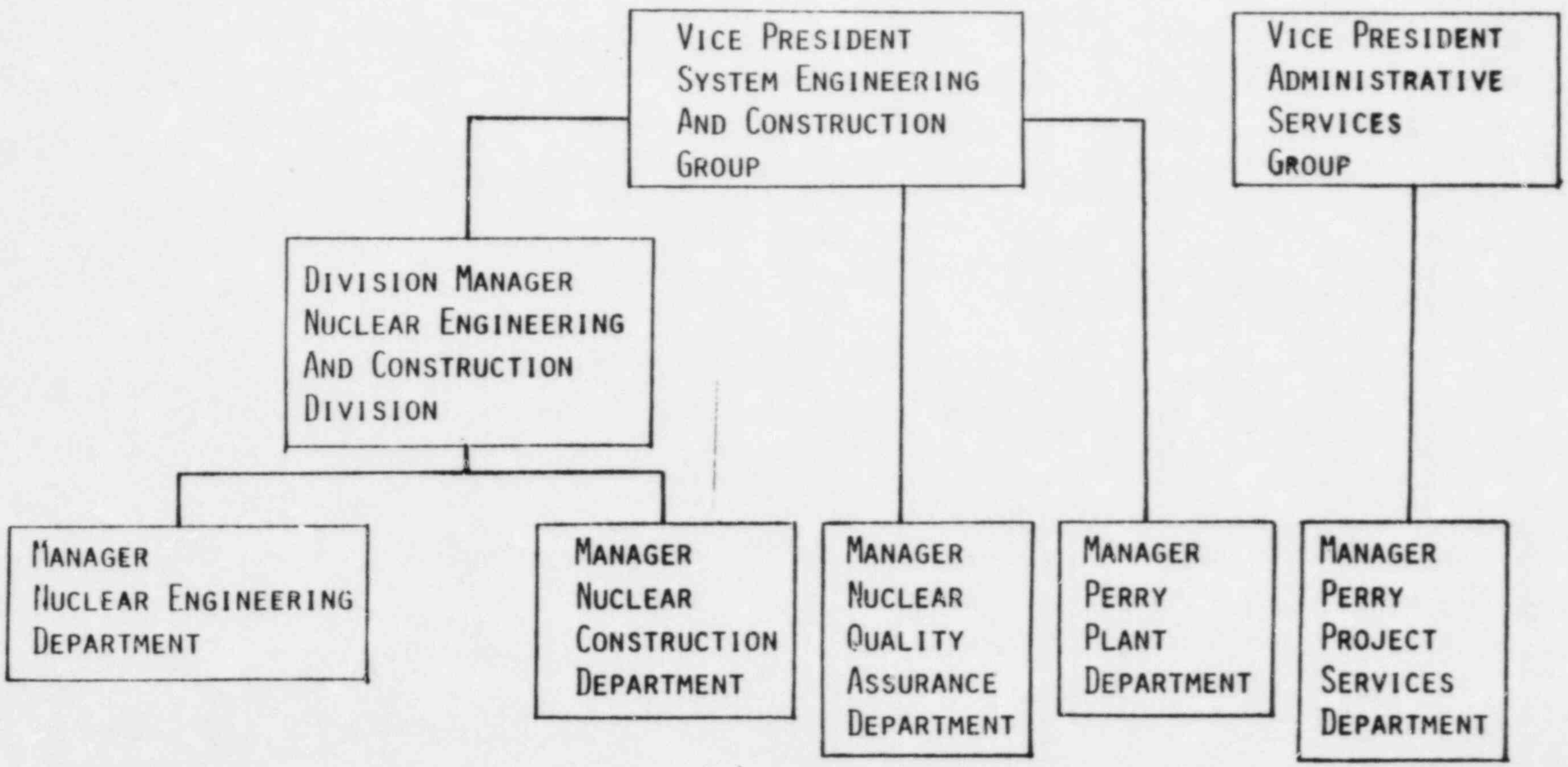
B. COMPLETED SINCE SER ISSUANCE.

C. NRC REVIEW AND COMMENT NEEDED, NO FURTHER CEI ACTION AT THIS TIME.

D. APPLICANT DEVELOPING RESPONSES.

E. GE STUDYING GENERICALLY FOR ALL BWR'S.

ORGANIZATIONAL CHANGES SINCE JANUARY 1982 NRC MANAGEMENT AUDIT AT PNPP



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APPENDIX VIII
CEI PRESENTATION - MANAGEMENT PRESENTATION

PERRY NUCLEAR POWER PLANT
SITE ORGANIZATION:
PROFESSIONAL STAFF EXPERIENCE

DEPARTMENT	# OF TOTAL PROFESSIONALS**	# OF ENGINEERING AND RELATED SCIENCE DEGREES	# OF ADVANCED ENG./SCI. DEGREES	PROFESSIONAL EXPERIENCE YEARS	NUCLEAR EXPERIENCE YEARS
OFFICE OF VP	2	2	1	73	23
PERRY PLANT DEPARTMENT	27	24	4	236	161
NUCLEAR ENGINEERING AND CONSTRUCTION DIVISION	100	95	20	1004	599
NUCLEAR QUALITY ASSURANCE DEPARTMENT	30	30	2	249	77
TOTAL	159	151	27	1562	860

**N.B. THE NUMBER OF TOTAL PROFESSIONALS INCLUDES SOME NON-TECHNICAL DEGREED PERSONNEL

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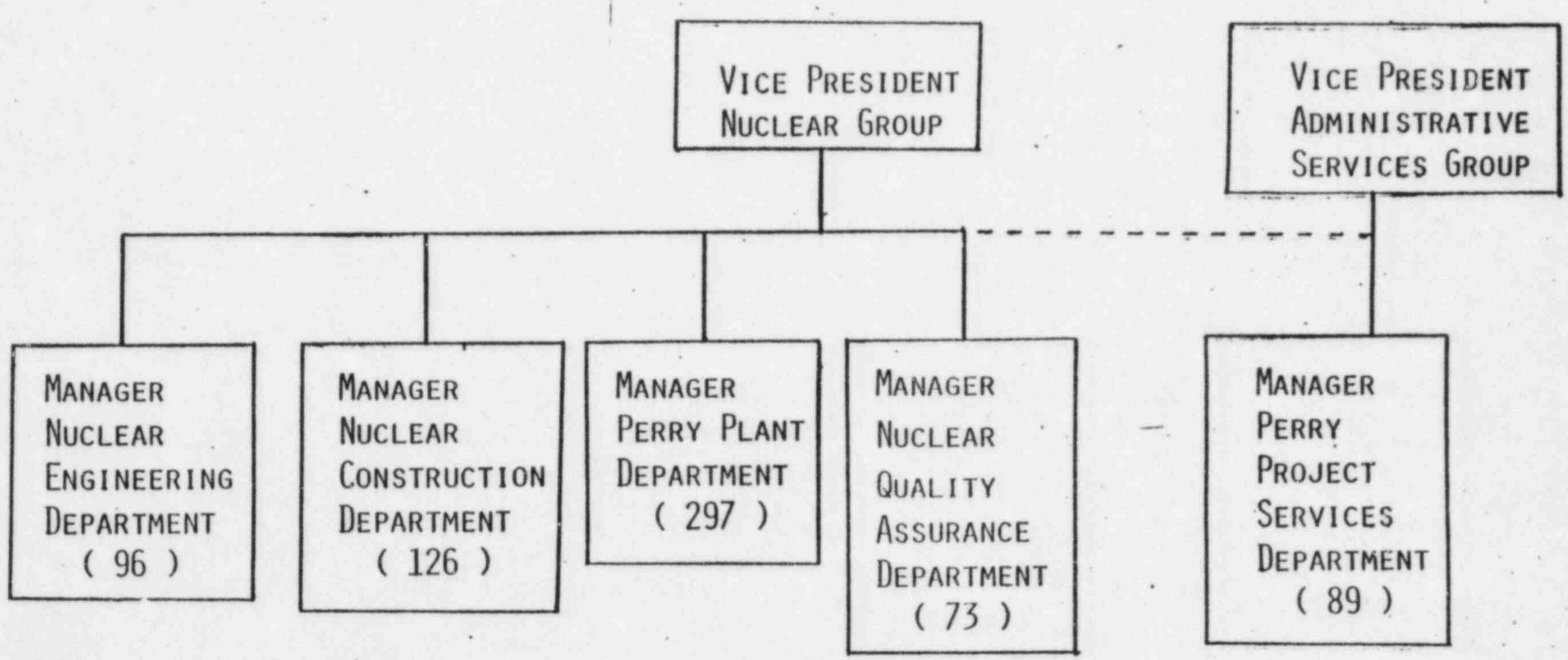
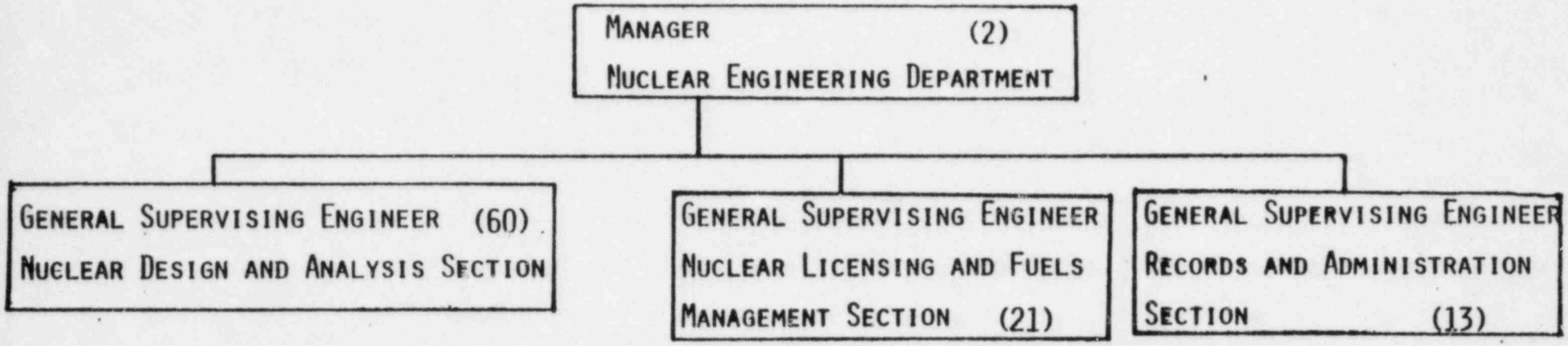
PNPP SITE ORGANIZATION AT
FUEL LOAD (11/83)**



** PROJECTED MANNING LEVELS FOR 11/83 SHOWN IN PARENTHESES:
TOTAL MANNING OF NUCLEAR GROUP (INCLUDING OFFICE OF THE VICE-PRESIDENT)= 595


MANAGER (2)
NUCLEAR ENGINEERING DEPARTMENT
GENERAL SUPERVISING ENGINEER (60)
NUCLEAR DESIGN AND ANALYSIS SECTION
GENERAL SUPERVISING ENGINEER
NUCLEAR LICENSING AND FUELS
MANAGEMENT SECTION (21)
GENERAL SUPERVISING ENGINEER
RECORDS AND ADMINISTRATION
SECTION (13)

INDEPENDENT SAFETY ENGINEERING GROUP

- FIVE INDIVIDUALS AS MEMBERS OF ISEG
- STAGGERED TERMS OF SERVICE; ASSURES CONTINUITY OF EXPERIENCE
- STAFFED BY ENGINEERS AND OTHER TECHNICALLY-ORIENTED PERSONNEL
- QUALIFICATIONS TO ANSI/ANS 3.1, SECTIONS 4.1 AND 4.2 (REV. 1981)
- ISEG CHAIRMAN REPORTS TO MANAGER, NUCLEAR ENGINEERING DEPARTMENT

ISEG: EVALUATION AND AUDIT FUNCTIONS

- USE INPO SEE-IN PROGRAM, NUCLEAR NOTEPAD
- EVALUATE LIS, NRC, INPO, ISSUANCES FOR SAFETY DESIGN IMPLICATIONS
- MAKE DESIGN EVALUATIONS AND RECOMMENDATIONS TO IMPROVE PLANT SAFETY
- PERIODICALLY REVIEW AND AUDIT PNPP OPERATIONAL QA PROGRAM

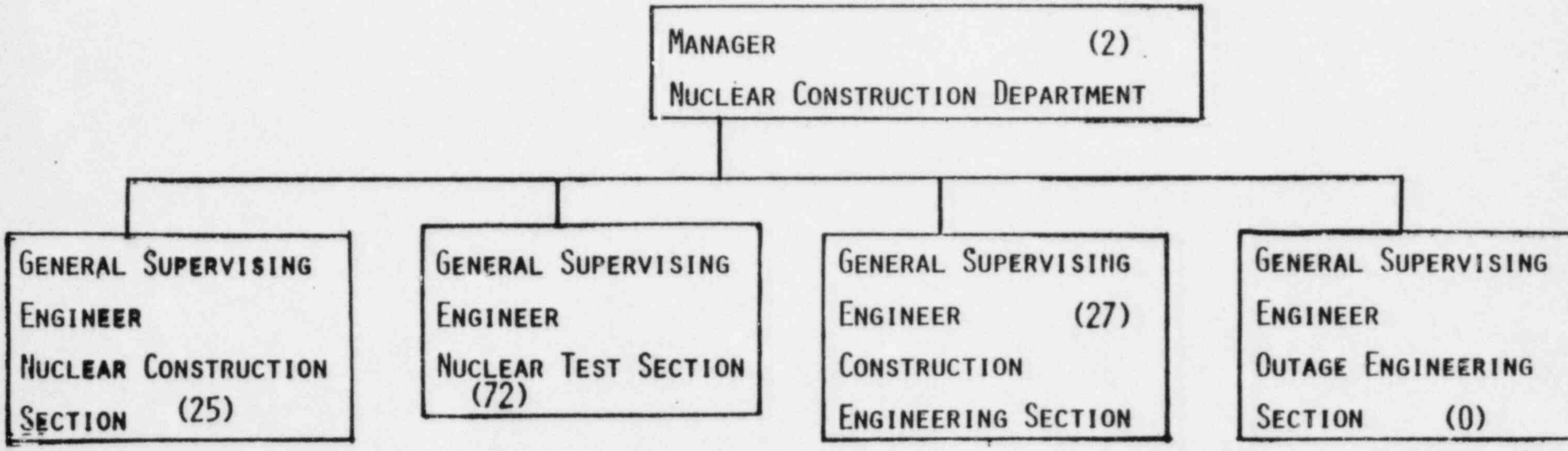
NON-CEI TECHNICAL RESOURCES

Gilbert Associates, Inc.
- Architect-Engineer
- Engineering and Design Services for Unit 1 Start-Up and Operation

General Electric
- Nuclear Steam System, Fuel and Turbine Supplier
- Technical Assistance for Unit 1 Start-Up and Operation

NUS
- Environmental Monitoring and Evaluation

Other Consultants

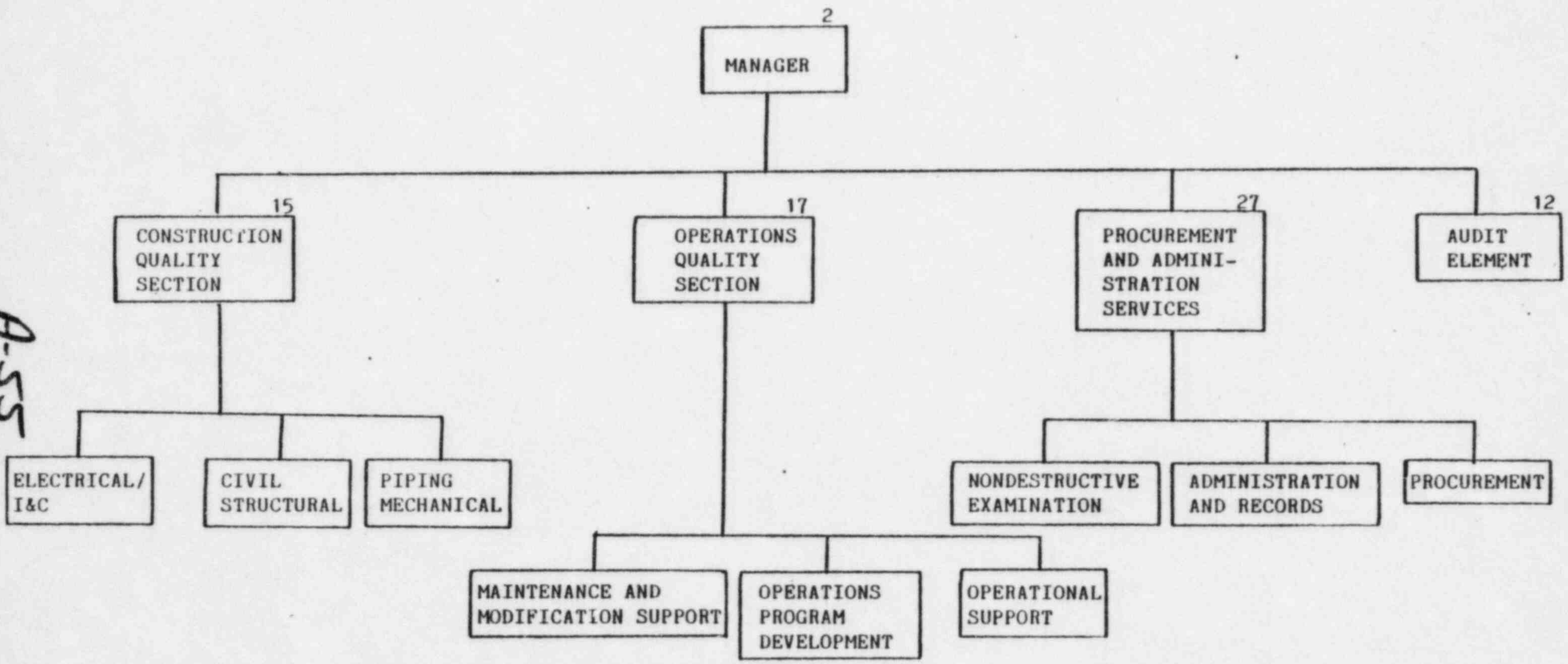

MANAGER (2)
NUCLEAR CONSTRUCTION DEPARTMENT
GENERAL SUPERVISING ENGINEER NUCLEAR CONSTRUCTION SECTION (25)
GENERAL SUPERVISING ENGINEER NUCLEAR TEST SECTION (72)
GENERAL SUPERVISING ENGINEER (27) CONSTRUCTION ENGINEERING SECTION
GENERAL SUPERVISING ENGINEER OUTAGE ENGINEERING SECTION (0)

A-54

73 CEI

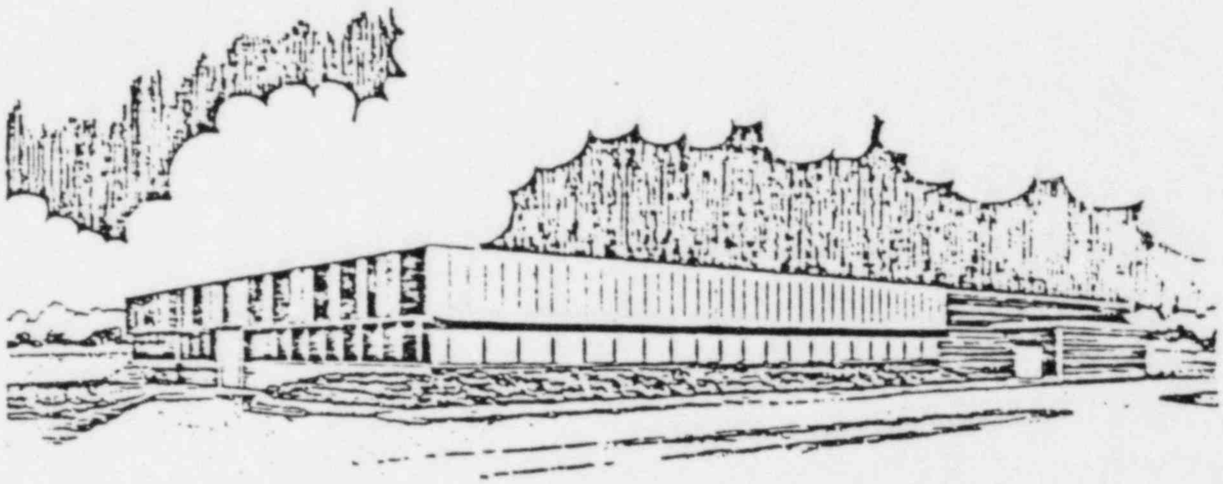
NUCLEAR QUALITY ASSURANCE DEPARTMENT

UNIT I FUEL LOAD



A-55

PERRY TRAINING FACILITY



NUCLEAR PROJECT TRAINING SECTION
PERSONNEL QUALIFICATIONS

1 - BSEE, MS Eng. Admin.

1 - BA Psych., MA Education

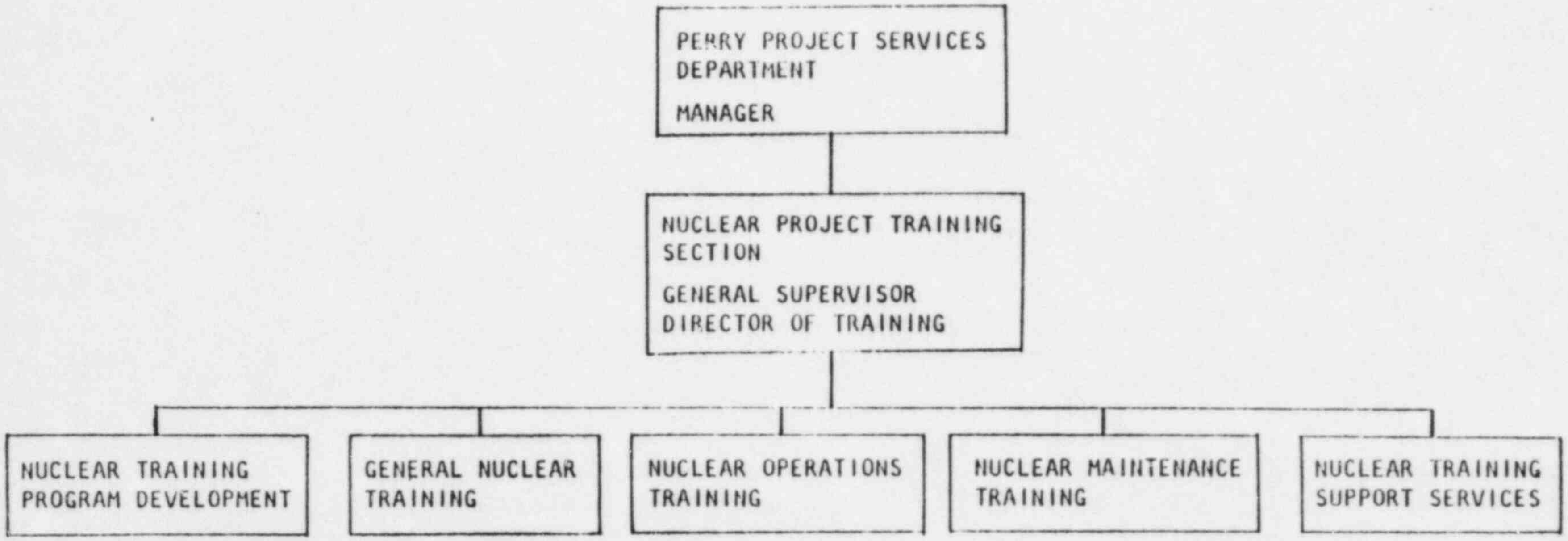
1 - BA Political Science, Experience in Training Administration

1 - BA Education, MA Education, Advanced courses toward PHD Curriculum Supervision

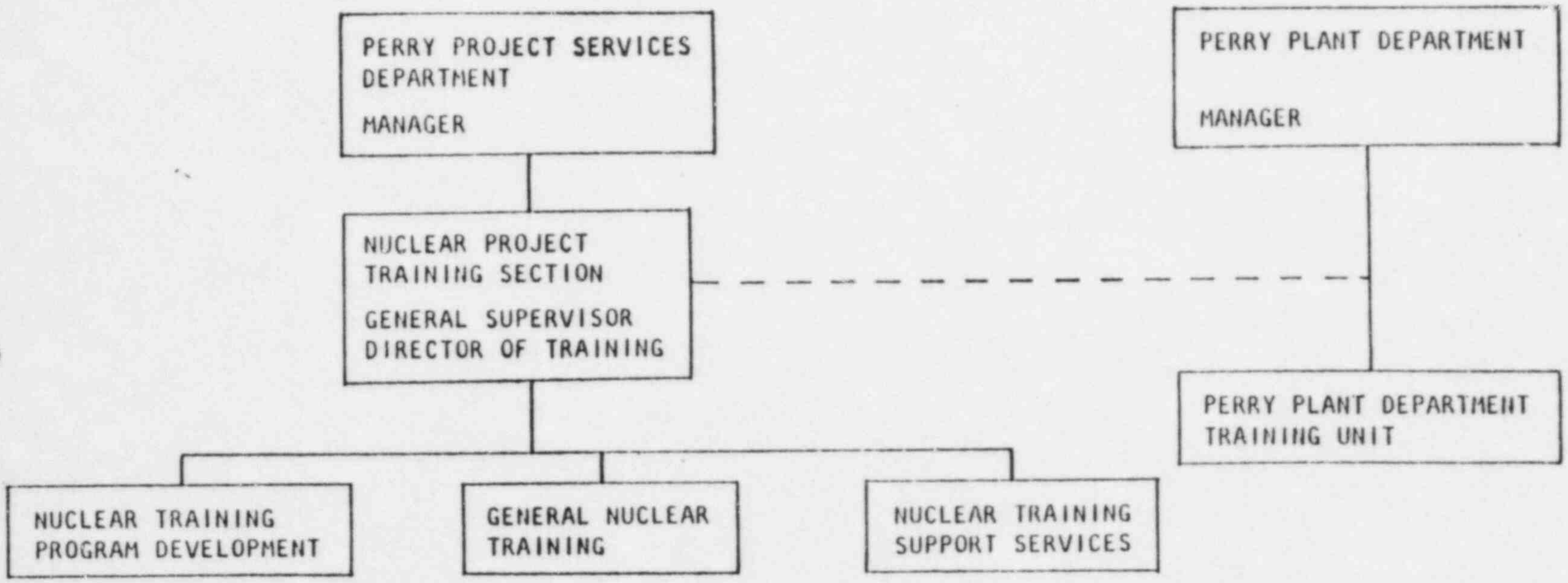
1 - BSME

1 - BS Broadcast, MA Communications

PNPP SITE TRAINING ORGANIZATION
AT FUEL LOAD 11/83

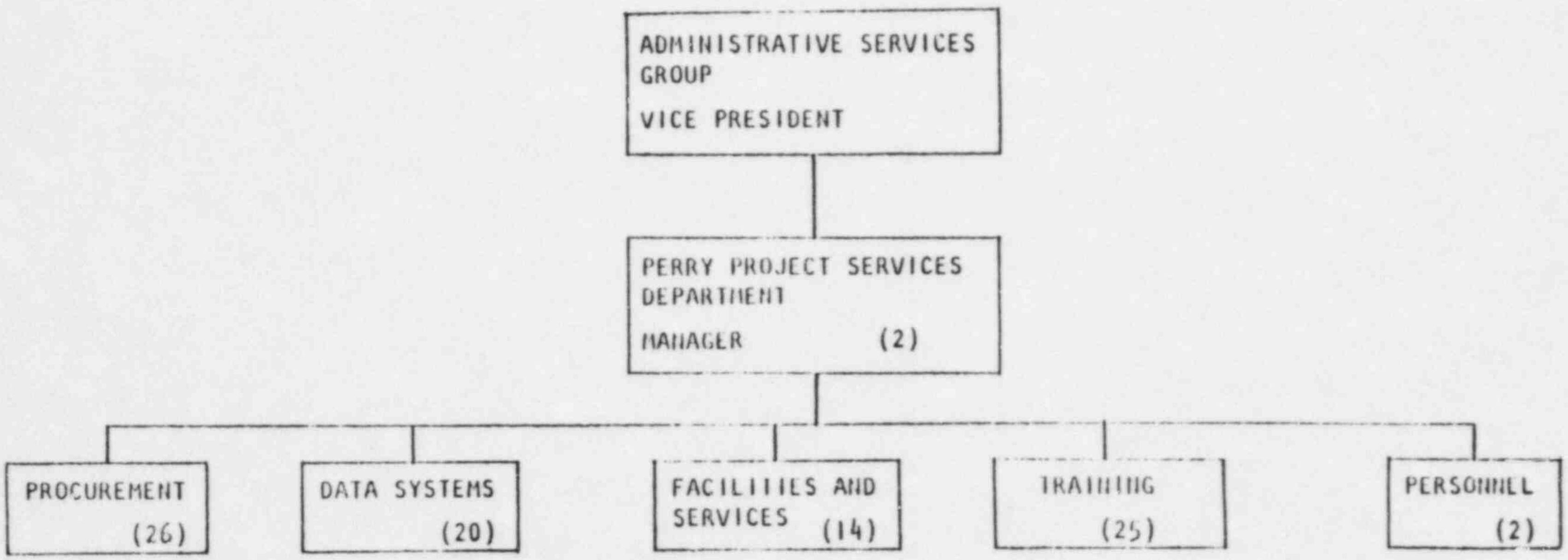


PNPP SITE TRAINING ORGANIZATION
PRESENT

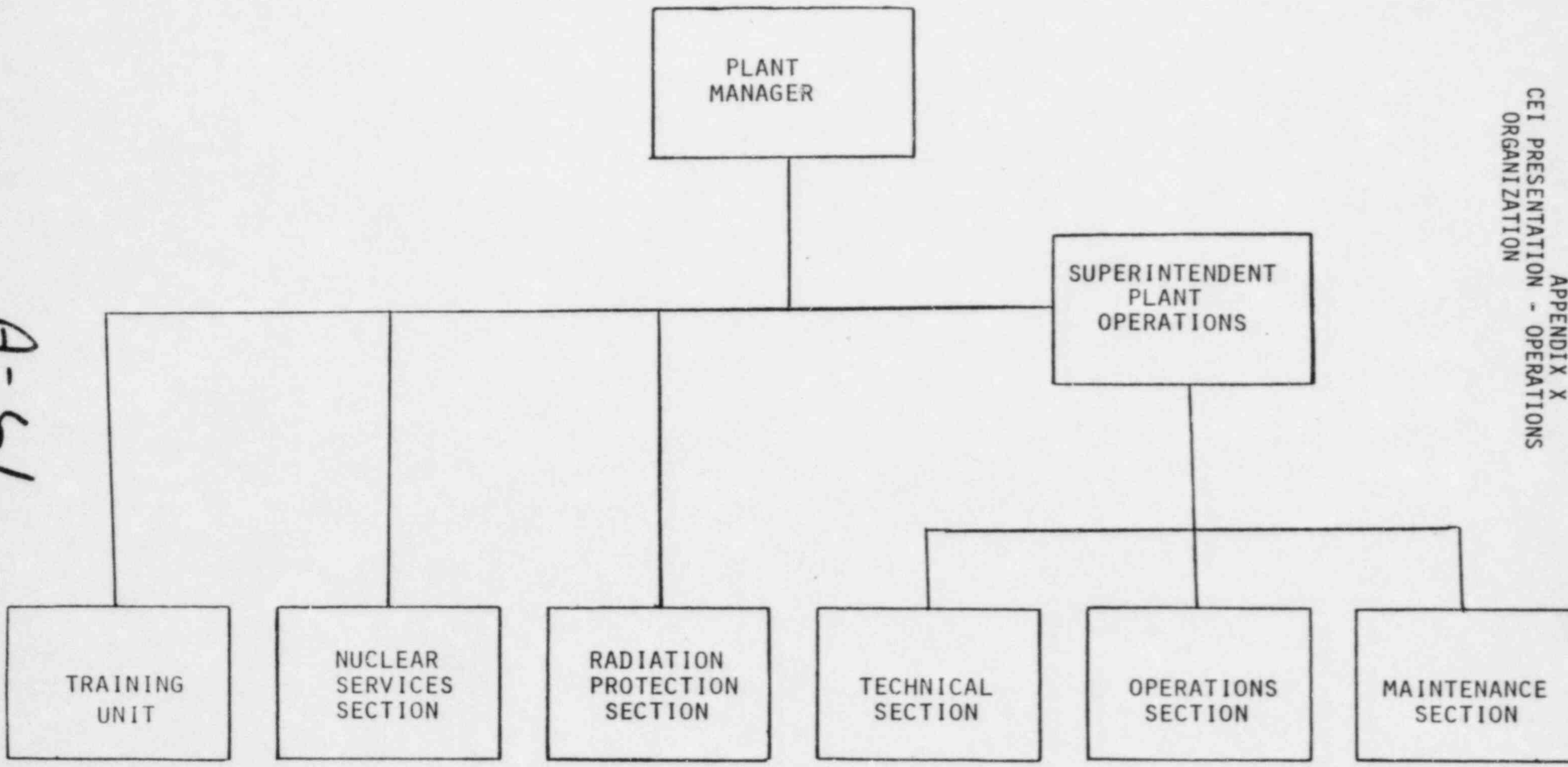


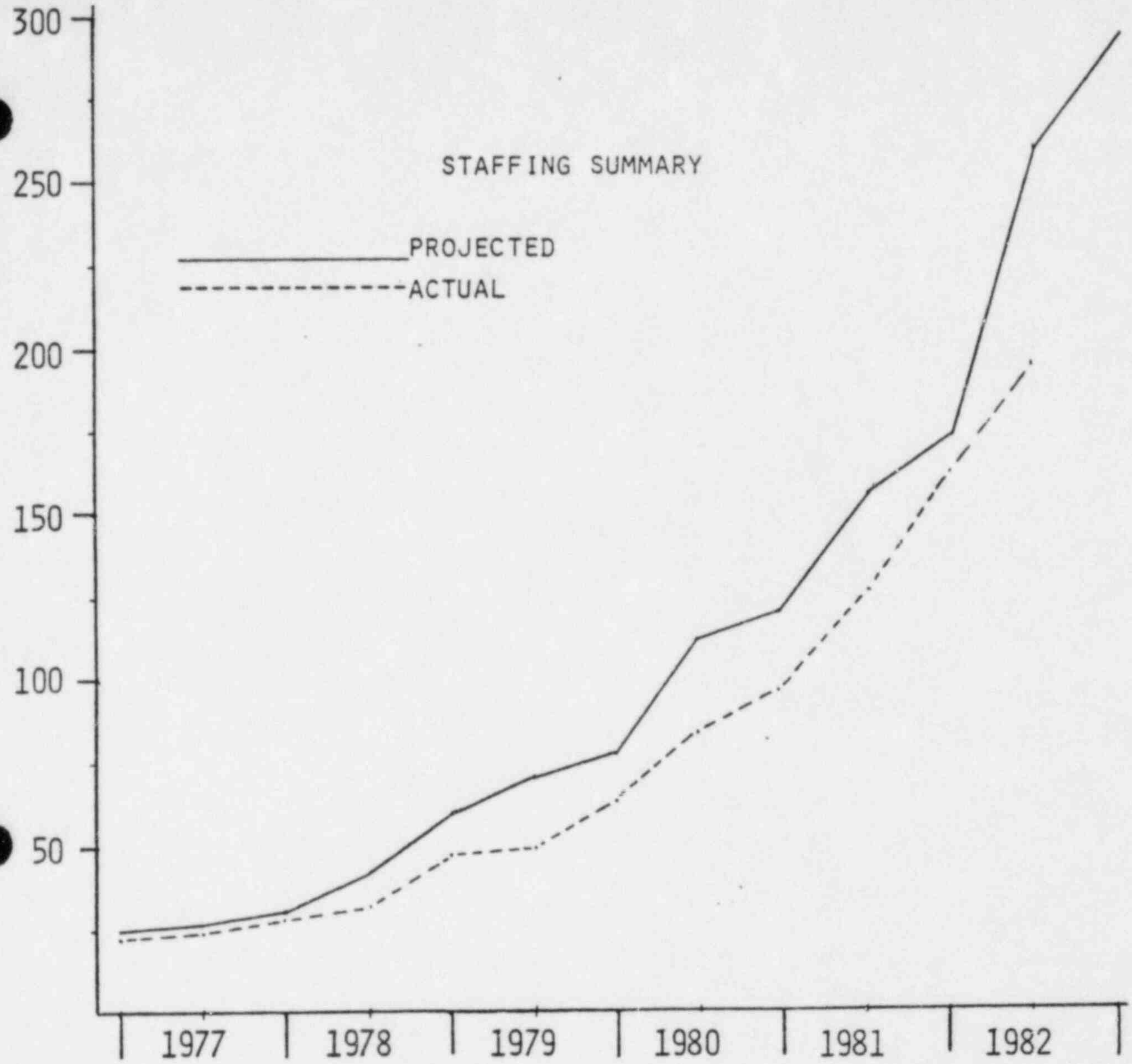
A-59

AT FUEL LOAD, UNIT 1



PERRY PLANT DEPARTMENT





	TO DATE	PROJECTED
OPERATIONS	52	63
MAINTENANCE	47	85
TECHNICAL	41	53
RADIATION PROTECTION	22	43
OTHER (MANAGEMENT, ADMINISTRATIVE, TRAINING, SECURITY)	33	53
	195	297

