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

April 23, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
265TH ACRS MEETING
May 6 - 8, 1982
WASHINGTON, DC

Thursday, May 6, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. — Opening Session (Open)
- Opening Remarks
- Chairman's Report (PGS/RFF)
  - Topical Subcommittee meetings in support of ACRS report on proposed NRC Safety Research Program Budget for FY 1984-85

2) 8:45 A.M. - 12:45 P.M. — Quantitative Safety Goals (Open)
2.1) Discuss proposed ACRS report to NRC regarding proposed Safety Goals for Nuclear Power Plants (NUREG-0880) (DO/JMG/GRQ)

12:45 P.M. - 1:45 P.M. — LUNCH

3) 1:45 P.M. - 2:45 P.M. — Robert E. Ginna Nuclear Plant (Open)
3.1) Report by NRC Staff regarding repair of steam generator tubes and resumption of operations at the Robert E. Ginna Nuclear Plant
(Note: Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter and information the premature release of which would be likely to significantly frustrate performance of the Committee's statutory function.)

4) 2:45 P.M. - 6:30 P.M. — Wolf Creek Generating Station Unit 1 (Open)
4.1) 2:45 P.M. - 3:15 P.M.: Report of ACRS Subcommittee and consultants who may be present regarding proposed operation of this unit (JJR/RKM/DRB)
4.2) 3:15 P.M. - 6:30 P.M.: Meeting with NRC Staff and applicant
(Note: Portions of this session will be closed as necessary to discuss Proprietary Information applicable to this project.)

Friday, May 7, 1982, Room 1046, 1717 H Street, NW, Washington, DC

5)	8:30 A.M. - 9:30 A.M.	Quantitative Safety Goals (Open) 5.1) Discuss proposed ACRS report to NRC regarding proposed Safety Goals for Nuclear Power Plants (NUREG-0880) (DO/JMG/GRQ)
6)	9:30 A.M. - 12:00 Noon	Emergency Response Capability in Nuclear Power Plants (Open) 6.1) 9:30 A.M. - 10:00 A.M.: Report of ACRS Subcommittee and consultants who may be present regarding proposed requirements for emergency facilities and response capability in nuclear power plants (SECY 82-111, "Requirements for Emergency Response Capability," dated March 11, 1982) (DAW/RKM/DCF) 6.2) 10:00 A.M. - 12:00 Noon: Meeting with NRC Staff
	12:00 Noon - 1:00 P.M.	LUNCH
7)	1:00 P.M. - 1:30 P.M.	ACRS Future Activities (Open) 7.1) 1:00 P.M. - 1:10 P.M.: Anticipated Subcommittee Activities (MWL) 7.2) 1:10 P.M. - 1:30 P.M.: Proposed ACRS activities (RFF)
8)	1:30 P.M. - 5:00 P.M.	Palisades Plant (Open) 8.1) 1:30 P.M. - 2:00 P.M.: Report of ACRS Subcommittee and consultants who may be present regarding the systematic evaluation and integrated reliability assessment for this plant (CPS/RKM) 8.2) 2:00 P.M. - 5:00 P.M.: Meeting with NRC Staff and licensee as appropriate (Note: Portions of this session will be closed as necessary to discuss Proprietary Information related to this plant.)

9) 5:00 P.M. - 6:30 P.M.. Qualification Program for Safety Related Equipment (Open)

9.1) 5:00 P.M. - 5:30 P.M.: Report of ACRS Subcommittee and consultants who may be present regarding the proposed NRC Rule (10 CFR 49) regarding Qualification of Electrical Equipment for Nuclear Power Plants

9.2) 5:30 PM. - 6:30 P.M.: Meeting with NRC Staff and industry representatives as appropriate

Saturday, May 8, 1982, Room 1046, 1717 H Street, NW, Washington, DC

10)	8:30 A.M. - 12:30 P.M.	Preparation of ACRS Reports to NRC (Open/Closed) 10.1) 8:30 A.M. - 10:30 A.M.: Quantitative Safety Goals 10.2) 10:30 A.M. - 11:30 A.M.: Emergency Response Capability at Nuclear Plants 10.3) 11:30 A.M. - 12:30 P.M.: Wolf Creek Generating Station Unit 1 (Portions of this session will be closed as required to discuss Proprietary Information and information which will be involved in an adjudicatory proceeding.)
	12:30 P.M. - 1:30 P.M.	LUNCH
11)	1:30 P.M. - 3:30 P.M.	Preparation of ACRS Reports to NRC (Open) 11.1) 1:30 P.M. - 2:30 P.M.: Palisades Nuclear Plant 11.2) 2:30 P.M. - 3:30 P.M.: Qualification Program for Safety Related Equipment (Note: Portions of this session will be closed as necessary to discuss Proprietary Information related to these items.)
12)	3:30 P.M. - 4:30 P.M.	ACRS Subcommittee Activity (Open) 12.1) Reports of Subcommittee activity regarding: 12.1-1) 3:30 P.M. - 4:00 P.M.: Extreme Environmental Phenomena - Flooding potential/methodology (DO/RS/WB) 12.2) 4:00 P.M. - 4:30 P.M.: LOFT Cooperative Research Program (MSP/PAB) (Note: Portions of this session will be closed to discuss information the premature release of which would be likely to significantly frustrate performance of the Committee's statutory function.)

For the Nuclear Regulatory Commission.

B. J. Youngblood,

Chief, Licensing Branch No. 1, Division of Licensing.

[FR Doc. 82-11229 Filed 4-22-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket Nos. 50-445; 50-446]

Texas Utilities Generating Co., et al., (Comanche Peak Steam Electric Station, Units 1 and 2), (Application for Operation License); Continuation of Evidentiary Hearing

April 19, 1982.

Please take notice that a continuation of an evidentiary hearing will be held in this operating license proceeding before an Atomic Safety and Licensing Board (Board), pursuant to the Atomic Energy Act of 1954 as amended (the Act), and the regulations in Title 10, Code of Federal Regulations (CFR), Part 50, "Licensing of Production and Utilization Facilities," Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," and Part 2, "Rules of Practice." The prior portion of the evidentiary hearing was held December 2, 1981.

This continuation of the evidentiary hearing will commence on June 7, 1982, at 9:00 a.m., local time,[1] at the Fort Worth Hilton Hotel, located at 1701 Commerce Street, Fort Worth, Texas 76101 and will continue until completion of taking evidence on the issues and contentions described hereafter. This evidentiary hearing will address the matters in controversy resulting from Contention 5 (QA/QC), and from Board Questions 1 and 3, if necessary.

A final prehearing conference, pursuant to 10 CFR 2.752, will be held at the same location immediately prior to the resumed evidentiary hearing.

On February 5, 1979, the Nuclear Regulatory Commission (NRC) issued a notice in the **Federal Register** of the "Availability of Applicants' Environmental Report, Consideration of Issuance of Facility Operating Licenses, and Opportunity for Hearing" for Comanche Peak (44 FR 6995). The notice stated that a petition for leave to intervene must be filed by March 5, 1979. Timely petitions were received from the State of Texas for participation as an interested state under 10 CFR 2.715(c), and from Citizens Association for Sound Energy (CASE), Citizens for Fair Utility Regulation (CFUR) and the Texas Association of Community Organizations for Reform Now/West Texas Legal Services (ACORN).

By its Order Relative to Standing of Petitioners to Intervene, entered June 27, 1979, the Board admitted these petitioners as Intervenors in this proceeding. Subsequently, ACORN's motion for its voluntary dismissal as a party was granted by Memorandum and Order entered July 24, 1981. CFUR's motion for withdrawal as a party was granted by an April 2, 1982 Order.

Any person who wishes to make an oral or written statement in this proceeding but who has not filed a petition for leave to intervene, may request permission in writing to make a limited appearance pursuant to the provisions of 10 CFR § 2.715 of the Commission's Rules of Practice. Limited appearances will be permited in this proceeding at the discretion of the Board, but at times, within such limits and on such conditions as may be determined by the Board. Persons desiring to make a limited appearance are requested to inform in writing the Secretary of the Commission, United States Nuclear Regulatory Commission, Washington, D.C. 20555, not later than May 24, 1982. A person permitted to make a limited appearance does not become a party, but may state his or her position and raise questions which he or she would like to have answered to the extent that the questions are within the scope of the hearing as specified above. A member of the public does not have the right to participate unless granted the right to intervene as a party or the right of limited appearance.

Written limited appearance statements may be submitted to the Board at any time prior to closing the record in this phase of the proceeding. Oral statements will only be received at times designated by the Board in order not to interfere with the taking of evidence in this adjudicatory proceeding. Oral limited appearance statements may be made on Tuesday, June 8, 1982, at 9:00 a.m., and at such other times as the Board shall specify. Both oral and written statements will be made a part of the official record of this proceeding.

It is so ordered.

Dated at Bethesda, Maryland, this 19th day of April 1982.

For the Atomic Safety and Licensing Board.

Marshall E. Miller,

Chairman, Administrative Judge.

[FR Doc. 82-11230 Filed 4-22-82; 8:45 am]

BILLING CODE 7590-01-M

[1] Please note that the time for commencement of this hearing has now been advanced to 9:00 a.m., although the Revised Schedule entered March 25, 1982, set the time for 1:00 p.m. on June 7.

Advisory Committee on Reactor Safeguards; Meeting

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on May 6–8, 1982, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published in the **Federal Register** on April 13, 1982.

The agenda for the subject meeting will be as follows:

Thursday, May 6, 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.: Proposed NRC Quantitative Safety Goals (Open)—The Committee members will discuss a proposed ACRS report to the NRC regarding proposed quantitative safety goals to be used in the design, siting, construction, and operation of nuclear power plants.

1:45 P.M.–2:45 P.M.: Robert E. Ginna Nuclear Plant (Open)—The members will hear a briefing from the NRC Staff regarding steam generator tube repairs and restart of the Robert E. Ginna Nuclear Plant.

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

2:45 P.M.–6:30 P.M.: Wolf Creek Generating Station Unit 1 (Open)—The Committee members will hear and discuss the reports of its Subcommittee and consultants who may be present regarding the request of the Kansas Gas & Electric Company, et al. for a license to operate the Wolf Creek Generating Station Unit No. 1.

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

Friday, May 7, 1982

8:30 A.M.–9:30 A.M.: Proposed NRC Quantitative Safety Goals (Open)—The members will continue discussion of a proposed ACRS report to the NRC regarding quantitative safety goals.

9:30 A.M.–12:00 Noon: Emergency Response Capability in Nuclear Power Plants (Open)—The Committee will hear the report of its Subcommittee and consultants who may be present regarding proposed requirements for emergency facilities and response capability in nuclear power plants

(SECY 82-111, "Requirements for Emergency Response Capability," dated March 11, 1982).

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

1:00 P.M.-1:30 P.M.: ACRS Future Activities (Open)—The members will discuss the scope and scheduling of anticipated and proposed Subcommittee and full Committee activities.

1:30 P.M.-5:00 P.M.: Palisades Plant (Open)—The members will hear and discuss the report of its Subcommittee and consultants who may be present regarding the Systematic Evaluation and Integrated Plant Safety Assessment for this plant. Representatives of the NRC Staff, the licensee, and the nuclear industry as appropriate will make presentations and respond to questions.

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

5:00 P.M.-6:30 P.M.: Qualification Program for Safety Related Equipment (Open)—The members will hear and discuss the report of its subcommittee and consultants who may be present regarding the proposed NRC rule 10 CFR 50.49, Environmental Qualification of Electrical Equipment for Nuclear Power Plants.

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

Saturday, May 8, 1982

8:30 A.M.-12:30 P.M.: Preparation of ACRS Reports (Open)—The members will discuss proposed ACRS reports to the NRC regarding matters discussed during this meeting.

Portions of this session will be closed as necessary to discuss Proprietary Information, information which will be involved in an adjudicatory proceeding, and information the premature release of which would be likely to significantly frustrate the performance of the Committee's statutory function.

1:30 P.M.-2:15 P.M.: ACRS Subcommittee Reports (Open)—The members will hear and discuss the reports of designated subcommittees regarding the status of assigned activities including safeguards and security provisions at nuclear power plants and the methodology related to flooding potential at nuclear facilities.

2:15 P.M.-4:00 P.M.: Preparation of ACRS Reports (Open)—The members will complete discussion of proposed ACRS reports to the NRC regarding matters discussed during this meeting.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 30, 1981 (46 FR 47903). In accordance with these procedures, oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Committee, its consultants, and Staff. Persons desiring to make oral 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, information which will be involved in an adjudicatory proceeding (5 U.S.C. 552b(c)(10)), and 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)).

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. EST.

Dated: April 19, 1982.

John C. Hoyle,

Advisory Committee Management.

[FR Doc. 82-11232 Filed 4-22-82; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Subcommittee on Qualification Program for Safety Related Equipment; Meeting

The ACRS Subcommittee on Qualification Program for Safety Related Equipment will hold a meeting on May 5, 1982, Room 762, 1717 H Street, NW, Washington, DC. The Subcommittee will discuss the proposed final version of the rule 10 CFR 50.49, "Environmental Qualification of Electrical Equipment for Nuclear Power Plants", and time permitting proposed rulemaking for the accreditation of qualification testing organizations.

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

The entire meeting will be open to public attendance except for those sessions which will be closed to protect proprietary information (Sunshine Act Exemption 4). One or more closed sessions may be necessary to discuss such information. To the extent practicable, these closed sessions will be held so as to minimize inconvenience to members of the public in attendance.

The agenda for subject meeting shall be as follows: Wednesday, May 5, 1982—8:30 a.m. until the conclusion of business.

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

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

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Dr. Richard Savio or Staff

Issue Date:
September 2, 1982

MINUTES OF THE
265TH ACRS MEETING
MAY 6-8, 1982
WASHINGTON, DC

CERTIFIED

The 265th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H Street N.W., Washington, DC, was convened by Chairman P. Shewmon at 8:30 a.m., Thursday, May 6, 1982.

[Note: For a list of attendees, see Appendix I. H. Etherington was not in attendance at the meeting; W. M. Mathis did not attend the meeting 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 Company, Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. Chairman's Report (Open to Public)

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

A. Nominations for Advisory Committee on Reactor Safeguards

The Chairman informed the Committee of Announcement 82-45 (see Appendix IV) requesting nominations for membership on the Advisory Committee on Reactor Safeguards. He indicated that the announcement solicits candidates with a background in one or more of the following areas:

- Experience in the design, construction, or operation of large, complex facilities and/or surveillance monitoring programs
- Experience with high pressure/high temperature systems or equipment
- Nuclear power plant operations, including the management of an operating organization

Chairman Shewmon announced the formation of a screening panel for selection of candidates for the upcoming vacancy on the ACRS full committee. The screening panel is to consist of J. J. Ray, W. M. Mathis, and D. W. Moeller, Chairman.

B. Review of FY 84-85 Research Program and Budget

The Chairman reminded the Committee of its responsibility with regard to the research report for the FY 1984-85 research program and budget (see Appendix V). C. P. Siess explained to the Committee that the scope and format of the report was yet to be worked out. But, he indicated that the Committee had previously agreed to include detailed comments regarding the program and budget in the February report to Congress with the July report to the Commission confined to the particular budget requests. The report to Congress would then also provide guidance to the NRC Staff regarding the preparation of the next year's budget and the next version of the proposed NRC Long-Range Research Program Plan. The ACRS Executive Director mentioned the results of discussions with R. Minogue and D. Ross regarding the interests of RES, particularly with respect to the proposed NRC Long-Range Research Program Plan. It was agreed that the report scope and content noted by C. P. Siess would be helpful to RES. R. Minogue and D. Ross did indicate, however, that they desired an opportunity in August to discuss the nature and scope of the Long-Range Research Program Plan, not necessarily the contents. D. Okrent suggested that the July report to the Commission should not preclude detailed comments about the research program and budget if Members desire to include them.

R. F. Fraley indicated that another result of the discussions with RES was a commitment by the Committee to consider the out years in the July report.

C. Statement by Nuclear Group on Equipment Qualification

The Chairman informed the Committee that the Nuclear Group on Equipment Qualification had requested time to make an oral statement during the discussion of the proposed NRC rule 10 CFR 49 regarding the Qualification Program for Safety Related Equipment.

D. RES Questionnaire

The Chairman discussed briefly the memorandum from the U. S. House of Representatives, Committee on Science and Technology, Subcommittee on Energy Research and Production. He noted a letter from John F. Duggan, Jr., Staff Director of the Subcommittee on Energy Research and Production, to R. B. Minogue which contained a memorandum and six pages of questions concerning the NRC Research Program (see Appendix VI).

E. Invitation to Testify at Committee on Science and Technology Hearings

The Chairman cited a press release by Congresswoman M.L. Bouquard dated April 27, 1982 announcing hearings of the Subcommittee on Energy Research and Production of the U.S. House of Representatives Committee on Science and Technology. The ACRS has been invited to send representatives to this hearing, The ACRS Chairman and C. P. Siess have been designated. He mentioned a letter from the Committee on Science and Technology to C. P. Siess dated April 30 which contained five discussion items of particular importance to the Subcommittee on Energy Research and Production (see Appendix VII).

II. NRC Staff Report Regarding the Robert E. Ginna Nuclear Plant (Open to Public)

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

J. Lyons, NRC Project Manager for the Ginna Plant, reviewed NUREG-0909, NRC Report on the January 25, 1982, Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant, and presented a brief background of the event, the event followup, and the current plant status (see Appendix VIII). He indicated that the Chairman of the NRC had set up a task force to investigate the event. The Staff-prepared factual report which serves as a data base for the event, has been sent to all plant licensees.

Technical issues being evaluated by the NRC Staff prior to restart of the Ginna Plant relate to

- Steam Generators
- Power Operated Relief Valves (PORV)
- Procedures/Human Factors
- Thermal Transient and the Reactor Vessel

J. Lyons indicated that the Licensee has performed many evaluations and has enlisted Westinghouse, Battelle Columbus Laboratories, EPRI, and Combustion Engineering to do various analyses for them on the failure mechanisms of the steam generator and other systems that were affected during the event. It was mentioned that Rochester Gas and Electric had installed a loose parts monitor after the incident. It was pointed out that most nuclear plants do not have loose parts monitors.

J.Lyons used a Ginna steam generator hot leg tube sheet map and other diagrams to explain the failure mechanism for the incident as well as inspection and repairs being done on the steam generator preparatory to restart of the plant. The Committee discussed the characteristics of the Ginna steam generator internals. Several Members expressed interest in axial loads, the process of tube plugging, locations of foreign objects found near the tube sheet, and the results of metallurgical examinations of damaged tubes. J. Lyons presented a series of color slides which depicted the tube rupture and the degradation of tubes in the general vicinity of the ruptured tube.

When a color slide of the ruptured tube was shown, D. A. Ward requested an explanation as to why the flow from this break was more than double-ended flow. G. Holahan, NRC Staff, indicated that the estimated flow is about what would be expected from a double-ended tube rupture which the Staff estimated at 750 gpm. He explained that the Licensee had estimated about 600 gpm, while the FSAR value for a double-ended steam generator tube failure is about 800 gpm.

G. Lainas, NRC Staff, pointed out that several steam generator tubes have been plugged at Ginna over a period of years. The debris that was found in the steam generator after the Ginna accident was the result of steam generator modifications which were made in 1975. D. W. Moeller questioned whether there were any changes in the plugging operations over that seven year period which might have led to the Ginna transient. J. Lyons explained that the degradation that led to the Ginna incident was probably a slow progression since leakage had been detected on a number of occasions in tubes adjacent to the tube that ruptured. Tubes that leaked were plugged approximately eight months to a year after adjacent tubes had been found leaking and had been plugged.

J. Lyons explained that as far as the area of human factors is concerned, the actual event was not covered in any single procedure that Rochester Gas and Electric had at that time. The Licensee used a number of procedures in order to handle the incident. J. Lyons indicated that the NRC Task Force concluded that the procedures, coupled with the training and plant staff experience, provided effective response to the event. Nevertheless, J. Lyons pointed out that the Licensee has made some changes to its procedures since the event. These changes facilitate inproved communications during plant emergencies.

J. Lyons outlined the Staff's future actions with regard to the incident (see the last page of Appendix VIII). D. Crutchfield, NRC Staff, indicated that there could be debris on the secondary side of steam generators at other plants. He added that part of the Staff's generic evaluation and conclusions will be to consider whether secondary side inspections ought to be done with video or fiber optics techniques to detect any debris which might be in a steam generator.

III. Operating License Review of the Wolf Creek Generating Station Unit 1

(Open to Public)

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

A. Report of the ACRS Subcommittee

J. J. Ray summarized the April 21-22, 1982 Subcommittee Meeting in Emporia, Kansas. A 5000 acre manmade lake is the source of cooling water. A corner of the lake is contained by a category I underwater dam which serves as the ultimate heat sink. He explained that should the main dam be lost, there would be a dam that would be uncovered that would bottle up in one portion of the lake sufficient water to continue to cool the plant.

J. J. Ray suggested omitting a formal presentation on a.c./d.c. power systems for Wolf Creek. He personally addressed the subject by discussing offsite power supply reliability and the switch yard. It was pointed out that the lines approaching the substation are well separated such that there is no congestion in the sense of right-of-way availability. Also pointed out was the fact that there are four 345kv lines which enhance the reliability of offsite power from the point of view of a.c. blackout potential. J. J. Ray pointed out that the Wolf Creek system has black start capability with gas turbines. These gas turbines can respond very quickly to help restore other electric power stations and provide power to Wolf Creek systems should they be separated from a power grid.

J. J. Ray indicated that two individuals are intervening in the licensing of this plant on the basis of emergency preparedness, and one group is intervening on the basis of financial qualifications. W. Kerr added that the Committee should pay particular attention to Kansas Gas and Electric's approach to the requirements of an STA and to the changes in the Staff approach to the required seismic analysis for the Wolf Creek Station. W. Kerr indicated that ACRS consultants thought differences between the construction permit requirement and the newer operating license stage seismic spectrum trivial and felt that reanalysis was unnecessary.

B. Description of Wolf Creek Plant

G. L. Koester, Vice President of Nuclear for Kansas Gas and Electric, explained the multiple ownership of the Wolf Creek project with Kansas Gas and Electric (KG&E) (owning 47%) the lead company during construction, and the operator during the operational phase of the faculty. G. L. Koester described the plant site, noted differences between the Wolf Creek and Callaway Plants (from a slide) and briefly discussed the project schedule (see Appendix IX). J. Ebersole expressed concern with the design of the ultimate heat sink, an underwater Category 1 dam. The Committee discussed the design and margin for the underwater dam with L. L. Holish of Sargent and Lundy. L. L. Holish explained that the dam was designed for uniform overtopping by placing armorplated riprap across both the upstream and downstream face, with bedding material below that. The dam itself was built out of impervious clay. D. W. Moeller requested some reassurance on the capabilities of this dam from the Staff, but the Staff was unable to answer the question. D. Okrent questioned the margin of the dam to withstand an earthquake load.

G. L. Koester briefly discussed KG&E's relationship with the SNUPPS' organization which will continue after operation. The SNUPPS' Design Assurance Program was mentioned as well as the Wolf Creek association with INPO.

D. Okrent asked whether KG&E was involved in a cooperative program with the British in their probabilistic risk assessment of a SNUPP's plant. G. Rathbun of KG&E indicated that they are not involved in the British PRA studies. But, he did indicate that KG&E is reviewing inhouse two studies done for the British by American consultants. D. Okrent noted changes to the SNUPP's PWR design that the British intended to make. F. Schwoerer of SNUPPS indicated that the British have different criteria for quantitative safety goals to deal with reactors located much closer to populated areas. He outlined a few of the design differences as follows:

- Two diesel generators out of four - increase in reliability of onsite a.c. power system
- Two turbine driven d.c. controlled charging pumps added - capability of providing water to the reactor cooling pump seals
- Secondary containment structure - better handle on leak rate from a containment structure
- Westinghouse Integrated Protection System, digital computer-based.

D. Okrent asked G. L. Koester if quality assurance audits uncovered design or review problems associated with construction of the Wolf Creek facility. E. Creel, QA Manager for KG&E, indicated that in the process of performing audits of design processes, programmatic type problems had been found but no design errors.

C. NRC Presentation of Status of Plant Review

J. B. Hopkins, NRC Project Manager for Wolf Creek, indicated that this plant is essentially a duplicate of the Callaway Unit 1 with the exception of certain features. The portions of the Wolf Creek Plant outside the scope of the duplicate plant design were shown in a slide (see Appendix X). J. B. Hopkins reviewed 11 open items from the safety evaluation report. D. Okrent asked if the Staff had done anything in the area of systems interaction in their review of the Wolf Creek Plant. G. Edison of the Staff indicated that systems interactions were handled thoroughly at Callaway and therefore, for the standard part of the plant, the SNUPPS part, the Staff did go into systems interactions in a detailed manner. D. W. Moeller asked the NRC Staff whether a reactor operator in the Wolf Creek control room could override the control room isolation valves which block the air inlets in the event of smoke detection in order to clean the smoke from the control room atmosphere. G. Rathbun offered to respond to this matter at a later date. J. Ebersole questioned the method of isolation with dampers, such as rooms which are isolated by damper arrangements which are contained within the ducting systems. G. Rathbun indicated that dampers have fusible links.