ATTRITION RATE LAST 12 MONTHS - 6%

ATTRITION RATE IN OPERATIONS - 2%

PERRY NUCLEAR POWER PLANT DEPARTMENT
STAFF POWER PLANT EXPERIENCE (IN MAN-YEARS)

TOTAL NUCLEAR EXPERIENCE	
SUPERVISORS AND MANAGEMENT	525
TECHNICIANS AND OPERATORS	400
TOTAL MAN-YEARS	925

	COMMERCIAL NUCLEAR POWER PLANT EXPERIENCE			NON-COMMERCIAL NUCLEAR PLANT EXPERIENCE	COMMERCIAL FOSSIL PLANT EXPERIENCE
	PNPP NUCLEAR EXPERIENCE	COMMERCIAL NUCLEAR EXPERIENCE (OTHER)	TOTAL COMMERCIAL EXPERIENCE		
SUPERVISORS AND MANAGEMENT	210	38	248	277	126
TECHNICIANS AND OPERATORS	97	10	107	293	35
TOTAL EXPERIENCE IN MAN-YEARS	307	48 143 320	355	570	161

TEMPORARY ASSIGNMENTS
TO OTHER UTILITIES (REPRESENTATIVE)

BRUNSWICK

DAVIS BESSE

DRESDEN

MONTICELLO

GRAND GULF

MILLSTONE

HATCH

LASALLE

PEACH BOTTOM

STARTUP ACTIVITY PARTICIPATION

- OPERATING PROCEDURE DEVELOPMENT
- SURVEILLANCE DEVELOPMENT
- EMERGENCY PLAN PREPARATION AND IMPLEMENTATION
- CONTROL ROOM HUMAN FACTORS STUDY
- SYSTEM WALKDOWN AND TURNOVERS
- FIRE PROTECTION PLAN
- SYSTEM OPERATING DESCRIPTION PREPARATION
- PLANT OPERATIONS
- STARTUP PROCEDURE DEVELOPMENT

DEDICATED TRAINING RESOURCES

ASSOCIATED SUPPORT ORGANIZATIONS

OHIO STATE UNIVERSITY -

1. OPERATOR COLLEGE UPGRADE INSTRUCTION
2. GENERAL EMPLOYEE TRAINING PROGRAM DEVELOPMENT
3. RADIOLOGICAL CONTROLS TRAINING (GENERAL) PROGRAM DEVELOPMENT

LAKELAND COMMUNITY COLLEGE -

1. OPERATOR COLLEGE UPGRADE INSTRUCTION

GENERAL ELECTRIC -

1. PERRY SPECIFIC CONTROL ROOM SIMULATOR OPERATOR CERTIFICATION SERVICES
2. NUCLEAR DISCIPLINE TRAINING SERVICES

NUCLEAR EDUCATION TRAINING SERVICES -

1. MISCELLANEOUS OPERATOR ACADEMIC TRAINING

UNIVERSITY OF WISCONSIN -

1. INSTRUCTOR TRAINING

GENERAL PHYSICS CORPORATION -

1. BASIC ACADEMIC TRAINING

2. LICENSE CANDIDATE TESTING AND EVALUATION

PERRY CONTROL ROOM SIMULATOR TRAINING

PERSONNEL ATTENDED - 46

SUCCESSFUL COMPLETION

RO - 9

SRO -30

TOTAL 39 85%

IN TRAINING - 15

PLANT ENGINEER STAFF

- 12 ENGINEERS COMPLETED CONTROL ROOM SIMULATOR TRAINING OF WHICH 10 ARE SRO CERTIFIED AND 2 RO CERTIFIED

NRC LICENSED SENIOR REACTOR OPERATOR COLLEGE PROGRAMS

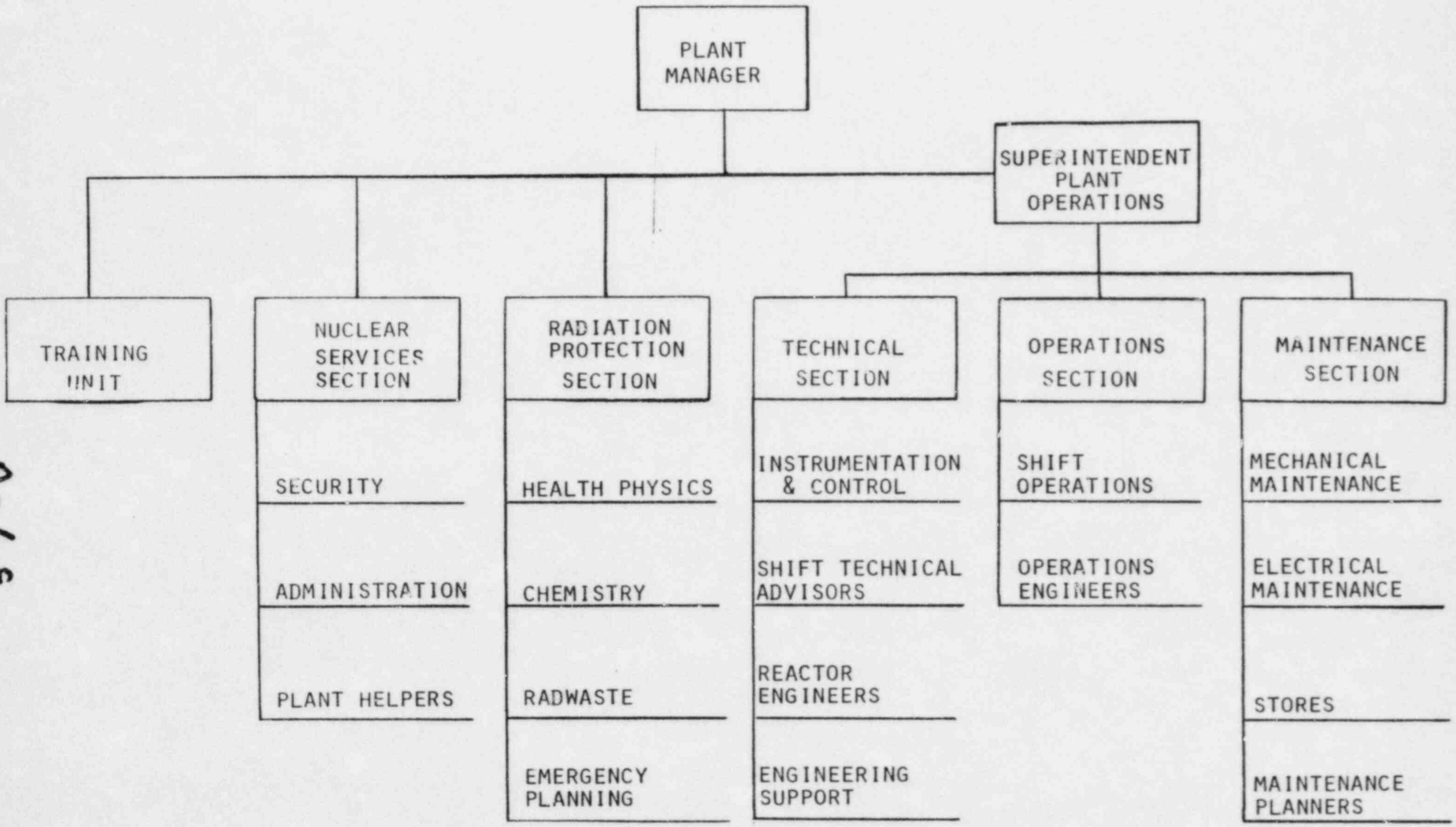
COURSE	QUARTER CREDIT HOURS COMPLETED
CALCULUS AND DIFFERENTIAL EQUATIONS	25
PHYSICS	15
CHEMISTRY	18
THERMAL SCIENCES	12
TOTAL	70

COMPLETION STATUS

13 OPERATOR CANDIDATES $\geq$ 66 QTR. HRS.

10 OPERATOR CANDIDATES $\geq$ 10 QTR. HRS.

PERRY PLANT DEPARTMENT



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PROJECTED STAFFING (EXCLUDING TRAINING ORGANIZATION AND GUARD FORCE)

UNIT 1	UNITS 1 & 2
297	457

PERRY NUCLEAR POWER PLANT

POWER PLANT OPERATIONS EXPERIENCE

PLANT MANAGER

10 YEARS PERRY NUCLEAR POWER PLANT

- VARIOUS MANAGEMENT POSITIONS DURING CONSTRUCTION OF PNPP
- S.R.O. CERTIFIED ON BWR-6 SIMULATOR

18 YEARS COMMERCIAL FOSSIL PLANT

PLANT OPERATIONS SUPERINTENDENT

- CURRENTLY UNFILLED -

OPERATIONS GENERAL SUPERVISOR

7 YEARS NUCLEAR NAVY

- ENGINEERING OFFICER OF THE WATCH ON NUCLEAR SUBMARINES

8 YEARS PERRY NUCLEAR POWER PLANT

- S.R.O. CERTIFIED ON BWR SIMULATOR

2 YEARS COMMERCIAL FOSSIL PLANT

MAINTENANCE GENERAL SUPERVISOR

4 YEARS PERRY NUCLEAR POWER PLANT

- R.O. CERTIFIED

10 YEARS COMMERCIAL FOSSIL PLANT

NUCLEAR SERVICES GENERAL SUPERVISOR

- CURRENTLY UNFILLED -

TECHNICAL GENERAL SUPERVISOR

6 YEARS PERRY NUCLEAR POWER PLANT

- S.R.O. CERTIFIED ON BWR-6 SIMULATOR

3 YEARS COMMERCIAL FOSSIL PLANT

RADIATION PROTECTION GENERAL SUPERVISOR

6 YEARS PERRY NUCLEAR POWER PLANT

4 YEARS DAVIS-BESSE NUCLEAR POWER PLANT

- TEST LEADER DURING PLANT START-UP

4 YEARS COMMERCIAL FOSSIL PLANT

SHIFT SUPERVISORS (5)

29 YEARS NUCLEAR NAVY

6 YEARS COMMERCIAL NUCLEAR PLANT OPERATION

28 YEARS PERRY NUCLEAR POWER PLANT

- ALL S.R.O. CERTIFIED ON BWR SIMULATORS

5 YEARS COMMERCIAL FOSSIL PLANT

AVERAGE POWER PLANT EXPERIENCE - 13.6 YEARS

UNIT SUPERVISORS (7)

41 YEARS NUCLEAR NAVY

0 YEARS COMMERCIAL NUCLEAR PLANT OPERATION

20.5 YEARS PERRY NUCLEAR POWER PLANT

- ALL S.R.O. CERTIFIED ON BWR SIMULATOR

14 YEARS COMMERCIAL FOSSIL PLANT

AVERAGE POWER PLANT EXPERIENCE - 10.8 YEARS

SUPERVISING OPERATORS (12)

54.5 YEARS NUCLEAR NAVY

0 YEARS COMMERCIAL NUCLEAR PLANT OPERATION

26 YEARS PERRY NUCLEAR POWER PLANT

39.5 YEARS COMMERCIAL FOSSIL PLANT

- 8 S.R.O. CERTIFIED, 4 R.O. CERTIFIED ON BWR SIMULATORS

AVERAGE POWER PLANT EXPERIENCE - 10 YEARS

LICENSED OPERATOR TRAINEES (23)*

149.5 YEARS NUCLEAR NAVY

2 YEARS COMMERCIAL NUCLEAR PLANT OPERATION

9.5 YEARS PERRY NUCLEAR POWER PLANT

10.5 YEARS COMMERCIAL FOSSIL PLANT

AVERAGE POWER PLANT EXPERIENCE - 7.5 YEARS

*ON BOARD OR ACCEPTED JOB OFFERS

PRA/SIA

PROBABILISTIC RISK ASSESSMENT SYSTEMS INTERACTION ANALYSIS

I. CEI'S VIEW OF PRA/SIA

II. HISTORY OF PRA/SIA AT CEI

III. MANAGEMENT ACTION PLAN (MAP)

IV. MINI-PRA

V. SIA

VI. CONCLUSION

APPENDIX XI
CEI'S PRA PRESENTATION

III. MANAGEMENT ACTION PLAN

- CYGNA ENERGY SERVICES
- RELATE PRA/SIA TECHNIQUES TO NEEDS
- A BASE FOR FUTURE PRA/SIA EFFORTS
- COMPLIMENTARY ASPECTS OF PRA AND SIA
- FURTHER STUDIES
  - MINI-PRA
  - SIA

A-74

IV. MINI-PRA

- GRAND GULF RSSMAP

- LIMERICK/WASH-1400/GE STD PLANT

- FIVE TASKS

A-75

TASK 1

GATHER SITE SPECIFIC DATA FOR EXPLANT CONSEQUENCES

TASK 2

EMERGENCY RESPONSE CAPABILITY INPUTS

TASK 3

RADIONUCLIDE RELEASE SOURCE TERM DETERMINATION

RSSMAP

PLANT UNIQUE SYSTEMS

TASK 4

DETERMINE EX-PLANT CONSEQUENCES

SUPPRESSION POOL DF'S

TASK 5

REPORT

V. SYSTEM INTERACTION ANALYSIS

- SEISMIC INTERACTION INSPECTION GROUP
  - SEISMIC CLEARANCES
  - FALL DOWN
- ENGINEERING REVIEW PROCEDURES
  - REDUNDANT, INDEPENDENT, SPATIALLY SEPARATED
- APPENDIX R SAFE SHUTDOWN ANALYSIS
- INTERNALLY GENERATED MISSILE STUDIES
- HIGH AND MODERATE ENERGY PIPE BREAK ANALYSIS
- EQUIPMENT QUALIFICATION
- NRC, INPO, OPERATING EXPERIENCE

A-77

- NON 1E I & C SYSTEMS
- ACHIEVE AND MAINTAIN COLD SHUTDOWN

ADVERSE SYSTEM INTERACTION -- THE OCCURENCE OF A SET OF DEPENDENT FAILURES THAT DEFEATS OR JEOPARDIZES THE PERFORMANCE OF A SAFETY FUNCTION.

OR

DEPENDENT FAILURES WHICH COULD CAUSE A TRANSIENT MORE SEVERE THAN THOSE ANALYZED IN THE FSAR.

SAFETY FUNCTIONS

- REACTIVITY CONTROL
- VESSEL WATER LEVEL
- VESSEL PRESSURE
- DECAY HEAT REMOVAL

1. FIND ADVERSE STATES

2. DEVELOP FAULT TREES USING ADVERSE SYSTEM STATES AS THE TOP EVENT.

3. TEST FOR INTERDEPENDENCIES BASED ON:

   - LOCATION

   - ELECTRIC POWER

   - SENSOR

4. EVALUATE POTENTIAL SI'S.

VI. CONCLUSION

- IN HOUSE EXPERTISE
- USEFUL IMPLEMENTATION
- FUTURE EFFORTS
  - NRC REQUIREMENTS
  - INDUSTRY GUIDELINES

"OPEN ISSUE" PRA/SIA PROGRAM ELEMENT VS NEED MATRIX

NEEDS

PRA/SIA PROGRAM ELEMENTS	EFFORT MAN-WEEKS	FSAR QUESTIONS	INTERVENOR ISSUE EMERGENCY PLAN	INTERVENOR ISSUE ASIATIC CLAMS
I) PRA/AVAILABILITY ELEMENTS	—	—	—	—
1) ASSESS EXISTING ANALYSIS	2		✓	
2) CONSTRUCT LOGIC MODELS	2			✓
3) QUANTIFY LOGIC MODELS	8			✓
4) DATA COLLECTION	6			✓
5) EX-PLANT CONSEQUENCES	—	—	—	—
1) ASSESS EXISTING DATA	2		✓	
2) PREPARE CRAC2 SITE DATA	9		✓	
3) RUN CRAC2 - MINI-PRA	13		✓	
II) SYSTEMS INTERACTION ELEMENTS	—	—	—	—
1) SPECIFIC TASKS USING SIA TECHNIQUES	—	—	—	—
a) ELECTRICAL POWER AND CONTROL SYSTEMS	37	✓		
2) GENERAL SIA TASKS	—	—	—	—
a) DETERMINE FAULT TREE TOP EVENTS	14			✓

PNPP

SRV

&

LOCA-RELATED

LOADS

A-82

PERRY EVALUATION

SRV ACTUATION

POOL BOUNDARY PRESSURES

19 VALVES

LOW-LOW SET

8 ADS

1-9-9 MULTI-VALVE ACTUATION

COMPLETE DYNAMIC ANALYSIS

GESSAR II, APP. 3B

A-83

SRV EFFECTS-PERRY

- HORIZONTAL "TEE" STIFFENERS

  CONTAINMENT "RINGING"

  FILLED ANNULUS FOR
  RESPONSE REDUCTION

- SRV LINE HORIZONTAL SUPPORT
  STRENGTHEN VERTICAL SUPPORT

-

A-84

PERRY EVALUATION

LOCA-RELATED LOADS

MINIMAL EFFECT OF
CO & CHUGGING

POOL SWELL (MAJOR IMPACT)

GESSAR II, APP. 3B

GE/NRC DISCUSSIONS

DRAFT ACCEPTANCE CRITERIA
"DESIGN VERIFICATION"

A-85

PERRY POOL SWELL

ORIGINAL ANALYSIS

VELOCITY OF 40 FPS

FROTH IMPACT OF 15 PSI

FROTH "DRAG" ($\triangle$P)

DRAFT CRITERIA

VELOCITY GRADIENT (Vmax=50)

FROTH IMPACT (Pmax f(HEIGHT)

PERRY PLANT UNIQUE$\triangle$P

A- 86

PERRY-POOL SWELL

SHIELDING

PIPING & VALVES

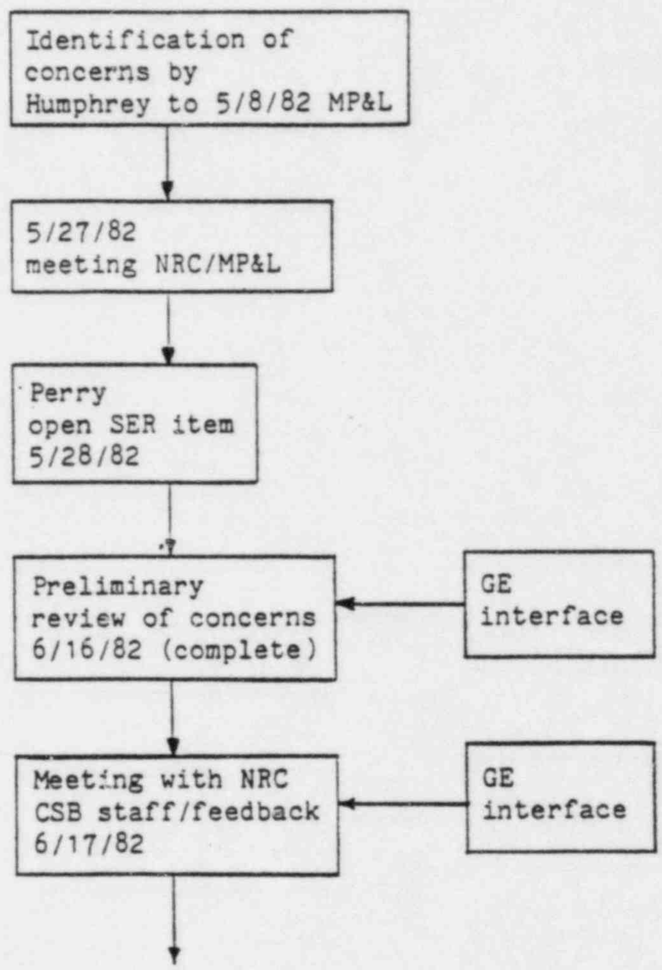
HATCHES & LOCKS

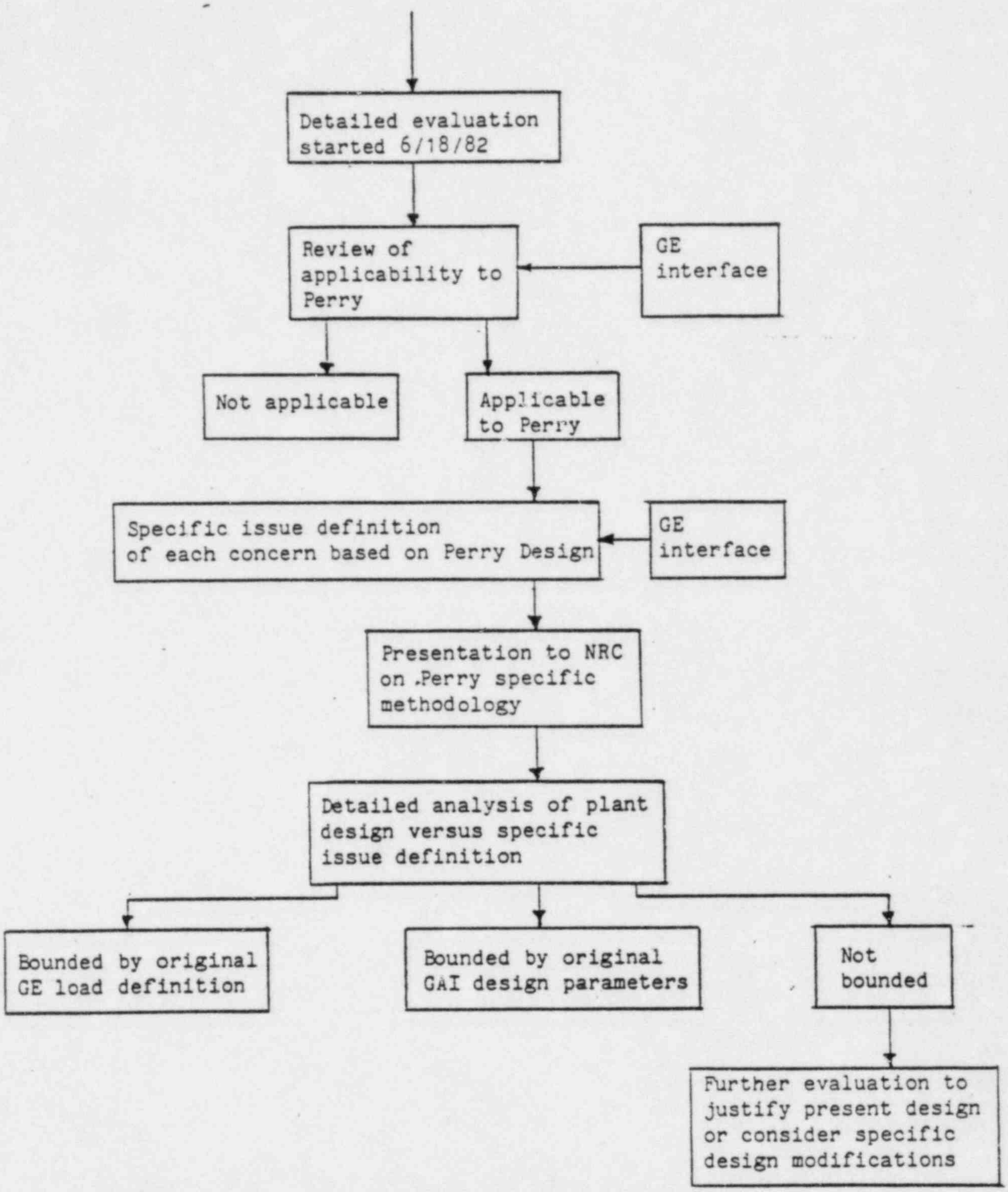
HCU FLOOR PLATING

PENETRATION REINFORCEMENT

INCREASED PIPING RESPONSE
& SUPPORT REACTIONS

EQUIPMENT QUALIFICATION


Identification of concerns by Humphrey to 5/8/82 MP&L
5/27/82 meeting NRC/MP&L
Perry open SER item 5/28/82
Preliminary review of concerns 6/16/82 (complete)
GE interface
Meeting with NRC CSB staff/feedback 6/17/82
GE interface


Detailed evaluation started 6/18/82
Review of applicability to Perry
GE interface
Not applicable
Applicable to Perry
Specific issue definition of each concern based on Perry Design
GE interface
Presentation to NRC on Perry specific methodology
Detailed analysis of plant design versus specific issue definition
Bounded by original GE load definition
Bounded by original GAI design parameters
Not bounded
Further evaluation to justify present design or consider specific design modifications

SUMMARY

GESSAR LOADS

NRC DRAFT ACCEPTANCE CRITERIA

EQUIPMENT QUALIFICATION

TOTAL COMMITMENT

A-90

AGENDA

HISTORY

PNPP CONFIGURATION

PNPP EVALUATION

SUMMARY

PNPP

HISTORY OF "HYDRODYNAMIC LOADS"

GE EARLY DEFINITION

FIRL

GESSAR (PRELIM)

GESSAR II -APP 3B

NUREGS

A-92

PERRY CONFIGURATION

GE STD 238 PLANT

- STEEL CONTAINMENT

- 19 SRV's

WETWELL PLATFORMS

- COLUMNS IN POOL

- HIGHER HCU PLATFORMS

VENT STRUCTURE

- 7.5′ SUBMERGENCE

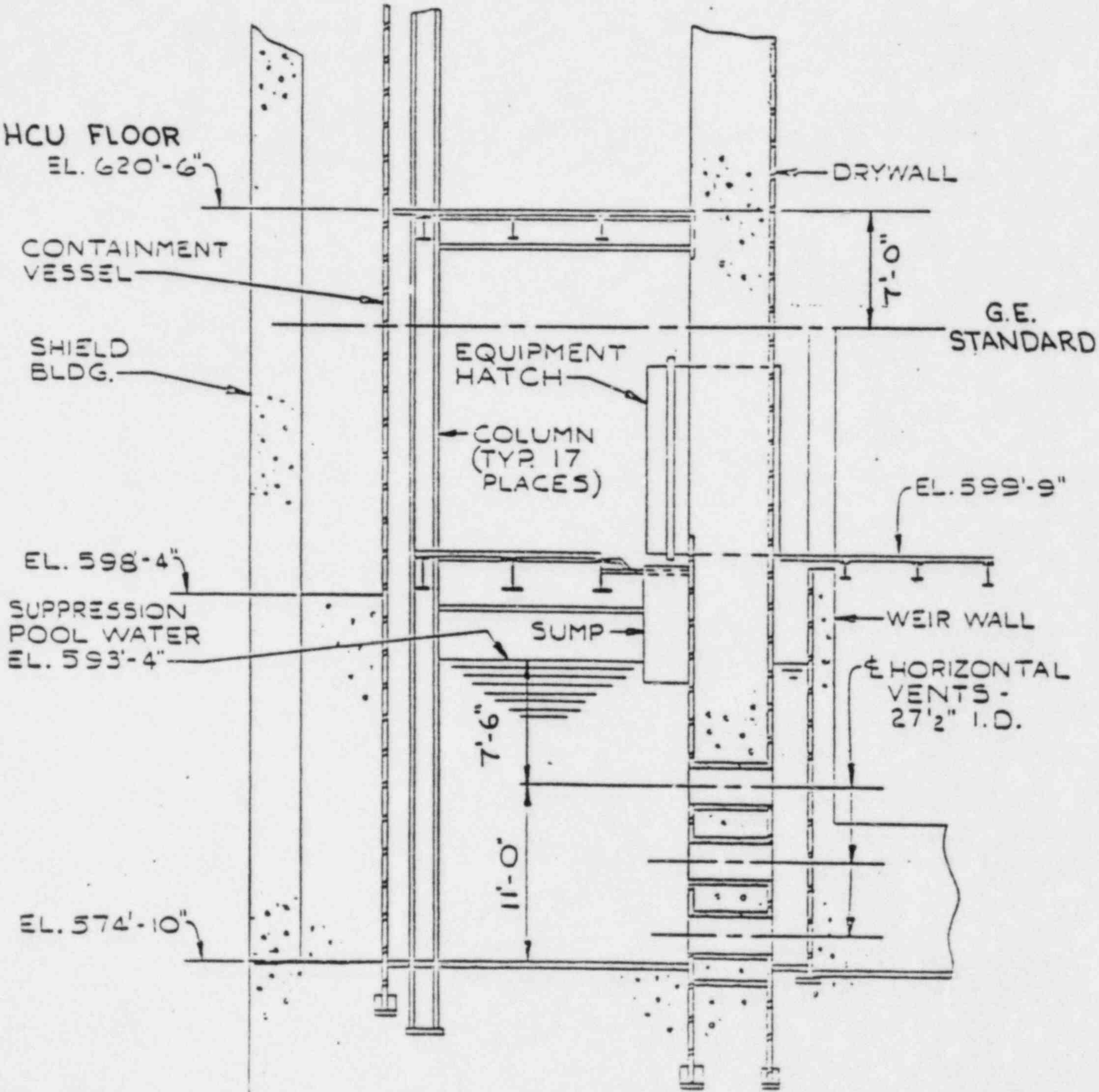
A-93

PERRY CONFIGURATION

CONT.

SEPARATE REACTOR
BLDG. BASE MAT

FILLED ANNULUS BETWEEN
CONTAINMENT & SHIELD BLDG.



SUPPRESSION POOL CROSS SECTION

PERRY NUCLEAR POWER PLANT

A-95

DEVELOPMENT OF THE EMERGENCY PLAN

DEVELOPMENT CRITERIA

- 10CFR50 Appendix E
- NUREG 0654
- NUREG 0696

STRONG CEI INVOLVEMENT

- Preparation of the PNPP Emergency Plan
- Preparation of Implementing Instructions
- Coordination Activities with other Organizations
- Training

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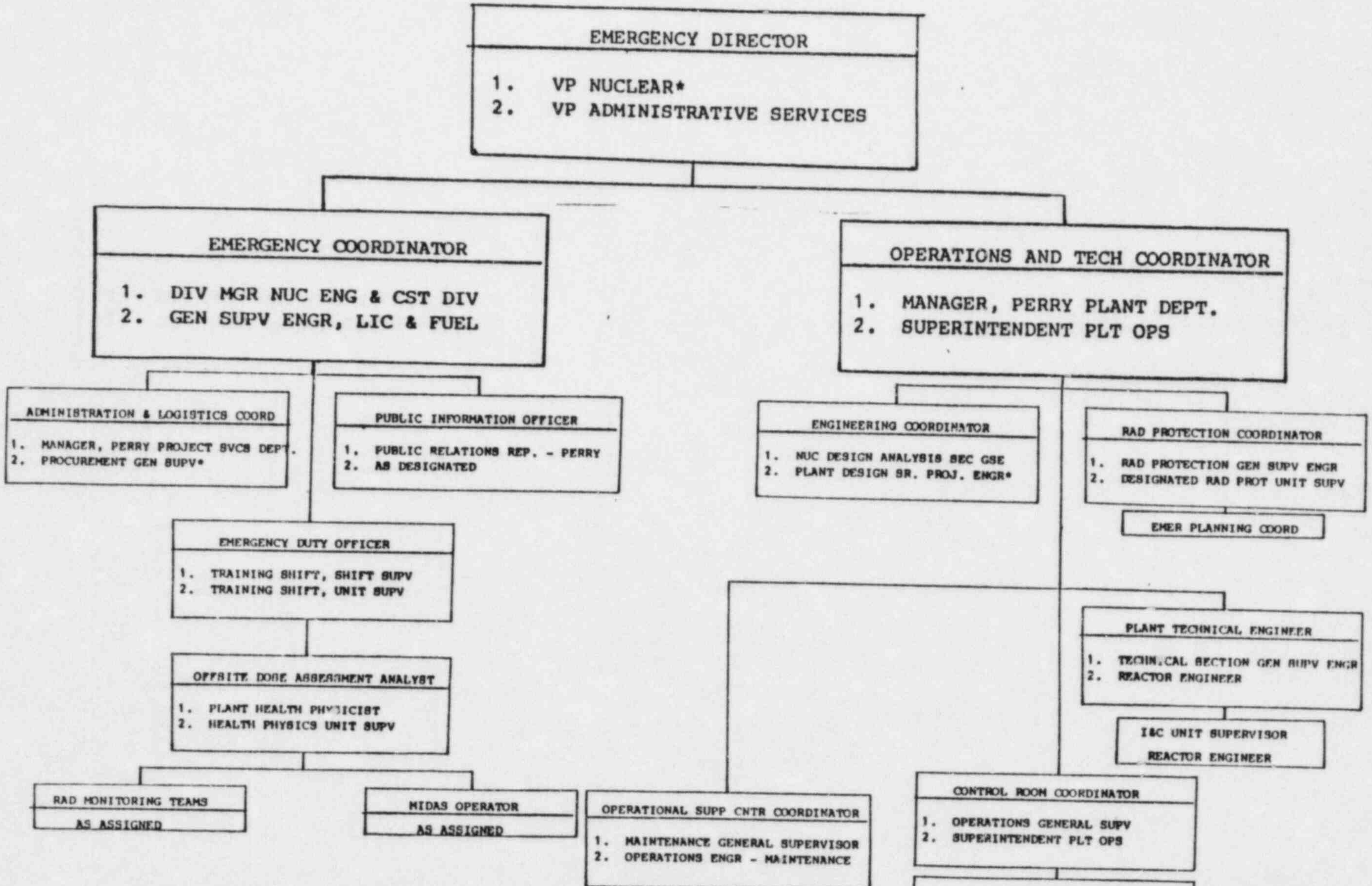

*POSITION TO BE NAMED 6 MONTHS PRIOR TO FUEL LOAD

FIGURE 5-3
EMERGENCY RESPONSE ORGANIZATION

2

EMERGENCY RESPONSE FACILITIES

- Technical Support Center
- Operational Support Center
- Emergency Operations Facility

TECHNICAL SUPPORT CENTER

Service Building Basement
 9500 sq ft

Close to Control Room
 within 2 minutes

Instrumentation
 SPDS & Dose Projection Capabilities

Communications
 Primary and Back up

Habitability
 Same as Control Room

OPERATIONAL SUPPORT CENTER

Control Complex Basement

Health Physics Area

Communications

Primary and Back up

Equipment

Readily Available

A-100

4

EMERGENCY OPERATIONS FACILITY

Training Center / EOF

8400 sq ft - EOF portion

Instrumentation

ERIS & DoseProjection Capabilities

Location

1/2 Mile from Control Room

Communications

Primary and Back up

Habitability

Filtered Ventilation

$\geq$ 5 for .7 MEV Gamma

A-102

STATE OF OHIO

- Comprehensive & Coordinated Response between all Political Units
- Covers All Nuclear Plants in Ohio
- Extension of other emergency activities
- Plan tested & found acceptable in 3 Full Scale Exercises
  - Davis - Besse
  - Zimmer

6

LOCAL GOVENMENTS

- Lake County
- Ashtabula County
- Geauga County

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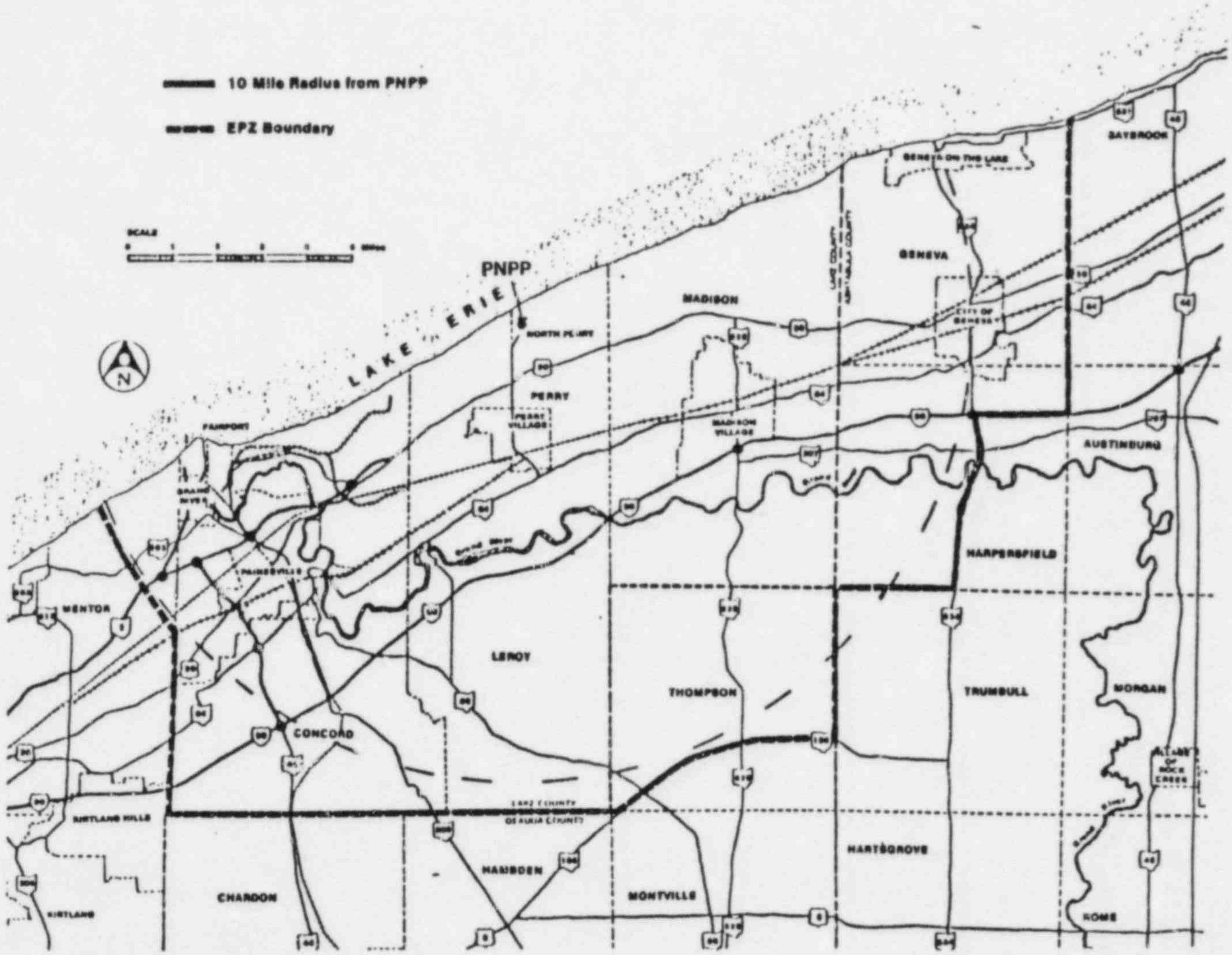
7

Emergency Planning Zone (EPZ) for PNPP



A-104

8

PROMPT PUBLIC ALERTING SYSTEM

SYSTEM DESIGN: MARCH 1982

ORDER EQUIPMENT: JUNE 1982

INSTALLATION: MARCH 1983

PRELIMINARY HUMAN FACTORS EFFORTS

— REDUCTION IN SIZE OF CONTROL BOARDS

— ATTENTION TO LAYOUT AND ORGANIZATION (INCLUDING ½ SIZE MOCK-UP)

— PROVIDE OPERATOR AIDES TO ASSIST IN DECISIONS

— MINIMIZE OPERATOR FATIGUE BY PROVIDING SIT-DOWN CONSOLES

— CONTROL ROOM ENVIRONMENT WHICH IS PLEASING AND ALLOWS FOR EFFICIENT COMPLETION OF TASKS.

— BWR 6 CONTROL ROOM OWNERS GROUP

GE EFFORTS RESULTED IN OPTION TO PGCC FOR SIT-DOWN OPERATOR CONSOLE.

— ANALYZED FOR HUMAN ENGINEERING WITH RESPECT TO ANTHROPOMETRICS AND FUNCTIONAL ALLOCATION.

— LAYOUT OF PANELS WAS BASED ON OPERATIONAL NEEDS.

POST TMI HUMAN FACTORS EFFORTS

- BWR OWNERS GROUP FORMED
- 1/80 SUBCOMMITTEE OF THE OG FORMED
  - REVIEW CONTROL ROOMS, AND
  - PROPOSE MODIFICATIONS BASED ON HED'S
  - CEI ACTIVE PARTICIPANT
- SUBCOMMITTEE DEVELOPED DETAILED CR CHECKLIST BASED ON HUMAN FACTORS CONSIDERATIONS.
- 9/81 OG SURVEY TEAM CONDUCTED DETAILED CR DESIGN REVIEW OF PERRY BASED ON CHECKLIST. (MEMBERS WERE OTHER UTILITY PERSONNEL AND HF CONSULTANTS).
  - SURVEY INCLUDES:

    OPERATOR INTERVIEWS

    PANEL LAYOUT

    CONTROL ROOM ENVIRONMENT

    TASK ANALYSIS

A-107

FAVORABLE ASPECTS OF CONTROL ROOM DESIGN

- GENERALLY CONFORM TO ANTHROPOMETRIC GUIDELINES

- DEMARCATION AND FUNCTIONAL GROUPINGS USED

- EXTENSIVE USE OF COLOR CODED MIMICS

- LABELS EASY TO READ AND IDENTIFY RELATED DEVICES

- COLOR CODING CONSISTENTLY APPLIED

- GOOD VISIBILITY OF CONTROL SURFACES

- ANNUNCIATORS GROUPED BY SYSTEM AND ABOVE RELATED CONTROLS AND DISPLAYS

- CRT DISPLAYS INCORPORATED INTO MAIN CONSOLE

- NORMAL ILLUMINATION GOOD

- CONSOLES AND ROOM DECOR PRESENT A HIGHLY PROFESSIONAL APPEARANCE

GENERAL RECOMMENDATIONS FOR ENHANCEMENTS

- ADDITIONAL FUNCTIONAL GROUPINGS COULD BE HELPFUL IN VERTICAL PANEL SECTIONS.
- SOME LABELS AND ANNUNCIATOR WINDOWS COULD BE WORDED MORE SUCCINCTLY OR ACCURATELY.
- MARK OR COLOR CODE DISPLAYS WITH NORMAL, ABNORMAL AND MARGINAL RANGES.
- USE LARGER INDICATORS FOR SOME PARAMETERS ON MAIN CONSOLE.
- USE ADDITIONAL CODING METHODS ON THE PANELS FOR CONTROLS DIFFERENTIATION.
- REVIEW ALARMS FOR FURTHER PRIORITIZATION.
- ANNUNCIATOR ENHANCEMENTS MAY INCREASE THE EFFECTIVENESS AS OPERATOR AID.
- EMERGENCY LIGHTING LEVEL IS BELOW RECOMMENDED.
- SOME INFORMATION POTENTIALLY USEFUL TO THE OPERATOR IS NOT AVAILABLE IN THE MAIN OPERATING AREA.
- SOME INSTRUMENT MODIFICATIONS MAY BE NECESSARY TO BETTER ENABLE THE OPERATOR TO EVALUATE THE STATE OF THE PLANT.

CURRENT STATUS RE: HUMAN FACTORS

— TASK FORCE FORMED TO EVALUATE HED'S

— 90% OF FINDINGS HAVE BEEN RESOLVED AND WILL BE IMPLEMENTED BEFORE FUEL LOAD. (MOST ARE "PAINT, LABEL, TAPE" FIXES).

— REMAINING 10% PRESENTLY BEING INVESTIGATED, AND

— PRELIMINARY DETAILED CONTROL ROOM DESIGN REVIEW HAS BEEN SUBMITTED TO NRC FOR THEIR REVIEW.

IN ADDITION:

— OG HAS RECEIVED COMMENTS FROM STAFF ON OG PROGRAM.

— NUREG 0700 HAS BEEN ISSUED.

— PERRY PRESENT PROGRAM INCLUDES MODIFICATION AND RE-REVIEW BASED ON DIFFERENCES BETWEEN OG AND NUREG 0700.

— NRC IS PLANNING ITS SURVEY FOR NEAR FUTURE.

— FINAL DCRDR WILL BE SUBMITTED 6 MONTHS PRIOR TO FUEL LOAD.

— AN ONGOING HF PROGRAM WILL BE IMPLEMENTED.

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Natural Resources Defense Council, Inc.

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202 223-8210

New York Office
122 EAST 42ND STREET
NEW YORK, N.Y. 10168
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415 421-6561

RECEIVED

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July 7, 1982

Dr. Paul Shewmon, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPENDIX XV
LTR TO P. SHEWMON FROM T. COCHRAN OF NATURAL RESOURCES DEFENSE COUNCIL, INC RE CLINCH RIVER BREEDER REACTOR

Dear Dr. Shewmon:

I understand that the full Advisory Committee on Reactor Safeguards (ACRS) is meeting tomorrow, July 8, 1982, to consider the suitability of the proposed site for the Clinch River Breeder Reactor (CRBR). I also am aware that the ACRS Subcommittee on CRBR has held several meetings this year* to discuss the CRBR licensing approach, core disruptive accidents, and the suitability of the proposed site. I have attended these meetings when possible and have reviewed the transcripts of each meeting.

As you may be aware, the Natural Resources Defense Council, Inc. (NRDC), is a principal intervenor in the CRBR licensing proceedings. Several of NRDC's contentions concern the suitability of the CRBR site and other safety issues under review by the ACRS Subcommittee on CRBR.

I am writing you to express my dismay over the inadequacy of the review to date of the CRBR licensing approach, CRBR design, and the proposed site by the ACRS Subcommittee. First, during eight meetings, the Subcommittee has invited only the Applicants and the NRC Staff to present their respective views on the CRBR safety and site issues. The Subcommittee has ignored completely the Intervenors in this matter. Not a single member of the Subcommittee has directly sought, even informally, the views of the Intervenors' experts regarding CRBR safety and site suitability issues, even though the Subcommittee is aware of at least some of Intervenors'

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*/ Feb. 2-3; March 30-31; May 4-5; May 24-25.

New England Office: 17 ERIE DRIVE · NATICK, MA. 01760 · 617 655-2656
Public Lands Institute: 1720 RACE STREET · DENVER, CO. 80206 · 303 377-9740

contentions in this case (Transcripts of ACRS CRBR Subcommittee meeting, March 31, 1982, pp. 123-124). Intervenors are in sharp disagreement with both the Staff and the the Applicants on several key issues under review by the ACRS, and the ACRS should be fully aware of all points of controversy before making a decision.

Second, it has become obvious that neither the Staff nor the Applicants are being completely candid with the ACRS CRBR Subcommittee. Neither party has informed the Subcommittee of the severe limitations that have been placed, at their request, upon the scope of the safety and site suitability reviews during the LWA-1 proceedings. I suggest that the Subcommittee and full ACRS review the transcript of the Atomic Safety and Licensing Board Prehearing Conference of April 5-6, 1982, and the depositions of the Staff and Applicants taken by NRDC in June 1982.* You will find the presentations made by the Applicants and Staff to the ACRS strikingly dissimilar to those made to the Licensing Board and the Intervenors.

A third impropriety concerns Dr. William E. Kastenberg, a consultant to the ACRS CRBR Subcommittee. Under contract to the Department of Energy, one of the Applicants in this licensing proceeding, Dr. Kastenberg prepared a report entitled "Anticipated Transients Without Scram for Light Water Reactors: Implications for Liquid Metal Breeder Reactors" (co-authored with Kenneth H. Solomon), RAND Note N-1188-DOE, July 1979. In this report, Dr. Kastenberg draws conclusions about the adequacy of the CRBR design which also bear directly on the suitability of the CRBR site. As a prior consultant to DOE on matters directly related to the CRBR, Dr. Kastenberg should not now be serving as an ACRS consultant on those same issues. I do not know Dr. Kastenberg and make no allegations concerning objectivity; yet I believe he should withdraw from the ACRS CRBR Subcommittee immediately to avoid any appearance of bias or impropriety.

Fourth, at the March 31, 1982, Subcommittee meeting, Dr. Carson Mark , an ACRS member whose opinions I respect but do not necessarily agree with, stated (Transcript, p. 124):

> ... it will be hilarious if the intervenors bring this up -- is [sic] the possibility of interrupting operations at K25, which they obviously would like to interrupt anyway. To raise that contention will really be great fun.

---

*/ Staff - May 6, 1982; Applicants - June 16, 21, 1982.

Had the ACRS shown Intervenors the courtesy of inviting our views on our contentions, I might be inclined to dismiss this statement as a little joke in bad taste but of no consequence. The fact that the ACRS continues to thumb its nose at Intervenors while making these remarks reflects a more serious problem; namely, that the ACRS displays a lack of independence and detachment necessary to function as an impartial reviewer of the CRBR.

I would be pleased to hear that you are taking steps to rectify this situation.

Sincerely,

Thomas B. Cochran, Ph.D.

CRBR PLANT SITE SUITABILITY REVIEW

- LWA-1s
- PROPOSED SITE PREPARATION ACTIVITIES
- APPROACH TO SITE SUITABILITY REVIEW
- SITE SUITABILITY REPORT

LWA-1s

- o ISSUANCE GOVERNED BY 10 CFR 50.10(E)
- o AUTHORIZES CONDUCT OF NON-SAFETY-RELATED SITE PREPARATION ACTIVITIES
- o REQUIRES COMPLETION OF ENVIRONMENTAL AND SITE SUITABILITY REVIEWS AND PUBLIC HEARINGS THEREON
- o ACTIVITIES UNDERTAKEN ENTIRELY AT RISK OF APPLICANTS
- o ISSUANCE HAS NO BEARING ON ISSUANCE OF CONSTRUCTION PERMIT
- o ISSUANCE REQUIRES FINDING THAT "...BASED UPON THE AVAILABLE INFORMATION AND REVIEW TO DATE, THERE IS REASONABLE ASSURANCE THAT THE PROPOSED SITE IS A SUITABLE LOCATION FOR A REACTOR OF THE GENERAL SIZE AND TYPE PROPOSED FROM THE STANDPOINT OF RADIOLOGICAL HEALTH AND SAFETY CONSIDERATIONS..." (10 CFR 50.10 (E)(2)
- o 27 ISSUED SINCE ESTABLISHED IN 1974

PROPOSED SITE PREPARATION ACTIVITIES

- GENERAL SITE CLEARING AND GRADING

  AREAS FOR ACCESS ROADS AND RAILROADS, TEMPORARY CONSTRUCTION FACILITIES, PARKING LOT, MAIN PLANT, COOLING TOWERS, SWITCHYARDS, STORAGE AREAS, ON-SITE QUARRY, RUNOFF TREATMENT PONDS, CONCRETE BATCHING AND MIXING PLANT AND BARGE UNLOADING FACILITY.

- EXCAVATION

  ACCESS ROADS AND RAILROADS, CONCRETE BATCHING AND MIXING PLANT, PARKING LOT, MAIN PLANT, COOLING TOWERS, SWITCHYARDS, STORAGE AREAS, TEMPORARY CONSTRUCTION FACILITIES AND BUILDINGS, RUNOFF TREATMENT PONDS AND QUARRY OPERATIONS.

- INSTALLATION OF TEMPORARY CONSTRUCTION FACILITIES

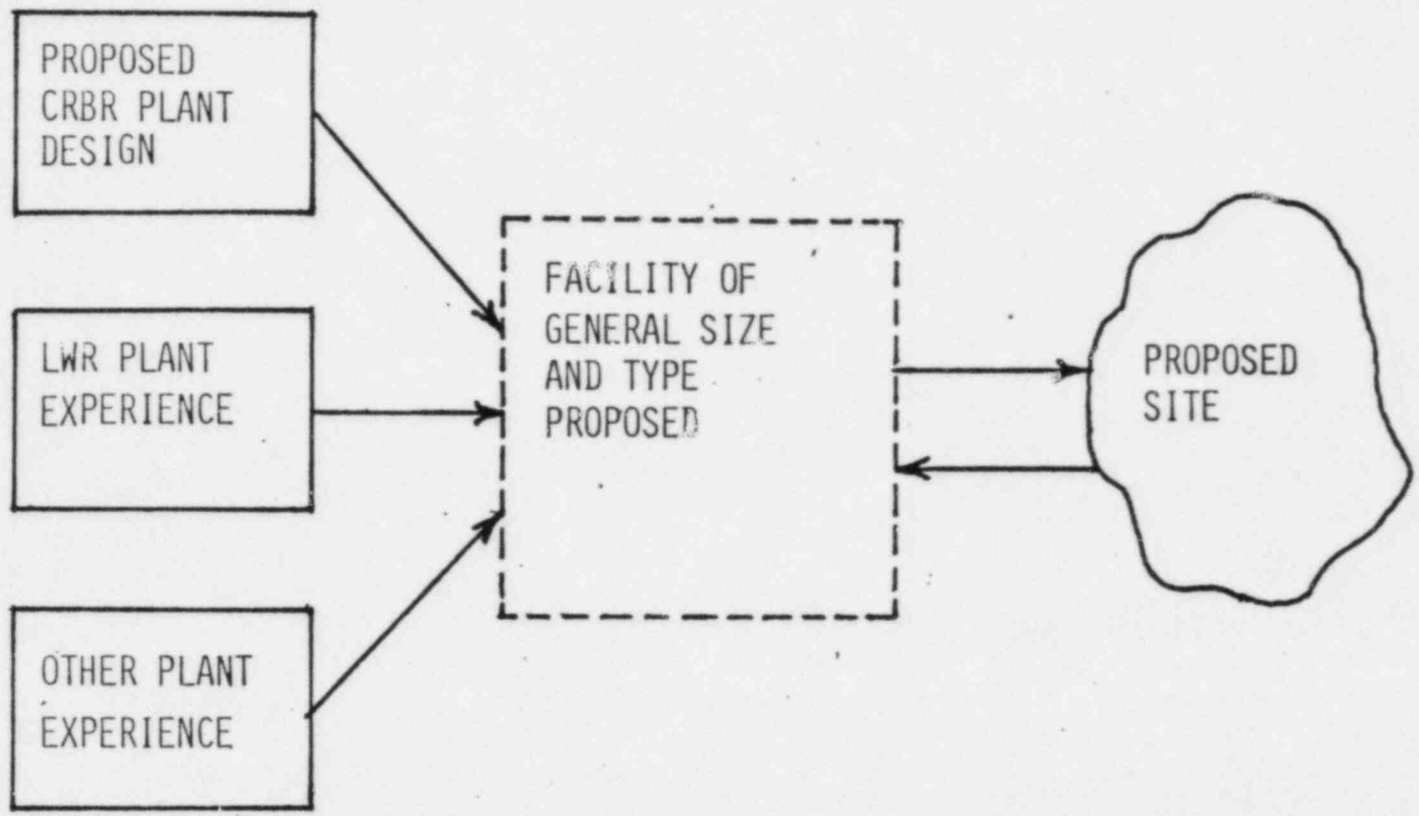
  TEMPORARY ONSITE ROADS, CONSTRUCTION PARKING AREAS, RAILROADS AND RAILROAD SPURS, CONTRACTOR WORK AND STORAGE AREAS, CONSTRUCTION UTILITIES, CONCRETE BATCHING AND MIXING PLANT, ONSITE QUARRY AND CRUSHING FACILITY SEWAGE TREATMENT PLANT AND CRAFT TOILET FACILITY, FIRE PROTECTION SYSTEM,

RUNOFF TREATMENT PONDS, STORM DRAINAGE SYSTEM, BARGE UNLOADING SYSTEM AND CONSTRUCTION BUILDINGS.

- OTHER ACTIVITIES

  PERMANENT ACCESS ROAD, RAILROAD SPUR, CONSTRUCTION PARKING AREA, TEMPORARY ROADS, CONTRACTOR WORK AND STORAGE AREAS, CONSTRUCTION UTILITIES, PERMANENT MAIN SURVEY CONTROL LINES AND BENCHMARKS AND QUARRY AND STOCKPILE AREAS.

APPROACH TO SITE SUITABILITY REVIEW



STEP 1: DEFINE CHARACTERISTICS OF FACILITY OF GENERAL SIZE AND TYPE PROPOSED RELEVANT TO SITE SUITABILITY.

STEP 2: DETERMINE CHARACTERISTICS OF PROPOSED SITE.

STEP 3: ASSESS COMPATIBILITY OF SITE AND FACILITY CHARACTERISTICS.

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SITE SUITABILITY REPORT

- o NUREG-0786 (UPDATES MARCH 1977 REPORT)
- o DOCUMENTS RESULTS OF STAFF'S EVALUATION OF SUITABILITY OF CLINCH RIVER SITE FOR FACILITY OF GENERAL SIZE AND TYPE AS PROPOSED CRBR PLANT.
- o CONCLUDES THAT BASED ON AVAILABLE INFORMATION AND REVIEW TO DATE, THERE IS REASONABLE ASSURANCE THAT THE CLINCH RIVER SITE IS A SUITABLE LOCATION FOR A FACILITY OF THE GENERAL SIZE AND TYPE AS THE PROPOSED CRBR PLANT FROM THE STANDPOINT OF RADIOLOGICAL HEALTH AND SAFETY CONSIDERATIONS.

CONTENTIONS RELATED TO SITE SUITABILITY REPORT

CONTENTION NO.	SUBJECT
1(A)* 2* 3(B)-(D)*	INCLUSION OF CDAs IN DBA SPECTRUM AND, HENCE, IN SITE SUITABILITY SOURCE TERM
5(A)**	ADEQUACY OF CLINCH RIVER SITE METEOROLOGY AND POPULATION DENSITY.
5(B)	LONG-TERM EVACUATION OF NEARBY FACILITIES
11(D)(1)	10 CFR 100.11 ORGAN DOSE EQUIVALENT LIMITS

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* LIMITED TO FEASIBILITY OF DESIGNING CRBR PLANT TO MAKE CDAs SUFFICIENTLY IMPROBABLE THAT THEY CAN BE EXCLUDED FROM DBA SPECTRUM

** CONTENTION MORE RELATED TO NEPA ALTERNATIVE SITE REVIEW probability that they may be excluded from the CRBR design bases.

(4) Applicants have not established that the test program used for their reliability program will be completed prior to Applicants' projected date for completion of construction of the CRBR.

2. The analyses of CDAs and their consequences by Applicants and Staff are inadequate for purposes of licensing the CRBR, performing the NEPA cost/benefit analysis, or demonstrating that the radiological source term for CRBRP would result in potential hazards not exceeded by those from any accident considered credible, as required by 10 CFR §100.1(a), fn. 1.

a) The radiological source term analysis used in CRBRP site suitability should be derived through a mechanistic analysis. Neither Applicants nor Staff have based the radiological source term on such an analysis.

b) The radiological source term analysis should be based on the assumption that CDAs (failure to scram with substantial core disruption) are credible accidents within the DBA envelope, should place an upper bound on the explosive potential of a CDA, and should then derive a conservative estimate of the fission product release from such an accident. Neither Applicants nor Staff have performed such an analysis.

APPENDIX I

ADMITTED AND RENUMBERED CONTENTIONS

1. The envelope of DBAs should include the CDA.

   a) Neither Applicants nor Staff have demonstrated through reliable data that the probability of anticipated transients without scram or other CDA initiators is sufficiently low to enable CDAs to be excluded from the envelope of DBAs.

   b) Neither Applicants nor Staff have established that Applicants' "reliability program" even if implemented is capable of eliminating CDAs as DBAs.

      (1) The methodology described in the PSAR places reliance upon fault tree and event tree analysis. Applicants have not established that it is possible to obtain sufficient failure mode data pertinent to CRBR systems to validly employ these techniques in predicting the probability of CDAs.

      (2) Applicants' projected data base to be used in the reliability program is inadequate. Applicants have not established that the projected data base encompasses all credible failure modes and human elements.

      (3) Even if all of the data described in Applicants' projected data base is obtained, Applicants have not establishd that CDAs have a sufficiently low

c) The radiological source term analysis has not adequately considered either the release of fission products and core materials, e.g. halogens, iodine and plutonium, or the environmental conditions in the reactor containment building created by the release of substantial quantities of sodium. Neither Applicants nor Staff have established the maximum credible sodium release following a CDA or included the environmental conditions caused by such a sodium release as part of the radiological source term pathway analysis.

d) Neither Applicants nor Staff have demonstrated that the design of the containment is adequate to reduce calculated offsite doses to an acceptable level.

e) As set forth in Contention 8(d), neither Applicants nor Staff have adequately calculated the guideline values for radiation doses from postulated CRBRP releases.

f) Applicants have not established that the computer models (including computer codes) referenced in Applicants' CDA safety analysis reports, including the PSAR, and referenced in the Staff CDA safety analyses are valid. The models and computer codes used in the PSAR and the Staff safety analyses of CDAs and their consequences have not been adequately documented, verified or validated by comparison with applicable experimental data. Applicants' and Staff's safety analyses do not establish that the models accurately represent the physical phenomena and principles which control the response of CRBR to CDAs.

g) Neither Applicants nor Staff have established that the input data and assumptions for the computer models and codes are adequately documented or verified.

h) Since neither Applicants nor Staff have established that the models, computer codes, input data and assumptions are adequately documented, verified and validated, they have also been unable to establish the energetics of a CDA and thus have also not established the adequacy of the containment of the source term for post accident radiological analysis.

3. Neither Applicants nor Staff have given sufficient attention to CRBR accidents other than the DBAs for the following reasons:

a) Neither Applicants nor Staff have done an adequate, comprehensive analysis comparable to the Reactor Safety Study ("Rasmussen Report") that could identify other CRBR accident possibilities of greater frequency or consequence than the accident scenarios analyzed by Applicants and Staff.

b) Neither Applicants' nor Staff's analyses of potential accident initiators, sequences, and events are sufficiently comprehensive to assure that analysis of the DBAs will envelop the entire spectrum of credible accident initiators, sequences, and events.

c) Accidents associated with core meltthrough following loss of core geometry and sodium-concrete interactions have not been adequately analyzed.

d) Neither Applicants nor Staff have adequately identified and analyzed the ways in which human error can initiate, exacerbate, or interfere with the mitigation of CRBR accidents.

4. Neither Applicants nor Staff adequately analyze the health and safety consequences of acts of sabotage, terrorism or theft directed against the CRBR or supporting facilities nor do they adequately analyze the programs to prevent such acts or disadvantages of any measures to be used to prevent such acts.

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a) Small quantities of plutonium can be converted into a nuclear bomb or plutonium dispersion device which if used could cause widespread death and destruction.

b) Plutonium in an easily usable form will be available in substantial quantities at the CRBR and at supporting fuel cycle facilities.

c) Analyses conducted by the Federal Government of the potential threat from terrorists, saboteurs and thieves demonstrate several credible scenarios which could result in plutonium diversion or releases of radiation (both purposeful and accidental) and against which no adequate safeguards have been proposed by Applicants or Staff.

d) Acts of sabotage or terrorism could be the initiating cause for CDAs or other severe CRBR accidents and the probability of such acts occurring has not been analyzed in predicting the probability of a CDA.

5. Neither Applicants nor Staff have established that the site selected for the CRBR provides adequate protection for public health and safety, the environment, national security, and national energy supplies; and an alternative site would be preferable for the following reasons:

   a) The site meteorology and population density are less favorable than most sites used for LWRs.

(1) The wind speed and inversion conditions at the Clinch River site are less favorable than most sites used for light-water reactors.

(2) The population density of the CRBR site is less favorable than that of several alternative sites.