G. Rathbun summarized the list of open items, confirmatory items, and license conditions from the point of view of the Applicant. He indicated KG&E's strong convictions with regard to their STA Program. He explained that providing the STA training to a person responsible for the in-line operation of Wolf Creek and upgrading of the control room would be the long-term solution for this deficiency identified at Three Mile Island.

G. P. Rathbun declared that with regard to license conditions, KG&E does not believe the NRC's proposed surveillance program is technically justified with regard to the design change of going from silver indium-cadmium to hafnium control rods. In answer to a question by P. G. Shewmon, G. B. Rathbun indicated that Comanche Peak is the lead plant using hafnium control rods. P. G. Shewmon inquired as to the Staff's position regarding inservice inspection of the hafnium control rods at Wolf Creek, and in particular, whether this inspection is on the critical path for the plant outage. G. Edison offered to respond regarding the definition and objectives of the visual inspection procedures at a later date.

G. L. Koester displayed a management organization chart and a table of education and experience level of his staff. D. Okrent asked if KG&E had individuals with a reasonable amount of background in the area of PWR systems behavior analysis. G. L. Koester cited his Director of Nuclear Operations, T. D. Keenan and the plant superintendent as having had considerable PWR experience. In response to further questioning, G. L. Koester indicated that there was no one on the KG&E staff with the background of J. Cermac of SNUPPS. G. Rathbun of KG&E added that a safety analysis group is being set up under the management of the nuclear services group in order to develop capability in the area of thermal hydraulic behavior of PWR systems. D. Okrent inquired if there was expertise in the area of probabilistic risk assessment on the KG&E staff. G. Rathbun cited the newly formed Safety Engineering Group.

G. L. Koester discussed the Nuclear Safety Review Committee (NSRC) which reports directly to him, which has nine members from disciplines such as operations engineering, quality assurance and metallurgy, six KG&E people and three outside participants. He mentioned a tenth member of the committee which will be the manager of safety engineering who will also serve as secretary to the NSRC. G. L. Koester also described a Quality Assurance Committee composed of the Vice-President of Nuclear and four other committee members, with three of the committee members outside of the nuclear department. He indicated that it is the job of this committee to measure the effectiveness of the Quality Assurance Program for KG&E and make necessary changes to adjust the effectiveness of the program to a desired level.

T. Keenan, Director of Nuclear Operations for KG&E, briefly discussed the Wolf Creek operating philosophy in the context of balancing quality assurance and nuclear safety. F. Rhodes, Plant Superintendent at Wolf Creek, presented the status of the Wolf Creek site organization. D. W. Moeller expressed concern for the educational background of the health physics supervisors and M. Nichols, the Manager, who will supervise these HP supervisors. J. J. Ray inquired whether consultants included on shift in the operations experience base will have hands-on responsibility for operating the plant. F. Rhodes indicated that they will serve primarily as consultants to the shift supervisor and not serve as ROs.

F. Rhodes briefly discussed the Plant Safety Review Committee (PSRC) whose purpose is to advise the Plant Superintendent on all nuclear safety related matters. He indicated that this committee was set up early on to review and approve all safety related procedures and the start-up program. D. W. Moeller expressed an interest in the interaction between the PSRC and the Nuclear Safety Review Committee. F. Rhodes indicated that they do not hold joint meetings, but that the minutes of the PRSC are submitted to the NRSC for their review.

F. Rhodes briefly described the training program at Wolf Creek as divided into two basic areas, the Licensed Operator Training Program and the Non-Licensed Operator Training Program. At the conclusion of the presentation, D. Okrent asked the Staff how they judge when an adequate degree of technical support capability is present on or off site. E. Johnson of the Staff indicated that in the review of the Licensee's capabilities, account was taken that KG&E is not a large organization, and fairly new to the nuclear business. He added that the Staff considered compensating factors such as outside contract help - contracts with SNUPPS, Westinghouse, and Bechtel. R. Benedict of the NRR Licensing Qualifications Branch, indicated that the Staff does not have firm criteria for what might be required in the way of detailed knowledge because particular events which might occur at a plant cannot be easily predicted.

R. Wescott of NRR Hydrologic and Geotechnical Engineering Branch discussed the performance characteristics of the ultimate heat sink (UHS) dam should the main dam fail at Wolf Creek. [R. Wescott indicated that the Staff did hand calculations using conservative assumptions of instantaneous failure of the main dam. They determined a maximum velocity close to the 9.5 ft. per second calculated by the Applicant for water that overruns the dam as it comes out of the main lake.] He concluded that the Applicant's analysis was conservative. D. W. Moeller asked whether the Staff's judgment was corroborated by experimental data. R. Wescott indicated that the Staff work was based upon a technical paper by the Army Corp of Engineers which experimentally evaluated the effects of overtopping of different dams with different amounts of riprap protection and filter blankets. The Committee discussed the construction characteristics of this dam under earthquake loadings with B. Jagannath, NRR Engineering Branch. D. Okrent was particularly interested in the margin of safety for the UHS dam for a .2g or greater earthquake.

The Applicant responded positively to an earlier question by D. W. Moeller concerning isolation of the control room because of smoke. An earlier question by J. Ebersole with regard to the main steam isolation valve and their capability to close was answered by the Applicant with a description of actuation design and verification testing on these valves. G. Rathbun also explained that the fusible link temperature on fire dampers is approximately 50^0 above the normal operating temperature for the rooms in question. J. Ebersole still expressed his concern that the circuit breakers in the protected rooms might not remain functional because the fusible link melts at a temperature which is probably very close to the failure point in the circuit breaker instrumentation.

D. Use of Shift Technical Advisors (STAs)

R. C. Coulthard, KG&E Manager of Nuclear Training, described a college level training program set up to qualify people to serve as Shift Technical Advisor. He mentioned a technical paper presented at the Tenth Biannual Topical Conference on Reactor Operating Experience in Cleveland, Ohio, in August 1981 (see Appendix XI). T. D. Keenan of KG&E discussed the simulator training programs and history of the Shift Technical Advisor (STA) concept from KG&E's point of view. He indicated that it was KG&E's assumption that the Shift Technical Advisor position was a temporary one which was eventually to be upgraded. T. D. Keenan quoted excerpts from several documents, including NUREG-0578, NUREG-0737, NUREG-0660, ANS-3.1 Draft Rev. 4 10/81, etc., with regard to the temporary nature of the STA position and KG&E's intent in meeting the requirement for the STA position through its Shift Technical Advisor Academic Program (see excerpts in Appendix IX). W. M. Mathis commented that he did not see much difference between STA and the position of Operating Consultant that KG&E had created. T. D. Keenan explained that these two positions were completly different in that the Shift Operating Consultant is primarily an advisor from the perspective of long-term plant operation to provide additional operating experience for the first year. He added that the independent STA is often an inexperienced person with no operating experience and merely academic training who is supposed to provide advice and consultation to an experienced operator. He indicated that KG&E finds this very objectionable from a human engineering prospective. R. L. Tedesco of the NRC Staff explained that the STA was conceived to be a person who would provide improved technical capability to the operating crews. He would not be involved in command functions provided to the Shift Supervisor. It is the STA who would also be used as a liaison with a number of groups involved with accident management involving the Technical Support Center. The Committee discussed the STA position with R. L. Tedesco. M. W. Carbon expressed his support of the Staff's approach and intent in regard to creation of the Shift Technical Advisor position.

E. Seismic Design of Plant and Equipment

P. Sobel, NRC Staff Seismologist, explained that the Staff concluded at the CP stage that an acceleration of .12g was not appropriate for the Wolf Creek site specific structures. The standard SNUPPS portion of the plant is constructed to an 0.2g SSE. She indicated that the Staff reexamined, during the OL review, the maximum earthquakes associated with the central stable region and the Nemaha Uplift. The Applicant elected to perform a seismic reevaluation of all Category 1 non-SNUPPS structures, using the 0.15g site specific spectrum and using the Lawrence Livermore 84th percentile site specific spectrum calculated for a magnitude 5.25 local earthquake.

C. J. Sprout, Technical Staff Engineer for KG&E mentioned a meeting with the Staff during which KG&E proposed and the Staff agreed that the evaluation would be based upon a Regulatory Guide 1.60 spectra anchored at .15g. She mentioned a March 25 report detailing the results of equipment exceedence evaluations for the ultimate heat sink whose results were accepted by the Geotechnical Branch of NRC as indicated in the SER. She added that the evaluation for the remainder of the essential service water structures had just been submitted to the NRC this week, the results of which show that the design of these structures is not affected by the 0.15g SSE. G. Rathbun added that KG&E has evaluated the ultimate heat sink dam and found it had adequate margin. He added that KG&E has evaluated plant structures and has found margin in the structures. He indicated that KG&E has not completed its evaluation of the equipment qualification items and is proceeding with that at this time. J. J. Ray asked whether the Applicant was concerned about a potential major expenditure associated with continuation of the seismic analysis. G. L. Koester indicated that KG&E was concerned with a major expenditure, but, suggested that they had already complied with the requirements from the NRC and had a built-in conservatism in their analytical method. D. Okrent asked if there were any equipment that might not be able to meet the seismic design basis at .15g. C. J. Sprout indicated that KG&E is looking at equipment associated with the service water pump house, the motor control centers, and traveling screens which are in the essential service water pump house.

F. Control Room

D. A. Ward asked KG&E whether they were in compliance with NUREG-0700, and whether they had done a task analysis. J. M. McKinstray, KG&E, indicated that a task analysis had not been done and certain areas such as environmental aspects of the control room and the communications area had not been checked because they were incomplete. F. Schoerer of SNUPPS indicated that the Westinghouse Owners Group is doing a task analysis specifically directed to the emergency procedures which will probably be applicable to the SNUPPS plants. He added that KG&E would be doing its own task analysis for the normal operating procedures which differ from plant to plant. J. M. McKinstray, in answer to a question from D. A. Ward, indicated that KG&E is not buying the Westinghouse SPDS, but has formed a Subowners Group of Westinghouse PWRs and contracted with Quadrex to develop software which is in effect a subset of the SAS system. D. A. Ward inquired whether the upgrade in the control room at Wolf Creek would fulfill the requirements to eliminate the need for an STA. G. Edison of NRC indicated that it was his personal opinion that KG&E was doing the proper upgrade to its control room but suggested that the matter of the STA was a separate item, subject to deliberations with the Staff.

G. Preparation of Emergency Operating Procedures

D. A. Ward asked the Applicant if he were participating with an Owners Group in the INPO coordinating effort for writing Emergency Procedure Guidelines. J. Zell, KG&E, indicated that they are participating with the Westinghouse Owners Group in the preparation, validation and use of the Emergency Procedure Technical Guidelines. J. Zell also indicated that Human Factors Guidelines are being handled in concert with the INPO effort.

H. Radiation Protection Program

D. W. Moeller suggested to KG&E that plant management work with the Health Physics Staff to try to keep the source term down to a reasonable level by working on decontamination procedures. D. W. Moeller pointed out certain inconsistencies in Table 11.4 on page 11-5 of the SER for the Wolf Creek Generating Station Unit 1 concerning liquid effluent dose design objectives for the plant and the site, radioiodines and other radionuclide releases to the atmosphere and the activity release rate for iodine 131. J. B. Hopkins, NRC Project Manager for Wolf Creek, offered to examine the apparent discrepancies in the SER table and respond at a later date.

I. Emergency Planning

D. A. Ward expressed interest in the Technical Support Center and what sort of environmental protection had been designed into it. R. F. Lewis, KG&E, indicated that the Technical Support Center would have the same accident protection factors as the plant control room. He indicated that there would also be installed fixed and portable survey equipment and provisions for protective clothing. D. Okrent asked whether the Westinghouse Owners Group had considered an emergency operating procedure to deal with earthquake-induced emergencies. F. Rhodes, the Plant Superintendent, indicated that the Owners Group was developing symptom-oriented procedures such that a specific accident would be found through the analysis procedure. He added that they are not preparing a specific guideline for earthquakes.

D. W. Moeller asked whether the Wolf Creek Plant had an emergency supply of potable water for drinking and sanitary facilities. G. Rathbun indicated that KG&E has potable water supplies and food available to operators and stored rations for the control room. P. G. Shewmon expressed interest in whether PG&E had noted any bolting fractures or problems with high strength bolts. The Applicant indicated that the high strength bolts used in the plant are ASTM A.194 (100,000 psi) specification bolts. It was indicated that KG&E had not experienced any problems with bolting fractures. D. W. Moeller cited intervenors' concerns with evacuation and questioned when FEMA will finish its assessment of the emergency plan for Wolf Creek. C. R. Van Niel from the NRC Division of Emergency Preparedness indicated that the State and local plans are about 90% complete and the State should be submitting their plan to FEMA for review shortly.

The Committee agreed that it could write a letter on the Wolf Creek Generating Station Unit 1. M. W. Carbon and D. Okrent expressed concern about the Staff and Applicant seismic evaluations of the ultimate heat removal capability.

IV. Palisades Plant Systematic Evaluation Program Review (Open to Public)

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

[Note: W. Kerr did not participate in the review of the Palisades Plant.]

A. Report of the ACRS Subcommittee

C. P. Siess indicated that the review of the Palisades Plant should result in advice to the Staff and Commission on essentially two subjects:

- Has the Systematic Evaluation Program been conducted so as to meet the objectives set out for it?
- Specific application of the SEP to the Palisades Plant.

C. P. Siess discussed the subcommittee meeting held on April 15, 1982 attended by D. A. Ward and himself in addition to ACRS consultants Catton, Lipinsky, and Fitzsimmons. Also mentioned was NUREG-0820, The Integraded Plant Safety Assessment of the Systematic Evaluation for Palisades, and a handout which contained the report of five consultants that the Staff asked to review their Palisades SER. (The report of the five consultants to the Staff is included in Appendix XII). C. P. Siess discussed the agenda for the Palisades review by the ACRS and noted that there were no formal responses item-by-item from the Applicant (see Appendix XIII).

C. P. Siess pointed out that the review for Palisades was incomplete in the sense that SEP topics identical with unresolved safety issues or TMI items that were under review by others on the Staff were deleted from the SEP review of Palisades in order to avoid duplication. In addition, he indicated that there are nine open items which require more data from the Applicant. C. P. Siess indicated that the plant is still operating with a provisional operating license. The granting of a full-term operating license may be discussed but can not be considered until the unresolved safety issues and TMI issues are fully addressed. The operating history of the Palisades Plant as reflected in LERs and its past poor regulatory performance was mentioned.

B. Overview of SEP

W. T. Russell, Chief of the Systematic Evaluation Program Branch in NRR, discussed the objectives of the three phases of the SEP and listed the objectives of the program (see Appendix XIV). He mentioned that the SEP/Integrated Assessment will be part of the basis necessary to convert plants with provisional operating licenses to full-term licenses. J. Ebersole noted a statement on one of the viewgraphs indicating that topics were deleted because they were not normally included in the review of light-water reactors. W. T. Russell indicated that these were generic issues related to environmental or NEPA issues. He explained that they were deleted because they were of lesser safety significance but were reviewed and presented before the Commission with safety evaluations. D. Okrent expressed concern that the Staff was not reviewing management structure and technical capability for the SEP plants. C. P. Siess reminded the Committee that the systematic evaluation program was primarily related to the physical design of the plant. He indicated that Inspection and Enforcement had done separate studies of management for all the plants. W. T. Russell explained that issues such as systematic appraisal of licensee performance and technical support capability were addressed. D. Okrent contended that these were different from management as a topic. W. T. Russell indicated that issues of onsite technical support were identified in 1977 as appropriate for review under conduct of operations even before the development of the TMI Action Plan. It was for this reason that no additional review for the SEP conduct of operations was done. C. P. Siess pointed out that while some TMI items have been eliminated as SEP topics, there are other TMI items that are new licensing issues that are being covered for all plants but are not part of the SEP program. W. T. Russell assured the Committee that the Staff will address the status of all TMI items, not just those related to the SEP plants particularly.

R. Vincent of Consumers Power (CPCO) briefly described the Palisades Plant and its history from the beginning of construction including application for conversion to the full-term operating license in January, 1974 (see Appendix XV). The Committee questioned the ultimate heat sink aspects of the design, CPCO's experience with steam generator problems, and identified problems with cracking in the turbine discs.

C. Staff Discussion of Seismic Review and Residual Heat Removal System Reliability

W. T. Russell described the seismic review of the Palisades Plant with regard to the general philosophy and scope, the overview of review approaches, determination of seismic hazard, the bases for reevaluation, and the conclusions (see Appendix XVI). He indicated that the NRC used Regulatory Guide 1.60 spectra anchored at 0.2g and showed that the design was appropriate and the plant seismically qualified at that level. He indicated that the site specific spectra developed by a uniform hazards approach is a probabilistic type approach for a typical soil site like Palisades. In answer to a question by D. Okrent, W. T. Russell indicated that all the systems related to decay heat removal including the auxiliary feedwater system were subject to a complete analysis including the buried piping for the systems. In answer to an inquiry by J. Ebersole, W. T. Russell indicated that batteries were found to have inadequate anchorage and all batteries as well as other electrical equipment were ordered reanchored.

D. Okrent asked whether a March 16, 1982 research information letter Number 130 from R. B. Minogue was taken into account by the Staff in its evaluation of estimated piping motions and strain in supports. W. T. Russell indicated that Palisades had been evaluated against current criteria in the piping and support areas. He added that if this research letter had called into question current criteria, then restudy of the matter would have to take place.

W. T. Russell explained that review of the residual heat removal reliability was combined with other topics involved with safe shutdown. He indicated that the Staff was concerned with trying to determine whether these older plants had the capability to get to cold shutdown using only safety grade equipment (see Appendix XVII). It was indicated that this topic was done at an early stage because it was a key topic in identifying aspects that were important with regard to backfits on the Palisades Plant. W. T. Russell indicated that the conclusion of the RHR review for Palisades was that the shutdown cooling system did have susceptibility to single failures, but the matter was resolved by the addition of two new pumps. It was also found that the condensate storage tank inventory as far as pure water was concerned was not adequate to allow the plant to remain at hot shutdown for four hours and then get to a point where the operators could initiate shutdown cooling.

D. Okrent expressed concern about Palisades' auxiliary feed water reliability. W. T. Russell indicated that the conclusion in this area was that a third auxiliary feedwater pump was required in an independent room because it was susceptible to single failures in the space it currently occupies. He added that the Applicant has proposed using a spare high pressure injection pump.

In answer to a question by Chairman Shewmon, W. T. Russell indicated that the Staff did not look at any issues related to high-strength bolts as far as the SEP is concerned.

C. P. Siess mention that he could not find RHR shutdown topics on a list of items that met current criteria handed out during the presentation (see Appendix XVIII). W. T. Russell indicated that they were covered in Chapter 4 of NUREG-0820.

D. Limited Risk Analysis

M. Rubin, Reliability and Risk Assessment Branch for Safety Technology for NRR, explained that the Staff had conducted a limited risk study in support of the integrated assessment program for Palisades. He indicated that since there was no plant specific probabilistic risk assessment of Palisades, another Combustion Engineering plant was utilized, Calvert Cliffs. He explained that the first part of the study using Palisades FSAR data and system specific details was to obtain some idea of the effect of the SEP on the issue of plant system availability. The second portion of the program was to use the Calvert Cliffs PRA to derive an estimation of the importance of specific systems and the importance of those systems to risk. D. Okrent questioned in what sense the systems were important and whether Calvert Cliffs was sufficiently representative of the importance of Palisades' systems to risk. W. T. Russell indicated that they were important in their contribution to core melt and the different release categories for the risks. D. Okrent expressed concern about the Staff's set of results which were developed on the basis of certain assumptions about reliability. A system which is very reliable from the calculation for Calvert Cliffs will have a small relative contri bution to risk for Palisades. C. P. Siess mentioned that he could not find residual heat removal (RHR) shutdown topics on a list of items that met current criteria handed out during the presentation (see Appendix XVIII). W. T. Russell indicated that they were covered in Chapter 4 of report NUREG-0820 and make a small relative contribution to risk or are unimportant to risk at Palisades. D. Okrent indicated that the logic of the Palisades study methodology is in question because the study is highly influenced by the Calvert Cliffs study. R. Axtmann indicated that he was not completely sure whether the risk referred to in Staff tabulations referred to the process or the risk to the public, especially with regard to containment leakage and safety injection actuation. M. Rubin indicated that this is risk contributing to core melt weighted by risk fractions.

E. Integrated Assessment Topics

T. Michaels of the SEP Branch explained that the integrated assessment for Palisades was concerned with classifying 31 topics into various groups for backfitting. Topics were classified as to whether they required backfitting, procedural backfits, hardware backfits, required further analysis and potential hardware backfits or whether these were topics in dispute between Consumers Power and the Staff (see Appendix XX). D. Okrent inquired how thoroughly the compressed air system was evaluated in regard to the topic of tornado missiles. W. T. Russell indicated that the compressed air system was important to safety and considered in the safe shutdown review.

He added that Palisades has a separate high pressure air system which is safety grade for safety related valves inside the containment. D. Okrent questioned whether Palisades in the case of an earthquake would have any other means of leak detection other than a system to detect radioactive materials within the containment. T. Michaels indicated that there are redundant means of measurement, diverse systems some of which are seismically qualified which will operate if nonseismically qualified systems do not function. He indicated that if they did have an earthquake in excess of an operating basis earthquake, they would be required to shut down per Appendix A of Part 100, and would have an adequate leak detection system from the seismic standpoint for leakage from the containment. He added that the

Staff did not recommend backfitting or upgrading additional seismic capabilities of the containment leak detection system. With regard to the subtopic of valve location under the topic of containment isolation system, D. Okrent questioned why the PRA results showed no difference in containment unavailability or failure of isolation whether two valves were outside or one was inside and one was outside the containment. He questioned whether this finding would become a precedent for future applicant submittals. W. T. Russell indicated that that result in the PRA was dominated by the higher probability of the failure of the valve and not the probability of the failure of the pipe between the containment and the valves on the outside of the containment. He added that the Staff considered it good design practice to put one valve inside and one valve outside the containment but could not justify requiring Palisades to make the major modification of moving a large number of valves. With regards to isolation of high and low pressure systems, T. Michaels indicated that the Staff tried to use procedural backfits wherever possible as a general rule. The Committee discussed alternatives that the Staff had for the satisfactory resolution of the problem of interfacing low and high pressure systems. D. Okrent inquired how the Staff judges what constitutes adequacy with regard to flooding of safety systems in the intake structure of the station service and cooling water systems. T. Michaels indicated that the Applicant will provide drainage in the intake structure. There will be alarms and assurance that the operator can act in sufficient time to prevent inundation of the service water pumps.

D. Okrent expressed concern with regard to the reliability of the alarms, especially in the event of a seismic event. W. T. Russell explained that the Staff is looking at the reliability of the alarm systems, their redundancy and postulated sizes of breaks that might cause flooding. W. T. Russell added that while this issue is clearly safety related, the Staff has specified no other criteria other than that there shall be alarms to alert to this event.

C. P. Siess brought the Committee's attention to the historical background behind an issue that concerned flooding potential and protection requirements, SEP topics all dealing with the seiche level. He pointed out that there is considerable difference between the Applicant calculated numbers and the Staff calculated numbers (the design value) with the Staff numbers sized for an 13 1/2 ft. surge. He indicated that the result would be flooding of the service water pumps and low pressure safety injection pumps.

J. Ebersole expressed concern for a critical ventilation item in the battery rooms. He indicated that once a month the Applicant invites a hydrogen problem unless he periodically does an equalizing charge on the batteries. J. J. Ray questioned battery charges on loss of transmission lines from the switchyard to the plant. W. T. Russell explained that the Applicant had replaced batteries with new ones which have a normal capacity of eight hours and, with action to strip load, they can extend well beyond that time and backfeed from the switchyard in six hours back through the main transformers. He indicated that the Staff had found this new installation acceptable.

The Committee discussed the matter of inservice inspection of the water intake control structures, intake cribs, with regard to potential damage from ice damage such as occurred in 1972. Since this structure was the only category 1 structure for getting water to the service water pump suction, some Committee Members were concerned how a malfunction would be detected since the plant no longer uses once-through cooling.

C. P. Siess asked why fire protection ended up as an SEP topic when there was a separate fire protection review being conducted by the Staff. W. T. Russell explained that this was a particular issue where integration occurred quite significantly. He explained that inability to meet fire protection requirements might expose problems in the area of wind and tornado loads.