(3) Alternative sites with more favorable meteorology and population characteristics have not been adequately identified and analyzed by Applicants and Staff. The analysis of alternative sites in the ER and the Staff Site Suitability Report gave insufficient weight to the meteorological and population disadvantages of the Clinch River site and did not attempt to identify a site or sites with more favorable characteristics.

b) Since the gaseous diffusion plant, other proposed energy fuel cycle facilities, the Y-12 plant and the Oak Ridge National Laboratory are in close proximity to the site an accident at the CRBR could result in the long term evacuation of those facilities. Long term evacuation of those facilities would result in unacceptable risks to the national security and the national energy supply.

6. The ER and FES do not include an adequate analysis of the environmental impact of the fuel cycle associated with the CRBR for the following reasons:

   a) The ER and FES estimate the environmental impacts of the fuel cycle based upon a scale-down of analyses presented in the LMFBR Program Environmental Statement and Supplement for a model LMFBR and fuel cycle. The analyses of the environmental impacts of the model LMFBR and fuel cycle in the LMFBR Program Statement and Supplement are based upon a series of faulty assumptions.

   b) The impacts of the actual fuel cycle associated with CRBR will differ from the model LMFBR and fuel cycle analyzed in the LMFBR Program Environmental Statement and Supplement. The analysis of fuel cycle impacts must be done for the particular circumstances applicable to the CRBR. The analyses of fuel cycle impacts in the ER and FES are inadequate since:

      (1) The impact of reprocessing of spent fuel and plutonium separation required for the CRBR is not included or is inadequately assessed;

(2) The impact of transportation of plutonium required for the CRBR is not included, or is inadequately assessed;

(3) The impact of disposal of wastes from the CRBR spent fuel is not included, or is inadequately assessed;

(4) The impact of an act of sabotage, terrorism or theft directed against the plutonium in the CRBR fuel cycle, including the plant, is not included or is inadequately assessed, nor is the impact of various measures intended to be used to prevent sabotage, theft or diversion.

7. Neither Applicants nor Staff have adequately analyzed the alternatives to the CRBR for the following reasons:

a) Neither Applicants nor Staff have adequately demonstrated that the CRBR as now planned will achieve the objectives established for it in the LMFBR Program Impact Statement and Supplement.

(1) It has not been established how the CRBR will achieve the objectives there listed in a timely fashion.

(2) In order to do this it must be shown that the specific design of the CRBR, particularly core design and engineering safety features, is sufficiently similar to a practical commercial size LMFBR that building and operating the CRBR will

demonstrate anything relevant with respect to an economic, reliable and licensable LMFBR.

(3) The CRBR is not reasonably likely to demonstrate the reliability, maintainability, economic feasibility, technical performance, environmental acceptability or safety of a relevant commercial LMFBR central station electric plant.

b) No adequate analysis has been made by Applicants or Staff to determine whether the informational requirements of the LMFBR program or of a demonstration-scale facility might be substantially better satisfied by alternative design features such as are embodied in certain foreign breeder reactors.

c) Alternative sites with more favorable environmental and safety features were not analyzed adequately and insufficient weight was given to environmental and safety values in site selection.

(1) Alternatives which were inadequately analyzed include Hanford Reservation, Idaho Reservation (INEL), Nevada Test Site, the TVA Hartsville and Yellow Creek sites, co-location with an LMFBR fuel reprocessing plant (e.g., the Development Reprocessing Plant), an LMFBR fuel fabricating plant, and underground sites.

8. The unavoidable adverse environmental effects associated with the decommissioning of the CRBR have not been adequately analyzed, and the costs (both internalized economic costs and external social costs) associated with the decommissioned CRBR are not adequately assessed in the NEPA benefit-cost balancing of the CRBR.

   a) There is no analysis of decommissioning in the Applicants' Environmental Report;

   b) Environmental Impact Statements (EIS) related to LWRs prepared by NRC have been inadequate due in part to recently discovered omissions (see below), and the FES for the CRBR is no different;

   c) A recent report "Decommissioning Nuclear Reactors" by S. Harwood; May, K.; Resnikoff, M.; Schlenger, B.; and Tames, P. (New York Public Interest Research Group (N.Y. PIRG), unpublished, January, 1976) indicates that (with the exception of the Elk River reactor) the isolation period following decommissioning of power reactors has been based on the time required for Co-60 to decay to safe levels. Harwood, et al. (p. 2) believe the previous analyses are in error because they have underestimated the significance of radionuclide, Ni-59. The time period for Ni-59 to decay to safe levels is estimated by Harwood, et al. (p. 2) for LWR to be at least 1.5 million years. The economic and societal implications of this 1.5 million year decay period are at present unknown.

d) Petitioner believes the NRC must systematically analyze all neutron activation products that may be produced in the proposed CRBR to determine the potential isolation period, following decommissioning, and then provide a comprehensive analysis of the costs (both economic and societal) of decommissioning.

9. Neither Applicants nor Staff have demonstrated that Applicants' plans for coping with emergencies are adequate to meet NRC requirements.

a) The PSAR contains insufficient information regarding Applicants' ability to identify the seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the Emergency Planning Zones.

b) Applicants and Staff have failed to account properly for local emergency response needs and capabilities in establishing boundaries for the plume exposure pathway and ingestion pathway EPZs for the CRBR.

c) The PSAR contains insufficient analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, nor does it note major impediments to the evacuation or taking of protective actions.

d) The PSAR contains insufficient information to ensure the compatibility of proposed emergency plans for both onsite areas and the EPZs, with facility design features, site layout, and site location.

e) The PSAR contains insufficient information concerning the procedures by which protective actions will be carried out, including authorization, notification, and instruction procedures for evacuations.

f) Applicants' proposed emergency plans fail to take into account the special measures necessary to cope with a CDA, including the need for increased protective, evacuation and monitoring measures, reduced response time and special protective action levels.

g) Applicants and Staff have failed to provide adequate assurance that the proposed emergency plans will meet the requirements and standards of 10 CFR §50.47(b).

10. Neither Applicants nor Staff have demonstrated that the facility will be provided with systems necessary to establish and maintain safe cold shutdown and maintain containment integrity that are capable of performing their functions during and after being exposed to the environmental conditions

    a) associated with postulated accidents, as required by General Design Criterion 4, 10 CFR Part 50, Appendix A; or

    b) created by sodium fires or the burning (or local detonation) of hydrogen.

11. The health and safety consequences to the public and plant employees which may occur if the CRBR merely complies with current NRC standards for radiation protection of the public health and safety have not been adequately analyzed by Applicants or Staff.

    a) Neither Applicants nor Staff have shown that exposures to the public and plant employees will be as low as practicable (reasonably achievable).

    b) Neither Applicants nor Staff have adequately assessed the genetic effects from radiation exposure including genetic effects to the general population from plant employee exposure.

c) Neither Applicants nor Staff have adequately assessed the induction of cancer from the exposure of plant employees and the public.

d) Guideline values for permissible organ doses used by Applicants and Staff have not been shown to have a valid basis.

(1) The approach utilized by Applicants and Staff in establishing 10 CFR §100.11 organ dose equivalent limits corresponding to a whole body dose of 25 rems is inappropriate because it fails to consider important organs, e.g., the liver, and because it fails to consider new knowledge, e.g., recommendations of the ICRP in Reports 26 and 30.

(2) Neither Applicants nor Staff have given adequate consideration to the plutonium "hot particle" hypothesis advanced by Arthur R. Tamplin and Thomas B. Cochran, or to the Karl Z. Morgan hypothesis described in "Suggested Reduction of Permissible Exposure to Plutonium and Other Transuranium Elements," <u>Journal of American Industrial Hygiene</u> (August 1975).

BASIS FOR STAFF's BELIEF THAT CRBR RISK WILL BE COMPARABLE TO LWR RISK

- CRBR WILL MEET ALL APPLICABLE LWR REGULATORY CRITERIA AND ADDITIONAL SPECIAL CRITERIA APPROPRIATE TO LMFBRs.
- CONSEQUENCES OF DBAs AND SSST WILL BE WITHIN 10 CFR 100 GUIDELINES.
- DESIGN MEASURES TO MAKE SEVERE ACCIDENTS (CDAs) VERY IMPROBABLE.
- DESIGN MEASURES TO ACCOMMODATE SEVERE ACCIDENTS (CDAs).
- PRELIMINARY EVALUATION OF ACCIDENT RISKS.
- PERFORMANCE OF PRA TO CONFIRM THAT CRBR MEETS SAFETY GOAL.

RISK COMPARABILITY
OF
CRBRP DESIGN
WITH LWR'S

<u>SIMILAR SOURCES AND CAUSES</u>

- RISK DOMINANT ACCIDENT SEQUENCES INVOLVE CORE
- CORE INVENTORIES ARE COMPARABLE PER MW (PLUTONIUM LARGER IN CRBRP)
- HEAT GENERATION <u>VS</u> HEAT REMOVAL IMBALANCE FOR FUEL DAMAGE TO OCCUR

<u>SIMILAR ACCIDENT TYPES</u>

- INTERNAL PLANT FAILURES
- EXTERNAL FORCES
- SABOTAGE

CORE DISRUPTION

INTERNAL PLANT FAILURE

LOCA

FLOW BLOCKAGE

LOHS

FAILED FUEL PROPAGATION

TRANSIENTS

PRIMARY COOLANT SYSTEM RESPONSE TO CORE DISRUPTION

MECHANICAL FAILURES - HEAD RELEASE

THERMAL FAILURES - RELEASE TO REACTOR CAVITY

CONTAINMENT RESPONSE

- CONTAINMENT ENVIRONMENTAL FACTORS AFFECTING FISSION PRODUCT BEHAVIOR AND EQUIPMENT OPERATION

  PRESSURE

  TEMPERATURE

  AIRBORNE MATERIALS

- CONTAINMENT FAILURE MODES

  FAILURE TO ISOLATE

  EARLY FILTERED VENTING

  OVERPRESSURE FAILURE

  PROMPT FAILURES

LOSS OF ALL OFF-SITE ELECTRIC POWER AT CRBRP

HEAT TRANSPORT SYSTEMS

- STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM
  STEAM DUMPING (SHORT TERM)
  PROTECTED AIR-COOLED CONDENSERS
- DIRECT HEAT REMOVAL SYSTEM

ELECTRICAL POWER

- TURBINE BYPASS SYSTEM
- DIESEL GENERATORS
- BATTERY POWER (SEVERAL HOURS)

CDA SEQUENCE CLASSES FOR SCOPING CRBR RISKS
FROM INTERNAL INITIATORS

INITIATION	PRIMARY SYSTEM FAILURE	CONTAINMENT FAILURE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	NONE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	OVERPRESSURE
GENERIC CORE DISRUPTION	SMALL HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION
GENERIC CORE DISRUPTION	LARGE HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION

RISK COMPARABILITY
OF
CRBRP DESIGN
WITH LWR'S

SIMILAR SOURCES AND CAUSES

- RISK DOMINANT ACCIDENT SEQUENCES INVOLVE CORE
- CORE INVENTORIES ARE COMPARABLE PER MW (PLUTONIUM LARGER IN CRBRP)
- HEAT GENERATION VS HEAT REMOVAL IMBALANCE FOR FUEL DAMAGE TO OCCUR

SIMILAR ACCIDENT TYPES

- INTERNAL PLANT FAILURES
- EXTERNAL FORCES
- SABOTAGE

Rumble
T4

CORE DISRUPTION

INTERNAL PLANT FAILURE

LOCA

FLOW BLOCKAGE

LOHS

FAILED FUEL PROPAGATION

TRANSIENTS

PRIMARY COOLANT SYSTEM RESPONSE TO CORE DISRUPTION

MECHANICAL FAILURES - HEAD RELEASE

THERMAL FAILURES - RELEASE TO REACTOR CAVITY

CONTAINMENT RESPONSE

- CONTAINMENT ENVIRONMENTAL FACTORS AFFECTING FISSION PRODUCT BEHAVIOR AND EQUIPMENT OPERATION

  PRESSURE

  TEMPERATURE

  AIRBORNE MATERIALS

- CONTAINMENT FAILURE MODES

  FAILURE TO ISOLATE

  EARLY FILTERED VENTING

  OVERPRESSURE FAILURE

  PROMPT FAILURES

LOSS OF ALL OFF-SITE ELECTRIC POWER AT CRBRP

HEAT TRANSPORT SYSTEMS

- STEAM GENERATOR AUXILIARY HEAT REMOVAL SYSTEM
  STEAM DUMPING (SHORT TERM)
  PROTECTED AIR-COOLED CONDENSERS
- DIRECT HEAT REMOVAL SYSTEM

ELECTRICAL POWER

- TURBINE BYPASS SYSTEM
- DIESEL GENERATORS
- BATTERY POWER (SEVERAL HOURS)

CDA SEQUENCE CLASSES FOR SCOPING CRBR RISKS FROM INTERNAL INITIATORS

INITIATION	PRIMARY SYSTEM FAILURE	CONTAINMENT FAILURE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	NONE
GENERIC CORE DISRUPTION	SMALL OR LARGE HEAD RELEASE & THERMAL FAILURE	OVERPRESSURE
GENERIC CORE DISRUPTION	SMALL HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION
GENERIC CORE DISRUPTION	LARGE HEAD RELEASE & THERMAL FAILURE	CONTAINMENT ISOLATION

A SCOPING COMPARISON OF SEVERE ACCIDENT RISKS DUE TO CRBRP WITH COMPARABLE SIZE LWRs AT CRBRP SITE

- USED CRAC CODE TO PERFORM THE CALCULATIONS TO GAIN A PERSPECTIVE OF RELATIVE RISKS OF CRBRP AND LWRs.

- THE CRBRP ACCIDENT SEQUENCES, PROBABILITIES, AND RELEASE FRACTIONS WERE BASED ON SCOPING ESTIMATES DESCRIBED TO YOU BY ED RUMBLE.

- THE BWR AND THE PWR ACCIDENT SEQUENCES, PROBABILITIES, AND RELEASE FRACTIONS WERE THE SAME AS USED IN OUR ACCIDENT EVALUATIONS FOR ENVIRONMENTAL STATEMENTS. (RSS REBASE-LINE)

  THE CORE INVENTORIES CORRESPONDED TO THE POWER LEVEL OF 1121 MWT. (INCLUDING THE CONSIDERATION OF THE DIFFERENCES IN CRBRP AND LWR CORES).

- FOR THIS COMPARISON WE USED THE CRBRP SITE CHARACTERISTICS (METEOROLOGY, POPULATION DISTRIBUTION, ETC.)

CONCLUSIONS OF THE COMPARISON

- BASED ON THE PRELIMINARY SCOPING ANALYSIS THE STAFF FINDS THAT THE CRBRP RISKS WILL NOT EXCEED THE RISKS FROM COMPARABLE LWRs.

- FURTHER WORK ON A FULL PRA IS IN PROGRESS AND WILL ESTABLISH BETTER ESTIMATES OF PROBABILITIES AND RELEASES AS DISCUSSED BY ED RUMBLE.

CRBR DOSE GUIDELINES

	LWR*		CRBR**	
	CP STAGE DOSE (REM)	OL STAGE DOSE (REM)	CP STAGE DOSE (REM)	OL STAGE DOSE (REM)
THYROID	150	300	150	300
WHOLE BODY	20	25	20	25
BONE SURFACE	-	-	150	300
RED BONE MARROW	-	-	37.5	75
LUNG	-	-	37.5	75
LIVER	-	-	75	150

ADDITIONAL GUIDELINES

Mortality risk equivalent whole body dose from any postulated design basis accident (on a calculated dose basis) should be no greater than the mortality risk equivalent whole body dose value of 10 CFR Part 100 for an LWR (i.e., 34 rem whole body risk equivalent at the O.L. stage, and 24.5 rem whole body risk equivalent at the CP stage).

*BASIS: 10 CFR PA T 100

**BASIS: SAME AS LWR FOR THYROID AND WHOLE BODY. THE LUNG AND BONE DOSES ARE BASED ON THE CRITICAL ORGAN CONCEPT.

Site Suitability Source Term Release from Core†

RADIOACTIVE SPECIES	LWR* SOURCE TERM	CRBR** SOURCE TERM
NOBLE GASES	100%	100%
HALOGENS	50%	50%
SOLIDS	1%	1%
PLUTONIUM	-	1%

* BASIS: TID 14844 NON-MECHANISTIC SOURCE TERM (i.e., SEQUENCE OF EVENTS NOT TAKEN INTO ACCOUNT.

** BASIS: SAME BASIS AS FOR LWR SOURCE TERM WITH INCLUSION OF PLUTONIUM

† FISSION PRODUCTS ARE ASSUMED TO BE RELEASED FROM THE CORE TO THE PRIMARY CONTAINMENT. THE ASSUMPTION IS THAT THE SOURCE TERM FISSION PRODUCTS ARE INSTANTANEOUSLY RELEASED TO AND UNIFORMLY DISTRIBUTED THROUGHOUT THE PRIMARY CONTAINMENT AND AVAILABLE FOR RELEASE TO THE ENVIRONMENT (EXCEPT IN THE CASE OF THE IODINES IT IS ASSUMED THAT ONE-HALF OF THE IODINES RELEASED ARE INSTANTANEOUSLY PLATED OUT AND THE REMAINDER IS UNIFORMLY DISTRIBUTED THROUGHOUT THE PRIMARY CONTAINMENT AND AVAILABLE FOR RELEASE TO THE ENVIRONMENT).

SITE SUITABILITY SOURCE TERM ASSUMPTIONS AND DOSE RESULTS

Power Level	1121 MWt
Core Fraction Released to Containment:	
Noble Gases	100%
Iodines	50%
Solid Fission Products	1%
Plutonium	1%
Primary Containment Free Volume	$3.7 \times 10^6 ft^3$
Primary Containment Leak Rate	0.1%/day
Bypass Fraction	0.001%/day
Annulus Filtration System Filter Efficiencies:	
Particulate Iodine, Solids and Plutonium	99%
Elemental and Organic Iodine	95%
Annulus Filtration System Flow Rates, cfm:	
Exhaust	3000
Recirculation	11000
Aerosol Fallout Coefficients in Containment, hr^1	
0-2 hours	.0853
2-8 hours	.0659
8-24 hours	.0571
Minimum Exclusion Area Boundary Distance	670 meters
Low Population Zone	4023 meters
Atmospheric Dispersion Parameters (5% meteorology), sec/m^3	
0-2 hours at exclusion area boundary	1.22×10^{-3}
0-8 hours at LPZ	1.2×10^{-4}
8-24 hours at LPZ	8.4×10^{-5}
24-96 hours at LPZ	3.9×10^{-5}
96-720 hours at LPZ	1.4×10^{-5}

Dose Consequences, rem

	Exclusion Area	Low Population Zone
Thyroid	12	7
Whole Body	0.6	0.3
Lung	0.4	0.4
Bone Surface	10	9
Red Bone Marrow	2.4	2.1
Liver	1.1	1.0
Mortality Risk Equivalent Whole Body	1.7	1.1

CRBRP SITE SUITABILITY

BRIEFING FOR:

ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS (ACRS)

SITE SUITABILITY SOURCE TERMS AND NON-RADIOLOGICAL EFFECTS OF SODIUM REACTION PRODUCTS AEROSOLS

A-152

PRESENTED BY:
GEORGE H. CLARE
LICENSING MANAGER, CRBRP PROJECT
WESTINGHOUSE
ADVANCED REACTORS DIVISION
JULY 9, 1982

APPENDIX XXI
LMFBR SOURCE TERMS USED IN OTHER COUNTRIES

THE CRBRP SITE SUITABILITY SOURCE TERM IS COMPARABLE TO THAT USED FOR SITING FOREIGN LMFBRs

	PERCENT RELEASED FROM PRIMARY COOLANT BOUNDARY		
	CRBRP (USA)	CDFR (UK)	MONJU (JAPAN)
• NOBLE GASES	100	100	100
• HALOGENS (AIRBORNE)	50 (25)	50	10
• SOLIDS	1	1	1
• FUEL	1	1	1

NO EQUIVALENT TO THE SSST IS KNOWN TO BE USED IN FRANCE OR GERMANY (FRG).

A-153

7-82-2898-8

THE NON-RADIOLOGICAL EFFECTS OF SODIUM REACTION PRODUCT AEROSOLS HAVE BEEN CONSIDERED

- $Na + O_2 \rightarrow NaO_x$
  $NaO_x + H_2O \rightarrow NaOH\ (+ O_2)$
  $NaOH + CO_2 \rightarrow Na_2CO_3\ (+ H_2O)$
- EFFECTS ON SAFETY RELATED EQUIPMENT ARE ADDRESSED
  - ENVIRONMENTAL QUALIFICATION
  - CONTROL ROOM
  - AEROSOL MITIGATION FEATURES

A-154

7-82-2898-6

ANY OFFSITE CONCENTRATION OF SODIUM REACTION PRODUCT AEROSOLS WILL BE LOW

A-155

ASSUME:

- STEAM GENERATOR BUILDING DESIGN BASIS LEAK
- 100% OF SPRAY REACTION PRODUCTS AIRBORNE
- ONLY ESF MITIGATION IS EFFECTIVE

EVALUATION:

- DEPLETION IN THE SGB; HAA-3 (440 LB/5 MIN)
- 50% METEOROLOGY; 1×10^{-3} SEC/m^3
- DEPLETION DURING TRANSPORT; 1/100

RESULTS: 7 MILLIGRAMS (NaOH) PER CUBIC METER

7-82-2898-7

CRBRP SITE SUITABILITY

CRBRP PROJECT

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

SITE DESCRIPTION

A-15C

PRESENTED BY:

HENRY B. PIPER
PUBLIC SAFETY
CRBRP PROJECT OFFICE

APPENDIX XXII
SITE DESCRIPTION

JULY 9, 1982

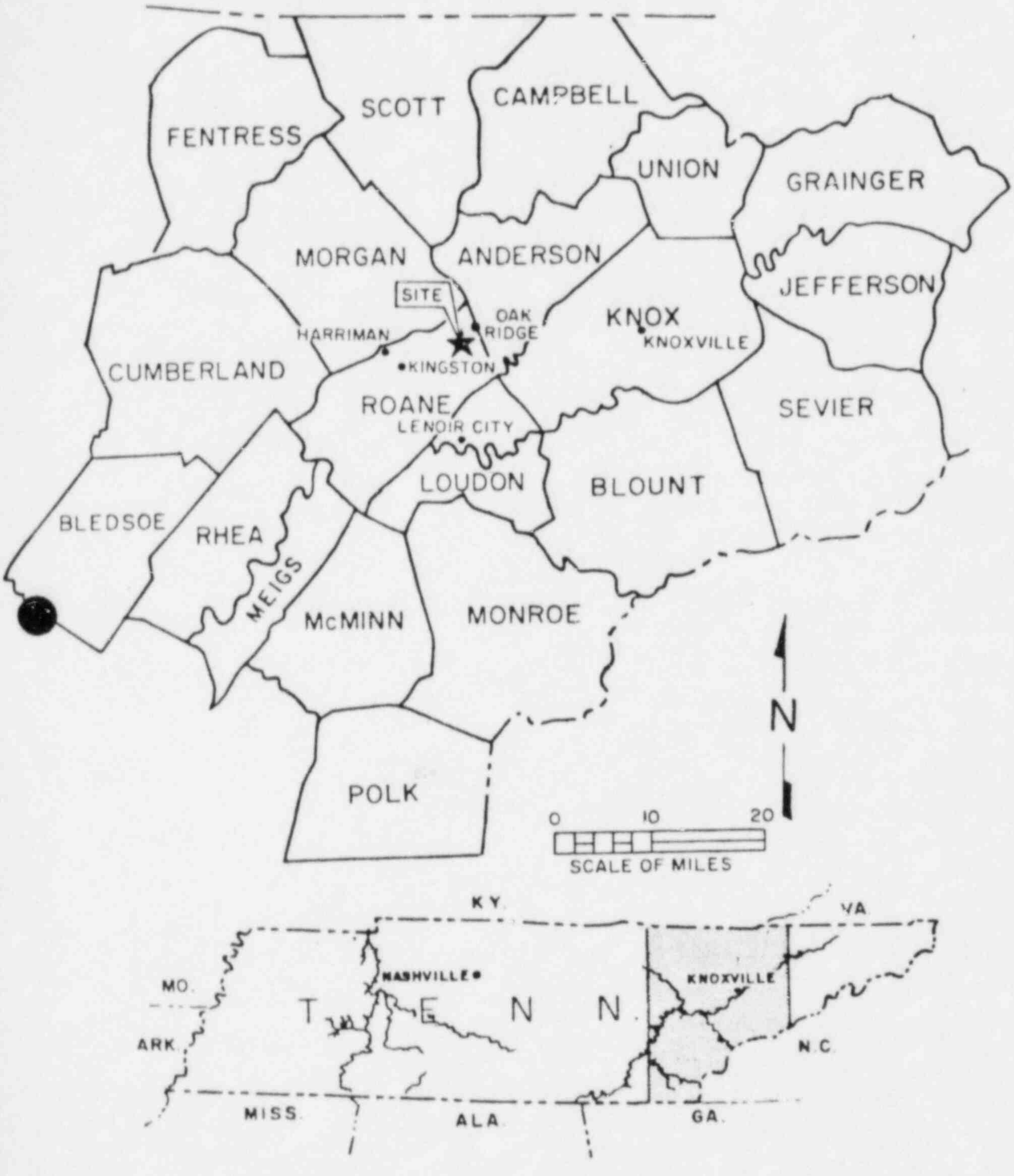

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Figure 2.1-1 LOCATION OF CLINCH RIVER SITE IN RELATION TO COUNTIES AND STATE

A-157

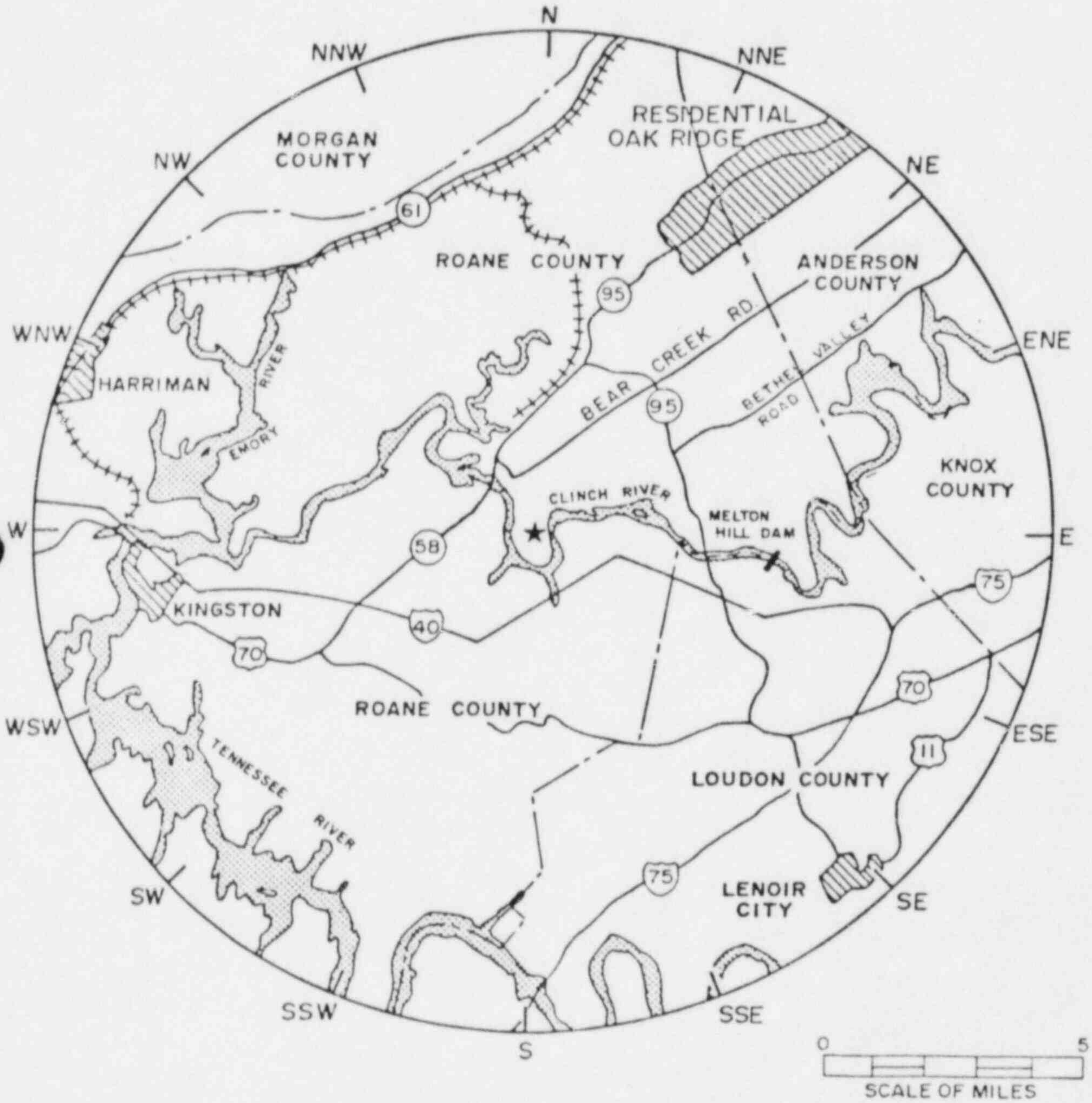

Figure 2.2-4 URBAN AREAS WITHIN 10 MILES OF THE CRBRP SITE.

10

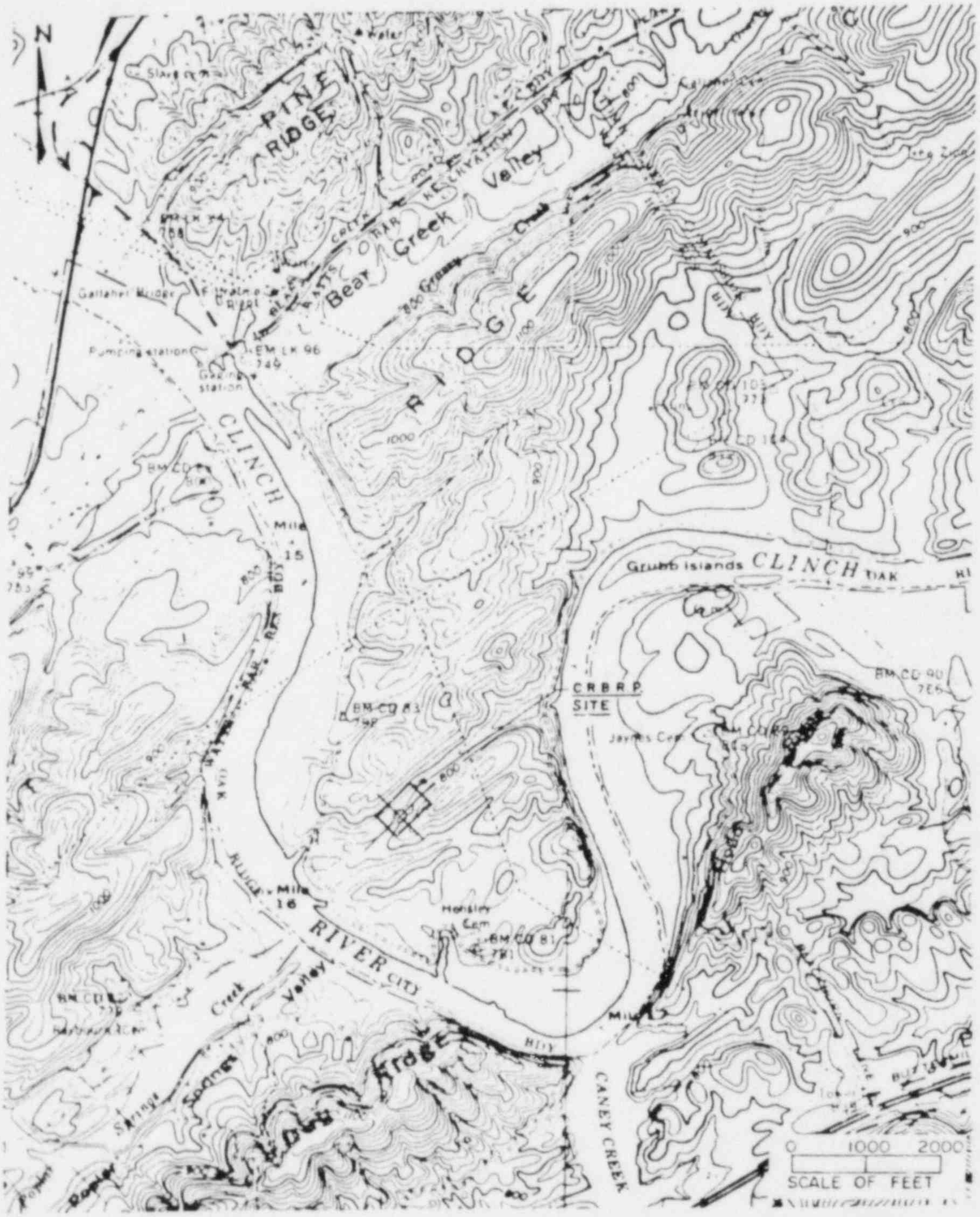

Figure 2.1-5 TOPOGRAPHY OF THE CRBRP SITE

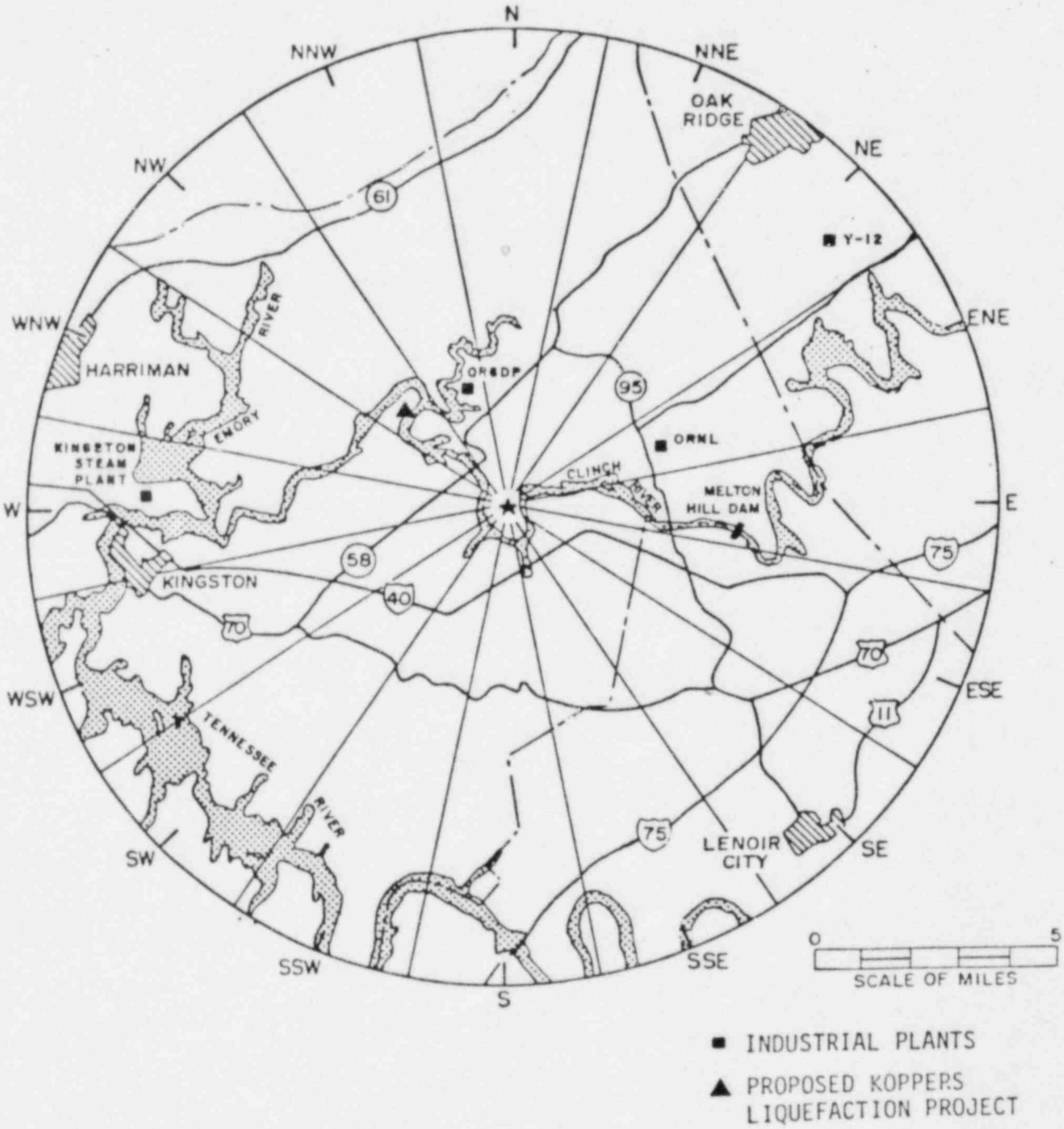

Figure 2.2-10 INDUSTRIAL PLANTS WITHIN 10-MILE RADIUS OF THE CRBRP SITE.