F. Consultants' Review of Palisades Review of SEP

W. T. Russell brought the Committees attention to a handout which contained excerpts from the letters and overall assessments by the Consultants and the Staff (see Appendix XXI). D. Okrent, with regard to the applicability of the SEP topics, questioned remarks by Consultant S. Bush, which indicated that some strong positions taken at the inception of the SEP have been weakened. W. T. Russell indicated that in response to that comment, the Staff would be upgrading the topic definitions from those written in 1977 when they were more hardware oriented and did not reflect current Staff philosophy. C. P. Siess expressed concern with regard to the LER experience at Palisades relative to the combined performance of the offsite and onsite power systems. He indicated that this plant has a history of partial blackouts with loss of one of the diesels at the same time the plant has lost offsite power. He questioned whether the Staff had been influenced by the fact that station blackout is an unresolved safety issue. W. T. Russell indicated that the Staff did not review the station blackout issue. He explained that the Staff had not yet come up with the criterion for imposing a station blackout requirement on licensees. He did indicate that the Staff had sent the Utility a letter requesting a response with regard to the Staff's concerns about station blackout at the Palisades Plant. The Committee discussed diesel reliability at the Palisades Plant.

D. W. Moeller indicated that the basis for deletion of SEP topics was to appear in Appendix A, however, Appendix A did not have any statements for the reasons for the deletion. He asked the Staff where the explanations for the deletions of certain SEP topics were. W. T. Russell indicated that topics that were deleted were reviewed and the safety evaluation or equivalent issued. He added that since the safety evaluations were so elaborate, it was not deemed cost effective for the Staff to include them in the SEP report on Palisades. C. P. Siess indicated that the Safety Evaluation Reports for the Palisades Plant are available to Committee Members on the proper distribution list.

B. Davis, NRC Staff, indicated that until recently Consumers Power Co. has performed below Staff expectation from a management point of view for a number of years. He indicated that evaluation of LERs since 1979 prompted I&E to request from this Licensee, and to finally approve in March of 1981, the initiation of a regulatory improvement program. He explained that Staff action was triggered by a combination of items taken collectively which included a systematic assessment of licensee performance or SALP covering the period of 1979, a long-term containment violation where the containment purge bypass was open for an extended period of time, a violation where the containment sump isolation valve was improperly open for a couple of days, and the event where both 125v. d.c. battery circuits were opened improperly during a surveillance test.

B. Jorgensen, I&E Region III, presented two bar charts showing the Palisades items of noncompliance and Palisades avoidable LERs from 1975 through 1982 (see Appendix XXII). He defined avoidable LERs as composed of personnel errors and the existence of a bad procedure in design, manufacturing or construction. He added that you could trace an equipment error back to a person's mistake if enough facts are known about the failure. B. Jorgensen explained that management changes involving increases in staffing levels, changes in management personnel and changes in organizational structure were brought about by the 1981 regulatory improvement program. In answer to a question by M. Bender, F. Buckman, Executive Director of Nuclear Activities for Consumers Power indicated that a consulting organization, Management Analysis Corp. of San Diego, was responsible for appraising

Palisades management structure in response to the NRC Staff request. B. Jorgensen indicated that while there have been some changes in the organization at Palisades, there has been considerable augmentation in the area of performance verification either improving the requirements for quality control or imposing new requirements for a separate person at an equal skill level to verify that a job is correctly done. M. Bender asked how the Staff could verify that these organizational changes represent some measurable improvement in the overall performance of plant personnel. B. Jorgensen indicated that it would be very difficult to tell which organizational changes had a salutory effect.

G. Licensee Comments

F. Buckman of Consumers Power addressed three questions:

- What is the result of SEP for Palisades?
- From the CPCO viewpoint, was the SEP effort worth it?
- What guidance might CPCO give for SEP Phase III?

F. Buckman indicated that Consumers Power Co. had invested about 39,000 man hours of its own, plus $2.3 million through the use of contractors. He indicated that, of about 5 or 6 modifications that are being proposed, only 2 are relatively significant from CPCO's point of view. These are the alteration of the configuration of the main steam isolation valves, the problem with regard to single failure, and flood protection for the service water intake structure. He added that the item involving the main steam isolation valves is significant in terms of cost and probably significant in terms of contribution to safety.

F. Buckman indicated his belief that the commitment to SEP has probably not been justified since it has been in direct competition with resources allocated for TMI modifications and electrical equipment qualification, fire protection, SEP at Big Rock Point, regulatory performance improvement, and training of people and staffing of the organization for the operation of the Midland Plant. F. Buckman indicated that the NRC Staff has expended little effort in trying to assist in the prioritization of the myriad of issues that it has required CPCO to address on a tight schedule. He suggested that the SEP at least to some degree has lacked systemization and integration. He explained that as far as the integrated assessment for Palisades has been concerned, there has not been much effort by the NRC Staff to try to solve more than one problem with one modification. He added that it was his belief that the program has not been particularly successful in identifying deficiencies but more successful in identifying differences between the Palisades Plant and current regulatory requirements.

F. Buckman indicated that it is his belief that an SEP assessment would be more effective with a plant-specific risk assessment at the outset prior to topic evaluation. He suggested that the plant-specific PRA be use in the integrated assessment to tie the whole program together. F. Buckman added that it was his belief that a PRA be used as a companion to the systematic evaluation program and not an imposed requirement upon a licensee before he enters an SEP evaluation. F. Buckman also added that the NRC should carefully evaluate and agree with the licensee on resource requirements necessary to carry out the activities which are made necessary by the SEP evaluation.

The Committee discussed Consumer Power Co.'s motivations in undertaking the intake structure modifications and the changes with regard to the main steam isolation valves.

The Committee agreed to report to the Commissioners on its review of the Systematic Evaluation Program, Phase II, and its application to the Palisades Plant stressing the following conclusions by the ACRS:

a. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Palisades Plant and should be achieved for the remaining plants in Phase II of the program.

b. The actions taken thus far by the NRC Staff in its SEP assessment of the Palisades Plant are acceptable.

c. The ACRS will defer its review of the FTOL for the Palisades Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI items.

V. Emergency Response Capability in Nuclear Power Plants (Open to Public) [D. C. Fischer was the Designated Federal Employee for this portion of the meeting.]

D. Ward explained that the purpose and scope of SECY-82-111, Requirements for Emergency Response Capability, was to identify a set of NRC proposed basic requirements that relate to emergency response capabilities at nuclear power plants. He explained that the document deals with the safety parameter display system (SPDS), the control room design review, Regulatory Guide 1.97 (Instrumentation Requirements During and Following An Accident), emergency plant operating procedures, and emergency response facilities which include the Technical Support Center, the Operational Support Center, and the Emergency Operations Facility (see Appendix XXXIII). D. Ward gave a historical perspective on the subject which included a summary of the May 5, 1982 Human Factors Subcommittee Meeting, OPE evaluation of the SECY paper, and a Human Factors Society letter (see Appendix XXIV). Also mentioned was a list of questions provided by D. Okrent for discussion at the May 5 Subcommittee Meetings. D. Ward pointed out that the methods for implementing and for developing the requirements in the SECY paper are of particular importance. He questioned why the SPDS has been singled out for development and installation in plants prior to conducting a control room review.

V. Stello, NRC Staff, defined the scope of the SECY document as involving emergency response facilities, control room improvements, and some specific issues related to operator capability (changing from event-oriented procedures to symptom-oriented procedures). He stessed that these are highly interrelated activities which should be brought together and integrated and coordinated in one place and from one point of view.

V. Stello indicated that the recommendations to the Commission of the Committee for Review of Generic Requirements' (CRGR) are to approve the Staff's proposed set of basic requirements, and to approve of the Staff's proposed implementation plan (see Appendix XXVI). He added that the Staff's implementation plan would encompass integration of activities, the SPDS control room design review, Regulatory Guide 1.97, emergency operating procedures, and emergency response facilities. He mentioned that a lesson from the TMI accident, an extremely important lesson, was that TMI was a process of cognitive errors. He indicated that the current SECY document offers an opportunity now to provide a mechanism to assist the operator in the comparison of conflicting information, and in interpretation of information in the control room during an emergency.

D. Okrent expressed concern that NRC is not placing enough emphasis on trying to provide an improved understanding of what the SPDS or the instruments that are there are doing, and on comparisons of conflicting information and interpretation of information. V. Stello attempted to answer the question by discussing symptom-oriented procedures, procedure development, and consideration of control room design review. R. Mattson, NRC Staff, discussed the guideline in NUREG-0696 that required that the SPDS information had to have online checking of instrumentation to confirm whether the data provided was good or bad information. The Committee discussed the various limitations of the SPDS and its relationship to emergency procedures.

V. Stello pointed out that there is an issue with respect to how Regulatory Guides and other documents are being used by the Staff. He asserted that NUREGs are not ever again to be used to promulgate requirements. He added that with the exception of probably a few NUREGs that are in the mill, this will be the new way the agency will operate in the future.

H. L. Thompson, Jr., Acting Director of the Division of Human Factors Safety, discussed SECY-82-111 from the perspective of how it affects areas within the responsibility of the Division of Human Factors Safety. He mentioned discussions held between his staff and the staff of the CRGR which identified concerns that possibly the human factors area had not received proper emphasis in the document. He defined his responsibilities as control room review, emergency operating procedures, and the SPDS installation. H. L. Thompson indicated that he had incorporated into SECY-82-111 those basic requirements which were identified in guidance documents NUREG-0700 and Draft NUREG-0801 which were issued by the Human Factors Division. He lauded a major aspect in the SECY document which involved the change to negotiated plant specific schedules vs. mandated schedules for control room review. He indicated that there was no basic change in the area of the detailed control room review in that licensees will still submit their program plan for review. H. L. Thompson compared several of the control room design review aspects in SECY-82-111 with requirements prior to the SECY document (see Appendix XXVII). In answer to a question by D. W. Moeller, he indicated that there is probably not a straightforward procedure for determining what human engineering defects in the control room have safety significance. J. Ebersole raised a question about enhancements licensees are free to implement as soon as identified without Staff approval. He indicated that these changes are often very costly because the licensee for the record is required to update all drawings from the design through the subsequent alteration process with these changes. R. Moore, NRC Staff, indicated that the cost of these changes usually does not multiply with a large number of changes because the utility will incorporate all of the changes into the drawings at one time. He added that the NRC would only get involved during its review in checking the higher level drawings that are used in the construction (modification) of the control room.

H. L. Thompson explained that SECY-82-111 singles out the SPDS for prompt implementation. He added that while SPDS installation and implementation would be integrated with the ongoing emergency operating procedures upgrade and the detailed control room design review, the emergency operation procedures upgrade and control room design review should not impact the schedule for the SPDS implementation. In answer to a question by D. Okrent, R. Mattson explained that the emergency operating procedures will be in place, in general, before SPDSs are installed.

R. Mattson NRC Staff, made four brief points:

- SECY-82-111 is an integration paper - a later integration step than the Action Plan in early 1980 and NUREG-0696.

- The CRGR has considered cost estimates in the same way that value impact was considered at the Action Plan stage.

- It has been recognized that there would be an SPDS validation period to assess operational information in SPDS design The proposed method of regulating SPDSs is novel, born of an attempt on the part of CRGR and NRR to speed the implementation of SPDS.

- Implementation of Regulatory Guide 1.97 through SECY-82-111 is the best implementation plan yet, despite the fact that it delays full environmental qualification of Reg. Guide 1.97 instruments and delays their qualification to be consistent with qualification of other safety equipment in the plant.

B. Grimes, Director of Division of Emergency Parparedness, discussed emergency response facilities, specifically the Technical Support Center in the emergency operations facility. He suggested that there is adequate basis in the existing regulations for achieving acceptable emergency response facilities provided that NUREG-0696 is made a Regulatory Guide and discussed with the Commission. He noted a substantial disagreement that still exists with the CRGR proposal relating to the fact that the Division of Emergency Preparedness believes that habitability of the Technical Support Center should essentially be equivalent to that of the plant control room exactly as specified in NUREG-0696 and that filtration systems need not be redundant nor automatically activated. P. G. Shewmon questioned the usefulness of the Technical Support Center during times of normal operation of the plant. B. Grimes indicated that the technical support center may be activated for dealing with potential emergency situations perhaps once a year. He added that the facility would also be used by the utility as a dual use facility for the conducting of exercises or training.

C. Hopkins, Technical Director of The Study Group of the Human Factors Society, spoke about a comprehensive long-range human factors plan for nuclear reactor regulation under study by his group for the past 16 months. He mentioned the first recommendations of the study group entitled, System Engineering of the Regulatory Process, which approves of the general approach taken by SECY-82-111 in terms of laying out specific items and attempting to integrate these in terms of overall program. He expressed the serious concerns of the study group that the document SECY-82-111 deemphasizes somewhat the importance of human factors (see Attachment 4 in Appendix XXIV).

C. Hopkins noted two items which he suggested were of particular importance:

- Overemphasis on the SPDS.
- Lack of requirement for approval of the plans for the control room review.

J. Ebersole expressed concern that there were no requirements on the SPDS for independent information source or sensor. The result of a problem reading in a control room instrument would not produce a confirmation of difficulty but an SPDS retransmission of faulty data and potential control room confusion. J. Rosenthal of the Instrumentation and Control Branch, explained online validation of data using an artificial intelligence, doing sorts on meter levels. J. Rosenthal pointed to this level of artificial intelligence provided by a minicomputer. R. Mattson indicated that a minicomputer is not a requirement for the SPDS. C. Mark asked C. Hopkins how the Human Factors Society might design its own control room.

C. Hopkins spoke of redesign of instruments, presenting information in a different format, integration of some information that is currently scattered over the control boards and, in general, reduction in the number of displays and controls.

B. Coley, Chairman of the AIF Subcommittee on Control Rooms and Emergency Response Facilities, applauded the approach of SECY-82-111 as a new approach to the industry regulatory interface (see Appendix XXVIII). He expressed industry concern for a balanced emphasis which should take account of the fact that the control room, the operator, and the procedures in effect constitute a system that must be addressed. He maintained that overemphasis on any one element of that triad such as implying that a single solution might lie in an SPDS or human factors expertise or a control room review would be a mistake and not produce positive change. He recognized that SECY-82-111 does recognize the functional approach in asking utilities to achieve concrete goals or objectives and not getting hardware prescriptive. B. Coley did indicate that it is the opinion of the AIF that the SPDS design should be derived and developed from the emergency operating procedure guidelines. In addition he contended that Regulatory Guide 1.7 compliance should flow from the entire process of the control room review. He suggested that the balanced approach that is represented in the SECY document should be extended to the way in which NRC implements the document and interfaces with industry. B. Coley pointed out that SECY-82-111 recognized that it is impractical to completely redesign all the control rooms in operating plants. He suggested that the same concept should be extended to software.

M. Bender asked whether the emergency operating procedure guidelines being developed by Babcock and Wilcox, Westinghouse, and Combustion Engineering would allow operator interchange between those three PWR systems. B. Coley indicated that operator interchange would be a problem, because an interchange of operators between different stations could not be effected without considerable training and licensing for those particular plants. He suggested that different approaches with regard to emergency operating procedures would normally follow from radically different designs or from the age of the plants and the technology employed at the different vendors.

M. Howard of KMC, Inc., which represents about 30 utilities, spoke negatively of the requirements documents which have previously been developed independent of each other and in many cases in isolation by NRR or the Division of Emergency Preparedness within I&E. He explained that these documents provided prescriptive requirements and established inflexible implementation dates. M. Howard indicated that the coordinating group for emergency preparedness strongly supports the CRGR and proposed SECY-82-111.

D. A. Ward indicated that certain consultants present during the Human Factors Subcommittee Meeting expressed positions that should be noted:

- R. Pearson endorsed the position of C. R. Hopkins of the Human Factors Society that the SPDS should flow out of a comprehensive control room analysis rather than precede it.
- G. Salvendy endorsed SECY-82-111, but thought additional research such as that done by EPRI ought to be done in the area of operator behavior.
- A. Debons expressed concern that there was inadequate analysis of the operation from the standpoint of the overall information flow through the system. D. A. Ward indicated that it was the over all concensus of the Subcommittee to endorse SECY-82-111.

VI. Qualification Program for Safety Related Equipment (Open to Public) [Note: G. R. Quittschreiber was the Designated Federal Employee for this Portion of the meeting.]

J. J. Ray explained that the purpose of the meeting on May 5, 1982 of the Subcommittee on Qualification Program for Safety Related Equipment was to review and discuss the final version of section 50.49 to 10 CFR 50 entitled, Environmental Qualification of Electrical Equipment for Nuclear Plants (see Appendix XXX). He indicated that the new rule deals with the environmental qualification of Class 1E electrical equipment and certain non-Class 1E electrical equipment such as post accident monitoring systems. J. J. Ray presented a historical perspective on the new addition to 10 CFR 50 (see Appendix XXX). He indicated that Subcommittee members were favorably impressed by the response of the Staff to public comments and the nature of changes that were made to comply with industry comments. He mentioned that a major provision of the rule will grandfather those operating plants and plants under licensing review, for which qualification is in process and will use either the DOR guidelines or NUREG-0588 for assessing plants which commenced such qualification within the 90 day period which followed the effective date of the rule.

J. J. Ray brought the Committee's attention to major residual concerns expressed by industry with which the Subcommittee concurs:

- The revision and issuance of Regulatory Guide 1.89 should be expedited so that the rule and associated guide will be concurrently available to industry.
- The rule should be revised before issuance to include seismic qualification
- The rule should include a statement specifically including equipment qualification for NTOL plants in the grandfather provision.

J. J. Ray noted that, in response to industry comments, a requirement in the rule that equipment, needed to complete one path of achieving and maintaining a cold shutdown condition, be environmentally qualified was deleted in the final rule. It was also mentioned that in response to a concern from the Electrical Systems Subcommittee of the ACRS which noticed absense of an "interpretive designation of the equipment which will be affected" by the regulatory action, the rule has been narrowed in scope and changed to specify the equipment by functions rather than by specific listing of equipment components.

S. K. Aggarwal explained why the requirements in the area of seismic and dynamic qualification were excluded from the final rule (see Appendix XXXI). He indicated that after lengthy discussion with the Commission and the ACRS, the Commission decided to pursue this issue by means of an advanced notice of rulemaking. W. Kerr indicated that it is his belief that industry is concerned with the possibility of having to take out equipment that has been environmentally qualified and replace it with equipment that would also now have to be seismically qualified. S. K. Aggarwal indicated that it is a Staff proposal to reassure industry that the statement covering this subject will be included as a statement of consideration. J. J. Ray added that, in effect, the Staff will grandfather existing installations seismically as they have grandfathered them environmentally in the present rule. W. Kerr suggested that if licensees are responsible through the general design criteria for environmental and seismic qualification, they should have the option to perform both qualifications at the same time. S. K. Aggarwal in an attempt to clarify the situation, explained that, if the licensees have followed the requirements of Regulatory Guide 1.100 on seismic requirements which endorses IEEE 344-1975, and if they have qualified equipment according to the DOR guidelines and NUREG-0588 as far as the environmental qualifications are concerned, this situation is acceptable to the Staff. In response to a question by J. Ebersole, S. K. Aggarwal indicated that NTOL plants are required by Regulatory Guide 1.100 to meet the requirement regarding seismic qualification.

S. K. Aggarwal indicated that the Staff seeks ACRS concurrence with the Commission in favor of deletion of the cold shutdown requirement - 50.49 (c). D. Okrent questioned the Staff concerning statements which implied that there is little incremental risk of serious release of radioactivity to the environs in going from hot shutdown to cold shutdown. R. Mattson explained that there can be risks in the inability to get to cold shutdown, a subject which is now under study as Unresolved Safety Issue A-45 (see Appendix XXXII). He explained that the reason for deletion of the cold shutdown requirement was somewhat different from that. He stated that the rule focuses on design basis events (DBE). Therefore, equipment (such as the ECCS) required for the DBE would already be qualified. Other occurrences such as normal shutdown, events beyond the design basis, core melt, etc. were not covered by the rule. Therefore, the requirement would be deleted. Equipment required for these other occurrences would be reviewed under USI A-45.

S. K. Aggarwal indicated that the Staff is seeking some legal advice on how to assist Licensees of operating plants which are required to satisfy the provisions of I&E Bulletin 79-01B and Licensing Generic Letter 82-09 for equipment located in a mild environment, if this statement is deleted from the final rule.

D. W. Moeller requested an explanation of apparently contradictory statements with regard to the requirement in the rule on aging. He indicated that the rule implies that you cannot qualify the equipment by letting it naturally age or use accelerated aging to define a qualified life because it is not technically feasible. S. K. Aggarwal indicated that these are really two separate comments which have been addressed by revision of the paragraph on aging. S. K. Aggarwal indicated that the Staff has taken the Subcommittee's advice and will expedite Regulatory Guide 1.89 which will be available to the public in the near future. He pointed out that it is very important that this rule be issued prior to June 30, 1982 because the date of qualification under the Commission's memorandum and order is June 30, 1982. This could put the Commission in an impossible situation necessitating some legal actions independent of this rule unless this rule is issued in final rule prior to that date.

J. J. Ray indicated that the Subcommittee agrees in general with the changes proposed by the Staff. However, he indicated that the Subcommittee was concerned that industry would have difficulties with seismic requirements unless the NRC grandfathered the relevent equipment in the new rule. He added that in the opinion of the Subcommittee, there appears to be justification for the concern of industry to have Reg. Guide 1.89 revised concurrently with the new rule because Reg. Guide 1.89 as revised is to guide the industry on how to comply with the rule, itself.

M. Bender suggested that the seismic area is likely to be a problem because of electrical equipment attached to valves and other mechanical equipment in the containment. Z. R. Rosztoczy of the Staff indicated that the Staff in the seismic area definitely requires sequential testing of every piece of equipment that is now being qualified. He indicated that it is a major concern for industry because the NRC Staff is not set up to review the seismic tests separate from the rest of the sequential testing process. Z. R. Rosztoczy spoke in favor of putting the seismic requirement back in the rule, or at minimum including a statement of consideration specifying that the Staff is following requirements that were established in 1974. S. K. Aggarwal disagreed. He stated that the Staff has grandfathered operating plants if they meet the requirements of the DOR guidelines or NUREG-0588 which does not require sequential testing. He added that Regulatory Guide 1.89, which has been released for public comment, specifically states that Licensees of operating plants will not be required to do a test sequence.

W. V. Johnston, NRC Staff, agreed with Z. Rosztoczy that when dealing with new plants such as NTOLs, the Staff expects these plants to use a full sequence of testing which includes environmental and seismic testing. He continued that the Staff wishes to reassure the ACRS and industry that for plants which have been qualified up to the present time, there will not be a requirement to redo the environmental qualification along with any seismic qualifications which they may have to do under a rule which does not exist. J. J. Ray indicated that it was his belief that the concern of industry involves not knowing what the seismic requirement is going to be in the new rule. R. G. LaGrange of the Equipment Qualifications Branch agreed with Z. Rosztoczy and disagreed with S. K. Aggarwal that the seismic requirements should be part of the rule.

Z. Rosztoczy stated his concern that the deletion of the mild environment requirement from the rule would create a major problem for the nuclear industry. W. V. Johnston reiterated S. K. Aggarwal's position with regard to deletion of the mild environment requirement from the rule. M. Bender suggested that the problem was more of a procedural matter than a technical matter, one with which the Committee should not be involved. Chairman Shewmon indicated that the Committee has sufficient information for its report and terminated the discussion.

VII. Quantitative Safety Goals (Open to Public)

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

The Committee continued its discussion and review of NUREG-0880, Proposed Policy Statement on Safety Goals for Nuclear Power Plants, A Discussion Paper. Chairman P. G. Shewmon focused the discussion by suggesting options for the ACRS report to the Commission. He noted that the ACRS statement on goals could endorse narrow functional goals (e.g., containment and ECCS functional specifications) or global goals which deal with the impact of nuclear facilities on the public health/risk. Goals could be of a qualitative or quantitative nature in either case.

Several points of view were presented by individual ACRS Members and were the subject of considerable debate. One group suggested that a quantitative goal which deals with public risk would not be particularly useful because the capability does not exist to assess compliance with the goal or the quantitative assessment of risks involved. These Members favored functional numerical quidelines which address design specifications for the major systems important to safety (e.g., containment and ECCS). Other Members suggested that the Quantitative Safety Goals should include a quantitative statement of what is meant by the phrase "undue risk to the public" which would form the basis for any set of specific plant design specifications.

Members spoke of the regulatory problems associated with implementation of the goals and the fact that use of probabilistically based goals may divert the regulatory process unnecessarily.

Several Members spoke against the use of an ALARA concept explaining that health risks are not well suited to a quantitative methodology. H. W. Lewis spoke against the use of goals limiting risk to individuals and suggested instead that the goal be directed only to limiting societal risks.

D. Moeller urged that a statement should be made regarding genetic as well as short and long term somatic effects since this is an area of considerable public interest and concern.

The difficulty of quantifying sabotage and seismic events was also cited. Some Members expressed concern regarding the use of probabilistic risk assessment (PRA) in decisionmaking at this early stage in its development. Members did note the inevitability of the use of PRA in decisionmaking and suggested that the ACRS make a statement calling for the Commission to properly control the use of PRA. One group of members suggested including accident risks from other parts of the nuclear fuel cycle as opposed to limiting the safety goals only to accidents in nuclear power plants. Some discussion occurred regarding the specifics of dealing with aspects of risk to the public such as "the hypothetical nearby individual", the man-rem concept, regional or societal effects, etc.

Chairman Shewmon summed up the discussion by assigning Members to draft paragraphs dealing with the global aspects of safety goals and the problems associated with implementation. The Committee expressed positive inclinations toward statements regarding other portions of the fuel cycle, individual risk criteria, and incentives for nuclear power plant siting in low-population areas.

VIII. Executive Sessions (Open to Public)

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

A. Subcommittee Assignments

1. Safeguards and Security

C. Mark advised the Committee that he had, as Chairman of the ACRS Safeguards and Security Subcommittee, reviewed the "Insider Package"

[Proposed NRC rules regarding Access Authorization, Search Requirements, and Miscellaneous Safeguards Matters such as Access Controls, Vital Area Designation, etc.] to determine if review by the full Committee is appropriate and concluded that ACRS action is not warranted at this time. He concluded that a Subcommittee briefing by the NRC is therefore unnecessary, but does wish to be kept informed of further developments regarding this subject.

2. Reactor Operators

The Committee agreed to have the Reactor Operations Subcommittee review proposed changes in the inspection and enforcement policy of the NRC including proposed changes in criteria for levying fines and coordination with INPO activities.

B. ACRS Reports, Letters, and Memoranda

1. ACRS Report on the Wolf Creek Generating Station, Unit No. 1

The Committee prepared a report to the Commissioners of its review of the Wolf Creek Generating Station Unit 1 and concluding that, if due consideration is given to the recommendations in the body of the report, and subject to satisfactory completion of construction, staffing, training, and preoperational testing, the facility can be operated at power levels up to 3425 Mwt without undue risk to the health and safety of the public.

2. ACRS Report on the Systematic Evaluation Program, Phase II, and its Application to the Palisades Plant

The Committee prepared a report to the Commissioners of its review of the Systematic Evaluation Program, Phase II, as it has been applied to the Palisades Plant. The ACRS concluded the following:

a. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Palisades Plant and should be achieved for the remaining plants in Phase II of the program.

b. The actions taken thus far by the NRC Staff in its SEP assessment of the Palisades Plant are acceptable.

c. The ACRS will defer its review of the FTOL for the Palisades Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI items.

[Note: W. Kerr did not participate in the review of the Palisades Plant or the drafting of the Palisades report.]

3. ACRS Report on Emergency Response Capabilities at Nuclear Power Plants

The Committee prepared a report to the Commissioners regarding the subject matter of SECY-82-111, "Requirements for Emergency Response Capability," taking into account reviews of this report at ACRS subcommittee meetings on January 5, March 17, and May 5, 1982 as well as the review by the full Committee at this meeting.

4. Rulemaking on Environmental Qualification of Electric Equipment

The Committee prepared a report to the Commissioners of its review of the proposed final rule, Environmental Qualification of Electric Equipment for Nuclear Power Plants, and recommended approval of the rule subject to considerations regarding deferment of the seismic response and cold shutdown requirements, priority revision of Regulatory Guide 1.89, and review of the practicality and safety value of current qualification reviews using "DOR Guidelines" and NUREG-0588.

5. Control of Occupational Exposures

The Committee approved a memorandum to the EDO regarding applicant and licensee radiation protection programs and implementation of the ALARA criterion urging attention to the reduction of operational exposures, to methods for the removal of radionuclide deposits if preventive measures are not successful, and to the minimization of the failure and subsequent need for the repair and/or replacement of major plant components.

6. ACRS Report of Ad Hoc Subcommittee on Foundation Problems and Remedial Action at Midland Plant Units 1 and 2

The Committee approved a memorandum from the ACRS Executive Director to the EDO regarding the matter of soils-related structural settlement problems at the Midland Plant Units 1 and 2 site.

The ACRS recommended:

a. That the Midland Plant Subcommittee review the adequacy of the seismic input criteria and the SSRS and their relation to the proposed permanent site dewatering as a means of reducing the probability of liquefaction due to an earthquake.

b. That, subject to a finding by the Midland Plant Subcommittee regarding the adequacy of the seismic input criteria, the ACRS recognizes the adequacy of the NRC Staff's efforts as outlined in this report and considers the proposed remedial measures as a matter that can and should be resolved in a manner satisfactory to the NRC Staff.

c. That the EDO be informed at this time that the ACRS has found the Staff's approach to be acceptable.

C. Generic Safety Items

1. ACRS Comments on NRC Proposed Safety Goal Policy Statement

The Committee was unable to complete its review of the NRC Proposed Safety Goal Policy STatement. Consequently, continued discussion will be scheduled during the 266th ACRS Meeting (June 1982).

D. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 266th ACRS Meeting, June 3-5, 1982 (see Appendix II).

2. Future Subcommittee Activities

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

E. Committee on Science and Technology Hearing on the NRC Research Program

The Committee endorsed the assignment of C.P. Siess and P. G. Shewmon to testify on May 18, 1982 before the U.S. House of Representatives Committee on Science and Technology, Subcommittee on Energy Research and Development regarding aspects of the NRC Safety Research Program.

F. Nominations for a New ACRS Member

Chairman Shewmon announced the formation of a screening panel for selection of candidates for the upcoming vacancy on the ACRS full Committee. The screening panel is to consist of J. J. Ray, W. M. Mathis, and D. W. Moeller, Chairman.

The 265th ACRS Meeting was adjourned at 3:10 p.m., Saturday, May 8, 1982.

APPENDIXES
TO
MINUTES OF THE 265TH ACRS MEETING
MAY 6-8, 1982

ATTENDEES
265TH ACRS MEETING
MAY 6-8, 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
William Kerr
Harold W. Lewis
Carson Mark
William M. Mathis
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

CONSULTANTS

G. Irwin
I. Catton
H. Kouts
F. Binford
M. Wechsler
T. Teofanous
Z. Zudans

NRC ATTENDEES

265TH ACRS MEETING

Thursday, May 6, 1982

Division of Licensing (NRR)

G. E. Edison
J. B. Hopkins
K. L. Kiper
E. L. Doolittle

Division of Systems Integration

M. Spangler
B. LeFave

Division of Engineering

B. Jagannath
C. Tan
R. L. Ferguson
R. Eberly
R. Wescott
A. Lee
P. Sobel

Nuclear Materials Safety & Safeguards

R. E. Zimmerman
D. Kunze

Region IV

E. H. Jonson

Nuclear Reactor Regulation

P. Shemanski, EQB
R. Benedict, DHFS
R. L. Rothman, GB
R. Gramann, NMSS
E. F. Goodwin
B. J. Youngblood
W. V. Johnston
J. Lyons
W. T. Russell
Z. R. Rosztoczy

Research

J. E. Richardson
S. K. Aggarwal

Of. of Emergency Preparedness

C. R. Van Niel
B. Grimes

NRC ATTENDEES

265TH ACRS MEETING

Friday, May 7, 1982

Nuclear Materials Safety & Safeguards

R. Gramann

Nuclear Reactor Regulation

E. Goodwin
W. Minners, DST
C. I. Grimes
G. Staley, HGEB
J. Rosenthal, ICSB
T. V. Wambach, DL
G. C. Lainas, DL
N. H. Wagner, DSI
M. H. Fliegel, DE
R. Pichumani, DE
O. Rothberg, DE
M. Rubin
T. Michaels
H. L. Thompson
R. Mattson
Z. Rosztoczy
R. G. LaGrange

Inspection & Enforcement

D. C. Boyd

Region III

M. Phillips
A. B. Davis
B. Jorgensen

EDP

V. Stello

APPLICANT ATTENDEES

265TH ACRS MEETING

Thursday, May 6, 1982

KANSAS CITY GAS & ELECTRIC

R. Lems
J. M. McKinstray
L. Morgan
D. R. Smith
G. L. Koester
M. D. Hall
J. M. Pippin
F. Rhodes
E. D. Tarver
G. D. Boyer
M. M. Nichols
G. Rathbun
L. Koerper
R. Terrill
E. W. Creel
D. Green
T. D. Keenan
J. A. Zell
M. L. Johnson
B. Hagan
J. A. Bailey
R. C. Coulthard
C. J. Sprout

KANSAS CITY POWER & ELECTRIC

J. Miller
D. Crawford

SNUPPS

F. Schwoerer
R. L. Stright
J. H. Riley
S. J. Seikey
N. A. Petrick
J. O. Cermak

UNION ELECTRIC

A. C. Passwater

Bechtel Power Corporation

J. H. Smith
D. Grove
F. W. Thomas
P. A. Ward
C. R. Klee
D. Kohler
K. Blodnikar
F. M. Roddy
D. C. Gasda
T. Diperha
J. Prebult
B. L. Meyers
K. Lee
J. M. Small

Westinghouse Electric Corporation

D. G. Maire
R. H. Mark
J. C. Mesmeringer
G. Lang
D. H. Rawkins
S. Phillips
B. Loreng
C. Wirst
J. W. Swigger

DAMES & MOORE

C. Scawthorn
D. F. Fenster
R. Rosdenda

SARGENT & LUNDY

L. L. Holish
J. Keter

Rochester Gas & Electric

R. C. Meredy

APPLICANT ATTENDEES

265TH ACRS MEETING

Friday, May 7, 1982

Consumers Power Co.

F. Buckman
R. A. Vincent
W. J. Beckius
R. B. Dewich
J. G. Lelois

Westinghouse

E. Murphy

PUBLIC ATTENDEES

265TH ACRS MEETING

Thursday, May 6, 1982

J. Drikina, Associated Press
D. Helli, KARD-TV
T. Tyitan, Atomic Industrial Forum
L. Cama, Self
J. C. Debunkett, NUS
K. Peterson, Staffer Publication (press)
C. Watt, KARD TV
J. Silberg, Shaw Pittman
R. Leyse, NSR
R. Borsum, Babcock & Wilcox
R. Contte, ScP
R. Farber, GRS
R. Boyd, KMC
J. Evans, TSI
O. Berry, Sandia
L. Knudsen, E&I
C. Ader, SWEC
C. Grochnal, SWEC
R. Ross, Doub & Muntzing
T. J. Sullivan, Consumers Power
K. E. Dreid, Consumers Power
R. W. Huston, Consumpers Power
K. Gage, NIRS
M. Kamada, KEPCo
J Rubio, CNSNS
G. Sanders, SANDIA
R. Marcello, Yankee Atomic Electric Company
J. Berga, EPRI
M. Horrell, EBASCO Services Inc.
T. R. Trumm, Commonwealth Edison
E. (Morris) Howard, KMC

PUBLIC ATTENDEES

265TH ACRS MEETING

Friday, May 7, 1982

R. L. Kershner, ARD Corporation
R. E. Howard, Commonwealth Edison Company
H. D. Thornburg, AROS
H. Steein-Lause, Public Service Ind.
U. Knudsen, EEI
J. McEwen, TSI
K. Malloy, Allen Corporation
C. O. Hopkins, HFX
R. Leyse, NSAC
C. Grochorse, Stone & Webster
J. W. Miller
R. G. Smith, SLP
J. Mosier, ESSEX Corporation
K. Watkins, Lowenstein-Newman
W. Cooley, Duke Power
T. Meek, Duke Power
E. Sliger, TVA
R. Rosee, Doub & Muntzing
S. Tremaine, NUS Corporation
L. S. Gifford, GTE
P. Tremblay, NUS
L. Cannon, Doc-Search Associates
M. Horrell, EBASCO
T. A. Kevern, NUTECH
G. Wrobel, Rochester Gas
M. H. Fletcher, Phoenix Power Services
R. Lewis, Newhouse Newspaper
R. J. Ross, Doub & Muntzing
S. Kumar, Gibbs & Hill, Inc.
R. Ross, Dames & Moore
R. Lewis, Newhouse
R. C. Wilson, NUS Corporation
M. H. Philips, Debevoise & Liberman
K. H. Smith, McGraw-Hill
B. Coley, AIF/INPO

APPENDIX II

FUTURE AGENDA

JUNE

Midland Plant Unit 2 -- OL (tentative)

ACRS comments/recommendations regarding proposed NRC Quantitative Safety Goals

ACRS comments/recommendations regarding NRC proposed plan of action on "Thermal Shock" of Reactor Pressure Vessels — 4 hrs

Discuss proposed ACRS report to NRC regarding proposed NRC Safety Research Program for FY 1984-85 and the long-range nature of the program in the out-years (FY 1986-88) (initial discussion) — 2 hrs

ACRS comments regarding proposed NRC rule on Application of TMI-2 Lessons Learned to OLs and relaxation of technical specifications for nuclear power plants in the event of emergency situations — Tentative

Discuss NRC policy regarding consideration of seismic events in emergency planning — 1/2 hr

ACRS reply to Commissioner Gilinsky regarding proposed changes in seismic design methodology proposed by P. Jennings (tentative) — 1/2 hr

Discuss final version of Task Action Plan A-45, Evaluation of Alternate Decay Heat Removal Systems — 3/4 hr

Meeting with the Commissioners

- Discuss status of ACRS comments/recommendations regarding proposed NRC Quantitative Safety Goals
- Discuss status of ACRS review of NRC Staff plan of action to address the matter of reactor PV thermal shock
- Discuss schedule and scope of ACRS review of the CRBR
- Discuss ACRS comments/recommendations regarding the proposed NRC Long-Range Research Program Plan (see ACRS Report dated 4/5/82)
- Discuss ACRS comments/recommendations regarding the installation and use of instrumentation for detection of inadequate core cooling (see ACRS report dated 4/6/82 regarding instumentation for monitoring water level or inventory)

JULY

Grand Gulf Unit 1 -- OL

Perry Nuclear Plant Unit 1 -- OL

WNP-2 -- OL

Ginna Nuclear Plant -- SEP Review

ACRS report to NRC on the Proposed NRC Safety Research Budget for FY 1984-85 and the long-range aspects of the out-years (FY 1986-88)

Clinch River Breeder Reactor -- Site Approval Review

AUGUST

Watts Bar Units 1 and 2 -- OL

Subcommittee report on proposed changes in Regulatory Guides (e.g., snubbers on systems important to safety) (Task No. SC 708-4)

ACRS comments on proposed NRC Interim rule on control of combustible gases

Future Meetings

ACRS comments regarding DOE National Plan for Siting High-Level Radioactive Waste Repositories and Environmental Assessment (DWM/RCT) - The Subcommittee on Waste Management, D. Moeller, Chairman, will handle this matter.

APPENDIX III

5/8/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

MAY

11 & 12 — Ad Hoc Metal Components Subgroup (Igne) - Bender, Shewmon, Etherington (tent). Purpose: To review pressurized thermal shock.

14 — Reactor Radiological Effects (Alderman/McKinley) - Moeller, Axtmann, Ebersole, Ray. To review control room habitability and to discuss the design basis for normal and abnormal conditions, testing, and research needs.

17 — Human Factors Briefing (Atlanta, GA) (Fischer/Major) - Ward, Bender. Purpose: To discuss the organization and management of nuclear power plants (particularly construction management programs) and to be briefed by INPO on the work it is doing related to human factors, the SPDS, and control room design.

~~17 & 18~~ CANCELLED — Advanced Reactors (Argonne, IL) (Boehnert) - Carbon, Mark. Purpose: To continue the discussion on a draft of the proposed report on LMFBR Safety Philosophy and Issues.

19 & 20 — Qualification Program for Safety Related Equipment (Albuquerque, NM) (Cappucci/Savio) - Ray, Kerr, Ebersole*, Ward (tent). Purpose: To tour the Sandia testing lab and to continue the review of the NRC Staff's development of equipment qualification criteria.

20 & 21 — Midland (Midland, MI) (Fischer) - Okrent, Siess, Mathis, Ebersole*, Moeller. Purpose: To review the application for an OL and to conduct a site visit.

24 & 25 — CRBR (Boehnert) - Carbon, Mark, Ray (tent). Purpose: To discuss threats to containment for CRBR.

26 — Advanced Reactors (Boehnert) - Carbon, Mark, Ray (tent). Purpose: To review NRC RES programs on Advanced Reactors for the ACRS Report on the Long Range Research Plan.

27 — Electrical Systems/Qualification Programs for Safety-Related Equipment (Savio/Cappucci) - Kerr, Ray, Ebersole, Mark, Mathis, Ward, Bender. Purpose: To review the RES proposed FY 84-85 research funding and programs in this area for the Long-Range Research Plan.

28 — Class 9 Accidents (Beal/Quittschreiber) - Kerr, Okrent, Shewmon, Ward, Moeller (tent), Siess (tent). Purpose: To review severe accident research plan and proposed rule on hydrogen control.

*Conflict to be resolved

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JUNE

~~1 (tent.)~~ CANCELLED	Regulatory Activities (Duraiswamy) - Siess, Bender, Carbon, Ward, Kerr, Ray. Purpose: To review proposed Regulatory Guides and Regulations.
1 (afternoon) 1-5 p.m.	Human Factors (Fischer) - Ward, Lewis, Mathis*, Moeller, Ray*, Kerr*, Mark*. Purpose: To review the NRC's Proposed FY 84-85 programs and budget, and to develop specific comment on the Long-Range Research Plan.
1 & 2	CRBR (Boehnert) - Carbon, Bender, Mark*, Ray*, Okrent (tent.)*, Mathis (tent.)*. Purpose: To begin the review of the seismicity and associated seismic design for the CRBR plant.
2	Safety Research Program (Duraiswamy) - Siess, Okrent*, Moeller, Bender*, Mathis*, Ward, Mark*, Plesset, Kerr*. Purpose: To discuss FY 84-85 NRC Safety Research Program and Budget and to gather information for the ACRS Report to the Commission on the FY 84-85 NRC Safety Research Program Budget.
2	Reactor Operations/TMI-2 Action Plans (Major) - Mathis*, Ray*. Purpose: To review the proposed rule on "Licensing Requirements for Pending Operating License Applications;" and the proposed rulemaking on 10 CFR 50 - Proposed Rule to Clarify Applicability of License Conditions and Technical Specifications in an Emergency.
3-5	266th ACRS Meeting
7	Metal Components (Igne) - Shewmon, Bender, Etherington, Ward, Mathis. Purpose: To be given a status report by owners group on research results and any changes made on steam generator design/ operation, and status report on USI A-3, -4, and -5.
8	Waste Management (Tang/McKinley) - Moeller, Axtmann, Carbon, Ray, Mark, Mathis, Plesset. To review and comment on DOE Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment; provide input for the Waste Management Chapter of the FY 84-85 Safety Research Program Review; review Staff waste management activities; and discuss advances in waste management practices.
16 & 17	ECCS (Idaho Falls, ID) (Boehnert) - Plesset, Ebersole, Ward. Purpose: To discuss GE's request for change in Appendix K decay heat requirements, and an update and status of selected RES LOCA/ECCS Research Programs.

* Conflict to be resolved

5/8/82

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

JUNE

23	Reactor Radiological Effects (Tang/McKinley) - Moeller, Ray, Axtmann, Okrent (tent). Purpose: (a.m.) To discuss NRC Staff proposed revision to 10 CFR 20. (p.m.) To discuss the use of KI for thyroid blocking in the event of a radiation accident.
23 & 24	WPPSS 2 (Hanford, WA) (Griesmeyer/Quittschreiber) - Plesset, Ebersole, Mark, Mathis, Ward (tent.). Purpose: To review application for an operating license.
24 & 25	Joint CRBR and Site Suitability (Boehnert/Alderman) - Carbon, Moeller, Bender, Okrent (tent.), Shewmon, Ebersole, Ray. Purpose: To discuss site suitability for CRBR.
28 & 29	Perry (Cleveland, OH) (Cappucci/Quittschreiber) - Ray, Axtmann, Bender, Okrent. Purpose: To review the application for an OL and to conduct a site visit.

JULY

1 (morning) (4 hours)	Grand Gulf 1 (Alderman) - Okrent, Bender, Ebersole, Siess. Purpose: To complete the operating license review.
1 (afternoon) (~1 hour)	Extreme External Phenomena (Savio) - Okrent, Bender, Siess, Mark. To review the RES proposed FY 84-85 research funding and programs in this area for Long-Range Research Plan.
1 (afternoon) (~1 hour)	Reliability and Probabilistic Assessment (Griesmeyer) - Okrent, Kerr, Bender, Ebersole, Mark, Siess, Lewis. Purpose: To review the RES proposed FY 84-85 research funding and programs for the SARA decision unit.
6 (tent.)	Regulatory Activities (Duraiswamy) - Siess, Bender, Carbon, Ward, Ray, Kerr. Purpose: To review proposed Regulatory Guides and Regulations.
7	Safety Research Program (Duraiswamy) - Siess, Okrent, Moeller, Shewmon, Bender, Carbon, Mathis, Ward, Mark, Plesset, Kerr. Purpose: To continue discussion on the FY 84 NRC Safety Research Program Budget and prepare comments for use by the ACRS in its report to the Commission.
8-10	267th ACRS Meeting

* Conflict to be resolved

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

AUGUST

11	Safety Research Program (Duraiswamy) - Siess, Okrent, Plesset, Ward, Shewmon, Bender, Kerr, Moeller, Mark, Carbon. To provide early input to the RES Staff for their preparation of the Long-Range Research Plan for FY 85-89.
12-14	268th ACRS Meeting

DATES TO BE DETERMINED

Date to Be Determined (June)	Systematic Evaluation Program (Alderman) - Siess, et al. Purpose: To review the completion of the Systematic Evaluation Program review on Ginna.
Date to Be Determined (July)	Watts Bar (Beal/Quittschreiber) - Ebersole, Bender, Ward. Purpose: To complete the review of the application for an OL.
Date to Be Determined	Transporation of Radioactive Materials (location to be determined) (Duraiswamy) - Siess, Bender, Mark. Purpose: To continue the review of the adequacy of the NRC package certification procedures.
Date to Be Determined	Metal Components (Igne) - Shewmon, Ward, Axtmann, Bender, Etherington, Mathis, Plesset. Purpose: To continue the review of pressurized thermal shock.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 11 & 12	Ad Hoc Metal Components Subgroup	(IGNE) Bender, Shewmon, Etherington (tent.),
		Cons: Binford, Catton, Kouts, Gall, Theofanous, Wechsler Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Bender

Purpose: To review the pressurized thermal shock matter with the NRC and industry.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 14	Reactor Radiological Effects	(ALDERMAN/MCKINLEY) Moeller, Axtmann, Ray, Ebersole, Consultants: M. First, C. Burschsted

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: D. Moeller

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

"Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19" - K. G. Murphy and Dr. Kim Campe, 13th AEC Air Cleaning Conference.

"Control Room Ventilation Intake Selection for the Floating Nuclear Power Plant" - D. H. Walker et. al. - 14th ERDA Air Cleaning Conference.

"Evaluation of Control Room Radiation Exposure" - T. Y. Byoun 14th ERDA Air Cleaning Conference

"Fission 2120: A program for assessing the need for Engineering Safety Feature Grade Air Cleaning Systems in post accident Environments - G. Martin et. al. 15th DOE Nuclear Air Cleaning Conference..

"A Consistent Approach to Air Cleaning System Duct Design - W. H. Miller et. al 16th DOE Nuclear Air Cleaning Conference.

Control Room Habitability Study for Quad Cities 1 & 2

Control Room Habitability Study for Dresden 2 & 3

NUREG-0570 "Toxic Vapor Concentrations in the Control Room Following a Postulated Accident Release

General Design Criterion 19, 10 CFR 50 Appendix A

Standard Review Plan 6.4 Control Room Habitability System

Section III. D.3.4 of NUREG 0737 - Control Room Habitability Requirements

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Control Room Habitability Study for Quad Cities 1 & 2

Control Room Habitability Study for Dresden 2 & 3

NUREG-0570 "Toxic Vapor Concentrations in the Control Room Following a Postulated Accident Release

Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants."

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 17, 1982	HUMAN FACTORS BRIEFING	(FISCHER/MAJOR) Ward, Bender

LOCATION: INPO Headquarters (Atlanta, GA)
10:00 a.m. to approxi. 5:00 p.m.