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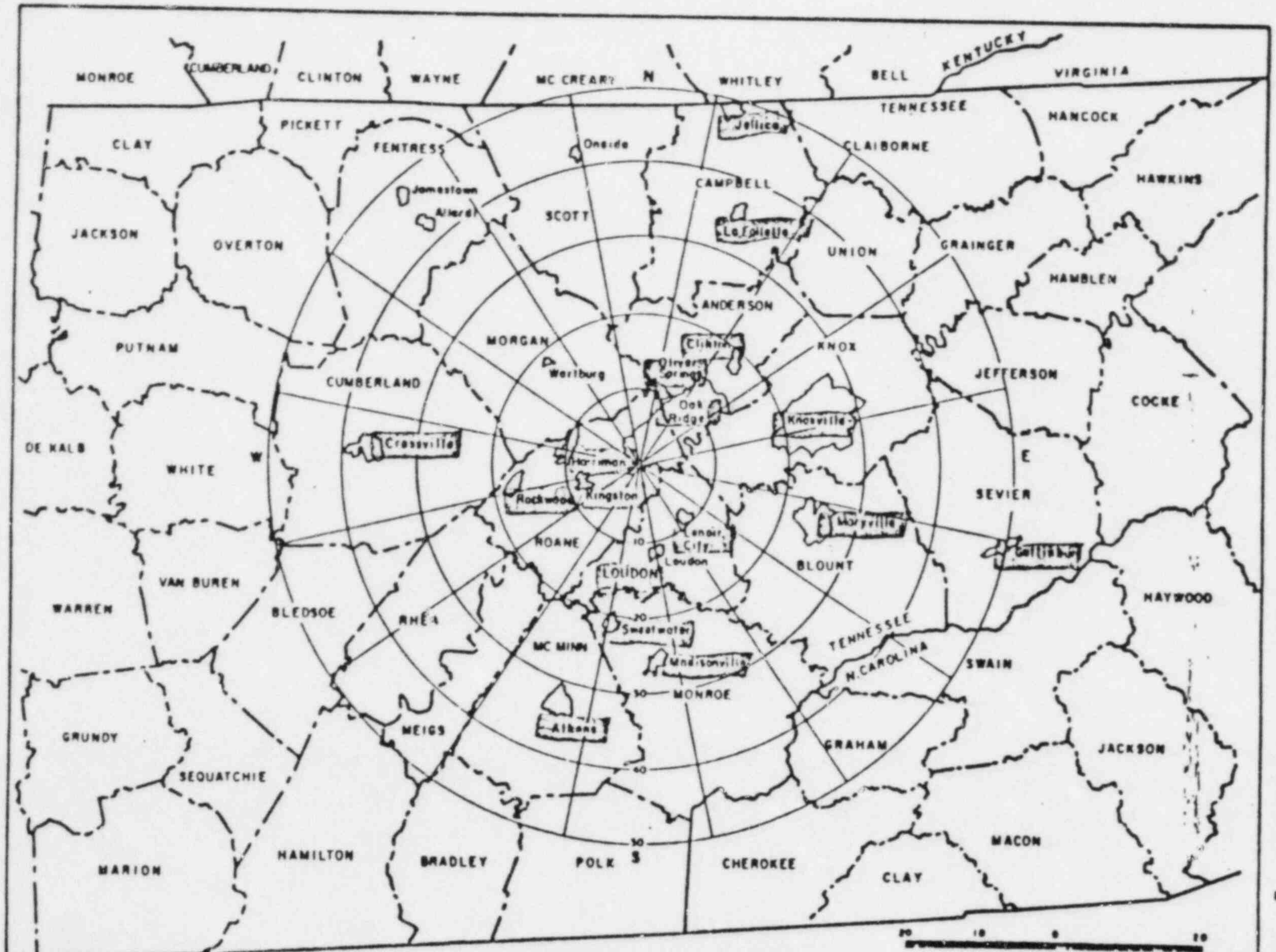
CIRCLE AND SECTOR FIGURE AND STUDY AREA WITHIN
50-MILES OF CRBRP SITE



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A-161

1980 RESIDENT POPULATION DISTRIBUTION

0 TO 10 MILES FROM THE CRBRP SITE

Direction	Distance (miles) 0 to 1	1 to 2	2 to 3	3 to 4	4 to 5	5 to 10	10-mile Total
N	0	0	0	0	0	2,000	2,000
NNE	0	0	0	0	0	4,400	4,400
NE	0	0	0	0	0	4,500	4,500
ENE	10	10	0	0	0	3,900	3,920
E	20	30	50	10	20	4,300	4,430
ESE	20	30	50	140	120	2,300	2,660
SE	0	20	50	140	110	7,200	7,520
SSE	0	30	40	90	320	2,000	2,480
S	0	50	50	120	160	1,100	1,480
SSW	10	30	50	80	90	800	1,060
SW	20	80	80	110	140	700	1,130
WSW	20	70	80	140	340	2,800	3,450
W	0	130	100	110	500	4,400	5,240
WNW	10	80	170	10	60	4,400	4,730
NW	30	30	0	10	40	1,700	1,810
NNW	10	0	0	0	120	1,100	1,230
Total	150	590	720	960	2,020	47,600	52,040
Cumulative Total	150	740	1,460	2,420	4,440	52,040	

10

A-162

1980 RESIDENT POPULATION DISTRIBUTION

10 TO 50 MILES FROM THE CRBRP SITE

Direction	Distance (miles) 10-mile Total	10 to 20	20 to 30	30 to 40	40 to 50	50-mile Total
N	2,000	4,700	2,200	6,400	7,000	22,300
NNE	4,400	9,100	6,300	17,500	10,300	47,600
NE	4,500	22,100	10,900	6,200	5,100	48,800
ENE	3,920	22,100	100,900	41,800	12,800	181,520
E	4,430	34,400	102,600	34,600	21,300	197,330
ESE	2,660	9,600	43,100	7,000	4,800	67,160
SE	7,520	5,300	6,300	3,700	2,300	25,120
SSE	2,480	5,200	2,400	4,100	6,500	20,680
S	1,480	5,600	7,200	6,200	5,500	25,980
SSW	1,060	3,300	11,200	22,800	9,900	48,260
SW	1,130	2,200	3,600	6,000	10,500	23,430
WSW	3,450	3,400	5,000	6,500	7,200	25,550
W	5,240	12,300	2,600	11,100	4,800	36,040
WNW	4,730	7,800	3,100	5,500	4,500	25,630
NW	1,810	2,400	2,100	3,900	7,900	18,110
NNW	1,230	3,800	1,700	5,600	5,000	17,330
Total	52,040	153,300	311,200	188,900	125,400	830,840
Cumulative Total	52,040	205,340	516,540	705,440	830,840	

10

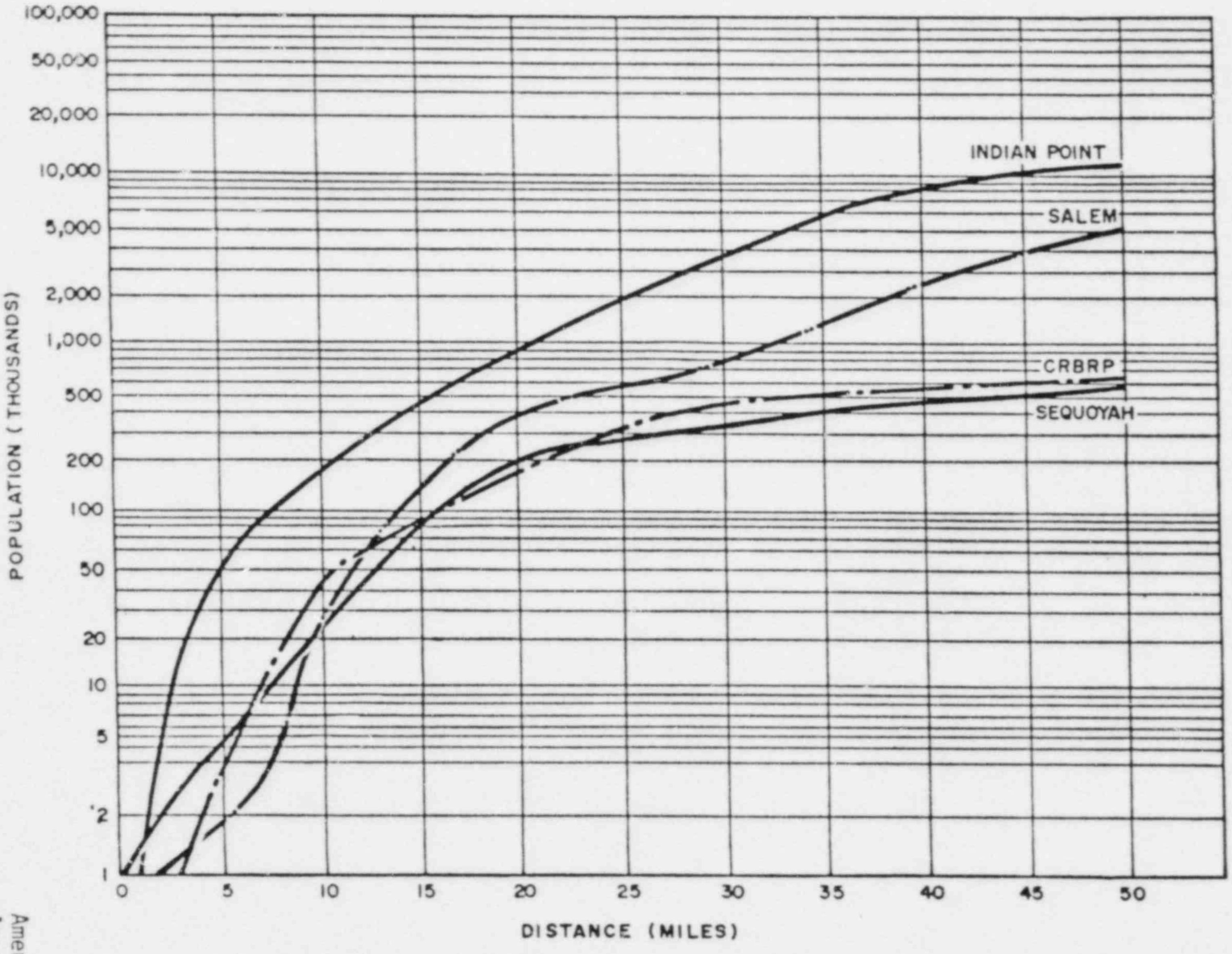

Figure 2.1-6

COMPARATIVE POPULATION DISTRIBUTION SURROUNDING NUCLEAR PLANT SITES

Amend. 15
Apr. 1976

A-164

EVENTS ANALYZED IN

DESIGN BASIS FLOOD DETERMINATION

PROBABLE MAXIMUM FLOOD

SEISMIC FAILURE

OBE CONCURRENT WITH ½ PMF

SSE CONCURRENT WITH 25-YEAR FLOOD

PROBABLE MAXIMUM PRECIPITATION

RAINFALL DEPTH (FOR A PARTICULAR SIZE BASIN) THAT APPROACHES THE UPPER LIMIT THAT THE PRESENT CLIMATE CAN PRODUCE.

PMP - CRBR

9 DAY STORM

*3-DAY ANTECEDENT STORM -	6.8 INCHES
*3-DAY DRY PERIOD -	0
*3-DAY MAIN STORM -	17.2 INCHES
*TOTAL	24.0 INCHES

*AVERAGE ON WATERSHED ABOVE WATTS BAR

FLOOD ELEVATIONS

PLANT GRADE ELEVATION = 815

EVENT	CRBR ELEVATION	
	MILE 16	MILE 18
PMF	776.0	777.5
OBE FAILURE WITH ½ PMF	798.2	804.3
SSE FAILURE WITH 25-YR. FLOOD	790.5	796.3

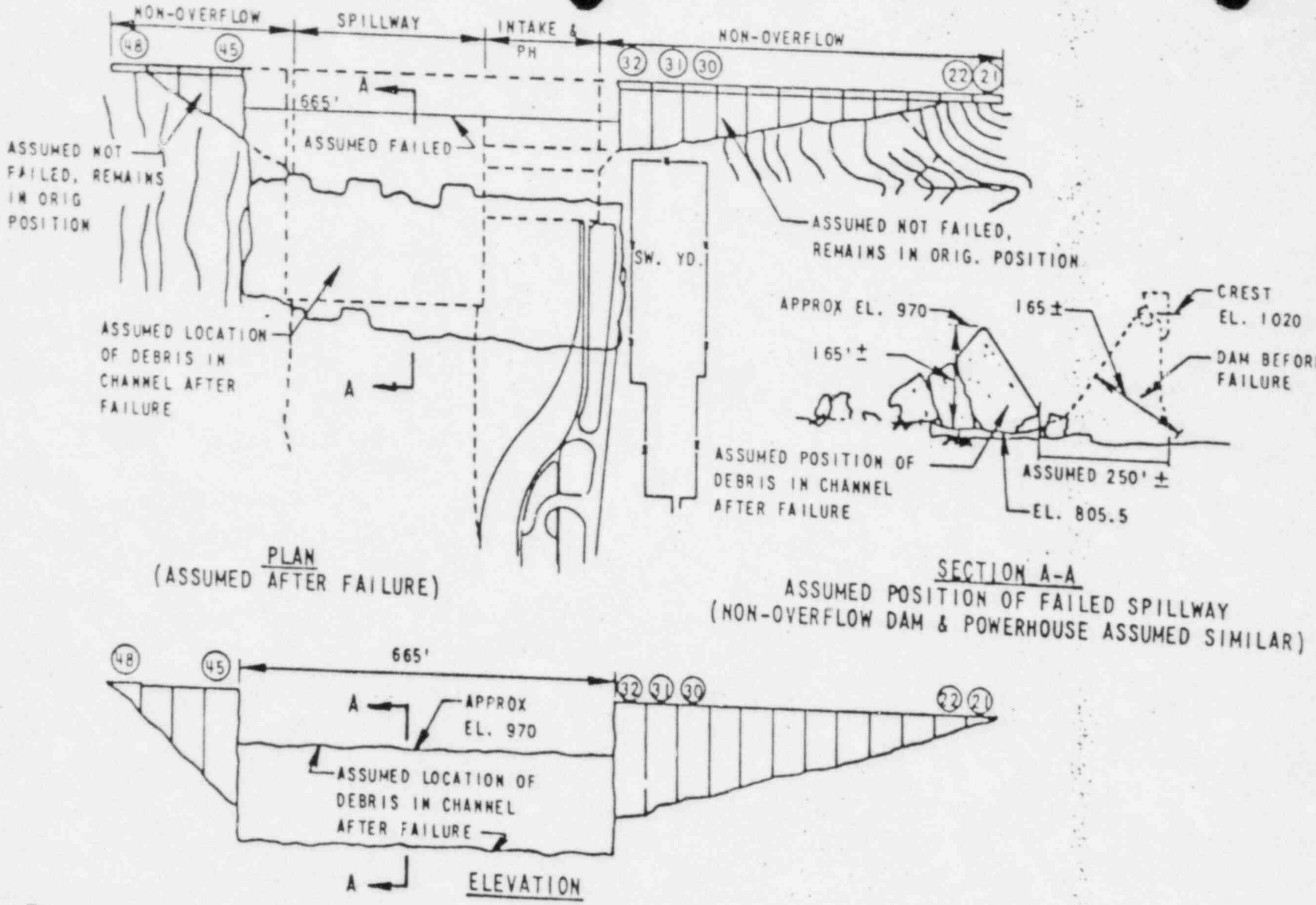

Figure 2.4-30 Norris Dam - Analysis for OBE & One Half PMF-Assumed Condition of Dam After Failure

SENSITIVITY RUNS

POSTULATED FAILURE MODE	CRBR ELEVATION MILE 16	CRBR ELEVATION MILE 18
OBE CONDITIONS WITH ½ PMF		
INSTANT VANISHMENT OF ENTIRE DAM (NO DEBRIS)	811.0	818.0
VANISHMENT OF THREE BLOCKS (38-40) TO GROUND LEVEL	802.2	808.4
OVERTURNING OF BLOCKS 33-44 (665-FOOT WIDTH) WITH 945 DEBRIS LEVEL	802.6	808.9
OVERTURNING OF BLOCKS 37-43 (370-FOOT WIDTH) WITH 925 DEBRIS LEVEL	805.3	811.9

CRBRP PROJECT

CRBRP SITE SUITABILITY

BRIEFING FOR:

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

HYDROLOGY

A-171

- **PROBABLE MAXIMUM FLOOD** — **R. LEE, TVA**
- **IMPACT OF NORRIS DAM SITE** — **T. J. ABRAHAM, TVA**
- **EFFECT OF CORE MELT ON GROUNDWATER** — **H. B. PIPER, CRBRP/PO**

JULY 9, 1982

LIQUID PATHWAYS EVALUATION

RADIOLOGICAL CONSEQUENCES ASSOCIATED WITH RELEASE OF RADIOACTIVE MATERIAL TO THE GROUNDWATER FOLLOWING AN HCDA HAVE BEEN EVALUATED IN:

A-172

- CRBRP-3, VOLUME 2, "HYPOTHETICAL CORE DISRUPTIVE ACCIDENT CONSIDERATIONS IN CRBRP; ASSESSMENT OF THERMAL MARGIN BEYOND THE DESIGN BASE"
- ER QUESTION/RESPONSE E240.2R

7-82-2898-2

ANALYSIS OF MELTED-FUEL-MASS LEACH

CRBRP LIQUID PATHWAY ANALYSES SIMILAR TO WASH-1400, WITH THE FOLLOWING EXCEPTIONS:

- CRBRP SITE SPECIFIC FLOW SYSTEM DATA WAS USED
- NO WATER WAS ASSUMED TO BE AVAILABLE FROM THE REACTOR CONTAINMENT VESSEL TO ADD TO GROUNDWATER AT MELT-THROUGH

COMPARISON OF CALCULATED GROUNDWATER EFFLUENT CONCENTRATIONS FOR MOST SIGNIFICANT ISOTOPES AT ENTRANCE TO CLINCH RIVER

	CRBRP		LWR		
NUCLIDE	CONCENT. (μci/cc)	TIME OF PEAK (YRS)	CONCENT. (μci/cc)	TIME OF PEAK (YRS)	MPC (10 CFR 20)
• Sr-90	3.6×10^{-9}	336	7.1×10^{-4}	5.9	3×10^{-7}
• Tc-99	6.8×10^{-8}	45	3.6×10^{-6}	.9	2×10^{-4}
• Pu-239	7.1×10^{-7}	3580	8.0×10^{-7}	535	5×10^{-6}

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7-82 2898-4

CRBRP/NRC LIQUID PATHWAY GENERIC STUDY (NUREG-0440) COMPARISON

- CRBRP CONTAINED RADIONUCLIDE SOURCE SIGNIFICANTLY LESS THAN SOURCE USED IN NUREG-0440
  - GENERALLY 2 TO 40 TIMES LESS
- SITE SPECIFIC PARAMETERS ARE SIMILAR.
  - NUREG-0440 USED CLINCH-TENNESSEE-OHIO-MISSISSIPPI RIVER SYSTEM

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7-82-2898-5

RADIONUCLIDE SOURCE TERM COMPARISON

A-176

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO $\frac{\text{NUREG VALUE}}{\text{CRBR VALUE}}$
• $^{3}\mathrm{H}$	5.9×10^{4}	2.34×10^{4}	3
• $^{39}\mathrm{Sr}$	9.2×10^{7}	1.60×10^{7}	6
• $^{90}\mathrm{Sr}$	6.1×10^{6}	6.79×10^{5}	9
• $^{90}\mathrm{Y}$	6.4×10^{6}	7.11×10^{5}	9
• $^{91}\mathrm{Y}$	1.2×10^{8}	2.04×10^{7}	6
• $^{95}\mathrm{Nb}$	1.7×10^{8}	3.48×10^{7}	5
• $^{103}\mathrm{Ru}$	1.4×10^{8}	5.26×10^{7}	3
• $^{103m}\mathrm{Rh}$	1.4×10^{8}	5.26×10^{7}	3
• $^{105}\mathrm{Rh}$	6.7×10^{7}	3.85×10^{7}	2
• $^{106}\mathrm{Rh}$	7.6×10^{7}	1.96×10^{7}	4
• $^{106}\mathrm{Ru}$	5.1×10^{7}	1.96×10^{7}	3
• $^{110m}\mathrm{Ag}$	3.5×10^{5}	4.33×10^{4}	8

RADIONUCLIDE SOURCE TERM COMPARISON

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO NUREG VALUE / CRBR VALUE
• ^{111m}Ag	4.3×10^6	2.57×10^6	2
• ^{113m}Cd	1.0×10^3	1.91×10^3	1/2
• ^{115m}Cd	6.2×10^4	3.55×10^4	2
• ^{115}Cd	8.8×10^5	5.46×10^5	2
• ^{123}Sn	9.4×10^5	3.62×10^5	3
• ^{125}Sn	1.5×10^6	7.58×10^5	2
• ^{125}Sb	7.4×10^5	3.96×10^5	2
• ^{125m}Te	2.5×10^5	7.88×10^4	3
• ^{127}Sb	8.3×10^6	3.76×10^6	2
• ^{127m}Te	1.6×10^6	5.40×10^5	3
• ^{127}Te	8.1×10^6	3.69×10^6	2
• ^{129m}Te	6.6×10^6	2.65×10^6	2
• ^{129}Te	3.9×10^7	9.71×10^6	4

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7-82-2898-18

RADIONUCLIDE SOURCE TERM COMPARISON

ISOTOPE	NUREG-0440 LWR CORE INVENTORY (Ci)	CRBRP CORE INVENTORY END OF CYCLE (Ci)	RATIO NUREG VALUE / CRBR VALUE
• 129_{I}	2.9	6.7×10^{-1}	4
• 131_{I}	1.0×10^{8}	3.00×10^{7}	3
• 132_{Te}	1.4×10^{8}	4.00×10^{7}	4
• 133_{I}	1.9×10^{8}	5.15×10^{7}	4
• 134_{Cs}	2.1×10^{7}	6.60×10^{5}	32
• 136_{Cs}	5.8×10^{6}	2.65×10^{6}	2
• 137_{Cs}	8.6×10^{6}	1.70×10^{6}	5
• 140_{Ba}	1.8×10^{8}	4.19×10^{7}	4
• 140_{La}	1.8×10^{8}	4.22×10^{7}	4
• 141_{Ce}	1.7×10^{8}	4.29×10^{7}	4
• 144_{Ce}	1.1×10^{8}	2.02×10^{7}	5
• 144_{Pr}	1.1×10^{8}	2.02×10^{7}	5
• 238_{Pu}	2.5×10^{5}	3.29×10^{5}	4/5
• 239_{Np}	2.1×10^{9}	9.48×10^{8}	2

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7-82-2693-19

SITE SPECIFIC PARAMETER COMPARISON

PARAMETER	CRBRP SITE SPECIFIC VALUE	NUREG 0440 VALUE
• LENGTH IN FEET FROM CORE BASEMAT MELT POINT TO RIVER.	1600	1500
• AVERAGE SOIL POROSITY	.3	.2
• PERMEABILITY (FLOW VELOCITY)	2000 FT/YR	2446 FT/YR

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7-82-2898-9

CONCLUSION

RADIOLOGICAL CONSEQUENCES ASSOCIATED WITH RELEASE OF RADIOACTIVE MATERIAL TO THE GROUNDWATER FOLLOWING A HCDA ARE:

- LESS THAN THOSE HYPOTHESIZED FOR AN LWR IN NUREG-0440 AND WASH-1400
- COMPARABLE TO 10 CFR 20 EFFLUENT RELEASE LIMITS FOR ROUTINE RELEASES

G-182

7-82-2898-10

NORRIS BACKROUND INFORMATION

GRAVITY DAM APPROXIMATELY 1800 FEET WITH A MAXIMUM HEIGHT OF 265 FEET.

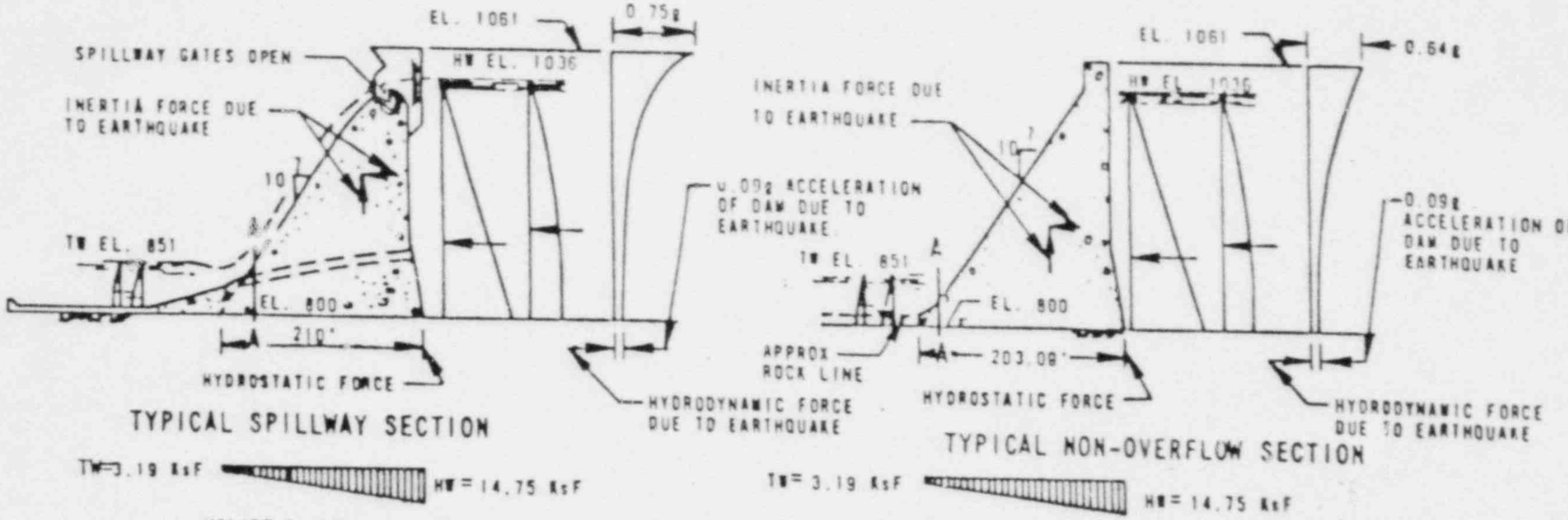
THE DAM IS A SOLID CONCRETE MASS CONCRETE STRUCTURE WITH AN OVERFLOW SPILLWAY, SLUICES AND NONOVERFLOW SECTIONS ON EACH SIDE.

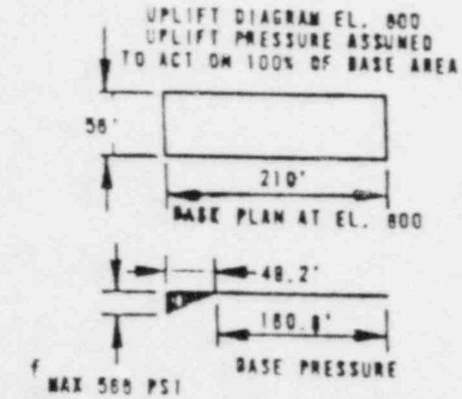
THE DAM WAS COMPLETED IN 1936.

NORRIS DAM WAS ORIGINALLY DESIGNED FOR AN EARTHQUAKE ACCELERATION OF 0.1G THROUGHOUT ITS HEIGHT.

TO ENSURE THE SAFETY OF ITS DAM TVA HAS A WELL DEVELOPED INSPECTION AND MAINTENENCE PROGRAM.

6650-40

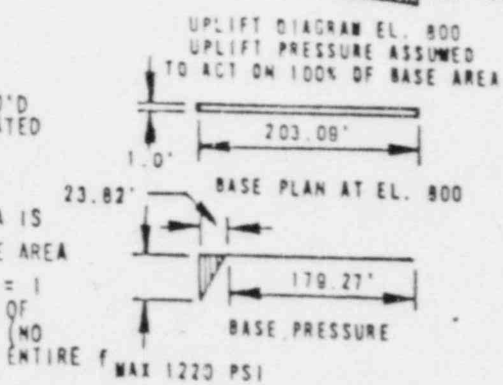




*SHEAR, S, THAT IS REQ'D FOR Q = 1 IS CALCULATED FROM SHEAR-FRICTION FORMULA

$$Q = \frac{0.65\ \Sigma V + SA}{\Sigma H},$$ A IS ASSUMED TO BE ENTIRE AREA

**SHEAR, S, REQD FOR Q = 1 CONSIDERING PORTION OF BASE IN COMPRESSION (NO TENSION) INSTEAD OF ENTIRE AREA.



NOTES: 1. VERTICAL ACCELERATION OF NON-OVERFLOW AND SPILLWAY AT BASED ASSUMED TO BE 0.06g. BY DYNAMIC ANALYSIS, AMPLIFICATION OF ACCELERATION ABOVE THE BASE WAS DETERMINED TO BE 0.14g AT THE TOP FOR THE NON-OVERFLOW SECTION AND 0.14g AT THE TOP FOR THE SPILLWAY.

2. HORIZONTAL ACCELERATION OF NON-OVERFLOW AND SPILLWAY AT BASE ASSUMED TO BE 0.09g. BY DYNAMIC ANALYSIS, AMPLIFICATION OF ACCELERATION ABOVE THE BASE WAS DETERMINED TO BE 0.64g AT THE TOP FOR THE NON-OVERFLOW SECTION AND 0.75g AT THE TOP FOR THE SPILLWAY.