BACKGROUND:

Who proposed action: D. Ward

Purpose: To discuss the organization and management of nuclear power plants (particularly construction management programs) and to be briefed by the Institute of Nuclear Power Operations (INPO) on the work that it is doing which relates to human factors.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
~~MAY 17 & 18, 1982~~ CANCELLED	ADVANCED REACTORS	(BOEHNERT) Carbon, Mark,
		Cons: Avery, Hartung, Lipinski, Siegel

LOCATION: Argonne National Laboratory

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To continue the discussion on a draft of the proposed report on LMFBR Safety Philosophy and Issues.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A revised draft of the subject report will be available prior to the meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 19-20	Qualification Program for Safety Related Equipment	(CAPPUCCI/SAVIO) Ray, Kerr, Ebersole, Ward (tent.) Cons: Catton, Lipinski

LOCATION: Albuquerque, NM

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To tour the Sandia testing laboratory and to continue the review of the NRC Staff's development of equipment qualification criteria.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

A Status Report will be issued two weeks prior to the scheduled meeting.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 20 & 21, 1982	MIDLAND	(FISCHER) Okrent, Siess, Mathis, Ebersole, Moeller Consultants: Epler, Lipinski, Osterberg, Zudans, F. Parker, Scavuzzo, P. Davis

LOCATION: Midland, MI - Site Visit and Subcommittee Meeting (Flights in are most convenient through either the Detroit Metropolitan Airport or Tricities (Saginaw - Flint - Midland Airport)

BACKGROUND:

Who proposed action: NRC Staff and ACRS

Purpose: To review the application for an OL and to conduct a site visit. The Midland plant is 1 miles south of the city of Midland and across the river from a Dow Chemical plant.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

NRR published the Midland SER on May 7, 1982. The SER contains 16 open items, 31 confirmatory issues, and 11 license conditions. In addition, the SER does not address the soils issues.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 24-25, 1982	CRBR	(BOEHNERT) Carbon, Mark, Ray (tent) Cons: Lipinski, Kastenberg, Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To discuss threats to containment for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in the near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 26, 1982	ADVANCED REACTORS	(BOEHNERT) Carbon, Mark, Ray (tent) Cons: Lipinski, Kastenberg, Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: To review NRC RES programs on Advanced Reactors for the ACRS Report on the Long Range Research Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
MAY 27, 1982	ELECTRICAL SYSTEMS/QUALIFICATION PROGRAMS FOR SAFETY-RELATED EQUIPMENT	(SAVIO/CAPPUCCI) Kerr, Ray, Ebersole, Mark, Mathis, Ward, Bender

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the RES proposed FY 1984-1985 research funding and programs in this area and to develop comments on the Long-Range Research Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposed funding and program plans in this area as are available and the Long Range Research Plan.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
May 28, 1982	Class 9 Accidents	(BEAL/QUITTSCHREIBER) Kerr, Okrent, Shewmon, Moeller (tent.), Ward, Siess (tent.)

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: C. Kelber, NRC/RES

Purpose: To review the NRC and Industry research plan for severe accidents.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Documents: Severe Accident Research Program Plan provided in April 1982.
Other documents will be provided.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
~~JUNE 1, 1982 (tent.)~~ CANCELLED	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Bender, Carbon, Ward, Kerr, Ray

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
JUNE 1, 1982 (evening) 1 p.m. - 5 p.m.	HUMAN FACTORS	(FISCHER) Ward, Lewis, Mathis, Moeller, Ray, Kerr, Mark Cons: Arnold, Buck, Debons, Keyserling, Pearson, Salvendy, Catton

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Ward

Purpose: The review the NRC's proposed FY 84-85 programs and budget, and to develop specific comments on the Long-Range Research Plan as they relate to Human Factors. RES will present its FY 84-85 Human Factors research budget proposal to the ACRS. The Subcommittee will then have discussions with NRC/RES and user offices.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

RES proposed budget as submitted to the EDO (scheduled available: 5/1/82).

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 1-2, 1982	CRBR	(BOEHNERT) Carbon, Bender, Mark, Ray, Ckrent (tent), Mathis (tent) Cons: Kastenberg, Lipinski, Luco, Pomeroy, Trifunac, White, Zudans

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: M. Carbon

Purpose: Begin Review of seismicity and associated seismic design for the CRBR plant.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. Chapter 2.5 and 3 of CRBR PSAR.
2. W Topical Report "Seismic Design Criteria for the CRBR" WARD-D-0037)

A-27

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 2, 1982	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Okrent, Moeller, Bender, Mathis, Ward, Mark, Plesset, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Routine Process

Purpose: To discuss FY 1984 NRC Safety Research Program and Budget and to gather information for the ACRS Report to the Commission on the FY 1984 and FY 1985 NRC Safety Research Program Budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposed FY 1984 - FY 1985 NRC Safety Research Program Budget information was distributed to the ACRS on May 6, 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June 2, 1982 (evening) 1 p.m. - 5 p.m.	Reactor Operations/TMI-2 Action Plans (Rules & Regulations)	(RKM) MATHIS, Ray

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: W. Mathis

Purpose:

1) To review the proposed rule on Licensing Requirements for Pending Operating License Applications (10 CFR 50.34, Rule contains the Basic Requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements"). This will be the second meeting with the Staff on this Rule. Public comments should have been evaluated and incorporated into the final form of the Rule prior to the Subcommittee meeting. The Rule should have been reviewed by the CRGR.

2) The Subcommittee will also review proposed rulemaking on 10 CFR 50 - Proposed Rule to Clarify Applicability of License Conditions and Technical Specifications in an Emergency. The rule will be reviewed in its "for comment" form prior to receiving public and industry comment.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. The final form of 10 CFR 50.34 is expected to be available by mid-May.

2. SECY-82-99 contains the proposed rule to clarify applicability of license conditions and technical specifications in an emergency in essentially the form it will go out for public comment.

A-29

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 7, 1982	Metal Components	(IGNE) Shewmon, Bender, Etherington, Ward, Mathis Consultants: Dillon, Kassner, Berger

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: P. Shewmon

Purpose: Status report by owners group on research results and any changes on steam generator design/operation made. The NRC Staff will present a status report on USI A-3, 4 and 5.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY: NUREG-0886

A-30

Rev 1

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
June 8, 1982	WASTE MANAGEMENT	(Tang, McKinley, Donoghue) Moeller, Axtmann, Carbon Ray, Mark, Mathis, Plessett
		Consultants: F. Parker, D. Orth, M. Steindler, S. Philbrick, G. Thompson, R. Foster

LOCATION: Room 1046, 1717 H St., NW, Washington, D.C.
8:30 am

BACKGROUND:

Who proposed action: D. W. Moeller

Purpose:

1. Review and comment on DOE Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment.

2. Provide input for Waste Management Chapter of FY 84-85 Safety Research Program Review.

3. Review Staff Waste Management activities.

4. Discuss advances in waste management practices.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. a) "National Plan for Siting High-Level Radioactive Waste Repositories and Environmental Assessment", (DOE/NWTS-4), February, 1982.
   b) Donoghue memo to Moeller on DOE Siting Plan, 4/29/82.

2. a) Siess memo to Safety Research Subcommittee on FY 84-85 Safety Research Program Review, 4/12/82.
   b) ACRS report to Palladino, "Comments on NRC Long-Range Research Plan, FY 1984-1988 (Draft NUREG-0784)", 5/5/82.
   c) "Comments on the NRC Safety Research Program Budget for FY 83" (NUREG-0795), 7/17/81, Ch. 7, pages 37-39.
   d) "Long-Range Research Plan, FY 1984-1988", (Draft NUREG-0784), 3/12/82, pages 8-1 to 8-19.
   e) ACRS Report to Congress on the NRC FY 1983 Safety Research Program (NUREG-0864), Ch. 7, pages 43-47, 2/12/82.

3. a) ACRS letter to Chairman Palladino on proposed rule for HLW disposal, 10 CFR 60, 9/16/81.
   b) Donoghue memo to Waste Management Subcommittee on HLW Management Bill, S-1662, 5/6/82.

4. a) Recent EPRI reports on radwaste processing techniques for nuclear power plants (EPRI NP-2334, EPRI NP-2335, EPRI NP-2338).

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 16-17, 1982	ECCS	(BOEHNERT) Plesset, Ebersole, Ward Cons: Catton, Acosta, Dukler, Garlid, Theofanous, Zudans

LOCATION: Idaho Falls, ID

BACKGROUND:

Who proposed action: M. Plesset/C. Siess

Purpose: To discuss (1) GE's request for change in Appendix K decay heat requirements; (2) Update and status of selected RES LOCA/ECCS Research Programs for Long Range Research Plan report by ACRS.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

To be provided in near future.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 23, 1982	Reactor Radiological Effects	(TANG/McKINLEY) Moeller, Ray, Axtmann, Okrent (tent)
		Cons: Healy, Bair, Morgan Muller

LOCATION: Washington, DC (Room 1046)

BACKGROUND:

Who proposed action: D. Moeller

Purpose: (Morning) To discuss NRC Staff proposed revision to 10 CFR 20.

(Afternoon) To discuss the use of Potassium Iodide for thyroid blocking in the event of a radiation accident.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. 10 CFR Part 20
2. Draft Revision to Part 20 (March 1982)
3. SECY-82-77 (Policy Issue on Potassium Iodide)
4. NRC Staff's testimony for Congressman Markey's hearing regarding Potassium Iodide (March 5, 1982)

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 23 & 24, 1982	WPPSS-2	(GRIESMEYER/QUITTSCHREIBER) Plesset, Ebersole, Mark, Mathis, Ward (tent)

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
JUNE 24-25, 1982	Combined CRBR and Site Suitability	(BOEHNERT/ALDERMAN) Carbon, Moeller, Bender, Okrent (tent), Shewmon, Ebersole, Ray Cons: A meteorologist F. Parker, R. Foster

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To discuss site suitability for CRBR.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 28 & 29	Perry	(CAPPUCCI/QUITTSCHREIBER) Ray, Axtmann, Bender, Okrent

LOCATION: Cleveland, Ohio

BACKGROUND:

Who proposed action: NRR

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

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (morning - 4 hrs.)	GRAND GULF 1	(ALDERMAN) Okrent, Bender, Ebersole, Siess

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To complete the OL review. ACRS Interim Report dated Oct. 20, 1981 gave conditional approval for operation up to 5% of full power. Questions regarding the ability of the Mark III containment to withstand certain dynamic loads and regarding hydrogen control required answers before the ACRS would approve full power operation. The Committee was also concerned regarding the depth and experience of the operating and support staffs.

Two major issues to be resolved: Hydrogen Control
Seismic Qualification Review Team
(Equipment Qualification)

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

1. SER issued 9/11/81.
2. SSER scheduled to be issued 5/15/82.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (p.m. - ~1 hr.)	EXTREME EXTERNAL PHENOMENA	(SAVIO) Okrent, Bender, Siess, Mark

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Subcommittee Chairman

Purpose: To review the RES proposed FY 1984 - 1985 research funding and programs in this area and to develop comments on the Long-Range Research Plan.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

Proposed funding and program plans in this area as are available and the Long-Range Research Plan.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1, 1982 (p.m. --~1 hr.)	RELIABILITY AND PROBABILISTIC ASSESSMENT	(GRIESMEYER) Okrent, Kerr, Bender, Ebersole, Mark, Siess, Lewis

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: D. Okrent

Purpose: To review the RES proposed FY 1984-FY 1985 research funding and programs for the SARA decision unit.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 6, 1982 (tentative)	REGULATORY ACTIVITIES	(DURAISWAMY) Siess, Bender, Carbon, Ward, Ray, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRC Staff

Purpose: To review proposed Regulatory Guides and Regulations.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 7, 1982	SAFETY RESEARCH PROGRAM	(DURAISWAMY) Siess, Okrent, Moeller, Shewmon, Bender, Carbon, Mathis, Ward, Mark, Plesset, Kerr

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: Routine Process

Purpose: To continue discussion on the FY 1984 and FY 1985 NRC Safety Research Program and Budget and to prepare comments for use by the ACRS in its report to the Commission on the FY 1984 and FY 1985 NRC Safety Research Program Budget.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

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:

A-1/2

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JUNE 1982	SYSTEMATIC EVALUATION PROGRAM	(ALDERMAN) Siess,et al

LOCATION: Washington, DC

BACKGROUND:

Who proposed action: NRR

Purpose: To review the completion of the Systematic Evaluation Program on Ginna.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
JULY 1982	WATTS BAR	(BEAL/QUITTSCHREIBER) Ebersole, Bender, Ward

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 due June 1982.

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

DATE	SUBCOMMITTEE	STAFF ENGR. & MEMBERS
To Be Determined	TRANSPORTATION OF RADIOACTIVE MATERIALS	(DURAISWAMY) Siess, Bender, Mark Cons: Langhaar, Shappert, Zudans

LOCATION: To be determined

BACKGROUND:

Who proposed action:

Purpose: To continue review of the adequacy of the NRC package certification procedures.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

SCHEDULE OF ACRS SUBCOMMITTEE MEETING

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

LOCATION: Washington, D.C.

BACKGROUND:

Who proposed action: P. G. Shewmon

Purpose: To continue the review regarding pressurized thermal shock.

PERTINENT PUBLICATIONS AND THEIR AVAILABILITY:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
Office of Public Affairs
Washington, D.C. 20555

No. 82-45
Tel. 301/492-7715

FOR IMMEDIATE RELEASE
(Friday, April 2, 1982)

NRC INVITES PUBLIC TO SUBMIT NOMINATIONS FOR
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Nuclear Regulatory Commission is anticipating a vacancy on its Advisory Committee on Reactor Safeguards and again is inviting the public to submit nominations. Nominations from technical, environmental, and public interest organizations are specifically sought.

The ACRS, a 15-member panel of experts representing a variety of scientific and technical disciplines, advises the Commission on the design features of nuclear facilities as they may affect public health and safety, the adequacy of proposed standards, and other aspects of nuclear regulation.

In the conduct of its activities, the ACRS evaluates items related to nuclear facilities including handling and disposal of radioactive wastes; the suitability of proposed sites with respect to the effect the proposed plant may have on the public health and safety and the effect of site-related features on the safety of the reactor plant itself; the design of the facility including engineered safety features, plant security provisions and safeguards for protecting special nuclear material; the competence of the design, construction, and operating organizations; the training and qualification of operating personnel; the quality assurance program; operating and emergency plans; periodic test and inspection programs for the facility; and the results of operating experience at the plant.

Currently, candidates with a background in one or more of the following areas would be most useful in conducting the work of the Committee.

- Experience in the design, construction or operation of large, complex facilities and/or surveillance-monitoring programs.

- Experience with high pressure/high temperature systems or equipment.

- Nuclear power plant operations including the management of an operating organization and/or the training/certification of operating personnel and/or the maintenance and operational support of reactor facilities.

Candidates should have a high degree of technical competence and experience which can be applied to the evaluation of nuclear safety matters. All qualified nominees will receive full consideration, and appointment will be made without regard to factors such as race, color, religion, sex, age, national origin, political affiliation, marital status, physical handicap, membership or non-membership in an employee organization, personal favoritism or other non-merit factor. The Commission's final selection will be based on individual qualifications as well as the need to balance the technical disciplines required by the Committee to carry out its functions. Nominations received prior to June 15 will be considered for the forthcoming vacancy on the Committee. Nominations received after that date will be considered for future vacancies as they develop.

Nominations should be sent to the:

Secretary of the Commission
Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Advisory Committee Management Officer

A resume or, as a minimum, a brief statement describing the educational and professional experience of the nominee as well as his/her current address should be included. Candidates must be U.S. citizens able to devote about 100 to 130 days per year to Committee business. An indication of the candidates' availability will facilitate consideration of individuals nominated. Appointments are for four-year terms.

Additional details regarding the duties and functions of the Committee and its members can be obtained by telephoning the Office of the Advisory Committee on Reactor Safeguards (202) 634-3265.

#

The following pages A-49 thru A-51 has been deleted as 1.

DELETION

April 15, 1982

TABLE 1

ASSIGNMENT OF RESPONSIBILITIES FOR THE PREPARATION OF ACRS REPORT TO THE COMMISSION ON NRC'S SAFETY RESEARCH PROGRAM AND BUDGET FOR FY 1984 AND FY 1985

OVERALL RESPONSIBILITY FOR THE ACRS REPORT:

Safety Research Program Subcommittee

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C. P. Siess (Chairman)
M. Bender
M. W. Carbon
W. Kerr
J. C. Mark
D. W. Moeller
D. Okrent
M. S. Plesset
P. G. Shewmon
D. A. Ward
S. Duraiswamy (Staff)

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

October 20, 1981

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

SUBJECT: ACRS REVIEW AND REPORTS ON SAFETY RESEARCH PROGRAMS

Dear Dr. Palladino:

Since 1977, the ACRS has been required by the Congress to report to it annually on the NRC Safety Research Program. This report is prepared each year, after OMB has transmitted the budget request to the Congress in November, and is submitted in February before the appropriate Congressional committees complete their recommendations on the authorization bill.

Since 1979, we have provided a report to the Commission on the research program and its budget, usually just before the EDO budget goes to the Commission for final action in July. This report has been similar in scope to the Report to Congress, although the original request from the Commission was for comments on the budget rather than a complete review of the safety research program.

In 1981, we prepared a report to the Commission on the draft Long Range Research Plan (LRRP). This report was in the form of a letter rather than the format of the other two reports noted above. This report, too, was requested by the Commission, and existing procedures call for similar reviews and reports on the yearly updates of the LRRP.

We believe that our reviews of the safety research program in general, and of individual areas and projects, have been useful to both us and the RES Staff. We believe that the Staff has been responsive in large part to our comments and recommendations.

However, we do not believe that the benefits from our reviews and reports justify the expenditure of resources by the ACRS, its Staff and consultants, and by the RES Staff, that has been required to make three separate reviews each year and prepare three separate reports. We understand that Mr. Minogue agrees with this evaluation.

We propose to ameliorate this situation, without reducing the extent or effectiveness of our review of the program and our interaction with the RES Staff, by the following procedures:

Report to Congress. We will continue to prepare this report, as before. It will be relatively long and relatively comprehensive, and will provide comments on the nature, scope and effectiveness of the program as well as on needs and proposed funding levels. This report will continue to be available in February, and thus can be used by the RES Staff as a basis for its update of the LRRP and its preparation of the next budget cycle.

Report to the Commission. If requested, we will, of course, provide comments or advice to the Commission on the RES budget request or on specific portions of the safety research program or on funding levels in detail or in general. However, we prefer not to provide evaluations and comments of the kind and scope already included in the Report to Congress. Such a report to the Commission would be brief and in letter form.

Long Range Research Plan. The first LRRP developed was little more than a five-year projection of current programs and current needs, and provided little to review in addition to the reviews we had already made of ongoing programs and those planned for the next one or two years. We believe, therefore, that reviewing the LRRP would not be an effective use of our time unless a more meaningful plan is developed.

We would be pleased to have your comments on these proposed changes in procedures, and we will be willing to discuss them with you and the Commissioners at your convenience.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

RSR01Q002

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

CHAIRMAN

December 10, 1981

Dr. J. Carson Mark
Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Mark:

Your letter of October 20 on the ACRS's several annual reviews of the RES program highlights a concern over the amount of time and effort that both the ACRS and the RES staff expend on the reviews. RES has estimated that it expends on the order of 150 to 300 man weeks per year preparing for, participating in, and doing follow-up analysis for the ACRS reviews of its program and budget. Your letter clearly expresses concern for the amount of ACRS time also spent in these reviews. We share with you a desire to significantly reduce the time spent on these reviews, while at the same time not reducing the benefit that we and the Congress receive from the ACRS input and guidance to our research program.

We agree that ACRS should not have to perform three separate reviews each year. We concur with your proposed approach to develop a plan in which the Committee would conduct only one thorough review each year, with possibly the need for some updating by RES to keep you abreast of important changes. We agree with your recommendations regarding the report to the Commission on the RES budget request and the preparation of a comprehensive report to the Congress in February of each year. However, we believe that in view of the timing of the annual report to Congress on the Research program, it would also benefit the Commission, Congress and the RES staff if this review included consideration of the Long Range Research Plan (LRRP).

It is our intention that the annual development and refinement of the LRRP constitute the foundation for the planning of our research program. Preparation of the LRRP permits us to lay out directions of the research program for the coming years and to obtain user office endorsement of these program directions. At this time, the research programs are in the formative stage and are more amenable to guidance and advice than at any other stage. The planning effort can benefit from the perspective of a group of experts in nuclear safety problems who are in intimate contact with the current regulatory challenges. A thorough review by ACRS at this stage should provide all of the background and material needed to

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RSR-1

allow the fulfillment of your obligations to the Congress and would be sufficient to provide my fellow Commissioners and me the benefit of your advice for our review of the RES budget in the summer. Normally, the plan would be available for your review in December with sufficient time for you to hold subcommittee meetings in January. The report for the Congress could also include an update of any changes (mostly deletions) to the program for the budget year being considered by the Congress that may have occurred as a result of the NRC-internal and OMB budget review process.

The LRRP for 1983-1987 (NUREG-0740) was the first attempt at preparing a plan under the current criteria. (Five-year plans were prepared in 1976 and 1977.) RES has received constructive criticism on that plan; the next version, which is now being prepared, will have the benefit of that advice. We expect that it will include more detailed program descriptions, discussion of need and expected use of results.

Thus, we concur with the ACRS recommendations contained in your letter of October 20, 1981 with the exception that an ACRS review of the LRRP be included in the comprehensive review of the research program which forms the basis for your annual report to Congress. This would give us the benefit of your advice at the earliest and most productive stage and, we believe, result in the most efficient use of your and our RES staff time.

Commissioner Ahearne agrees to the ACRS reducing their level of budget review, but would have preferred to retain some level of ACRS review and comment on the more significant items. The report to Congress is on the budget submitted to Congress. This review does not duplicate the report to the Commission on the budget being considered for future submittal. However, given the nature of the Long Range Research Plan, he agrees to relieving ACRS of their review of the Long Range Research Plan.

Please let us know if you have any additional thoughts on this matter.

Sincerely,

Joe

Nunzio J. Palladino

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

December 14, 1981

Rehm
Austin
Stello
[redacted]
Davis
Shapar
DeYoung
Michelson
Central Files

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

SUBJECT: ACRS REVIEW AND REPORTS ON NRC SAFETY RESEARCH PROGRAMS

Dear Dr. Palladino:.

In our letter of October 20, 1981 we expressed our belief "that reviewing the LRRP would not be an effective use of our time unless a more meaningful plan is developed." Although we anticipate significant improvements in the LRRP, it is perhaps too late to use the new LRRP as a basis for our report to Congress on the FY 1983 program since that report is well under way, and we have not yet received the new plan. Nevertheless, we intend to review the plan and, to the extent needed and practicable, provide you and the Commissioners with our comments. It is likely that our comments this year can be based primarily on the reviews we have carried out in preparation for our report to Congress; extensive interaction with the RES Staff should not be necessary. Nevertheless, we will consider ways in which our review of the FY 1984 Safety Research Program can be carried out in order to provide you with timely and useful comments on the LRRP and, at the same time, provide us with the information and insights we need to prepare our report to the Congress.

With regard to a review and report to the Commission in July on the RES budget request, we said in our letter of October 20, 1981 that we will continue to provide comments on funding levels, in detail or in general, and on specific portions of the program. In doing so, however, we would expect to limit our interaction with the RES Staff; this would be possible if there is an easily identifiable relation between their budget request and the needs and programs described in the LRRP. Moreover, we would not intend to elaborate on the bases for our recommendations if it is possible to relate them to comments made previously in connection with the LRRP and our report to Congress.

We will continue to make both general and specific recommendations to the Commission and to the RES Staff. It would be helpful to us in our continuing review of the Safety Research Program, if RES would respond in writing to each recommendation, general or specific, made in our report to the Congress.

In summary, we believe that procedures can be developed to provide the information requested in your letter of December 10, 1981.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

TABLE 1

ASSIGNMENT OF RESPONSIBILITIES FOR THE PREPARATION OF ACRS REPORT TO THE COMMISSION ON THE NRC SAFETY RESEARCH PROGRAM AND BUDGET FOR FY 1984 AND FY 1985

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
PLESSET (Boehnert)	1	1. LOCA AND TRANSIENT RESEARCH	SHEWMON (Beal)	1. f. Fuel Behavior Under Operational Transients	--	--
	2	2. LOFT	--	--	--	--

9-5-8

TABLE 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
KERR (Beal)	3	3. ACCIDENT EVALUATION AND MITIGATION	SHEWMON (Beal)	3. a. Behavior of Damaged Fuel	--	--
			MOELLER (Tang)	c. Fission Product Release and Transport	--	--
			OKRENT (Beal)	d. Accident Mitigation	--	--
KERR (Savio)	--	--	--	--	WARD (Fischer)	6. b. Plant Instrumentation and Control

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TABLE 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
CARBON (Boehnert)	4.	4. ADVANCED REACTORS	--	--	--	--
BENDER (Igne)	5	5. REACTOR AND FACILITY ENGINEERING	SIESS (Igne)	5. a. Structural Engineering Portion of Subelement 5.a	--	--
			OKRENT (Savio)	SSMRP Portion of Subelement 5.a	--	--
			SHEWMON (Igne)	b. Primary System Integrity	--	--
			RAY (Cappucci)	c. Electrical Equipment Qualification	--	--
			MOELLER (Alderman)	d. Fuel Cycle Facility Safety	--	--
			MOELLER (Tang)	e. Effluent Control and Chemical Systems	--	--
				f. Decommissioning	--	--

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TABLE 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
WARD (Fischer)	6	6. FACILITY OPERATIONS AND SAFEGUARDS	MOELLER (Tang)	6. c. Occupational Protection d. Emergency Preparedness	--	--
			KERR (Savio)	b. Plant Instrumentation and Control	--	--
			MARK (Bucci)	e. Safeguards	--	--

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.E 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY: Chapter	PRIMARY RESPONSIBILITY: Decision Unit Corresponding to Chapter	RECEIVES HELP FROM: ACRS Member (Staff Engr.)	RECEIVES HELP FROM: On Subelements	PROVIDES HELP TO: ACRS Member (Staff Engr.)	PROVIDES HELP TO: On Subelements
MOELLER (Tang/ Donoghue)	7	7. WASTE MANAGEMENT	--	--	KERR (Beal)	3. c. Fission Product Release and Transport
					BENDER (Igne)	5. e. Effluent Control & Chemical Systems f. Decommis-sioning
MOELLER (Alderman)	--	--	--	--	BENDER (Igne)	d. Fuel Cycle Facility Safety
					OKRENT (Griesmeyer)	9. c. Trans-portation and Material Risk
MOELLER (Tang)	8	8. SITING AND ENVIRONEMENTAL RESEARCH	OKRENT (Savio)	8. a. Earth Sciences	WARD (Fischer)	6. c. Occupa-tional Protection d. Emergency Prepared-ness

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TABLE 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY: Chapter	PRIMARY RESPONSIBILITY: Decision Unit Corresponding to Chapter	RECEIVES HELP FROM: ACRS Member (Staff Engr.)	RECEIVES HELP FROM: On Subelements	PROVIDES HELP TO: ACRS Member (Staff Engr.)	PROVIDES HELP TO: On Subelements
OKRENT (Griesmeyer)	9	9. SYSTEMS AND RELIABILITY ANALYSIS	MOELLER (Alderman)	9. c. Transportation Materials Risk	--	--
OKRENT (Beal)	--	--	--	--	KERR (Beal)	3. d. Accident Mitigation
OKRENT (Savio)	--	--	--	--	MOELLER (Tang)	8. a. Earth Sciences
					BENDER (Igne)	5. a. SSMRP Portion of Subelement 5.a

A-6.3

TABLE 1 (Cont'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
SHEWMON (Beal)	--	--	--	--	PLESSET (Boehnert)	1. f. Fuel Behavior Under Operational Transients
					KERR (Beal)	3. a. Behavior of Damaged Fuel
SHEWMON (Igne)	--	--	--	--	BENDER (Igne)	5. b. Primary System Integrity

TABLE 1 (Con'd)

ACRS Member (Staff Engr.)	PRIMARY RESPONSIBILITY		RECEIVES HELP FROM		PROVIDES HELP TO	
	Chapter	Decision Unit Corresponding to Chapter	ACRS Member (Staff Engr.)	On Subelements	ACRS Member (Staff Engr.)	On Subelements
SIESS (Igne)	--	--	--	--	BENDER (Igne)	5. a. Structural Engineering Portion of Subelement 5.a.
RAY (Cappucci)	--	--	--	--	BENDER (Igne)	5. c. Electrical Equipment Qualification
MARK (Bucci)	--	--	--	--	WARD (Fischer)	6. e. Safeguards

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April 15, 1982

TABLE 2

RESPONSIBLE PERSONNEL AND SUBCOMMITTEES FOR REVIEWING DECISION UNITS, SUBELEMENTS AND THE BUDGET ASSOCIATED WITH THE NRC'S SAFETY RESEARCH PROGRAM

OVERALL RESPONSIBILITY FOR THE ACRS REPORT:

Safety Research Program Subcommittee

C. P. Siess (Chairman)
M. Bender
M. W. Carbon
W. Kerr
J. C. Mark
D. W. Moeller
D. Okrent
M. S. Plesset
P. G. Shewmon
D. A. Ward
S. Duraiswamy (Staff)

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TABLE 2

RESPONSIBLE PERSONNEL AND SUBCOMMITTEES FOR REVIEWING DECISION UNITS, SUBELEMENTS AND THE BUDGET ASSOCIATED WITH THE NRC'S SAFETY RESEARCH PROGRAM

	DECISION UNITS		SUBELEMENTS			
CHAPTER NUMBER	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
1	1. LOCA AND TRANSIENT RESEARCH	Plesset (Boehnert)	1. a. Semiscale	ECCS	Plesset (Boehnert)	Experimental Programs *(H. Sullivan)
			b. Separate Effects Experiments and Model Development	ECCS	Plesset (Boehnert)	
			c. 2-D/3-D Program	ECCS	Plesset (Boehnert)	
			d. Code Improvement and Maintenance	ECCS	Plesset (Boehnert)	Analytical Models *(L. Shotkin)
			e. Code Assessment and Application	ECCS	Plesset (Boehnert)	
			f. Fuel Behavior Under Operational Transients	Reactor Fuel	Shewmon (Beal)	Fuel Behavior (M. Silberberg)

*Acting Branch Chief.

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TABLE 2 (Cont'd)

CHAPTER NUMBER	DECISION UNITS: Number and Name of Decision Unit	DECISION UNITS: Primary Responsibility ACRS Member (Staff Engineer)	SUBELEMENTS: Number and Name of the Subelement	SUBELEMENTS: Responsible Subcommittee	SUBELEMENTS: Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	SUBELEMENTS: Responsible NRC Branch and Branch Chief
2	2. LOFT	Plesset (Boehnert)	2.			
			a. Test Operation and Facility Support	ECCS	Plesset (Boehnert)	Experimental Programs *(H. Sullivan)
			b. Analysis and Engineering	ECCS	Plesset (Boehnert)	
			c. Fuel Processing and Examination	ECCS	Plesset (Boehnert)	
			d. Project Close-out	ECCS	Plesset (Boehnert)	
			e. Stand-by Activities	ECCS	Plesset (Boehnert	
			f. Decontamination and Decommissioning	ECCS	Plesset (Boehnert)	

*Acting Branch Chief.

TABLE 2 (Con'd)

	DECISION UNITS		SUBELEMENTS			
CHAPTER NUMBER	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
3	3. ACCIDENT EVALUATION AND MITIGATION	Kerr (Beal)	3. a. Behavior of Damaged Fuel	Reactor Fuel	Shewmon (Beal)	Fuel Behavior (M. Silberberg)
			b. Fuel Melt Behavior	Class 9	Kerr (Beal)	Severe Acc. Assessment (R. Curtis)
			c. Fission Product Release and Transport	Reactor Radiological Effects	Moeller (Tang)	Fuel Behavior (M. Silberberg)
			d. Accident Mitigation	Class 9	Okrent* (Beal)	Severe Acc. Assessment (R. Curtis)

*Subcommittee Member provides assistance on the Subelement 3.d "Accident Mitigation".

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TABLE 2 (Cont'd)

CHAPTER NUMBER	DECISION UNITS		SUBELEMENTS			
	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
4	4. ADVANCED REACTORS	Carbon (Boehnert)	4. a. Fast Reactors	Advanced Reactors	Carbon (Boehnert)	Severe Acc. Assessment (R. Curtis)
			b. Gas-Cooled Reactors	Advanced Reactors	Carbon (Boehernt/ McKinley)	Experimental Programs *(H. Sullivan)

*Acting Branch Chief.

CHAPTER NUMBER	DECISION UNITS: Number and Name of Decision Unit	DECISION UNITS: Primary Responsibility ACRS Member (Staff Engineer)	SUBELEMENTS: Number and Name of the Subelement	SUBELEMENTS: Responsible Subcommittee	SUBELEMENTS: Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	SUBELEMENTS: Responsible NRC Branch and Branch Chief
5	5. REACTOR AND FACILITY ENGINEERING	Bender (Igne)	5.			
			a. Mechanical and Structural Engineering	Combination of Dynamic Loads (Mechanical Eng. Portion)	Bender (Igne)	Mechanical and Structural Engineering (W. Anderson)
				Extreme External Phenomena (SSMRP Portion)	Okrent (Savio)	
				Structural Engineering (Structural Eng. Portion)	Siess (Igne)	
			b. Primary System Integrity	Metal Components	Shewmon (Igne)	Materials Engineering (C. Serpan)
			c. Electrical Equipment Qualification	Qualification Programs for Safety-Related Equipment	Ray (Cappucci)	Electrical Engineering (D. Sullivan)
			d. Fuel Cycle Facility Safety	Fuel Cycle	Moeller (Alderman)	Chemical Engineering (K. Steyer)
			e. Effluent Control and Chemical Systems	Reactor Radiological Effects	Moeller (Tang)	
			f. Decommissioning	Reactor Radiological Effects	Moeller (Tang)	

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TABLE 2 (Cont'd)

CHAPTER NUMBER	DECISION UNITS		SUBELEMENTS			
	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
6	6. FACILITY OPERATIONS AND SAFE-GUARDS	Ward (Fischer)	6. a. Human Engineering and Man-machine	Human Factors	Ward (Fischer)	Human Factors (J. Norberg)
			b. Plant Instrumentation and Control	Electrical Systems	Kerr (Savio)	Inst. & Control (E. Wenzinger)
			c. Occupational Protection	Reactor Radiological Effects	Moeller (Tang)	Occupational Protection (R. Alexander)
			d. Emergency Preparedness	Reactor Radiological Effects	Moeller (Tang)	Human Factors (J. Norberg)
			e. Safeguards	Safeguards and Security	Mark (Bucci)	Safeguards (J. Durst)

TABLE 2 (Cont'd)

	DECISION UNITS		SUBELEMENTS			
CHAPTER NUMBER	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
7	7. WASTE MANAGEMENT	Moeller (Tang/ Donoghue)	7.			
			a. High Level Waste	Waste Management	Moeller (Tang/ Donoghue)	Waste Management (J. J. Davis)
			b. Low Level Waste	Waste Management	Moeller (Tang/ Donogue)	
			c. Uranium Recovery	Waste Management	Moeller (Tang/ Donoghue)	

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TABLE 2 (Cont'd)

	DECISION UNITS		SUBELEMENTS			
CHAPTER NUMBER	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
8	8. SITING AND ENVIRONMENTAL RESEARCH	Moeller (Tang)	8. a. Earth Sciences	Extreme External Phenomena/ Reactor Radiological Effects	Moeller/ Okrent (Tang/ Savio)	Earth Sciences (L. Beratan)
			b. Siting	Site Evaluation	Moeller (Tang)	Siting & Environmental (E. Conti)
			c. Health Effects	Reactor Radiological Effects	Moeller (Tang)	Health Effects (W. A. Mills)
			d. Environmental Impacts	Reactor Radiological Effects	Moeller (Tang)	Siting & Environmental (E. Conti)

TABLE 2 (Cont'd)

	DECISION UNITS		SUBELEMENTS			
CHAPTER NUMBER	Number and Name of Decision Unit	Primary Responsibility ACRS Member (Staff Engineer)	Number and Name of the Subelement	Responsible Subcommittee	Responsible Personnel ACRS Subcommittee Chairman (Staff Engineer)	Responsible NRC Branch and Branch Chief
9	9. SYSTEMS AND RELIABILITY ANALYSIS	Okrent (Griesmeyer)	9.			
			a. Risk Methods and Data Evaluation	Reliability & Probabilistic Assessment	Okrent (Griesmeyer)	Risk Methodology & Data (B. Buchbinder)
			b. Reactor Risk and Reliability Analysis	Reliability & Probabilistic Assessment	Okrent (Griesmeyer)	Reactor Risk and Regulatory Analysis (G. Burdick & J. Tomlin)
			c. Transportation and Material Risk	Fuel Cycle	Moeller (Alderman)	Transportation & Materials (J. Malaro)

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COMMITTEE ON SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES

SUITE 2321 RAYBURN HOUSE OFFICE BUILDING
WASHINGTON, D.C. 20515
(202) 225-6371

May 4, 1982

Dr. Robert B. Minogue
Director
Office of Nuclear Regulatory Research
Nuclear Regulatory Commission
Washington, DC 20555

ATTN: FRANK
GILL...

Dear Dr. Minogue:

As you know, the Subcommittee is holding hearings on the NRC research program on May 18. To facilitate this evaluation, I am enclosing a copy of questions the Subcommittee staff has concerning the program. I hope this additional information can be made available to us by May 13, 1982.

Thank you for your assistance.

Sincerely,

John V. Dugan Jr.

JOHN V. DUGAN, JR.
Staff Director
Subcommittee on Energy Research
and Production

JVD:Pma
Enclosure

A-56

QUESTIONS FROM
ENERGY RESEARCH & PRODUCTION SUBCOMMITTEE ON
NRC RESEARCH PROGRAM

1. Please provide a cost breakdown of FY'83 research funding to individual program elements.

2. Please describe the PWR instrumentation being evaluated at Semiscale.

3. What design changes have caused delays at UPTF in Germany?

4. Describe the improved instrumentation and advanced display systems to be evaluated at LOFT in FY'83. Who proposed the designs?

5. What contribution is being made by industry to the NRC work on guidelines for management of severe accidents?

6. Is NRC doing any research on seismic designs which depend more heavily on the ability of large pipes to absorb energy when not held rigidly in place? Is any research at NRC addressing the tradeoffs between design and operational constraints in the use of snubbers and restraints?

7. Please describe the SSMRP program in more detail.

8. Please describe the steel and concrete containment experiments with large penetrations being conducted under the Reactor & Facility Engineering decision element.

9. Describe the improved real-time ultrasonic test method under development in FY'82 and its performance characteristics.

10. What reactor surveillance techniques will be evaluated in FY'83? Who proposed these techniques? What methods will be used to ensure that modifications in the operator-machine interface will actually improve safety? What work will be initiated to improve quantitative estimates of human reliability?

11. Describe NRC interaction with utilities on improvement of the impact on safety of human performance problems.

12. Who developed the requirements for SPDS? Is NRC conducting any continuing research on operator informational needs?

13. Is NRC conducting any research on reducing the impact of maintenance problems on overall plant safety?

14. What does NRC consider to be "an adequate degree of radiation protection for workers?" (NRC Budget Estimates - Fiscal Year 1983, p. 72)

15. Please describe any research that NRC is doing to address decontamination of systems contaminated as a result of accidents.

16. Describe cooperative studies with the state of Kentucky to assist the state in decommissioning the Maxey Flats site. Is this work generically applicable?

17. What engineering designs for managing uranium mill tailings are being evaluated by NRC? Who proposed them?

18. Please describe NRC research programs to protect "the health and safety of the public and workers from potentially harmful effects...of ionizing radiation and radioactive materials." (NRC Budget Estimates - Fiscal Year 1983, p. 75)

19. Please describe in detail work to be funded in the Systems and Reliability Analysis decision unit. How are the results of this work being used by the licensing and inspection staff in prioritizing and resolving safety issues? Please cite examples. How does this work help to stabilize regulatory practices? Please cite examples.

20. Please describe NRC's operations research as applied to licensee safety assurance practices.

21. Are there any results to date on analyses of the regulatory structure and safety evaluation practices? Please describe the methodology used in this work.

22. What programs are planned or under way to investigate alternative techniques of system safety assurance drawn from other industries? Please describe results to date.

23. What improvements were suggested for B&W auxiliary feedwater systems using risk analysis techniques in 1980?

24. Describe the NRC effort "to identify off-target or unnecessary regulations" using Systems and Reliability Analysis. What regulations are under consideration in this work? (NRC Budget Estimates - FY 1983, p. 77)

25. How do NRC researchers develop their expertise on reactor operation? What percent of research staff has utility experience? Reactor manufacturing experience?

26. Describe the process used by the NRC research organization to set priorities for the use of research dollars. To what extent are relative risk comparisons used in establishing the research budget request? Cite examples.

27. Are annual and monthly routine reports described in Regulatory Guide 1.16 used in setting research priorities? What analysis of these data is conducted by NRC research? What about analysis and use of reportable occurrences described in the same source? For instance, are causes of potentially serious incidents analyzed for trends? What percent of potentially serious incidents show human action to be a direct or collateral cause? How often does incorrect or tardy maintenance contribute to a potentially serious incident?

28. If the NRC were to reopen proceedings related to the Generic Environmental Statement on Mixed Oxide Fuel (GESMO), would additional research be required? If so, please describe the scope and magnitude of that research.

29. Why is the FY'83 safeguards program based on the assumption that nuclear fuel reprocessing will not be proposed by the industry? What adjustments or additions would be necessary if this assumption were removed?

30. The February draft of the Long Range Research Plan for FY 1984-88 states that "Human performance is now generally recognized to be a significant and perhaps dominant determinant of the risk resulting from operating nuclear facilities." How much of the NRC research budget directly addresses this problem? Please describe the program elements included.

31. Please describe the respective roles of NRC and DOE in nuclear safety research. What is the nature of NRC's participation on the DOE working groups formed in response to P.L. 96-567?

32. Is NRC research currently addressing questions involving the use of computers at the primary operator-reactor interface? If so, describe the research. If not, why not?

33. Is NRC currently developing methods for judging the competency of a utility, its management, and personnel? If so, please describe this work. If not, why not?

34. What is the breakdown, in terms of money and manpower, of the waste management program for (a) commercial high level waste, (b) commercial low level waste, (c) waste from uranium recovery operations, and (d) any other waste management R&D programs?

35. In uranium tailings research, is probabilistic risk analysis used in assessing the alternatives to covers for attenuating radon emanations?

36. What is the nature of the present interaction with EPA personnel? How do you propose to improve the interagency cooperation between NRC and EPA? Likewise, DOE and NRC?

37. How do you incorporate the results of research by other agencies, particularly DOE and EPA, or other research organizations, in your high-level waste program?

38. What are NRC's research activities regarding transuranic waste in the range of 10-100 nanocuries?

39. What is the status of the NRC effort to develop models for estimating quantitatively the performance of high level waste repositories?

40. Would NRC need to do additional research regarding the disposal of high level waste created by reprocessing? How long would such R&D take to complete?

41. In the Review and Evaluation of the Nuclear Regulatory Commission's Safety

Research Program for fiscal year 1983, the ACRS stated that "the DOE schedule for site selection and repository construction may be accelerated, perhaps by as much as 2 years. Final DOE site recommendations may therefore occur as early as January 1983. In order to evaluate properly the recommended sites, the NRC should have final siting criteria prior to that time. While the proposed FY 1983 budget for this Decision Unit will permit the NRC to meet this schedule, it will necessitate some curtailment in the level of effort originally planned." Does this reference to a site recommendation mean site characterization? In reference to "curtailment" and "level of effort originally planned," what is the original schedule for this research program? In audition, the same report goes on to state, "As a result, the NRC staff will have a reduced level of confidence that the criteria they develop will assure an acceptable disposal facility." What won't be done under the accelerated schedule to result in such a loss of staff confidence? What level of confidence does the current program assure? Is this adequate? If this is not adequate, what is needed to assure confidence?

42. Please describe the allocation of the $13 million request for advanced reactors for FY'83. Why do these two new reactor designs require relatively little funding in comparison with the better known light water reactors?

43. Is there any chance that research needs for CRBR will delay the regulatory review and licensing of that plant?

44. Describe the interface between DOE and NRC research for CRBR.

45. What changes have occurred in the FY'83 program in response to the ACRS review?

46. Have changes been made in response to the NSOC Reactor Safety Research Review Group report in September 1981? Please enumerate them.

47. The ACRS pointed out in an April 5 letter to Chairman Palladino concerning the Long Range Research Plan that "much still remains to be done in identifying... problems that represent the greatest potential contributors to risk." Is the NRC considering developing a systems approach for setting long-range research priorities which would respond to such critiques?

48. The ACRS also commented on April 5 that "the (Long Range Research) Plan does

6 A-81

not address research on LMFBRs or other advanced-reactor types beyond the CRBR." Please describe the NRC position on such research. If the NRC were to conduct sufficient research on LMFBRs or HTGRs to support commercial licensing, how many man-years of effort would be required? Does the NRC have contingency plans in the out-years of their program to address advanced reactor designs if such research becomes appropriate?

RECORD copy

265th ACRS Meeting

May 7, 1982.

APPENDIX VII
NRC R&D HEARINGS ANNOUNCEMENT - SUBCMTE ON ENERGY RESEARCH AND PRODUCTION

LARRY WINN, Jr.
Ranking Minority Member

#97-202
FOR IMMEDIATE RELEASE
April 27, 1982

CONGRESSWOMAN MARILYN L. BOUQUARD ANNOUNCES
HEARINGS ON NUCLEAR REGULATORY COMMISSION R&D

Congresswoman Marilyn L. Bouquard (D-TN), Chairman of the Subcommittee on Energy Research and Production, announced today that the Subcommittee will initiate oversight hearings on the research program of the Nuclear Regulatory Commission. The hearings will be held on Tuesday, May 18, 1982 from 9:00 a.m. to 12:00 noon and on Wednesday, May 19, 1982 from 1:00 p.m. to 5:00 p.m. in Room 2318 of the Rayburn House Office Building.

The Subcommittee will investigate the formulation of research priorities, the balance and technical effectiveness of the program and its contribution to the Commission's goal of protecting public health and safety and the common defense and security in the construction and operation of commercial nuclear powerplants.

In announcing the hearings, Mrs. Bouquard said "The $220 million NRC research program represents more than seventy-five percent of all nuclear safety R&D in this country. In the current economic climate, it is essential that the program be subjected to critical technical review. It is also essential that the results of the research be used effectively by all segments involved in nuclear power generation - the NRC, DOE, reactor manufacturers, and utilities which operate nuclear reactors. Real improvements in nuclear safety can best be attained and implemented through the cooperative efforts of all parties."

The Subcommittee on Energy Research and Production is one of seven subcommittees of the Committee on Science and Technology, chaired by the Honorable Don Fuqua (D-FL).

The schedule and tentative witness list for the hearings are as follows:

<u>Tuesday, May 18, 1982</u>

Dr. Robert B. Minogue
Director
Office of Nuclear Regulatory Research
Nuclear Regulatory Commission

Mr. Gordon Chipman
Depty. Asst. Secty. for Nuclear
Reactor Programs
Office of Nuclear Energy
U.S. Department of Energy

Dr. Chester P. Siess
Chairman
Nuclear Safety Research Program
Advisory Committee on Reactor Safeguards

<u>Wednesday, May 19, 1982</u>

Dr. John Taylor
Director
Nuclear Power Division
Electric Power Research Institute

Dr. Norman C. Rasmussen
Chairman
Reactor Safety Research Review Group
Nuclear Safety Oversight Committee

Dr. Herbert J.C. Kouts
Vice Chairman
Reactor Safety Research Review Group
Nuclear Safety Oversight Committee

Dr. Edwin Zebroski
Vice President
Engineering
Institute of Nuclear Power Operators

Additional witnesses will be announced when the list is finalized.

Staff Contacts: Louis Ventre, Jr. at 225-2981
Ray Pennotti at 225-3557

ATTACHMENT 1

COMMITTEE ON SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES
SUITE 2321 RAYBURN HOUSE OFFICE BUILDING
WASHINGTON, D.C. 20515
(202) 225-6371

April 30, 1982

Dr Chester P. Siess
University of Illinois
Urbana, Illinois 61801

Dear Dr. Siess:

I am pleased to invite you or your designee to testify on Tuesday, May 18, before the Subcommittee on Energy Research and Production. The hearing will address the research program conducted by the Nuclear Regulatory Commission and will be held at 9:00 a.m. in Room 2318 of the Rayburn House Office Building.

The purpose of these oversight hearings is to examine the management of this $220 million research program with particular emphasis on the following issues:

- Mechanism for establishing priorities for program components
- Relationship of NRC research to the regulatory process
- Relationship of NRC research to nuclear safety research at DOE and in industry
- Correspondence of funding levels to relative risk and known incidents in actual operating experience
- Impact of NRC safety research on actual safety in commercial powerplants.

Your testimony should address any of these issues you believe to be appropriate to your interests and expertise, as well as any additional issues you may consider significant.

Although your written statement may be as long and as detailed as you feel necessary, we ask that your oral testimony be limited to 15 minutes in order to provide sufficient time for questioning by the Subcommittee Members.