3. SPILLWAY GATES ASSUMED OPEN FOR THIS ANALYSIS.

ΣV	ΣH	$\frac{\Sigma H}{\Sigma V}$	AVG SHEAR	S REQD FOR Q=1	f_{MAX}	$FS = \frac{\Sigma MR}{\Sigma MO}$	VERT. SHEAR ON PLANE A-A
112,616K	143,587K	1.28	85psi (ENT. BASE)	49psi* (177 psi)**	586 psi	1.25	247psi

ΣV	ΣH	$\frac{\Sigma H}{\Sigma V}$	AVG SHEAR	S REQD FOR Q=1	f_{MAX}	$FS = \frac{\Sigma MR}{\Sigma MO}$	VERT. SHEAR ON PLANE A-A
2101K	2786K	1.33	85psi (ENT. BASE)	49psi* (415 psi)**	1220 psi	1.03	535psi

Figure 2.4-29. Norris Dam-Spillway & Non-Overflow Results of Analysis for OBE + 1/2 PMF

A-182

CONSERVATISMS IN THE SEISMIC ANALYSIS

1. THE CONSERVATIVE PSUEDO STATIC METHOD OF STABILITY ANALYSIS WAS USED. THIS ASSUMES A SUSTAINED RATHER THAN OSCILLATING FORCE.

2. THE AMPLIFICATION OF THE BASE ACCELERATION WAS TAKEN AS THE MAXIMUM FOR ALL PARTS OF STRUCTURE ALTHOUGH THEY ALL DO NOT OCCUR SIMULTATIC ONLY.

3. THE CONCRETE WAS ASSUMED INCAPABLE OF TAKING ANY TENSION.

4. ALTHOUGH THE DAM WAS ASSUMED TO OVERTURN THERE IS INSUFFICIENT ENERGY GENERATED OVER THE SHORT DURATION OF THE LOAD TO OVERTURN THE STRUCTURE.

5. CONSERVATIVE JUDGEMENT WAS USED IN ASSESSMENT OF FAILURE RECOGNIZING NUCLEAR PLANT SITING.

6. TVA'S ASSESSMENT OF NORRIS REGARDING ITS SAFETY PROGRAM IS THAT NORRIS CAN SAFELY WITHSTAND THE MAXIMUM CREDITABLE EARTHQUAKE.

7. OTHER GRAVITY DAMS HAVE BEEN SUBJECTED TO MUCH HIGHER EARTHQUAKE ACCELERATIONS AND HAVE NOT FAILED. FOR EXAMPLE, KONYA DAM IN INDIA. TVA MADE AN ANALYSIS OF KONYA USING THE PSUEDO-STATIC METHOD. RESULTS INDICATED THE DAM TO BE STRESSED MUCH WORSE THAN NORRIS.

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

July 6, 1982

APPENDIX XXV
PROPOSED SUMMARY OF SYSTEMATIC
EVALUATION PROGRAM SUBCOMMITTEE MTG.

MEMORANDUM FOR: C. P. Siess, Chairman
ACRS Subcommittee on the Systematic Evaluation Program

FROM: Richard K. Major, Senior Staff Engineer
Richard Major

SUBJECT: SUBCOMMITTEE ON THE SYSTEMATIC EVALUATION PROGRAM MEETING OF JUNE 30, 1982 (INTEGRATED SAFETY ASSESSMENT OF THE R. E. GINNA NUCLEAR POWER PLANT)

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

Attachment:
As stated

cc: ACRS Members
ACRS Technical Staff
I. Catton, ACRS Consult.
D. Fitzsimmons "
W. Lipinski "
E. Case, NRR
W. Russell, SEP Branch
A. Wang, SEP Branch
E. Goodwin, ACRS/NRC Liaison

PROPOSED SUMMARY OF THE JUNE 30, 1982 MEETING OF THE
SUBCOMMITTEE ON THE SYSTEMATIC EVALUATION PROGRAM
(INTEGRATED SAFETY ASSESSMENT OF THE R.E. GINNA NUCLEAR POWER PLANT)

Purpose:

The purpose of this meeting was to review the integrated plant safety assessment, systematic evaluation program review performed at the R.E. Ginna Nuclear Power Plant. Ginna is the second plant to complete an integrated plant safety assessment. (Palisades in May 1982 was the first.) The integrated plant safety assessment will be factored into the Staff's deliberations for a provisional operating license conversion to a full-term operating licsense for Ginna. The integrated assessment will form only part of the Staff's basis for a license conversion. The implementation or status of TMI requirements and unresolved safety issues will also be a part of the basis for the license conversion. (A supplement to the Ginna integrated assessment report will be issued by the Staff to support the license conversion. The Committee will be asked for its comments, again, when the license conversion is proposed at a later date.) The goal of this meeting was to judge the adequacy of the integrated assessment performed at Ginna. This case is being brought before the full Committee in July, 1982 for a report commenting on the integrated assessment.

Principal Attendees:

ACRS:

C. Siess, Chairman
W. Kerr, Member
H. Lewis, Member
J. Ebersole, Member
W. Lipinski, Consultant
I. Catton, Consultant
R Major, Staff (DFE)
H. Alderman, Staff

NRC Staff

W. T. Russell
C. Grimes
A. Wang
T. Cheng
R. Scholl, Jr.
M. Boyle
R Zimmerman

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Rochester Gas & Electric

R. Mecredy
G. Wrobel
T. Weis
R. Smith
G. Larizza

Meeting Highlights, Agreements and Requests:

1. The R. E. Ginna Nuclear Power Plant is operating under a provisional operating license. The S.E.P. review will form a part of the conversion to a full term operating license. There will be a hearing in connection with the license conversion.

2. The current schedule for the POL to FTOL license conversions has not been set. It is expected that they will occur in the same order as the S.E.P. reviews. Dr. Siess has suggested that the ACRS license conversion process revert back to the individual plant subcommittee rather than the S.E.P. Subcommittee. It is felt that the advantage of having one subcommittee review the license conversion will deminish, due to the more plant specific nature of the license conversions. The first license conversion will not occur until some time in 1983.

3. The Staff noted they considered the status of TMI Action Plan items and unresolved safety issues to the extent practical, in the integrated assessment. In future safety assessment reports the Staff will identify how these issues relate to the integrated assessment topics.

4. The Staff noted that where immediate fixes are required licensees have readily volunteered to make those fixes. (Ginna required no immediate fixes). Integrated assessment items proposed to the Commission for implementation could, if necessary, be mandated by order.
5. It was noted that some regulations (such as GDC 56 concerning containment isolation valves) allow the requirements to be implement on basis other than that defined in the regulation. Where differences do exist, the current design satisfies the Staff, and there is no flexibility in the letter of the regulations, those items will have to be identified as exceptions to the regulations.
6. The initial criticality for Ginna was in November of 1969 at a power level of 1300 MWt. The Plant began commercial operation in July 1970. In 1972, the power level was increased to 1520 MWt and RG&E applied for a full term operating license. In 1975 standby auxiliary feed-water systems were added and an upgrade to inservice inspection was made. Full flow condensate demineralizers were added and the SEP was started in 1977. Plant security upgrades were added in 1978. In 1980 TMI modifications including a technical support center, were added to the plant.
7. Ginna's performance statistics for plant life to date include: 33,853,098 MWe generated, a capacity factor of 69% and a plant availability of 75%.

8. The Ginna reactor is a two loop Westinghouse PWR. It produces 1520 MWt (490 MWe). See Schematic diagram on next page. A unique feature of the Ginna plant is the fact that there is a total of five auxiliary feedwater pumps. Two are motor driven and can be operated off the diesels, one is a turbine driven pump. In addition there are two standby auxiliary feedwater pumps that can inject water into the steam generators if normal auxiliary feedwater was not available. The standby AFW pumps take suction from the service water.

9. Of the 137 SEP topics, 24 were deleted since they were being reviewed generically under either the TMI Action Plan or as a USI. 21 additional topics were deleted that did not apply to the GINNA plant. Of the 92 remaining topics, 58 topics either met curent criteria or were acceptable on another defined basis. 7 more topics were found acceptable after modifications made during the review. 27 topics were considered for backfit in the integrated assessment.

10. The Staff described several topics which were found acceptable not by current criteria but on another defined basis.

    The seismic review of Ginna was started in mid 1979. A site specific spectrum was not available when the seismic review was started. Regulatory guide 1.60, horizontal ground response spectrum scaled to the original design PGA (0.2g) was used as the postulated SSE. This review spectrum is more conservative than either the site specific spectrum or the original design spectrum. The structural

490 MWe
1520 MWt

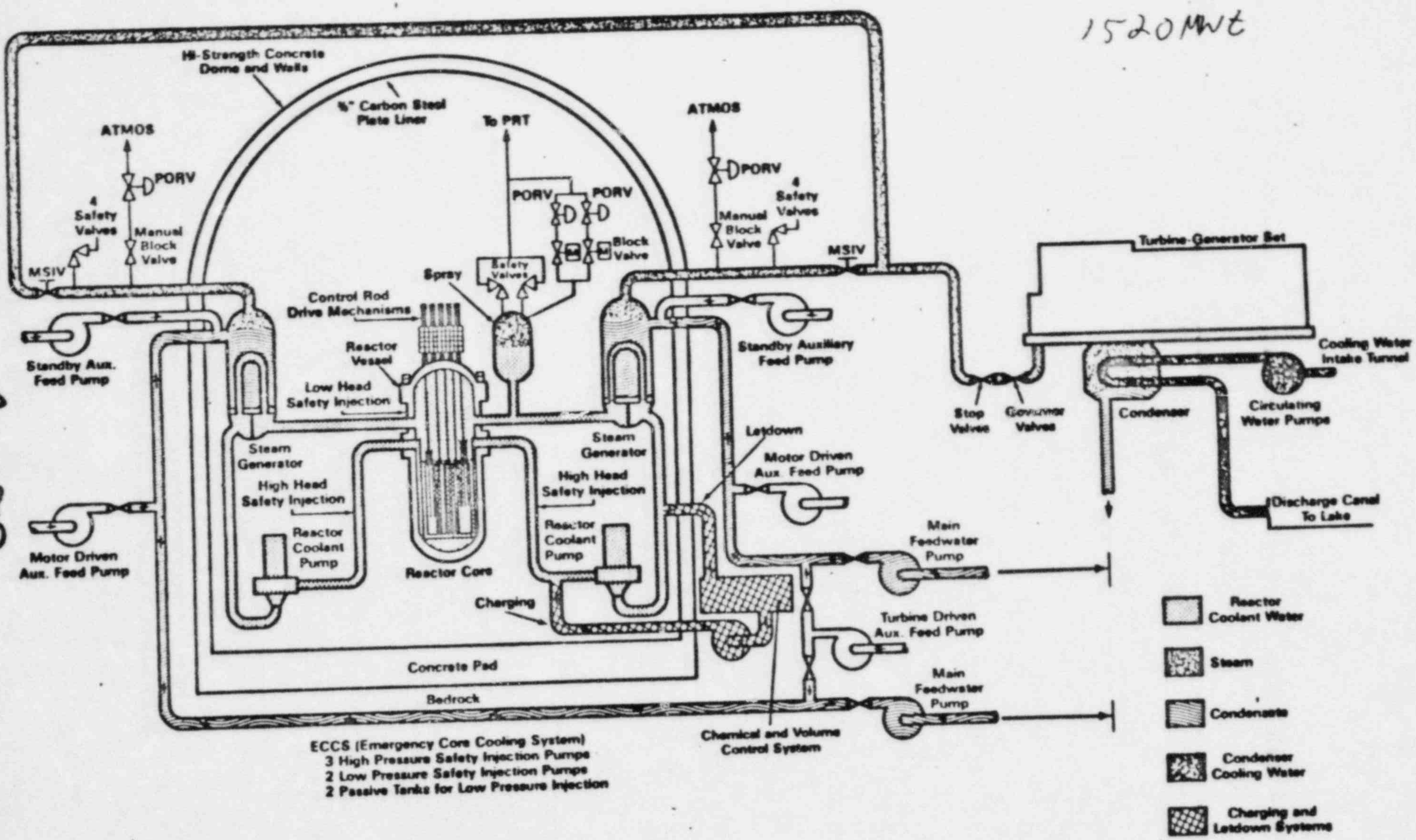

Figure 3.10 Schematic diagram of Westinghouse-designed pressurized water reactor of Rochester Gas & Electric Corporation's Ginna Nuclear Power Plant

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mechanical and functional integrity is now being covered under a non-SEP generic item. Two locations structure-wise were found to be over-stressed, the bracing of the auxiliary building and the bracing in the turbine building.

In the area of turbine missiles the Staff has reviewed the reliability of the over-speed turbine trip, and the adequacy of the inservice inspection of turbine discs. The Staff concluded, for an interim period until a decision is reached regarding the need for updated probabilistic analysis of the turbine missile hazard, the probability of damage from turbine missiles is acceptably low.

The subcommittee discussed a delay of 1 second (~ automatic) on the addition of NaOH to containment spray water. The benefit of this additive, which is used to scrub iodine from the containment atmosphere in a post-LOCA environment, was questioned. The risk of spraying the containment inadvertently with an alkaline solution versus the actual amount doses would be reduced in light of current understanding of the iodine source term following the TMI accident may be a nonconservative forced on a licensee by regulatory requirements. The licensee noted that in the event of an inadvertent containment spray actuation the difference between the clean-up problems presented by a boric acid or sodium hydroxide spray was negligable. (It would be a problem in either case.)

11. Copies of Comments from the NRC's Senior Review Group (J. Hendrie, S. Bush, Z. Zudans, H. Isbin, and R. Budnitz) were passed out to the Subcommittee. Attached to this summary is a copy of the review group comments. In general, the comments are favorable.

12. The Staff presented those items which were part of the integrated assessment which did not require backfit. These items were:

   - Topic II-4.D, Stability of Slopes
   - Topic III-4.A, Tornado Missiles
     (Section 4.11.3, Boric Acid Tanks)
   - Topic III-4.c, Internally Generated Missiles
     (Sections 4,12.1, Accumulator (CVCS) Letdown Lines and 4.12.4, Refueling Waster Storage Tank)
   - Topic III-6, Seismic Design Consideration
     (Section 4.15.2, Turbine Bldg.)
   - Topic III-8.A, Loose Parts Monitoring and Core Barrel Vibration Program
   - Topic V-5, Reactor Coolant Pressure Boundry Leakage to Containment
     (Section 4.15.1, Detection of Reactor Coolant Pressure Boundry Leakage to Containment)
     (Section 4.19.2, Monitoring of Reactor Coolant Inleakage)
     (Section 4.19.2, Technical Specifications Regarding Operability of Leakage Detection Systems)
     (Section 4.15.4, Reactor Coolant Inventory Balance)
   - Topic VI-4, Containment Isolation System
     (Section 4.22.1, Valve Location)
     (Section 4.22.2, Valve Number)
     (Section 4.22.3, Valve Actuation)
   - Topic IX-3, Station Service and Cooling Water Systems
     (Section 4.25,4, Pressure Sensor on Component Cooling Water Pumps)

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RG&E

HISTORY

GINNA STATION

1969	NOV.	INITIAL CRITICALITY AT 1300 MW
1970	JULY	COMMERCIAL OPERATION
1972		UPGRADE TO 1520 MW APPLIED FOR FULL TERM OPERATING LICENSE
1974		ARMOR STONE
1975		PIPE BREAKS OUTSIDE CONTAINMENT JET SHIELDS STANDBY AUXILIARY FEEDWATER SYSTEMS INSERVICE INSPECTION UPGRADE
1977		FULL FLOW CONDENSATE DEMINERALIZERS BEGIN SEP
1978		SECURITY
1980		TMI MODIFICATIONS INCLUDING TECHNICAL SUPPORT CENTER

RG&E

HISTORY

GINNA STATION

PERFORMANCE STATISTICS	(LIFE TO DATE)
MWE GENERATED:	33,853,048
CAPACITY FACTOR:	69%
AVAILABILITY:	75%

ANNUAL AVAILABILITY

	1976 - 58%
1981 - 82%	1975 - 77%
1980 - 76%	1974 - 62%
1979 - 73%	1973 - 95%
1978 - 81%	1972 - 69%
1977 - 86%	1971 - 76%
	1970 - 70%

FIRE AREA CODES

AB – AUXILIARY BUILDING

AVT – CONDENSATE DEMINERALIZER BUILDING

CC – CONTROL ROOM AREA

CT – CABLE TUNNEL

DG – DIESEL GENERATOR

H_2 – HYDROGEN STORAGE AREA

IB – INTERMEDIATE BUILDING

RC – REACTOR CONTAINMENT

SAF – STANDBY AUXILIARY FEEDWATER PUMP BUILDING

SB – SERVICE BUILDING

SH – SCREEN HOUSE

TB – TURBINE BUILDING

TO – TURBINE OIL STORAGE AREA

N_2 – NITROGEN STORAGE AREA

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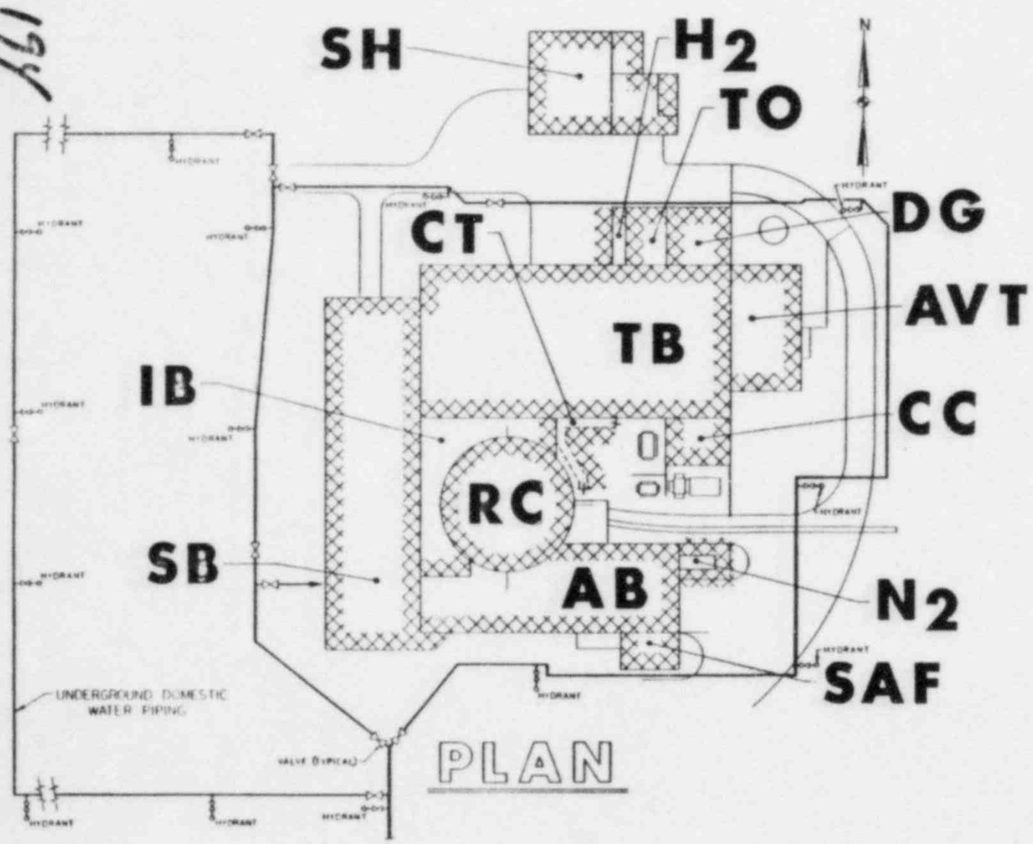

PLAN

LEGEND

1. WALL DESIGNATIONS
   - HOURLY RATE FOR RATED BARRIERS
   - FIRE AREA BOUNDARIES (HEIGHT ABOVE FLOOR)
   - NOT CONTINUOUS FROM FLOOR TO CEILING
   - CONTINUOUS FROM FLOOR TO CEILING
2. WIRE GATES – DOORS – RATINGS: LABEL (A OR B) HOURS PROTECTION ($1\frac{1}{2}$ OR 3)
3. SAFE SHUTDOWN EQUIPMENT
4. DAMPERS
5. HOSE REELS
   - (HOSE LENGTH)
6. FIRE EXTINGUISHERS
   - WATER
   - DRY CHEMICAL
   - CARBON DIOXIDE
   - HOSE CABINET
   - WALL HYDRANT
7. CABLE TRAYS/ENGINEERING SAFEGUARD
   - A
   - B
8. SAFE SHUTDOWN INSTRUMENTS
   - (NUMBER) - PRESSURE TRANSMITTER
   - (NUMBER) - PRESSURE SWITCH
   - (NUMBER) - FLOW SWITCH
   - (NUMBER) - FLOW TRANSMITTER
   - (NUMBER) - LEVEL INDICATOR
   - (NUMBER) - LEVEL TRANSMITTER
   - (NUMBER) - TEMPERTURE DETECTOR
   - (NUMBER) - E/P CONVERTER
9. SAFE SHUTDOWN VALVES
10. FIRE SUPPPESSION SYSTEMS
    - WATER
    - HALON

ROCHESTER GAS & ELECTRIC CORPORATION
GINNA STATION
FIRE PROTECTION EVALUATION
PLOT PLAN
GILBERT ASSOCIATES, INC.
04 4750 D-024-001 1

FIRE PROTECTION EVALUATION

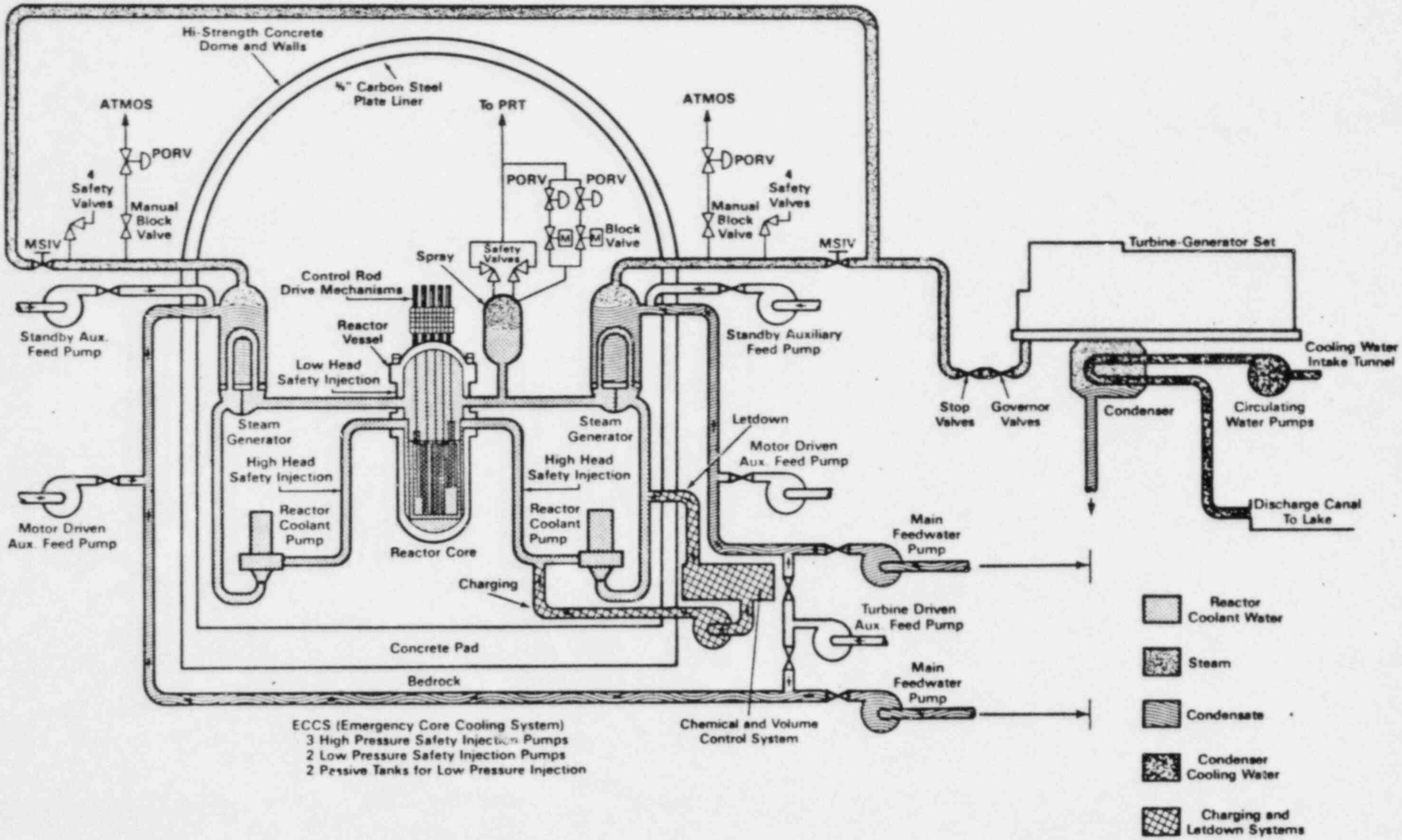

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Figure 2.1 **Schematic diagram of Westinghouse-designed pressurized water reactor of Rochester Gas & Electric Corporation's Ginna Nuclear Power Plant**

SEP PHASE II REVIEW
GINNA

- INTRODUCTION
- SEP OVERVIEW
    - PHASE I
    - PHASE II
    - PHASE III/NREP
- CURRENT PHASE II REVIEW SCHEDULE
- TOPIC REVIEW SUMMARY
- INTEGRATED ASSESSMENT RESULTS

CONTACT: C. GRIMES
X 28414

Table 2.1 Topic list selection and resolution

ORIGINAL PHASE I TOPIC LIST
800

↓

Many of these topics were deleted because they were duplicative in nature, were not normally included in the review of light-water reactors, were related to research-and-development programs, or were reviewed on a periodic basis in accordance with current criteria.

↓

FINAL LIST OF PHASE I TOPICS REVIEWED DURING PHASE II
137 (see Appendix A)

↓

Of the 137 topics, 24 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USIs) program or the Three Mile Island (TMI) NRC Action Plan (see Appendix B).

↓

REMAINING TOPICS AFTER DELETION OF USIs AND TMI-RELATED TOPICS
113

↓

Of the remaining 113 topics, 21 were deleted because the topics did not apply to Ginna (see Appendix C).

↓

FINAL NUMBER OF TOPICS REVIEWED FOR GINNA
92 (see Section 3.1 and Appendix E)

↓

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS
58 (see Section 3.1)

↓

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS AFTER MODIFICATIONS MADE DURING TOPIC REVIEW
7 (see Sections 3.1, 3.3.7, 3.3.8, 3.3.3, and 3.3.6-3.3.9, 3.3.11, and 3.3.12)

↓

TOPICS CONSIDERED FOR BACKFIT IN THE INTEGRATED ASSESSMENT
27 (see Table 4.1 and Sections 4.1-4.27)

NRR OPERATING PLAN*

SEP PHASE II SCHEDULE

PLANT	DATE	ACTUAL
PALISADES	MAY 82	APRIL 1, 1982
GINNA	MAY 82	MAY 27, 1982
OYSTER CREEK	SEPT. 82	
DRESDEN 2	OCT. 82	
MILLSTONE 1	NOV. 82	
YANKEE	JAN. 83	
LACROSSE	JAN. 83	
HADDAM NECK	FEB. 83	
BIG ROCK POINT	APR. 83	
SAN ONOFRE 1	APR. 83	

* 6/15/82 REVISION

TOPICS FOR WHICH THE PLANT DESIGN MEETS CURRENT CRITERIA

TOPIC	TITLE
II-1.B	Population Distribution
II.1-C	Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities
II.2-C	Atmospheric Transport and Diffusion Characteristics for Accident Analysis
II.3-A	Hydrologic Description
II-4.D	Site-Proximity Missiles (Including Aircraft)
III-7.C	Delamination of Prestressed Concrete Containment Structures
III-8.C	Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance
III-10.B	Pump Flywheel Integrity
IV-1.A	Operation with Less Than all Loops In Service
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures
VI-1	Organic Materials and Post-Accident Chemistry
VI-2.D	Mass and Energy Release for Possible Pipe Break Inside Containment
VI-3	Containment Pressure and Heat Removal Capability
VI-6	Containment Leak Testing
VI-7.A.1	Emergency Core Cooling System Reevaluation to Account For Increased Reactor Vessel Upper-Head Temperature
VI-7.A.2	Upper Plenum Injection
VI-7.A.3	Emergency Core Cooling System Actuation System
VI-7.C	Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves

TOPIC	TITLE
VI-7.C.2	Failure Mode Analysis (Emergency Core Cooling System)
VI-7.D	Long-Term Cooling Passive Failures (e.g., Flooding of Redundant Components)
VII-1.A	Isolation of Reactor Protection System From Nonsafety Systems, Including Qualification of Isolation Devices
VII-1.B	Trip Uncertainty and Setpoint Analysis Review of Operating Data Base
VII-2	Engineered Safety Features System Control Logic and Design
VII-6	Frequency Decay
VII-1.A	Potential Equipment Failures Associated with Degraded Grid Voltage
VIII-2	Onsite Emergency Power Systems (Diesel Generator)
IX-1	Fuel Storage
IX-4	Boron Addition System (PWR)
XIII-2	Safeguards/Industrial Security
XV-2	Spectrum of Steam System Piping Failures Inside and Outside Containment
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
XV-4	Loss of Nonemergency AC Power to the Station Auxiliaries
XV-5	Loss of Normal Feedwater Flow
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

TOPIC	TITLE
XV-8	Control Rod Misoperation (System Malfunction or Operator Error)
XV-10	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)
XV-12	Spectrum of Rod Ejection Accidents
XV-14	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-17	Radiological Consequences of Steam Generator Tube Failure (PWR)
XV-19	Loss-of-Cooling Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
XV-20	Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment
XVII	Operational Quality Assurance Program*

*The Operational Quality Assurance Program was reviewed to the criteria specified for operating reactors in 1974 (see Appendix A). NRC is currently evaluating all aspects of Nuclear PowerPlant Quality Assurance Programs. Additional review of this issue will be performed outside the context of SEP.

TOPICS FOR WHICH THE PLANT DESIGN WAS ACCEPTABLE ON ANOTHER DEFINED BASIS

TOPIC	TITLE
II-4	Geology and Seismology
II-4.A	Tectonic Province
II-4.B	Proximity of Capable Tectonic Structures in Plant Vicinity
II-4.C	Historical Seismicity Within 200 Miles of Plant
II-4.F	Settlement of Foundations and Buried Equipment
III-4.B	Turbine Missiles
III-7.D	Containment Structural Integrity Tests
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves
— V-6	Reactor Vessel Integrity
V-7	Reactor Coolant Pump Overspeed
V-11.A	Requirements for Isolation of High- and Low-Pressure Systems
V-11.B	Residual Heat Removal System Interlock Requirements
VI-7.C.1	Appendix K - Electrical Instrumentation and Control Re-reviews
VI-7.F	Accumulator Isolation Valves Power and Control System Design
VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing
VII-3	Systems Required for Safe Shutdown
VIII-3.A	Station Battery Capacity Test Requirements
VIII-4	Electrical Penetrations of Reactor Containment
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve
XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
XV-9	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
XV-15	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/ Relief Valve

TOPICS FOR WHICH PLANT DESIGN MEETS CURRENT CRITERIA OR EQUIVALENT BASED ON MODIFICATIONS IMPLEMENTED OR COMMITTED TO BY THE LICENSEE

- o TOPIC III-5.B, PIPE BREAK OUTSIDE CONTAINMENT
- o TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS
- o TOPIC V-11.A, REQUIREMENTS FOR ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS
- o TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM (ELECTRICAL)
- o TOPIC VI-7.B, ENGINEERED SAFETY FEATURE SWITCHOVER FROM INJECTION TO RECIRCULATION MODE (AUTOMATIC EMERGENCY CORE COOLING SYSTEM REALIGNMENT)
- o TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING RESPONSE TIME TESTING
- o TOPIC VIII-1.A, POTENTIAL EQUIPMENT FAILURES ASSOCIATED WITH DEGRADED GRID VOLTAGE
- o TOPIC VIII-3.A, STATION BATTERY CAPACITY TEST REQUIEREMENTS
- o TOPIC VIII-4, ELECTRICAL PENETRATIONS OF REACTOR CONTAINMENT
- o TOPIC IX-6, FIRE PROTECTION
- o TOPIC XV-17, RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE(PWR)
- o TOPIC XV-19, LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

INTEGRATED ASSESSMENT OF 27 TOPICS

CONSIDERED FOR BACKFIT

- TOPICS NOT REQUIRING BACKFIT
- TOPICS WITH PROCEDURAL BACKFITS
- TOPICS WITH HARDWARE BACKFITS
- TOPICS WITH ANALYSIS AND POTENTIAL HARDWARE BACKFITS
- TOPICS WITH DIFFERENCES BETWEEN RG&E AND STAFF

CONTACT: ALAN WANG
X24768

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TOPICS NOT REQUIRING BACKFIT

- TOPIC II-4.D, STABILITY OF SLOPES
- TOPIC III-4.A, TORNADO MISSILES
  (SECTION 4.11.3, BORIC ACID TANKS)
- TOPIC III-4.C, INTERNALLY GENERATED MISSILES
  (SECTIONS 4.12.1, ACCUMULATOR; 4.12.2, CVCS LETDOWN LINE AND
  4.12.4, REFUELING WATER STORAGE TANK)
- TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS
  (SECTION 4.15.2, TURBINE BUILDING)
- TOPIC III-8.A, LOOSE PART MONITORING AND CORE BARREL VIBRATION PROGRAM
- TOPIC V-5, REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE TO CONTAINMENT
- TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM
  (SECTIONS 4.22.1, VALVE LOCATION; 4.22.2, VALVE NUMBER; AND 4.22.3,
  VALVE ACTUATION)
- TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS
  (SECTION 4.25.4, PRESSURE SENSOR ON COMPONENT COOLING WATER PUMPS)
- TOPIC IX-5, VENTILATION SYSTEMS

o TOPIC II-4.D, STABILITY OF SLOPES

DIFFERENCE

SLOPE STABILITY WAS NOT DEMONSTRATED NOR WAS THE CONSEQUENCE OF SLOPE FAILURE SHOWN TO BE MOOT.