The Subcommittee will need sixty copies of your prepared statement 48 hours before the time of the hearing for advance distribution to the Subcommittee Members and staff. An additional seventy-five copies of your statement will be needed for distribution to the press at the time of the hearing. A brief biographical sketch suitable for inclusion in the hearing record should be attached. Please direct copies to Dr. Jack Dugan, Staff Director, Subcommittee on Energy Research and

Production, House Committee on Science and Technology, Room B374 Rayburn House Office Building, Washington, D. C. 20515.

If you have any questions regarding the hearing, please contact Dr. Raymond Pennotti, Technical Consultant, at 225-3557 or Mr. Louis Ventre, Jr., Counsel, at 225-2981.

Sincerely,

Marilyn L. Bouquard

MARILYN L. BOUQUARD, Chairman
Subcommittee on Energy Research
and Production

MLB:Pjs
Attachment

APPENDIX VIII
STAFF BRIEFING - GINNA RESTART SAFETY EVALUATION

BRIEFING OF ACRS

STATUS REPORT

GINNA RESTART SAFETY EVALUATION

J. Lyons
x-24362
May 6, 1982

A-87

OUTLINE

I. BACKGROUND

II. KEY TECHNICAL ISSUES RELATED TO RESTART OF GINNA

- STEAM GENERATOR
- PORV
- PROCEDURES/HUMAN FACTORS
- THERMAL TRANSIENT ON REACTOR VESSEL

III. FUTURE ACTIONS AND SCHEDULE

J. Lyons
x-24362

A-88

BACKGROUND

- EVENT SUMMARY
  - TUBE RUPTURE - JAN. 25, 1982
  - PORV AND SECONDARY RELIEF VALVE COMPLICATION
  - COLD SHUTDOWN - JAN. 26, 1982
- EVENT FOLLOWUP
  - CHAIRMAN'S TASK FORCE (NUREG-0909)
  - STAFF RESTART EVALUATION
  - LICENSEE ACTIVITIES
- CURRENT PLANT STATUS
  - TUBE REMOVAL AND INSPECTIONS COMPLETED
  - REFUELING COMPLETED
  - FINAL SG PREPARATION FOR RESTART UNDERWAY

J. Lyons
x-24362

KEY TECHNICAL ISSUES RELATED TO RESTART

- STEAM GENERATORS
- POWER OPERATED RELIEF VALVE
- PROCEDURES/HUMAN FACTORS
- THERMAL TRANSIENT ON REACTOR VESSEL

J. Lyons
x-24362

STEAM GENERATOR

INSPECTIONS

- LOCATED LEAK NEAR TUBE SHEET IN "B" S.G. HOT LEG
- PERFORMED ECT AND FIBER OPTIC INSPECTIONS OF FAILED TUBE
- PERFORMED ECT OF "A" AND "B" S.G.'S - HOT AND COLD LEGS
- BASED ON RESULTS OF PRIMARY SIDE EXAMINATIONS PERFORMED SECONDARY SIDE INSPECTIONS
- DISCOVERED DAMAGE TO PLUGGED TUBES
- DISCOVERED FOREIGN OBJECTS
- METALLURGICAL EXAMINATIONS

FAILURE MECHANISM

- LICENSEE CONCLUDES FOREIGN OBJECT MOST PROBABLE CAUSE OF INITIAL PLUGGED TUBE SEVERANCE
- SUBSEQUENT WEAR ON PLUGGED TUBES RESULTED IN TUBE SEVERANCE
- PHENOMENON CONTINUED UNTIL ACTIVE TUBE FAILED

REPAIRS

- FOREIGN OBJECTS REMOVED
- 2 - 3" PORTS DRILLED IN S.G. SHELL FOR ACCESS TO WEDGE AREA 4 AND 6
- 20 PREVIOUSLY PLUGGED AND STRUCTURALLY DEGRADED TUBES REMOVED FROM WEDGE AREA 4
- 6 TUBES REMOVED FROM WEDGE AREA 6

J. Lyons
x-24362

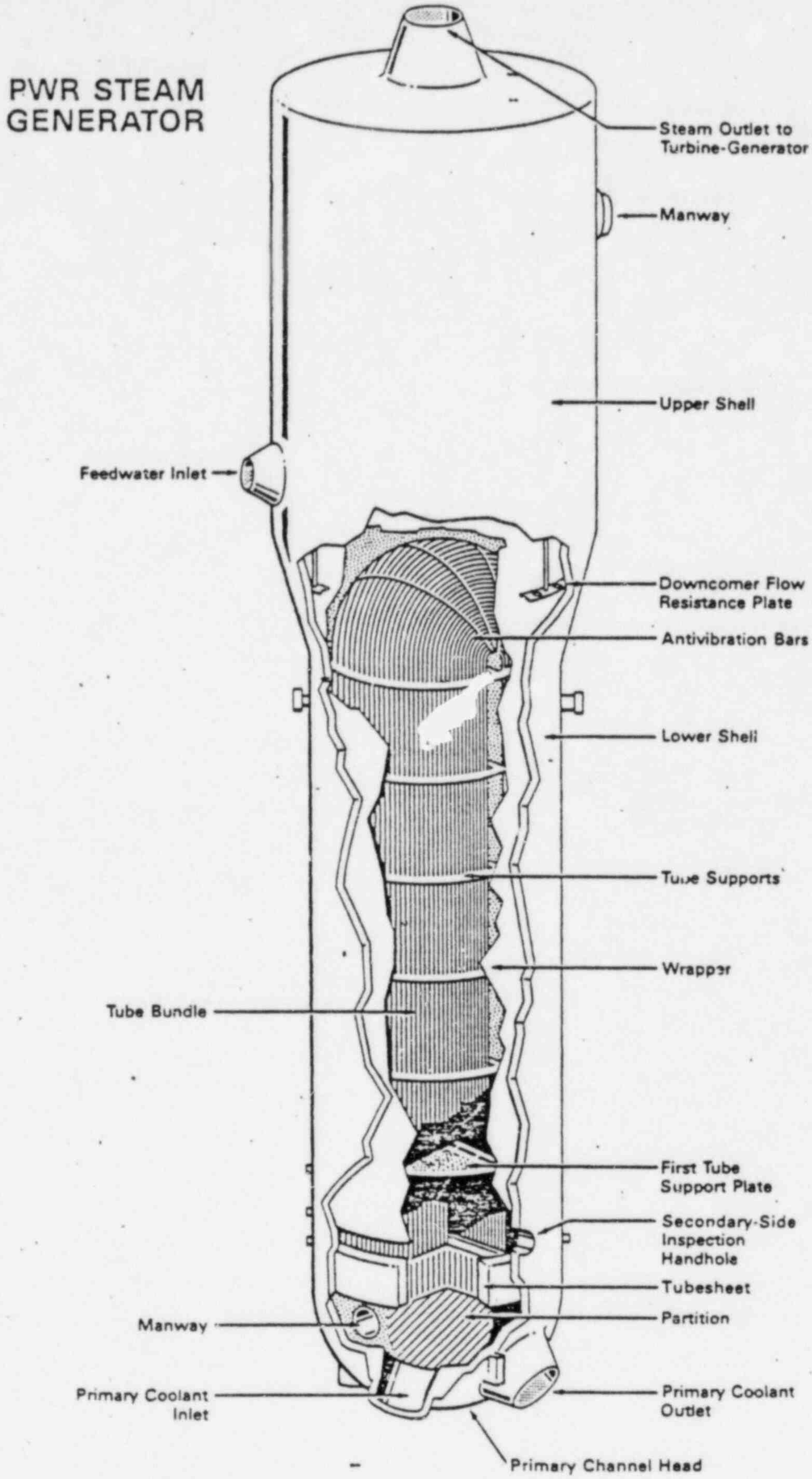

Figure 7.1 Westinghouse Model 44 steam generator

J. Lyons
x-24362

J. Lyons
x-24362

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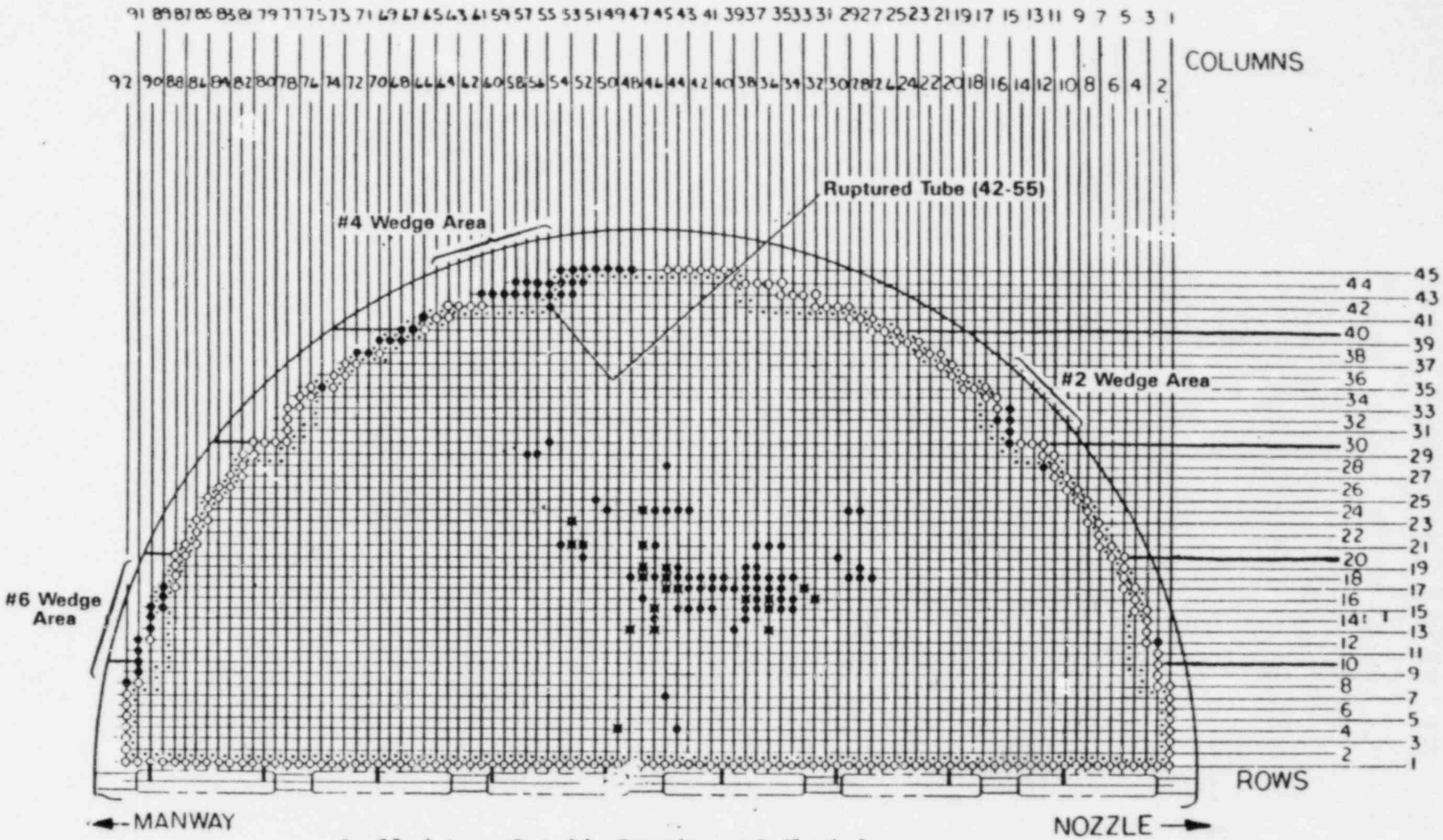
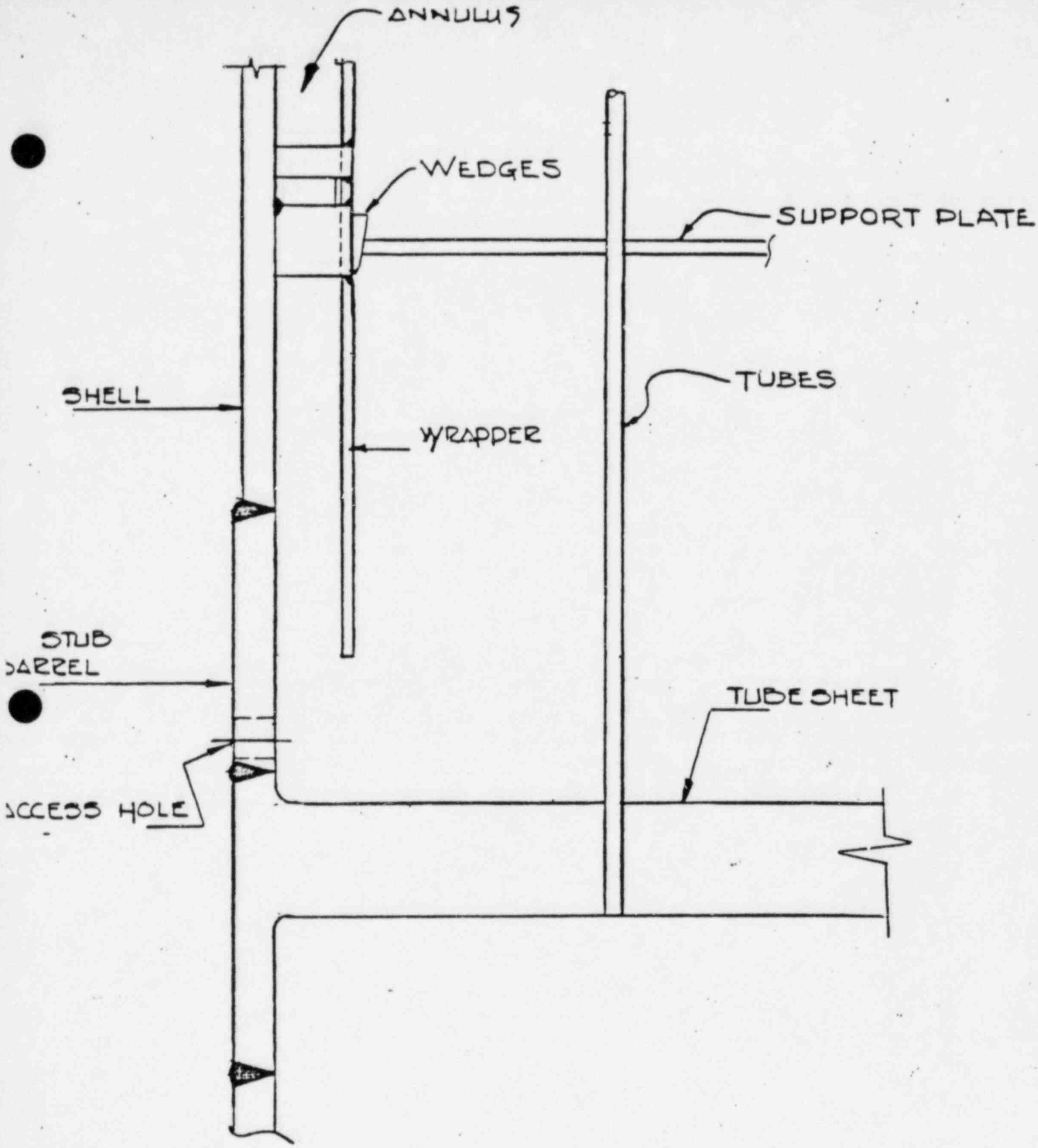


Small dots - Outside Boundary of flowholes
Large dots - Location of Plugged Tubes
X's - Sleeved tubes
Open Circles - Outside boundary of tube locations

Ginna B steam generator hot-leg tube sheet map before event of January 25, 1982



0	ORIGINAL INITIAL DATE				
NUMBER	REVISION	DRAWN BY	CHECKED BY	RESP. ENG.	ENG. MANG'R.
ROCHESTER GAS & ELECTRIC CORP. ROCHESTER, NEW YORK		"B" STEAM GENERATOR 3" ACCESS HOLE		SCALE NO.	

5.7-4 FIGURE 5.1 J. Lyons x-24362

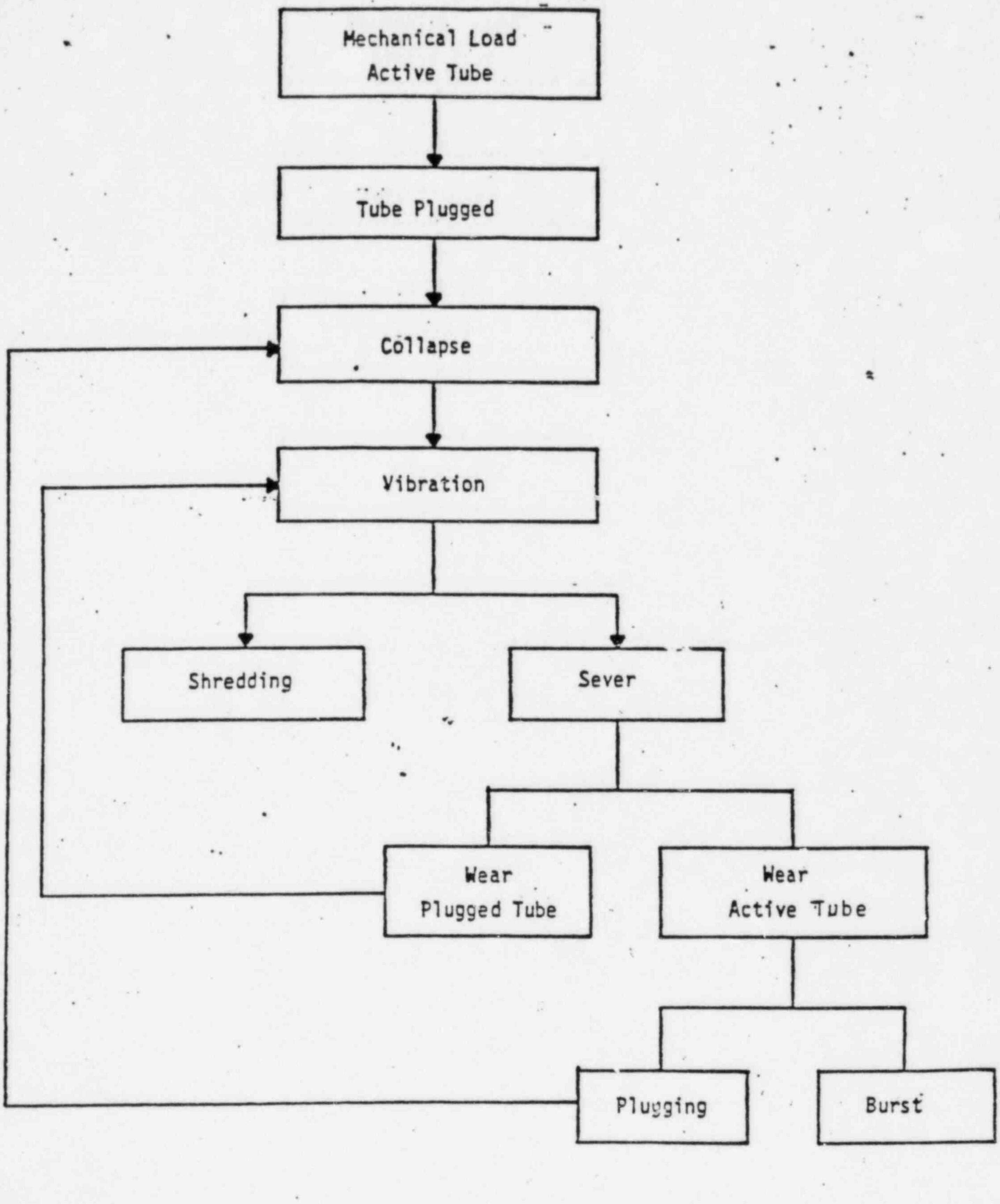
A-94

FOREIGN OBJECTS

B - STEAM GENERATOR

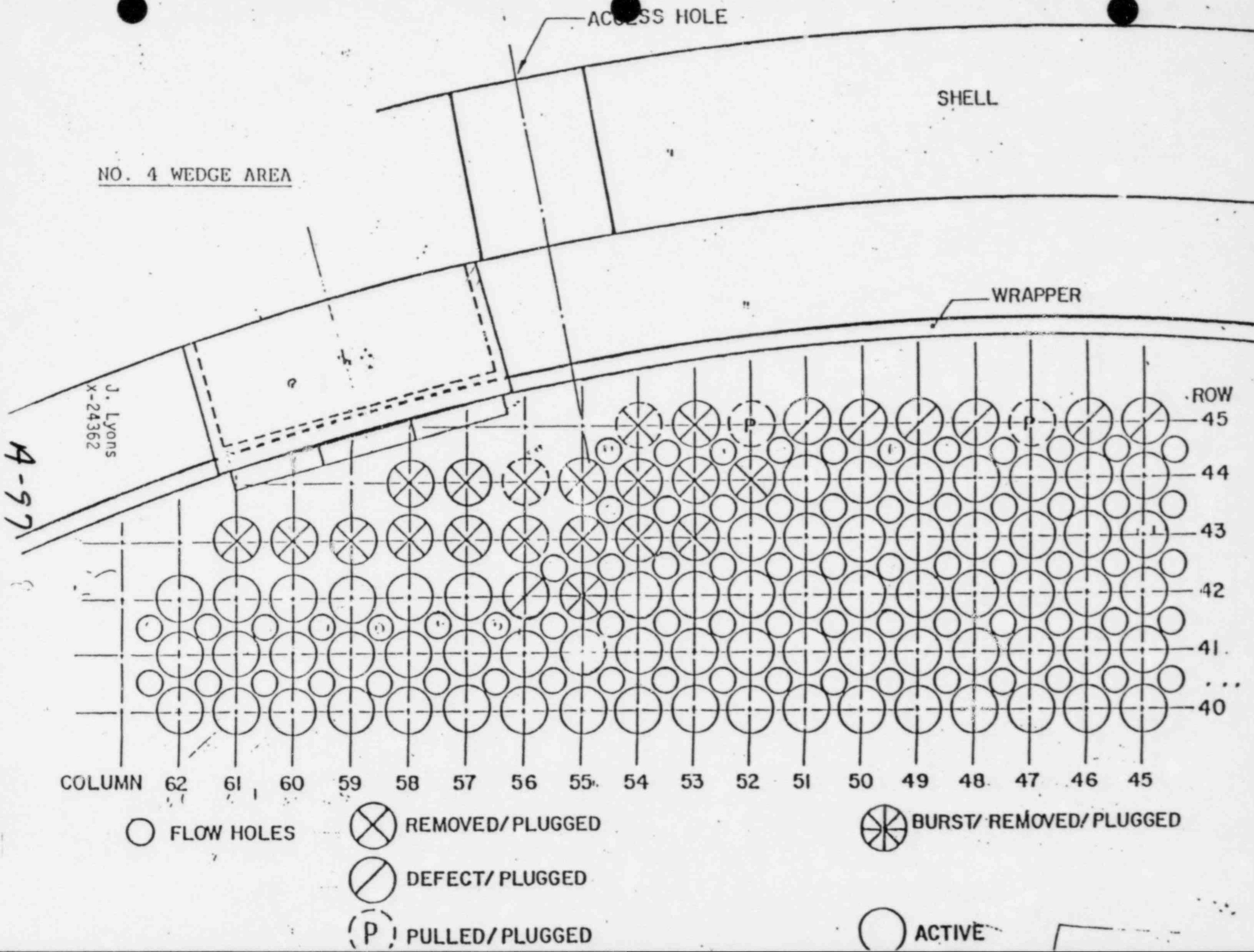
DESCRIPTION	DIMENSIONS	LOCATION
MAGNETIC CARBON STEEL PLATE	0.5" x 4.18" x 6.31"	R25 C85
MAGNETIC CARBON STEEL PLATE	0.5" x 1.5" x 3.5"	R45 C46
MAGNETIC CARBON STEEL PLATE (OVAL)	0.5" x 2.0" x 2.375"	LODGED BETWEEN R45 C53 AND R44 C53
MAGNETIC CARBON STEEL STRIP	0.05" x 0.6" x 4"	R25 C85
COPPER TUBING	0.25" DIAM. x 1.062" LONG	R45 C47
WELDING ELECTRODE	0.18" DIAM. x 2" LONG	R43 C34
SMALL PIECES OF INCONEL TUBING		WEDGE AREA #4 ALSO R30 L81 AND R33 C15