RESOLUTION

STAFF ANALYSIS DETERMINED FAILURE OF SLOPES WOULD NOT AFFECT ANY SAFETY-RELATED EQUIPMENT.

- TOPIC III-4.A, TORNADO MISSILES
  (SECTION 4.11.3, BORIC ACID TANKS)

  DIFFERENCE

  - BORIC ACID TANKS NOT PROTECTED FROM TORNADO MISSILES

  RESOLUTION

  - TANKS NOT REQUIRED FOR SAFE SHUTDOWN

  - FAILURE OF TANKS WOULD NOT CAUSE A FLOODING PROBLEM

- TOPIC III-4.C, INTERNALLY GENERATED MISSILES (SECTIONS 4.12.1, ACCUMULATOR; 4.12.2, CVCS LETDOWN LINE AND 4.12.4, REFUELING WATER STORAGE TANK)

DIFFERENCES

- VALVE 832B OF THE ACCUMULATOR SYSTEM WAS A POTENTIAL MISSILE
- VALVES HCV 133 AND RV 203 OF THE CVCS WERE POTENTIAL MISSILES
- THE RWST WAS A POTENTIAL TARGET FROM MISSILES GENERATED BY THE COMPONENT COOLING AND SERVICE WATER SYSTEMS.

RESOLUTIONS

- VALVE 832B's STEM WAS NOT ORIENTED TOWARD ANY SAFETY-RELATED EQUIPMENT
- VALVES HCV 133 AND RV 203 ARE REMOTELY LOCATED FROM ANY SAFETY-RELATED EQUIPMENT.
- THE COMPONENT COOLING AND SERVICE WATER SYSTEMS HAVE INSUFFICIENT INTERNAL ENERGY TO GENERATE ANY MISSILES OF CONSEQUENCE.

- o TOPIC III-6, SEISMIC CONSIDERATION (SECTION 4.15.2, TURBINE BUILDING)

DIFFERENCE

- STRESSES IN THE CROSS BRACINGS ABOVE THE TURBINE BUILDING OPERATING FLOOR HAS STRESSES THAT EXCEED YIELD.

RESOLUTION

- FURTHER ANALYSIS HAS SHOWN PRESENT DESIGN IS ADEQUATE.

o TOPIC III-8.A, LOOSE PARTS MONITORING AND CORE BARREL VIBRATION PROGRAM

<u>DIFFERENCE</u>

NO LOOSE PARTS MONITORING PROGRAM

<u>RESOLUTION</u>

- LOOSE PARTS BACKFITTING IS BEING CONSIDERED IN REVISION 1 TO REGULATORY GUIDE 1.133 BY CRGR

- LOOSE PARTS CAN BE DETECTED DURING REFUELING

- TOPIC V-5, REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE TO CONTAINMENT (SECTION 4.15.1, DETECTION OF REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE TO CONTAINMENT)

  <u>DIFFERENCES</u>

  - LOW SENSITIVITY OF SUMP LEVEL MONITOR
  - SOME SYSTEMS NOT SEISMICALLY QUALIFIED

  <u>RESOLUTIONS</u>

  - SUFFICIENT SYSTEMS ARE AVAILABLE TO DETECT A 1 GPM LEAK FROM THE RCPB TO CONTAINMENT WITHIN 1 HOUR.
  - WILL BE REVIEWED IN CONJUNCTION WITH HIGH ENERGY PIPE BREAK INSIDE CONTAINMENT

(SECTION 4.19.2, MONITORING OF REACTOR COOLANT INLEAKAGE)

DIFFERENCE

- SECONDARY SYSTEM AIR EJECTOR AND STEAM GENERATOR BLOWDOWN MONITORS ARE NOT SEISMICALLY QUALIFIED.

RESOLUTION

- SEISMIC UPGRADING OF REACTOR COOLANT LEAKAGE INTO STEAM GENERATOR NOT REQUIRED BECAUSE:

  A. SAMPLING FOR SECONDARY ACTIVITY CAN BE PERFORMED IF THE MONITORS FAIL.

  B. INSTRUMENTATION REQUIRED BY TMI ACTION PLAN ITEM II.F.1, "NOBLE GAS EFFLUENT MONITOR."

(SECTION 4.19.3, TECHNICAL SPECIFICATIONS REGARDING OPERABILITY OF LEAKAGE - DETECTION SYSTEMS)

DIFFERENCE

- LICENSEE'S TECHNICAL SPECIFICATIONS ONLY REQUIRES TWO LEAKAGE DETECTION SYSTEMS.

RESOLUTION

- LICENSEE HAS NINE VARIOUS METHODS AVAILABLE FOR LEAKAGE DETECTION.

- TOPIC V-5, REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION (CONTINUED)

  (SECTION 4.15.4, REACTOR COOLANT INVENTORY BALANCE)

  DIFFERENCE

  SENSITIVITY OF REACTOR COOLANT INVENTORY BALANCES NOT PROVIDED DURING TOPIC REVIEW

  RESOLUTION

  RCPB LEAKAGE DETECTION PROCEDURES WERE REVIEWED BY REGION I PERSONNEL AND FOUND ACCEPTABLE WITH RESPECT TO TECHNICAL SPECIFICATION REQUIREMENTS OF 1-GPM UNIDENTIFIED LEAKAGE FROM THE RCPB.

- o TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM
  (SECTION 4.22.1, VALVE LOCATION)

<u>DIFFERENCES</u>

- TWO ISOLATION VALVES OUTSIDE CONTAINMENT INSTEAD OF ONE INSIDE AND ONE OUTSIDE.
- CHECK VALVE OUTSIDE OF CONTAINMENT.

<u>RESOLUTIONS</u>

- PIPING BETWEEN CONTAINMENT AND CONTAINMENT ISOLATION VALVE IS RATED FOR AT LEAST CONTAINMENT DESIGN PRESSURE.
- PIPING RUNS REASONABLY SHORT.
- PIPING UP TO AND INCLUDING THE SECOND CONTAINMENT ISOLATION VALVE ARE SEISMIC CATEGORY I.
- ALL PIPING IS SUPPORTED AND DESIGNED FOR PIPE BREAK LOADS.
- MOST PIPING PENETRATIONS IN AREAS PROTECTED FROM TORNADO MISSILES.

SECTION 4.22.2, VALVE NUMBER

DIFFERENCE

- PENETRATIONS 100, 102, 106, AND 110 (REACTOR COOLENT PUMP SEAL INJECTION LINES) HAVE ONLY ONE CONTAINMENT ISOLATION VALVE.

RESOLUTIONS

- SYSTEM PIPING IS RATED AT 2250 PSI

- PIPING IS SEISMIC CATEGORY I

- POSITIVE DISPLACEMENT PUMPS MINIMIZE LEAKAGE BACK THROUGH PUMPS.

DIFFERENCE

- PENETRATIONS 105 AND 109 (CONTAINMENT SPRAY SYSTEM) HAVE ONLY ONE CONTAINMENT ISOLATION VALVE.

RESOLUTIONS

- CONTAINMENT SPRAY IS A CLOSED SYSTEM OUTSIDE CONTAINMENT.

- SEE SECTION 4.22.1

SECTION 4.22.3, VALVE ACTUATION

DIFFERENCE

- PENETRATION 112 (CVCS LETDOWN LINE) HAS ONLY ONE AUTOMATIC ISOLATION VALVE.

RESOLUTION

- CHANGED CONTROL CIRCUITRY FOR EXISTING VALVES AS PART OF TUBE RUPTURE INCIDENT.

DIFFERENCE

- PENETRATIONS 121C, 121D, 203A, AND 332A (PRESSURE SENSING LINES) ARE CAPPED OUTSIDE OF CONTAINMENT.

RESOLUTION

- CAPS ARE CLOSE TO CONTAINMENT AND LEAK TESTED.
- PENETRATIONS ARE SEISMIC CATEGORY 1.
- 1977 ASME BPV CODE, SECTION III, ARTICLE NE-3367, STATES THAT CLOSURE ON PENETRATIONS OF 2-INCH PIPE SIZE OR LESS CAN BE MADE WITH PIPE CAPS.

DIFFERENCE

- PENETRATIONS 205A, 206A, AND 207A (SAMPLING LINES) HAVE ONLY ONE AUTOMATIC ISOLATION VALVE ON EACH LINE.

RESOLUTION

- LICENSEE INFORMED THE STAFF THAT AN EXISTING VALVE HAD BEEN PREVIOUSLY CONVERTED TO AN AUTOMATIC ISOLATION VALVE ON EACH LINE.

o TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.25.4, PRESSURE SENSOR ON COMPONENT COOLING WATER PUMPS)

DIFFERENCE

- SINGLE OUTLET PRESSURE INDICATION FOR COMPONENT COOLING WATER PUMPS

RESOLUTION

- FLOW INDICATION IS AVAILABLE

o TOPIC IX-5, VENTILATION SYSTEMS

DIFFERENCE

- TWO SCENARIOS WHERE VENTILATION WOULD DIRECT AIR FROM AREAS OF HIGHER RADIOACTIVITY TO LOWER

RESOLUTION

- LEAKAGE IN BOTH CASES, IS TO A CONTROLLED AREA

TOPICS WITH PROCEDURAL BACKFITS

- o TOPIC II-1.A, EXCLUSION AREA BOUNDARY

- o TOPIC III-3.C, INSERVICE INSPECTION OF WATER CONTROL STRUCTURES (SECTION 4.10.1, CONFORMANCE WITH REGULATORY GUIDE 1.127)

- o TOPIC III-7.A, INSERVICE INSPECTION INCLUDING PRESTRESSED CONCRETE WITH EITHER GROUTED OR UNGROUTED TENDONS

- o TOPIC V-10.B, RESIDUAL HEAT REMOVAL SYSTEM RELIABILITY
  (SECTION 4.21.1, OVERPRESSURIZATION PROTECTION OF SHUTDOWN COOLING SYSTEM)
  (SECTION 4.21.2, USE OF SAFETY-GRADE SYSTEMS FOR SAFE SHUTDOWN)

- o TOPIC VI-7.B, ENGINEERED SAFETY FEATURE SWITCHOVER FROM INJECTION TO RECIRCULATION MODE

- o TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS
  (SECTION 4.25.1, TECHNICAL SPECIFICATIONS ON SERVICE WATER PUMPS)

- TOPIC II-1.A, EXCLUSION AREA AUTHORITY AND CONTROL

DIFFERENCE

THE CURRENT EXCLUSION AREA BOUNDARY (EAB) MAP HAS NOT BEEN REVISED IN THE T/S.

RESOLUTION

THE CURRENT EAB MAP WILL BE INCORPORATED IN THE GINNA T/S.

o TOPIC III-3.C, INSERVICE INSPECTION OF WATER CONTROL STRUCTURES (SECTION 4.10.1, CONFORMANCE WITH REGULATORY GUIDE 1.127)

DIFFERENCE

- LICENSEE INSPECTION PROGRAM DID NOT CONFORM TO REGULATORY GUIDE 1.127.

RESOLUTION

- LICENCEE WILL MODIFY THEIR INSPECTION PROGRAM AS RECOMMENDED BY THE STAFF.

o TOPIC III-7.A, INSERVICE INSPECTION INCLUDING PRESTRESSED CONCRETE CONTAINMENTS WITH EITHER GROUTED OR UNGROUTED TENDONS.

DIFFERENCE

- LICENSEE'S TENDON SURVEILLANCE PROGRAM DOES NOT SPECIFICALLY USE THE METHODOLOGY DESCRIBED IN REGULATORY GUIDE 1.35, REVISION 2.

RESOLUTION

- LICENSEE WILL MODIFY THEIR TENDON SURVIELLANCE PROGRAM AS RECOMMENDED BY THE STAFF.

o TOPIC V-10.B, RESIDUAL HEAT REMOVAL SYSTEM RELIABILITY

(SECTION 4.21.1, OVERPRESSURIZATION PROTECTION OF SHUTDOWN COOLING SYSTEM)

DIFFERENCE

- RHR CAN BE PLACED IN SERVICE BEFORE OPS

RESOLUTION

- PROCEDURES SHALL BE DEVELOPED TO PLACE THE OPS IN SERVICE BEFORE THE RHR IS IN SERVICE.

(SECTION 4.21.2, USE OF SAFETY-GRADE SYSTEMS FOR SAFE SHUTDOWN)

DIFFERENCE

- THERE IS LACK OF INFORMATION IN THE OPERATING PROCEDURES FOR THE USE OF ONLY SAFETY-GRADE SYSTEMS (WITHOUT ANY NON-SAFETY) TO SHUTDOWN THE REACTOR.

RESOLUTION

- PROCEDURES SHALL BE DEVELOPED FOR OPERATION OF SAFETY-GRADE SYSTEMS AND COMPONENTS TO ACHIEVE COLD SHUTDOWN IS NON SAFETY-GRADE SYSTEMS ARE UNAVAILABLE.

o TOPIC VI-7.B, ENGINEERED SAFETY FEATURE SWITCHOVER FROM INJECTION TO RECIRCULATION MODE

DIFFERENCE

- SWITCHOVER PROCEDURES DO NOT MEET CURRENT CRITERIA FOR OPERATOR ACTIONS.

RESOLUTION

- LICENSEE HAS CONTRACTED WESTINGHOUSE TO REVIEW PROCEDURES AND IMPROVE THE SWITCHOVER PROCEDURE.

o TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.25.1, TECHNICAL SPECIFICATIONS ON SERVICE WATER PUMPS)

DIFFERENCE

- PLANT MAY OPERATE WITH THE MINIMUM NUMBER OF SWS PUMPS ALIGNED TO ONE BUS.

RESOLUTION

- TECHNICAL SPECIFICATIONS WILL BE MODIFIED.

TOPICS WITH HARDWARE BACKFITS

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o TOPIC III-4.C, INTERNALLY GENERATED MISSILES (SECTION 4.12.3, STEAM GENERATOR BLOWDOWN)

DIFFERENCE

- VALVE CV 5738 IS A POTENTIAL MISSILE

RESOLUTION

- A RESTRAINT FOR THE OPERATOR FOR VALVE CV5738 WILL BE INSTALLED.

- o TOPIC II-2.A, SEVERE WEATHER PHENOMENA.
  TOPIC III-2, WIND AND TORNADO LOADINGS.
  TOPIC III-4.A, TORNADO MISSILES.
  (SECTION 4.11.2, SAFETY-RELATED EQUIPMENT)
  TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS.
  (SECTION 4.15.1, AUXILIARY BUILDING)
  TOPIC III-7.B DESIGN CODES, DESIGN CRITERIA, LOAD COMBINATIONS, AND REACTOR CAVITY DESIGN CRITERIA.

<u>DIFFERENCE</u>

- STRUCTURES IMPORTANT TO SAFETY DO NOT MEET CURRENT LICENSING CRITERIA.

<u>RESOLUTION</u>

- LICENSEE HAS PROPOSED THE FOLLOWING THREE STEP PROGRAM FOR RESOLUTION OF THE ABOVE TOPICS:

  1. DEVELOP DESIGN PARAMETERS AND CRITERIA FOR STRUCTURAL UPGRADE. THIS WOULD INCLUDE THE FOLLOWING:

     A. EVALUATION OF A SUITABLE RANGE OF INPUT PARAMETERS (WINDSPEEDS, ROOF LOADINGS, AND SO FORTH)

     B. SPECIFICATION OF ACCEPTANCE CRITERIA

     C. DEFINITION OF STRUCTURES AND SYSTEMS REQUIRING PROTECTION

     D. VALUE-IMPACT ASSESSMENT

  2. PERFORM THE STRUCTURAL ANALYSIS AND ENGINEERING DESIGN OF PROPOSED MODIFICATIONS USING THE PARAMETERS AND CRITERIA SHOWN IN ITEM 1.

  3. INSTALL THE MODIFICATIONS, AS REQUIRED, AS A RESULT OF THE ANALYSIS.

- o TOPIC III-5.B, PIPE BREAK OUTSIDE CONTAINMENT

  TOPIC III-6, SEISMIC DESIGN CONSIDERATION
  (SECTION 4.15.3, ESSENTIAL SERVICE WATER PUMP OPERABILITY)

  <u>DIFFERENCE</u>

  - SERVICE WATER PUMPS ARE SUSCEPTIBLE TO SEVERAL COMMON MODE FAILURES (SEISMIC, FIRE, WIND LOADING, PIPE BREAK AND FLOODING)

  <u>RESOLUTION</u>

  - LICENSEE PROVIDED A BACKUP COOLING WATER SOURCE FOR THE DIESEL GENERATORS

  - LICENSEE HAS AGREED TO UPGRADE THE ESSENTIAL SERVICE WATER SYSTEM

  - ALTERNATE WATER SYSTEM FOR [illegible]

- o TOPIC V-10.A, RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGER TUBE FAILURES

DIFFERENCE

- SERVICE WATER SYSTEM DOES NOT INCORPORATE A RADIATION MONITOR

RESOLUTION

- LICENSEE WILL INSTALL A RADIATION MONITOR FOR THE SERVICE WATER SYSTEM OR INCLUDE SURVEILLANCE AND OPERABILITY REQUIREMENTS FOR THE CCW SYSTEM RADIATION MONITOR.

- TOPIC VIII-3.B, DC POWER SYSTEM BUS VOLTAGE MONITORING AND ANNUNCIATION

DIFFERENCE

- DC POWER SYSTEM IS INSUFFICIENTLY MONITORED

RESOLUTION

- LICENSEE HAS AGREED TO PROVIDE ADDITIONAL DC SYSTEM MONITORING AND A "DC SYSTEM TROUBLE ALARM."

o TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.25.2, COMPONENT COOLING WATER SURGE TANK LEVEL AND INDICATION)

DIFFERENCE

- ONLY ONE SENSOR AND INDICATOR TO MEASURE COMPONENT COOLING WATER SURGE TANK LEVEL.

RESOLUTION

- SECOND TRANSMITTER AND LEVEL ALARMS WILL BE INSTALLED.

TOPICS WITH DIFFERENCES BETWEEN RG&E AND STAFF

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TOPIC II-3.B, FLOODING POTENTIAL AND PROTECTION REQUIREMENTS

TOPIC II-3.B.1, CAPABILITY OF OPERATING PLANTS TO COPE WITH DESIGN-BASIS FLOODING CONDITIONS

TOPIC II-3.C, SAFETY-RELATED WATER SUPPLY (ULTIMATE HEAT SINK)

TOPIC III-3.A, EFFECTS OF HIGH WATER LEVEL ON STRUCTURES
(SECTION 4.9.2, FLOODING OF DEER CREEK)

TOPIC III-3.C, INSERVICE INSPECTION OF WATER CONTROL STRUCTURES
(SECTION 4.10.2, DEER CREEK)

ISSUE	STAFF POSITION	LICENSEE POSITION
- FLOOD LEVEL OF DEER CREEK BASED ON PROBABLE MAXIMUM FLOOD.	- PROVIDE PROTECTION FOR STANDARD PROJECT FLOOD PLUS 1 FT.	

BASIS FOR STAFF POSITION

o INSUFFICIENT DATA TO CONCLUDE DEER CREEK WILL NEVER FLOOD BANKS.

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LEVELS PRODUCED BY	DEER CREEK CHANNEL CAPACITY	ELEVATION AT PLANT GRADE	HEIGHT ABOVE GRADE	AT SCREEN HOUSE GRADE	HEIGHT ABOVE GRADE
PMF-37,500 CFS (4-10-81 SER)	12,900	275	+4[1]	261	+7.5[1]
14,000 CFS (4-10-31 SER)	12,900	271	+0[1]	254.25	+0.75[1]
PMF 32,500 CFS (RG&E SUBMITTAL 8-18-81)	13,700	277	+6[1]	-	-
PMF 38,700 CFS (5-27-82 SER)	12,000	275.4	+4.4[1]	265.5	+12.5[1]
15,000 CFR, SPF (5-27-82 SER)	12,000	273	+2.0[1]	256	+2.5[1]
PMF 32,500 CFS (RG&E SUBMITTAL 6-25-82)	14,000	-		-	
13,000 CFS, SPF (RG&E SUBMITTAL 6-26-82)	14,000	BELOW CAPACITY OF DEER CREEK			

TOPIC III-3.A, EFFECTS OF HIGH WATER LEVEL ON STRUCTURES
(SECTION 4.9.1, EFFECTS OF GROUNDWATER LEVEL)

<u>ISSUE</u>	<u>STAFF POSITION</u>	<u>LICENSEE POSITION</u>
- LEVEL OF GROUNDWATER	- ASSUME GROUNDWATER LEVEL AT GRADE	- GROUNDWATER LEVEL AT 250

<u>BASIS FOR STAFF POSITION</u>

o INSUFFICIENT DATA TO CONCLUDE GROUNDWATER LEVEL IS LOWER THAN GRADE.

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TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM
(SECTION 4.22.2, VALVE NUMBER)

ISSUE	STAFF POSITION	LICENSEE POSITION
- PENETRATION 108 (REACTOR COOLANT PUMP SEAL RETURN AND EXCESS LETDOWN LINE) HAS ONLY ONE CONTAINMENT ISOLATION VALVE	INSTALL ANOTHER AUTOMATIC CONTAINMENT ISOLATION VALVE	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

o OPEN SYSTEM OUTSIDE CONTAINMENT

o LONG RUN OF PIPING BETWEEN THE CONTAINMENT AND THE VALVE

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(SECTION 4.22.2, VALVE NUMBER)

ISSUE	STAFF POSITION	LICENSEE POSITION
- PENETRATION 110B (SAFETY INJECTION TEST LINE), 121A (N_2 TO PRESSURIZER RELIEF TANK LINE), AND 129 (N_2 TO REACTOR-COOLANT-DRAIN-TANK LINE)	- LOCK CLOSE A SECOND MANUAL VALVE IN EACH LINE OR CONVERT IT TO AN AUTOMATIC ISOLATION VALVE.	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

o OPEN SYSTEMS OUTSIDE CONTAINMENT

(SECTION 4.22.3, VALVE ACTUATION)

ISSUE	STAFF POSITION	LICENSEE POSITION
- PENETRATIONS 120B, 123 (GAS ANALYZER LINES), AND 305A (CONTAINMENT AIR SAMPLE LINE) HAVE ONLY ONE AUTOMATIC ISOLATION VALVE	- INSTALL A SECOND AUTOMATIC ISOLATION VALVE ON EACH LINE	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

- o OPEN SYSTEM OUTSIDE CONTAINMENT.
- o LONG RUN OF NONSEISMIC PIPING BETWEEN THE CONTAINMENT AND THE VALVE IN EACH CASE.

(SECTION 4.23.3, VALVE ACTUATION)

ISSUE	STAFF POSITION	LICENSEE POSITION
- PENETRATIONS 201, 209 (REACTOR COMPARTMENT COOLING), 308, 311, 312, 315, 316, 319, 320, AND 323 (SERVICE WATER TO AND FROM FAN COOLERS) HAVE NO REMOTE ISOLATION VALVES.	- MANUAL VALVES SHOULD BE CONVERTED TO REMOTE MANUAL VALVES	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

o FOR CLOSED SYSTEMS GDC 57 REQUIRES 1 CIV FOR EACH PENETRATION

o SERVICE WATER LINES ARE LARGE (8")

o SYSTEM PRESSURE AT OUTLET LOWER THAN ACCIDENT PRESSURE

o NEED TO ISOLATE CAN BE DETERMINED DURING AN ACCIDENT

o ACCESS TO PRESENT MANUAL VALVES MAYBE LIMITED DUE TO HIGH RADIATION.

o TIME TO CLOSE THESE MANUAL VALVES MAYBE SIGNIFICANT.

o FAN COOLERS HAVE RECENTLY HAD MINOR LEAKS

o VALVES CLOSED DURING RECENT MAINTENANCE LEAKED.

o ASSUMPTIONS USED IN PRA MAY NOT BE APPLICABLE.

(SECTION 4.22.4, CLOSED SYSTEMS)

ISSUE	STAFF POSITION	LICENSEE POSITION
- VERIFY THAT THE REACTOR COMPARTMENT COOLING WATER SYSTEM IS A SAFETY-GRADE SYSTEM	- IF SAFETY-GRADE, SEE SECTION 4.22.3 - IF NOT SAFETY-GRADE, ONE AUTOMATIC VALVE INSIDE AND ONE AUTOMATIC VALVE OUTSIDE SHOULD BE ADDED ON EACH PENETRATION.	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

o OPEN SYSTEM OUTSIDE CONTAINMENT

o NON-SAFETY-GRADE SYSTEM INSIDE CONTAINMENT

(SECTION 4.22.4, CLOSED SYSTEMS)

ISSUE	STAFF POSITION	LICENSEE POSITION
- PENETRATIONS 301 AND 303 (AUXILIARY STEAM HEATING) HAVE ONLY ONE ISOLATION VALVE	- ADD A SECOND CLOSED VALVE TO EACH PENETRATION	- LICENSEE HAS NOT RESPONDED

BASIS FOR STAFF POSITION

- o NOT A SAFETY-GRADE SYSTEM
- o NOT SEISMICALLY QUALIFIED
- o NOT MISSILE PROTECTED

TOPICS WITH ANALYSIS AND POTENTIAL

HARDWARE BACKFITS

- TOPIC III-1, CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS (SEISMIC AND QUALITY GROUP)

STATUS

- LICENSEE HAS RESPONDED TO SOME IDENTIFIED DIFFERENCES

- o TOPIC III-4.A, TORNADO MISSILES
  (SECTION 4.11.1, COMPONENT COOLING WATER SYSTEM)

STATUS

RESOLUTION DEPENDENT ON ACCEPTABILITY OF USING STEAM GENERATORS AS ALTERNATE COLD SHUTDOWN METHOD.

o TOPIC III-5.A, EFFECTS OF PIPE BREAK ON STRUCTURES, SYSTEMS AND COMPONENTS INSIDE CONTAINMENT

STATUS

(SECTION 4.131.1, CHECK VALVES)
- LICENSEE HAS NOT COMPLETED REVIEW.

(SECTION 4.13.2, LETDOWN PIPING, STEAM GENERATOR BLOWDOWN PIPING AND ACCUMULATOR LEVEL)
- LICENSEE HAS NOT COMPLETED REVIEW.

(SECTION 4.13.3, ACCUMULATOR LINE AND PRESSURIZER SURGE LINE)
- LICENSEE HAS NOT COMPLETED REVIEW.

- o TOPIC III-6, SEISMIC DESIGN CONSIDERATION

(SECTION 4.15.4, SAFETY-RELATED TANKS)

STATUS

- LICENSEE HAS NOT RESPONDED

(SECTION 4.15.5, ELECTRICAL PANELS)

STATUS

- LICENSEE TO CONDUCT LOW-IMPENDENCE TEST FOR MAIN CONTROL BOARD.

(SECTION 4.16.6, ABILITY OF SAFETY-RELATED ELECTRICAL EQUIPMENT TO FUNCTION)

STATUS

- OWNERS GROUP REPORT DUE BY END OF YEAR.

(SECTION 4.15.7, QUALIFICATION OF CABLE TRAYS)

STATUS

- OWNERS GROUP REPORT DUE BY JUNE OF 1982.

- o TOPIC III-7.B, DESIGN CODES, DESIGN CRITERIA, LOAD COMBINATIONS, AND REACTOR CAVITY DESIGN CRITERIA (SECTION 4.17.1, CONTAINMENT LINER INSULATION)

  STATUS

  - LICENSEE IS PERFORMING ADDITIONAL ANALYSIS.

o TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.25.2, TECHNICAL SPECIFICATIONS ON SERVICE WATER PUMPS)

DIFFERENCE

- PLANT MAY OPERATE WITH THE MINIMUM NUMBER OF SWS PUMPS ALIGNED TO ONE BUS.

RESOLUTION

- TECHNICAL SPECIFICATIONS WILL BE MODIFIED.

- o TOPIC IX-6, FIRE PROTECTION

  STATUS

  - THIS TOPIC AND ASSOCIATED EXEMPTION REQUEST ARE BEING RESOLVED AS PART OF THE APPENDIX R REVIEWS.

III-3.A DESIGN BASIS GROUNDWATER LEVEL

BASED ON BORING DATA TAKEN PRIOR TO PLANT CONSTRUCTION, RG&E CONCLUDED THAT A DBGWL OF 250 ft msl WAS TO BE USED.

FOR SCREENHOUSE, USED GRADE OF 253'-6" FOR GROUNDWATER.

NRC EVALUATION (BY FRC) ARGUED THAT 1963-1965 WAS A PERIOD OF GREAT DROUGHT, LOW GROUNDWATER LEVELS. FOR CONSERVATISM, RG&E SHOULD ANALYZE STRUCTURES FOR GROUNDWATER AT GRADE, OR PROVIDE MORE DATA.

RG&E PROVIDED ADDITIONAL BORING DATA, TAKEN IN 1974, SUPPORTING THE EARLIER 250 DBGWL.

NRC ARGUED THAT ONE MORE SET OF DATA NOT ENOUGH. MAINTAINED POSITION THAT DBGWL SHOULD BE TAKEN AT GRADE.

PRESENT RG&E POSITION - DATA SUPPORTS RG&E SUBMITTALS.

- IN 13 YEARS OF OPERATION, NO UNEXPECTED LEAKAGE OF CRACKING NOTED.

PENETRATIONS 201, 209, 308, 311, 312, 315, 316, 319, 320, 323
SERVICE WATER TO CONTAINMENT FAN
COOLERS AND COMPARTMENT COOLERS

- ALL COOLING WATER PIPING AND COOLERS INSIDE CONTAINMENT ARE OUTSIDE MISSILE SHIELD.

- SERVICE WATER PRESSURE ON INLET TO COOLERS IS ABOVE POST-LOCA PEAK PRESSURE (60 PSIG). ON DISCHARGE, PRESSURE IS ~ 15 PSIG, WHICH IS HIGHER THAN CONTAINMENT PRESSURE EXCEPT FOR 2-3 HOURS IMMEDIATELY FOLLOWING ACCIDENT.

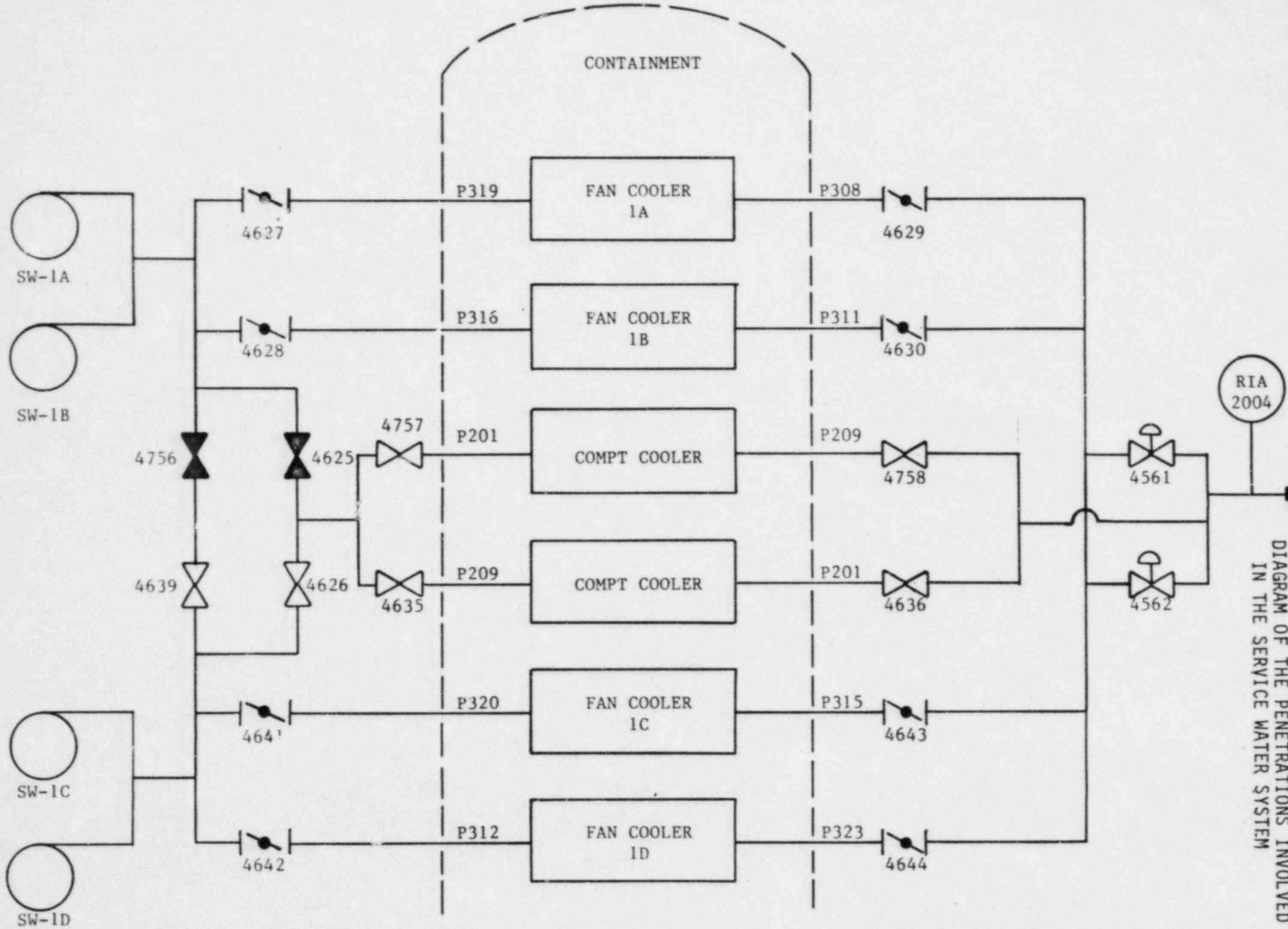
- NO SUBSTANTIAL LEAKAGE EXPECTED IN FAN COOLERS. GOOD OPERATING HISTORY (ONLY ONE INSTANCE OF MINOR LEAKAGE AT CARBON STEEL PLUGS. HAVE BEEN REPLACED WITH COPPER PLUGS).

- MANUAL VALVES OUTSIDE CONTAINMENT ARE ACCESSIBLE. RADIATION FIELD POST-LOCA CALCULATED AT 3 RAD/HR ASSUMING TMI SOURCE TERM. APPENDIX K LOCA SOURCE TERM ABOUT ORDER OF MAGNITUDE LOWER.

- COST OF PROPOSED MODIFICATIONS (REPLACE MANUAL VALVES WITH REMOTE-MANUAL) ESTIMATED AT 1/4 MILLION PER PENETRATION.

PENETRATIONS 201,209,308,311,312,315,316,319,320,323

SERVICE WATER



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APPENDIX XXIX
DIAGRAM OF THE PENETRATIONS INVOLVED
IN THE SERVICE WATER SYSTEM

SEP OBJECTIVES

1. REASSESS THE SAFETY MARGINS OF THE DESIGN AND OPERATION OF SELECTED OLDER OPERATING NUCLEAR POWER PLANTS.

2. ESTABLISH DOCUMENTATION WHICH SHOWS HOW EACH OPERATING PLANT REVIEWED IN THE SEP COMPARES WITH CURRENT CRITERIA ON SIGNIFICANT SAFETY CONSIDERATIONS, AND WHICH PROVIDES A BASIS FOR ACCEPTANCE OF ANY DEPARTURES FROM THESE CRITERIA.

3. PROVIDE THE CAPABILITY TO MAKE INTEGRATED AND BALANCED DECISIONS WITH RESPECT TO ANY REQUIRED SAFETY IMPROVEMENTS.

4. IDENTIFY AND RESOLVE SIGNIFICANT SAFETY DEFICIENCIES EARLY IN THE SEP, IF SUCH DEFICIENCIES EXIST.

5. EFFICIENTLY USE AVAILABLE PERSONNEL AND MINIMIZE NRC AND LICENSEE RESOURCE REQUIREMENTS TO PERFORM THE SEP.

6. PROVIDE SAFETY BASIS FOR CONVERSION OF PROVISIONAL OPERATING LICENSE TO FULL TERM OPERATING LICENSE.

NRC STAFF STATUS REPORT

ON UNRESOLVED SAFETY ISSUE (USI), TASK A-45

"SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

FOR THE

267TH ACRS MEETING

JULY 9, 1982

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ANDREW R. MARCHESE
TASK MANAGER FOR A-45
GENERIC ISSUES BRANCH
DIVISION OF SAFETY TECHNOLOGY, NRR
PHONE: 49-24712

APPENDIX XXXI
NRC STAFF STATUS REPORT ON USI TASK A-45

PRESENTATION OUTLINE

- PURPOSE
- OBJECTIVE
- BACKGROUND ON TASK A-45
- UPDATE ON TASK A-45
- MAIN ELEMENTS OF TASK ACTION PLAN A-45

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PURPOSE

- THE OVERALL PURPOSE OF TASK A-45 IS TO EVALUATE THE ADEQUACY OF CURRENT LICENSING DESIGN REQUIREMENTS TO ENSURE THAT NUCLEAR POWER PLANTS DO NOT POSE UNACCEPTABLE RISK DUE TO FAILURE TO REMOVE SHUTDOWN DECAY HEAT

A-25-8

OBJECTIVES

- TO DEVELOP A COMPREHENSIVE AND CONSISTENT SET OF DECAY HEAT REMOVAL (DHR) SYSTEM REQUIREMENTS FOR EXISTING AND FUTURE LWRs.
- TO EVALUATE ALTERNATIVE MEANS OF DHR AND OF DIVERSE "DEDICATED" SYSTEMS TO DEAL WITH A BROADER SPECTRUM OF TRANSIENT AND ACCIDENT SITUATIONS

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BACKGROUND

- COMMISSIONERS APPROVED SDHR REQUIREMENTS AS AN USI (REF., MEMO, S. J. CHILK TO W. J. DIRCKS, SECY-80-325, DATED DECEMBER 24, 1981
- TASK MANAGER ASSIGNED TO TASK A-45 ON FEBRUARY 17, 1981
- NUREG-0705 (MARCH 1981), "IDENTIFICATION OF NEW USIs RELATING TO NUCLEAR POWER PLANTS - SPECIAL REPORT TO CONGRESS, "PROVIDED AN EXPANDED DISCUSSION OF TASK A-45
- MEMORANDUM, A. R. MARCHESE TO T. E. MURLEY, "ACTIVITIES RELATED TO TASK A-45, "DATED APRIL 8, 1981
- DRAFT TASK ACTION PLAN (TAP) FOR TASK A-45 ISSUED ON MAY 22, 1981
- REVISION 0 OF TAP A-45 (APPROVED BY DST DIRECTOR) ISSUED ON OCTOBER 7, 1981
- REVISION 1 OF TAP A-45 ISSUED ON JUNE 2, 1982

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UPDATE ON TASK A-45 SINCE ACRS FULL COMMITTEE MEETING OF SEPTEMBER 10, 1981

- A TASK ACTION PLAN (REV. 0) FOR USI A-45 WAS ORIGINALLY APPROVED BY DIRECTOR, DST, ON OCTOBER 7, 1981
- THIS PLAN, WHICH AUTHORIZED A FOUR-YEAR PROGRAM WITH A COMPLETION DATE OF OCTOBER 1985, WAS NOT APPROVED BY DIRECTOR, NRR
- WE HAVE REASSESSED THIS PROGRAM TO DETERMINE IF THE PRIMARY GOALS COULD BE REALIZED ON A SHORTER SCHEDULE
- WE HAVE NOW DETERMINED THAT OUR PRIMARY OBJECTIVES CAN BE OBTAINED WITH A 30 MONTH PROGRAM
- WE ESTIMATE THAT A DRAFT NUREG REPORT CONTAINING OUR PROPOSED RECOMMENDATIONS INCLUDING ANY PROPOSED NEW REQUIREMENTS, ALONG WITH THE SUPPORTING TECHNICAL AND COST/BENEFIT BASIS, WILL BE AVAILABLE BY NOVEMBER 1984

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UPDATE (CONT.)

- REDUCED SCHEDULE OBTAINED BY:

  - DELETING MOST OF WORK ON FUTURE PLANTS, ALTHOUGH ACCEPTANCE CRITIERIA FOR DHRS FOR FUTUTE PLANTS WILL BE DEVELOPED

  - QUANTITATIVE ACCEPTANCE CRITERIA WILL BE BASED ON FREQUENCY OF CORE MELT DUE TO DHRS FAILURES RATHER THAN OVERALL RISK

  - RELYING MORE ON INDUSTRY TO PERFORM MORE PLANT-SPECIFIC EVALUATIONS OF ALTERNATIVE DHRS WHERE THE STAFF CAN SHOW SIGNIFICANT IMPROVEMENTS IN SAFETY

  - HAVING ONE CONTRACTOR WITH OVERALL RESPONSIBILITY FOR PROJECT MANAGEMENT, TECHNICAL DIRECTION AND INTEGRATION, INCLUDING SELECTION AND MANAGEMENT OF SUBCONTRACTORS

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UPDATE (CONT.)

- STEPS ACHIEVED TO START WORK ON PROGRAM:

  - RECEIVED APPROVAL BY DIRECTOR, NRR ON MARCH 15, 1982

  - RECEIVED APPROVAL BY SENIOR CONTRACT REVIEW BOARD ON APRIL 9, 1982

  - IMPLEMENTED A CONTRACT ON MAY 3, 1982 WITH SANDIA AS THE LEAD LAB. TO BEGIN WORK & PREPARE A DETAILED PROPOSAL

  - ISSUED REVIEW 1 OF TAP A-45 ON JUNE 2, 1982 THAT IS CONSISTENT WITH THE ABOVE

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MAIN ELEMENTS OF A-45 TASK ACTION PLAN-REVISION 1

- DEVELOP ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - DEVELOP QUANTITATIVE CRITERIA FOR EXISTING PLANTS
  - DEVELOP QUANTITATIVE CRITERIA FOR FUTURE PLANTS
  - DEVELOP QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"
- DEVELOP MEANS FOR IMPROVEMENT OF DHRS
  - PHENOMENOLOGICAL STUDIES
  - CONCEPTUAL DESIGN STUDIES
  - OPERATIONAL ASPECTS OF ALTERNATIVE DHR SYSTEMS
- ASSESS ADEQUACY OF DHRS IN EXISTING LWRs
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ON PROBABILISTIC BASIS
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

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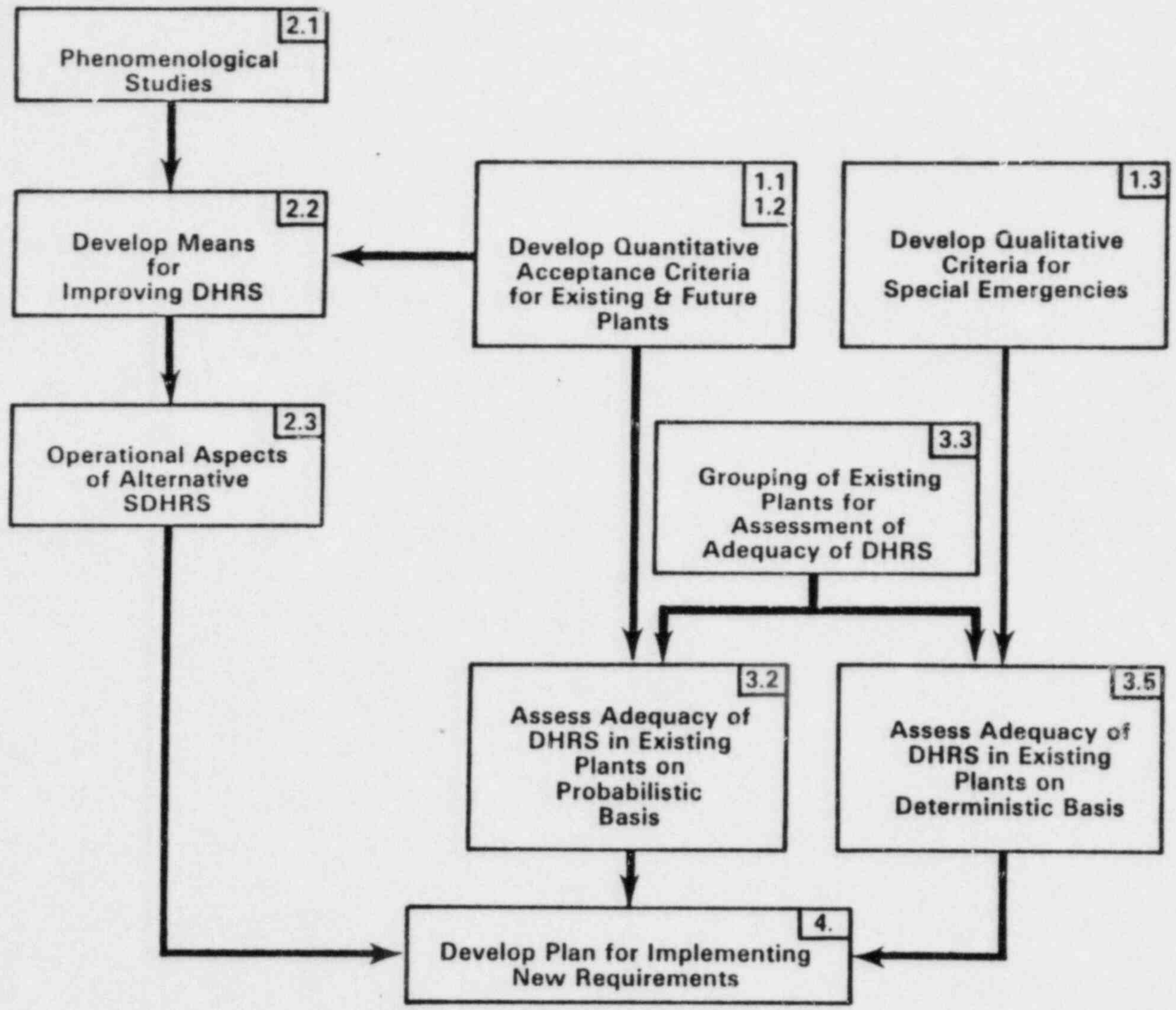

Figure 1. Inter-Relation of Sub-Tasks in Task Action Plan A-45

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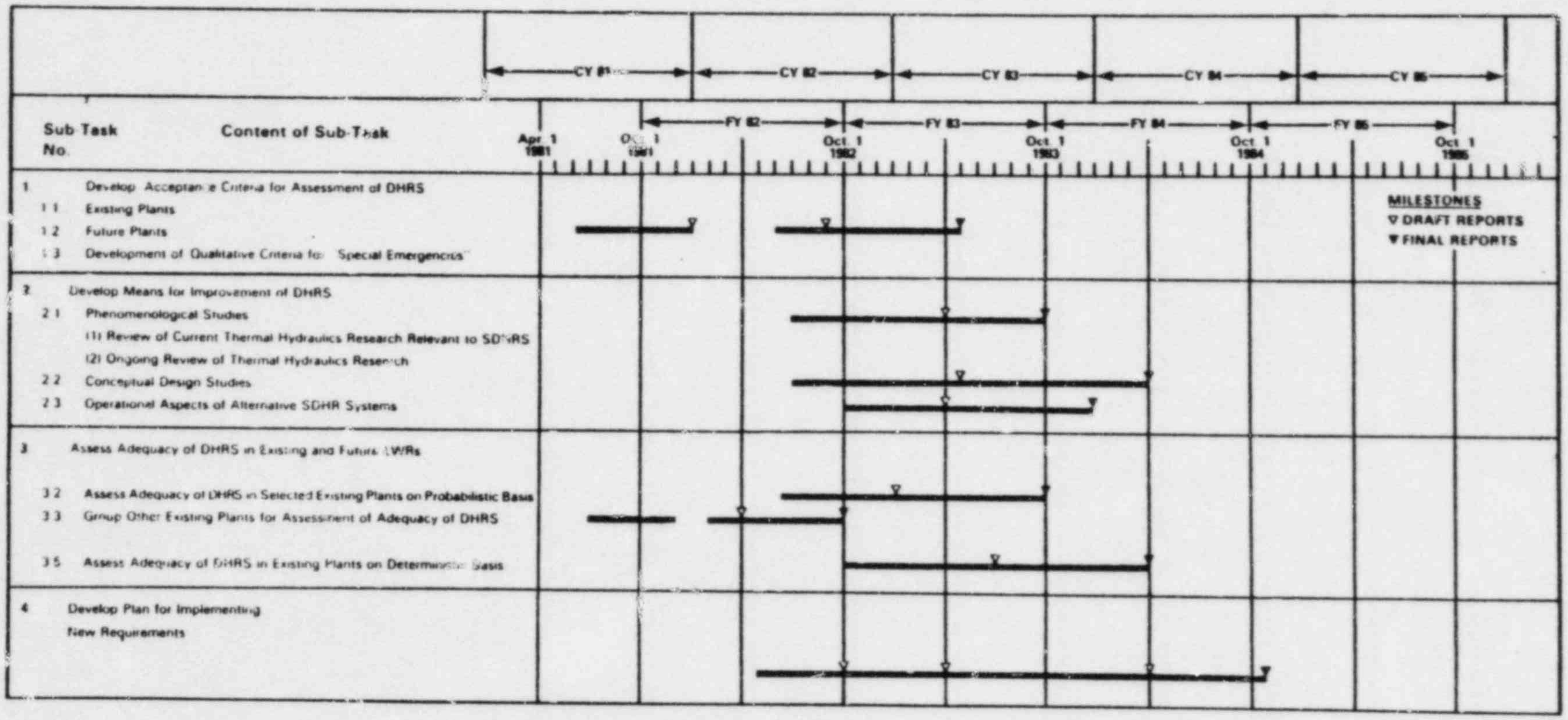
Figure B-1

DETAILED SCHEDULE FOR TASK A-45, "SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"



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BACK - UP

SLIDES

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DEFINITIONS USED IN TASK ACTION PLANT A-45

- REFLOOD PHASE (RFP): THE INITIAL PHASE OF A SEVERE LOCA, WHEN THE OBJECTIVE IS TO REFLOOD THE REACTOR

- SHUTDOWN DECAY HEAT REMOVAL (SDHR) PHASE: THE TRANSITION FROM REACTOR TRIP TO "HOT SHUTDOWN," EXCLUDING THE INITIAL REFLOODING PHASE IN A SEVERE LOCA

- RESIDUAL HEAT REMOVAL (RHR) PHASE: THE TRANSITION FROM "HOT SHUTDOWN" TO "COLD SHUTDOWN" AND MAINTAINING COLD SHUTDOWN CONDITIONS

- DECAY HEAT REMOVAL (DHR) PHASE: SDHR AND RHR PHASES COMBINED

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DEFINITION OF DECAY HEAT REMOVAL SYSTEM

IN THE CONTEXT OF TASK A-45, DHR SYSTEM IS DEFINED AS THOSE COMPONENTS AND SYSTEMS REQUIRED TO MAINTAIN PRIMARY AND/OR SECONDARY COOLANT INVENTORY CONTROL AND TO TRANSFER HEAT FROM THE REACTOR COOLANT SYSTEM AND CONTAINMENT BUILDING TO AN ULTIMATE HEAT SINK FOLLOWING SHUTDOWN OF THE REACTOR FOR NORMAL EVENTS, OFF-NORMAL TRANSIENT EVENTS (E.G., LOSS OF OFFSITE POWER, LOSS OF MAIN FEED-WATER) AND SMALL LOCAs (I.E., 1/2" TO 2"). DHR SYSTEM DOES NOT ENCOMPASS THOSE EMERGENCY CORE COOLING COMPONENTS AND SYSTEMS REQUIRED ONLY TO MAINTAIN COOLANT INVENTORY AND DISSIPATE HEAT DURING THE FIRST 10 MINUTES FOLLOWING MEDIUM OR LARGE LOCAs.

MAIN ELEMENTS OF A-45 TASK ACTION PLAN (Oct.'81)

- DEVELOP INTERIM ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - EXISTING PLANTS
  - FUTURE PLANTS
  - DEVELOPMENT OF INTERIM QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"

- DEVELOP MEANS FOR IMPROVEMENT OF SDHRS
  - PHENOMENOLOGICAL STUDIES
    (1) REVIEW OF CURRENT THERMAL-HYDRAULICS RESEARCH RELEVANT TO SDHRS
    (2) ON-GOING REVIEW OF THERMAL-HYDRAULICS RESEARCH
  - CONCEPTUAL DESIGN STUDIES (GENERIC)
  - OPERATIONAL ASPECTS OF ALTERNATIVE SDHR SYSTEMS

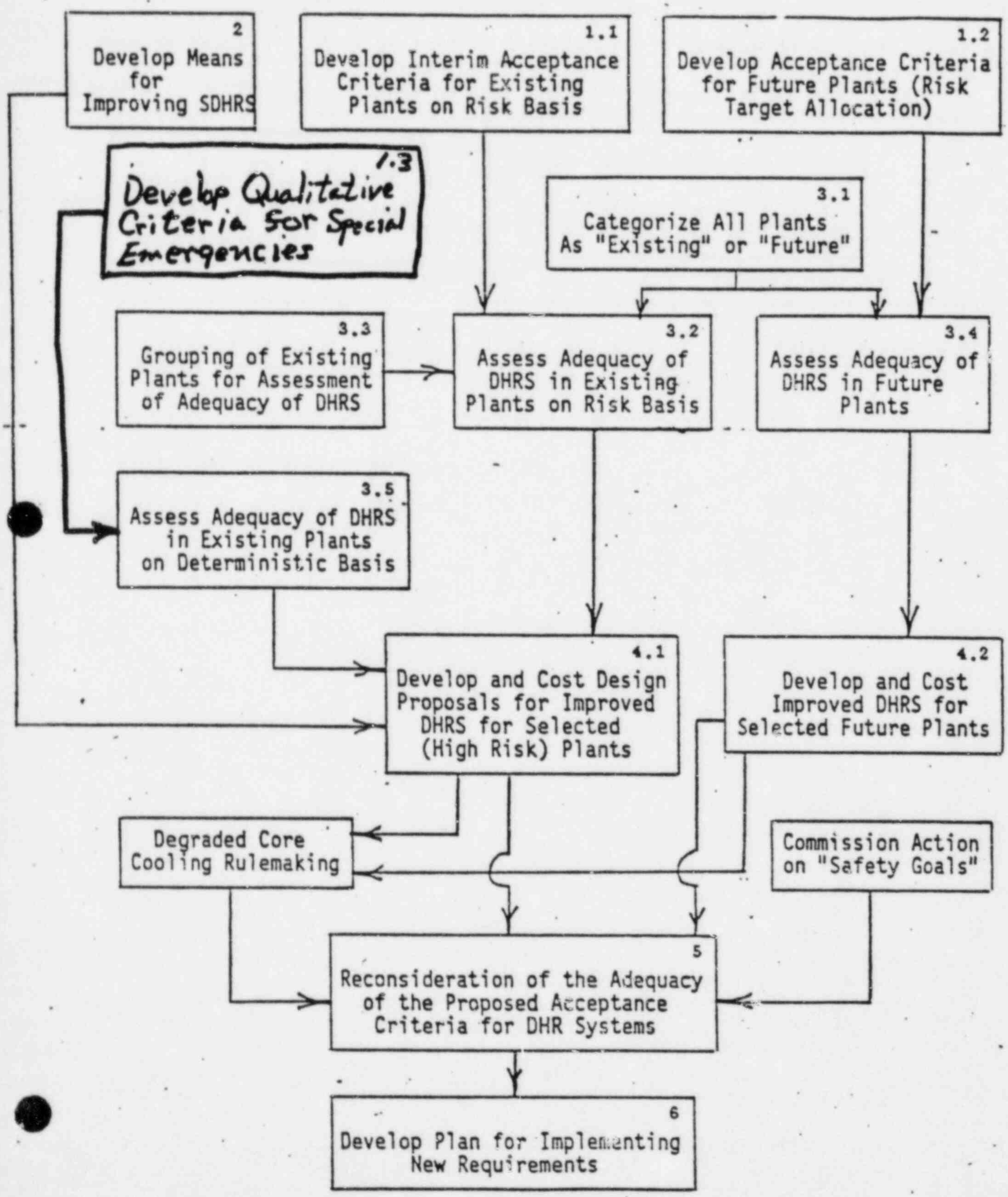
- ASSESS ADEQUACY OF DHRS IN EXISTING AND FUTURE LWRs
  - CATEGORIZE PLANTS AS "EXISTING" OR "FUTURE"
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ON RISK BASIS
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
  - ASSESS ADEQUACY OF DHRS IN SELECTED FUTURE PLANTS
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS

- DEVELOP AND COST IMPROVED DHRS IN SELECTED PLANTS
  - SELECTED EXISTING PLANTS
  - SELECTED FUTURE PLANTS

- RECONSIDER ADEQUACY OF ACCEPTANCE CRITERIA FOR DHRS
  - REVIEW INTERIM DHRS ACCEPTANCE CRITERIA AND TECHNICAL REQUIREMENTS, REVISE IF NECESSARY

- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

Figure 1. Inter-Relation of Sub-Tasks in Task Action Plan A-45 - (Oct. 1982)



Legend: [X] X - Identifies Sub-Task Number

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MAIN ELEMENTS OF A-45 TASK ACTION PLAN (Feb-82)

- DEVELOP ~~INTERIM~~ ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHRS
  - EXISTING PLANTS
  - FUTURE PLANTS
  - DEVELOPMENT OF ~~INTERIM~~ QUALITATIVE CRITERIA FOR "SPECIAL EMERGENCIES"

- DEVELOP MEANS FOR IMPROVEMENT OF SDHRS
  - PHENOMENOLOGICAL STUDIES
    (1) REVIEW OF CURRENT THERMAL-HYDRAULICS RESEARCH RELEVANT TO SDHRS
    (2) ON-GOING REVIEW OF THERMAL-HYDRAULICS RESEARCH
  - CONCEPTUAL DESIGN STUDIES (GENERIC)
  - OPERATIONAL ASPECTS OF ALTERNATIVE SDHR SYSTEMS

- ASSESS ADEQUACY OF DHRS IN EXISTING ~~AND FUTURE~~ LWRs
  - ~~CATEGORIZE PLANTS AS "EXISTING" OR "FUTURE"~~
  - ASSESS ADEQUACY OF DHRS IN SELECTED EXISTING PLANTS ~~ON RISK BASIS~~
  - GROUP OTHER EXISTING PLANTS FOR ASSESSMENT OF ADEQUACY OF DHRS
  - ~~ASSESS ADEQUACY OF DHRS IN SELECTED FUTURE PLANTS~~
  - ASSESS ADEQUACY OF DHRS IN EXISTING PLANTS ON DETERMINISTIC BASIS

- ~~DEVELOP AND COST IMPROVED DHRS IN SELECTED PLANTS~~
  - ~~SELECTED EXISTING PLANTS~~
  - ~~SELECTED FUTURE PLANTS~~

- ~~RECONSIDER ADEQUACY OF ACCEPTANCE CRITERIA FOR DHRS~~
  - ~~REVIEW INTERIM DHRS ACCEPTANCE CRITERIA AND TECHNICAL REQUIREMENTS, REVISE IF NECESSARY~~

- DEVELOP PLAN FOR IMPLEMENTING NEW REQUIREMENTS (E.G., PREPARE NUREG, REG. GUIDE)

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DISCUSSION WITH EPRI ON INDUSTRY INVOLVEMENT IN TASK A-45

- ENCOURAGE INDUSTRY COOPERATION AND INVOLVEMENT IN TASK A-45

- OPTIONS TO CONSIDER:

  - INDUSTRY SETS UP ITS OWN PARALLEL PROGRAM, OR
  - INDUSTRY DOES SPECIFIC PARTS OF A-45 ACTION PLAN (E.G., SUB-TASK 4 ON PLANT-SPECIFIC DESIGN OF ALTERNATIVE DHRS)
  - INDUSTRY PEER REVIEW GROUP FOR TASK A-45 MILESTONE REPORTS

- PRIORITY FOR DEVELOPMENT OF CONCEPTUAL DESIGNS FOR IMPROVED DHRS FOR A SPECIFIC PLANT WILL DEPEND ON:

  1. CORE MELT FREQUENCY DUE TO THAT PLANT AND ON THE EFFECTIVENESS OF IMPROVEMENT OF DHRS AS A MEANS OF REDUCING THAT FREQUENCY, AND/OR
  2. CAPABILITY FOR HANDLING "SPECIAL EMERGENCY" SITUATIONS

Pages A-274 thru ________ has been deleted as deleted I .

Pages A-275 thru A-280 has been deleted as deletion 1.

Radionuclide	Limit of NRC for µCi/kg Ingestion	µCi/cc corresponding to µCi/kg limit of NRC for ingestion (µCi/cc) SV	µCi/cc Limit from ICRP-2 for Ingestion 40 hr/wk (MPC)w	Limit of NRC for µCi/kg Inhalation	µCi/cc corresponding to µCi/kg limit of NRC for Inhalation	µCi/cc [illegible] given by NRC on [illegible] B-3 + B-4 [illegible]	µCi/cc Limit from ICRP-2 for Inhalation 40 hr/wk (MPC)a
H-3	8.1×10^4	0.1, 1	0.1 TB	8.1×10^4	1.1×10^{-5}, 1	2.2×10^{-5}	5×10^{-6} TB
F-18	5.4×10^4	0.07, 1	0.01 GIT	8.1×10^4	1.1×10^{-5}, 1	2.7×10^{-5}	3×10^{-6} GIT
Na-22	5.4×10^2	7×10^{-4}, 1	9×10^{-4} GIT	5.4×10^2	[illegible], 0.75	2.7×10^{-7}	2×10^{-9} L
Na-24	2.7×10^3	[illegible], 1	8×10^{-4} GIT	5.4×10^3	[illegible], 1	2.2×10^{-6}	10^{-9} GIT
P-32	5.4×10^2	1.8×10^{-3}, 0.375	5×10^{-4} B	8.1×10^2 / 2.7×10^2	[illegible], 0.33	2.7×10^{-7} / 1.6×10^{-7}	7×10^{-8} B
P-33	5.4×10^3	1.8×10^{-2}, 0.375		8.1×10^3 / 2.7×10^3	[illegible], 0.33	2.7×10^{-6} / 1.1×10^{-6}	
S-35	1.1×10^4 / 2.2×10^3	[illegible], 0.0013	2×10^{-3} T	1.6×10^4 / 8.1×10^{-7}	[illegible]	8.1×10^{-8}	3×10^{-7} T, L
Cl-36	1.6×10^3 / 2.4×10^2	2×10^{-3} / 3.0×10^{-4}, 1	2×10^{-3} TB	2.4×10^3 / 1.1×10^{-7}	[illegible], 0.75	1.1×10^{-6}	2×10^{-8} L
Cl-38	1.6×10^4	0.02, 1	0.01 GIT	5.4×10^4 / 5.4×10^4	9.9×10^{-6} / [illegible], 0.75	1.6×10^{-5} / 1.9×10^{-5}	2×10^{-6} GIT
Cl-39	2.2×10^4	0.03, 1		5.4×10^4 / 5.4×10^4	9.9×10^{-6} / [illegible], 0.75	2.2×10^{-5} / 2.4×10^{-5}	
Ar-37		–			–	1.9×10^{-4}	
Ar-41		–			–	2.7×10^{-6}	
K-40	2.7×10^2	3.4×10^{-4}, 1		2.7×10^2	3.7×10^{-8}, 1	1.6×10^{-7}	2×10^{-6} SB TB
K-42	5.4×10^3	6.7×10^{-3}, 1	6×10^{-4} GIT	5.4×10^3	7.4×10^{-8}, 1	1.9×10^{-6}	10^{-7} GIT
K-43	5.4×10^3	6.7×10^{-3}, 1		8.1×10^3	1.1×10^{-6}, 1	2.7×10^{-6}	
K-44	2.2×10^4	0.027, 1		5.4×10^4	[illegible], 1	2.7×10^{-5}	
K-45	2.7×10^4	0.034, 1		1.1×10^5	1.5×10^{-5}, 1	4×10^{-5}	
Ca-41	2.7×10^3	6.2×10^{-3}, 0.54		2.7×10^3	[illegible], 0.5	1.6×10^{-6}	
Ca-45	1.6×10^3	3.7×10^{-3}, 0.54	3×10^{-4} B	8.1×10^2	2.2×10^{-7}, 0.5	2.7×10^{-7}	3×10^{-8} B
Ca-47	8.1×10^2	1.9×10^{-3}, 0.54	10^{-3} B	8.1×10^2	2.2×10^{-7}, 0.5	2.7×10^{-7}	2×10^{-7} B, L, GIT
Cr-48	5.4×10^3	6.7×10^{-3}, 1		1.1×10^4	6.0×10^{-6}, 0.35	5.4×10^{-6}	
				8.1×10^3	4.4×10^{-6}	2.7×10^{-6}	
				8.1×10^3	4.4×10^{-6}	2.7×10^{-6}	

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(8)

Comments on Report to Dr. D. W. Moeller from Dr. R. C. Tang, June 9, 1982

This report rightfully points out the improvements brought about by the proposed changes in 20 CFR Part 20 but fails to even mention the all-important fact that this will mean an increase in most of the (MPC) values used by NRC in the past.

I sincerely hope the NRC will reconsider its plans and adopt ICRP-26 and subsequent ICRP-30 reports only insofar as they do not result in less conservatism, i.e., an increase in (MPC) values. The ICRP-26 is based on less supportable values, i.e. Wt and a cut-off of 50 rem/y to avoid non-stochastic effects than is the old ICRP-2 based on the critical organ concept.

I am confident that if this revision of 10 CFR Part 20 is adopted it will result in very serious consequences for the nuclear industry.

Respectively submitted,

Karl Z. Morgan
June 23, 1982

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ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. Memorandum, E. F. Goodwin to R. F. Fraley, Proposed NRR Agenda Items for the August, September and October ACRS Meetings, July 7, 1982

2. ACRS Presentation, Current Status of the FY 84-85 Chairman's Budget, Robert B. Minogue, July 7, 1982

3. Office of Nuclear Regulatory Research Program Budget for FY 1984 and FY 1985 - Table 2, Informational Decision Unit breakdown used as background material