J. Lyons
x-24362



Postulated Failure Mechanism Sequence

J. Lyons
x-24362


NO. 4 WEDGE AREA
ACCESS HOLE
SHELL
WRAPPER
J. Lyons
x-24362
ROW
45
44
43
42
41
40
COLUMN
62
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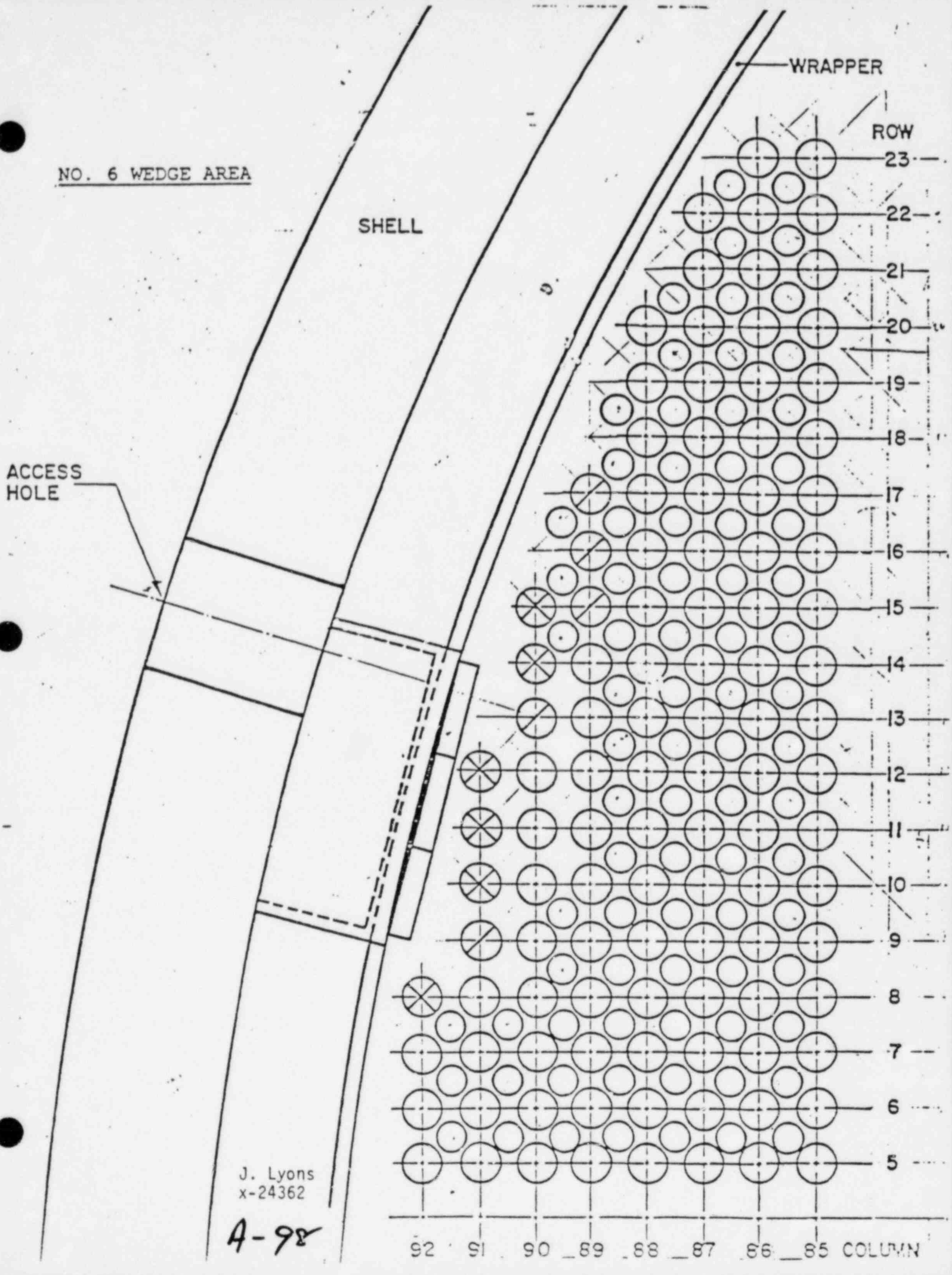
46
45
P
FLOW HOLES
REMOVED/PLUGGED
DEFECT/PLUGGED
PULLED/PLUGGED
BURST/REMOVED/PLUGGED
ACTIVE

A-97


NO. 6 WEDGE AREA
SHELL
WRAPPER
ROW
23
22
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16
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12
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10
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8
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6
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ACCESS HOLE
J. Lyons
x-24362
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92
91
90
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88
87
86
85
COLUMN

POWER OPERATED RELIEF VALVE

- TWO 3" VALVES MANUFACTURED BY COPES VULCAN (AIR TO OPEN)
- ONE VALVE CYCLES SUCESSFULLY 3 TIMES
- ON FOURTH CYCLE VALVE FAILED TO CLOSE
- CLOSED 3 DAYS LATER

FAILURE MECHANISM

- LICENSEE REMOVED AND TESTED PORV'S SUCCESSFULLY
- LICENSEE TRACED PROBLEM TO 3 WAY SOLENOID VALVE
- RESTRICTED EXHAUST PART

REPAIRS

- RESTRICTION ON SOLENOID REMOVED
- CHECK VALVE INSTALLED BETWEEN PORV AND SOLENOID VALVE
- AIR LINE FILTERS INSTALLED

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PROCEDURES/HUMAN FACTORS

- ACTUAL EVENT NOT SPECIFICALLY ADDRESSED IN ANY SINGLE PROCEDURE
- PROCEDURES USED TO MITIGATE EVENT INCLUDED
  - Action and Diagnostics for Spurious SI, LOCA, Loss of Secondary Coolant and SG Tube Rupture
  - Tube Rupture Procedure
  - Plant Shutdown Procedures
  - Draining RCS following tube rupture procedure.
- PROCEDURES COUPLED WITH TRAINING AND PLANT STAFF EXPERIENCE PROVIDED EFFECTIVE RESPONSE TO EVENT
- NEVERTHELESS LICENSEE CONCLUDED THAT CERTAIN MODIFICATIONS TO THE PROCEDURES ARE NECESSARY
- BASED ON EVENT RESPONSE, LICENSEE IS MAKING SOME MODIFICATIONS TO ENHANCE HUMAN FACTORS ASPECTS OF CONTROL ROOM

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Thermal Transient on Reactor Vessel

- Concern is that reactor vessel could have experienced a thermal transient that could affect its subsequent integrity.

- Licensee concludes that for the no mixing case that there is no flaw initiation at reactor vessel beltline welds.

- Staff analysis of transient and metallurgical aspects is still underway.

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LICENSEE'S PROPOSED ACTIONS

- COMPLETE METALLURGIAL EXAMINATIONS
- COMPLETE COLLAPSE AND FATIGUE TESTING
- INSTALL LOOSE PARTS ACOUSTIC MONITORS ON BOTH STEAM GENERATORS
- INTERMEDIATE OUTAGE - (APP. 120 EFPD)
  - EDDY CURRENT
  - FIBER OPTICS
  - VIDEO
  - VISUAL
- PROCEDURES
  - SHORT TERM
  - LONG TERM
- EQUIPMENT MODIFICATIONS
  - PRESSURIZER PORV CONTROL SYSTEM
  - LETDOWN ISOLATION SYSTEM
  - ALARM SETPOINTS FOR MAIN STEAM RADIATION MONITORS
  - POSITION INDICATION FOR MAIN STEAM PORV'S AND SAFETY VALVES
  - RCS SUBCOOLING MONITOR RANGE
  - QUALIFIED WIDE RANGE RCS PRESSURE INSTRUMENT
  - LOGIC FOR IC SAFETY INJECTION PUMP*
  - CONTROL ROOM RECORDERS FOR MAIN STEAM LINE PRESSURE*

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Future Actions

- Staff consultants reports on tube failure mechanism will be provided in restart SER.

- Independent calculations and review of reactor vessel thermal transient being performed.

- Significant findings from NUREG-0909 that affect Ginna restart are being evaluated and will be discussed.

- Generic aspects are being evaluated and will be reported separately.

- Restart SER currently scheduled to be issued about May 19, 1982.

J. Lyons
x-24362

AGENDA FOR THE ACRS FULL COMMITTEE MEETING ON WOLF CREEK GENERATING STATION, UNIT NO. 1

MAY 6, 1982 WASHINGTON, DC

		Name
2:45 p.m.	ACRS SUBCOMMITTEE CHAIRMAN'S REPORT TO FULL COMMITTEE	J. Ray Subcommittee Chairman
3:15 p.m.	DESCRIPTION OF PLANT - Overview of plant and site - Differences from Callaway - Construction schedule, estimated dates of fuel loading and commercial operation, schedule and description of start-up tests - Identification of contractors and their responsibilities; support from SNUPPS - QA/QC effectiveness to date	G. L. Koester Vice President - Nuclear
3:30 p.m.	STATUS OF PLANT REVIEW	
	- Applicability of Callaway approvals to Wolf Creek - Summary of open items, confirmatory items and license conditions	J. Hopkins, NRC
	- Applicant's Responses	G. P. Rathbun Manager Licensing
4:00 p.m.	ORGANIZATION AND MANAGEMENT	
	- Offsite Organization Functions of organizational components Independent Safety Engineering Group Key personnel experience Staffing; use of contractors Nuclear Safety Review Committee Quality Assurance Committee	G. L. Koester
	- Corporate policies towards safe plant operation	T. D. Keenan, Director Nuclear Operations
	- Onsite Organization Functions of organizational components Key personnel experience Staffing; use of contractors Plant Safety Review Committee Training of operators and non-licensed personnel	F. T. Rhodes Plant Superintendent

ACRS FULL COMMITTEE MEETING

4:40 p.m.	USE OF SHIFT TECHNICAL ADVISORS (STAs)	
	- Training courses for STAs; Use of simulators in training	R. C. Coulthard Manager Nuclear Taining
	- Understanding of original intent for STAs - Interpretation of NUREG-0737 - Viewpoint of need for STAs in light of training	T. D. Keenan
5:00 p.m.	SEISMIC DESIGN OF PLANT AND EQUIPMENT	
	- NRC status of review	P. Sobel, NRC
	- Applicant statement of analyses	C. J. M. Sprout Technical Staff Engineer
	- Prospective resolution	P. Sobel/C. J. M. Sprout
5:15 p.m.	CONTROL ROOM	
	- Remote Shutdown Panel - vulnerability to failures that also affect Control Room panels - Change in principles of design since original concepts (influence of TMI)	J. M. Pippin, Manager Instrumentation & Controls
	- Human factors review of design in process (influence on displays)	J. M. McKinstray Operations Coordinator
5:35 p.m.	PREPARATION OF EMERGENCY OPERATING PROCEDURES	J. A. Zell Operations Supervisor
5:50 p.m.	RADIATION PROTECTION PROGRAM	R. F. Lewis, Supervisor Radiological/Environmental Assessment
6:00 p.m.	EMERGENCY PLANNING	R. F. Lewis
6:15 p.m.	SECURITY	D. Kunze, NRC/J. M. Pippin
6:20 p.m.	SUMMARY: Future ACRS Action	ACRS Chairman

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WOLF CREEK

1. OVERVIEW OF PLANT SITE
2. DIFFERENCES FROM CALLAWAY
3. CONSTRUCTION SCHEDULE
4. CONTRACTOR SUPPORT
5. QA/QC EFFECTIVENESS

GLENN L. KOESTER - VICE PRESIDENT - NUCLEAR

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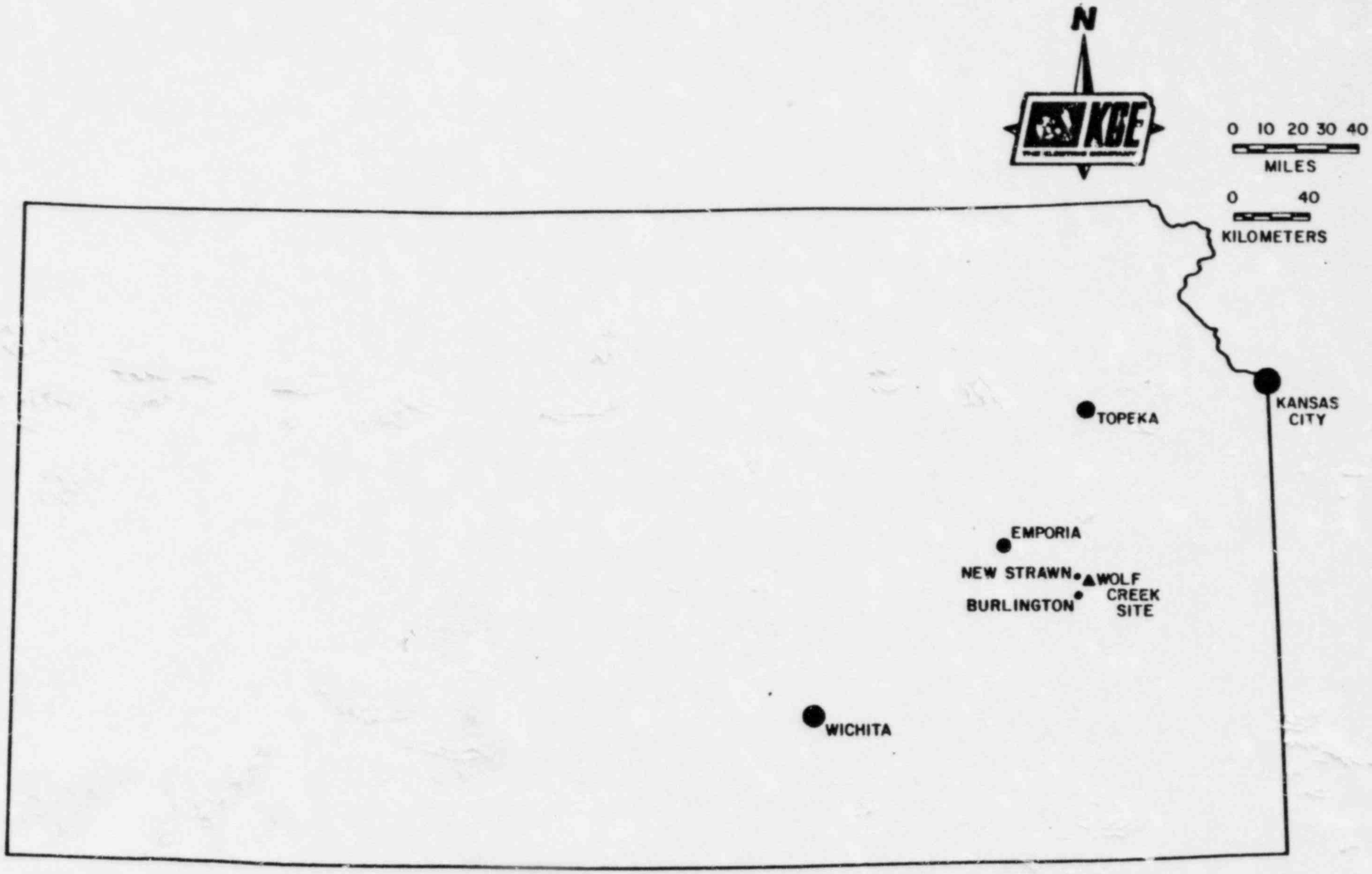
N
0 10 20 30 40
MILES
0 40
KILOMETERS
KANSAS CITY
TOPEKA
EMPORIA
NEW STRAWN
WOLF CREEK SITE
BURLINGTON
WICHITA

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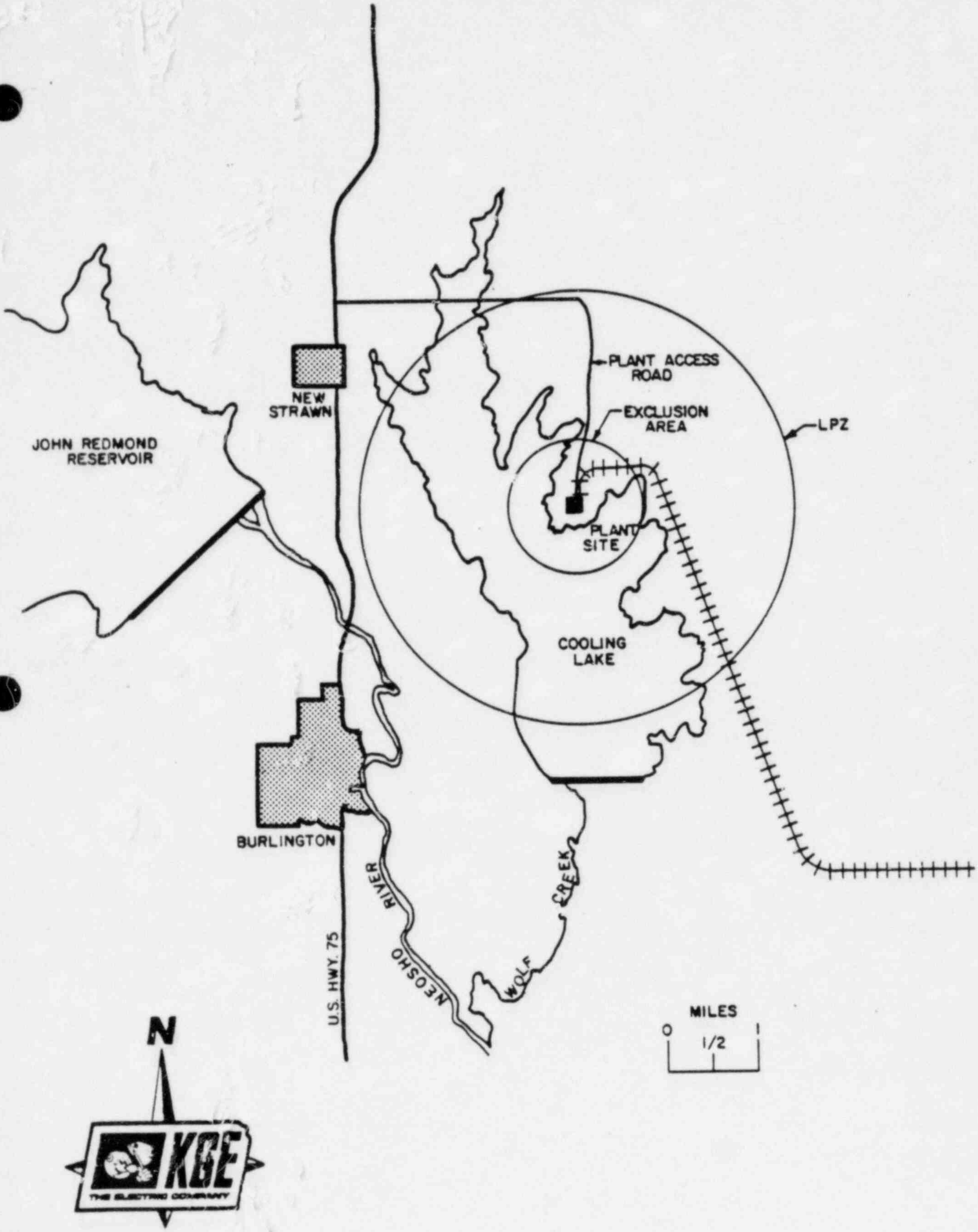
PLANT ACCESS ROAD
EXCLUSION AREA
LPZ
NEW STRAWN
JOHN REDMOND RESERVOIR
PLANT SITE
COOLING LAKE
BURLINGTON
U.S. HWY. 75
NEOSHO RIVER
WOLF CREEK
MILES
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1/2
1
N
KGE
THE ELECTRIC COMPANY

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PLANT GENERAL DESCRIPTION

* SNUPPS STANDARDIZED POWER BLOCK
* NSSS: WESTINGHOUSE 3425 MW_T
* TURBINE GENERATOR: GE TC 6F 38LSB
* POWER BLOCK AE: BECHTEL POWER CORPORATION
  GAITHERSBURG, MARYLAND
* SITE AE: SARGENT AND LUNDY
  CHICAGO, ILLINOIS
* GEOTECHNICAL SERVICES: DAMES & MOORE
* CONTRACTOR: DANIEL INTERNATIONAL CORPORATION
* NORMAL COOLING: COOLING LAKE
* BACKUP COOLING: ULTIMATE HEAT SINK IN THE COOLING
  LAKE, RETAINED BY AN UNDERWATER DAM

MAJOR DIFFERENCES

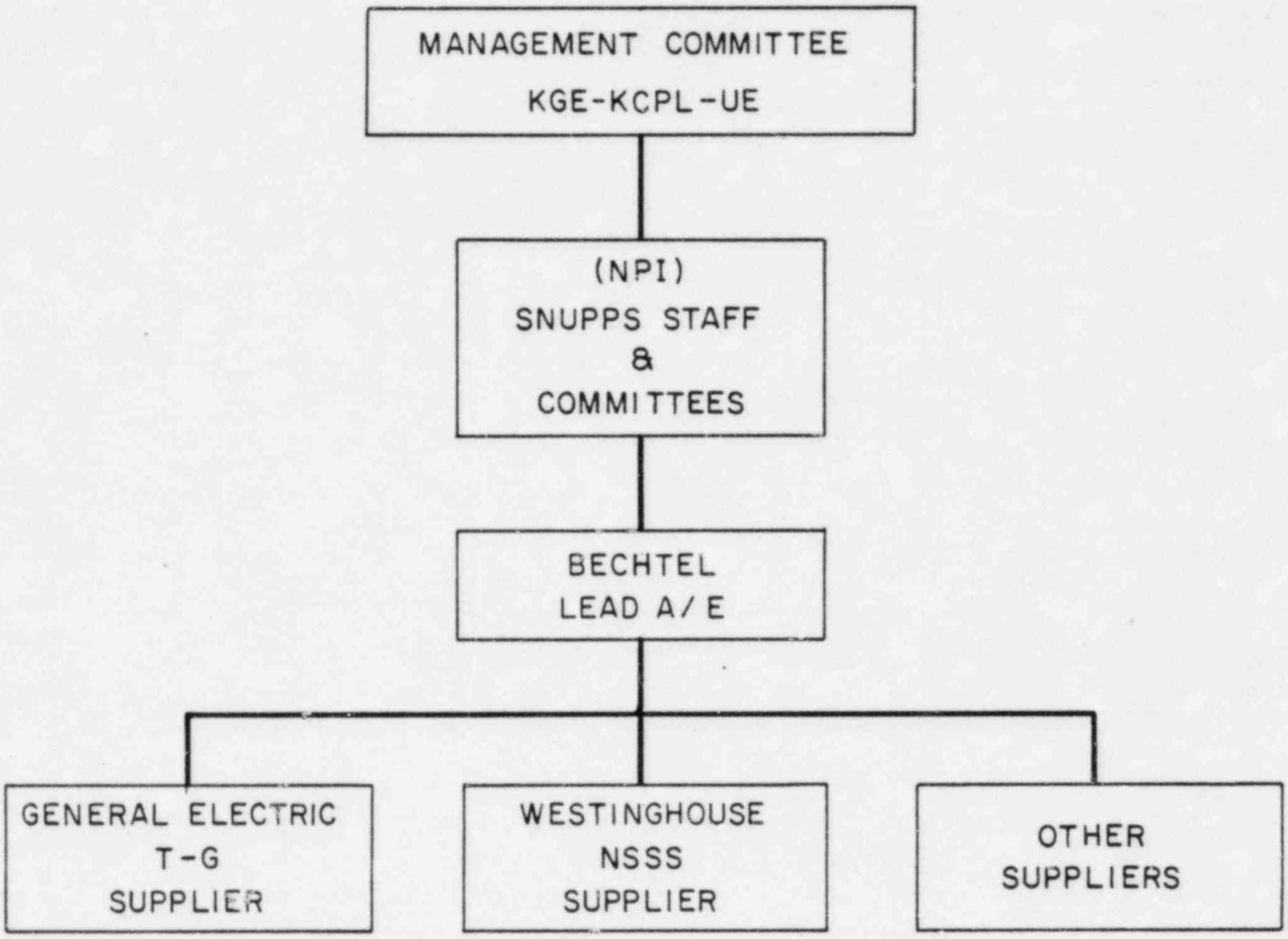
	WOLF CREEK	CALLAWAY
1.	Cooling Lake	Cooling Tower
2.	UHS Dam Submerged In Cooling Lake	UHS Retention Pond
3.	Circulating Water Screenhouse - Traveling Screens	Circulating Water Screenhouse - Fixed Screens
4.	Service Water From The Circulating Water System	Service Water From Hyperbolic Cooling Tower
5.	Water Intake And Discharge To Cooling Lake	Water Intake And Discharge To Cooling Tower Basin
6.	Makeup Water Screenhouse From Neosho River Makeup Discharge To The Cooling Lake	Makeup Water Intake And Discharge To Missouri River

PROJECT SCHEDULE

WOLF CREEK GENERATING STATION, UNIT NO. 1

ACTIVITY	DATE
Application For CP	April, 1974
ACRS Report	October, 1975
LWA Granted	January, 1977
<u>CP ISSUED</u>	May, 1977
First "Q" Concrete	June, 1977
Set Reactor Vessel	February, 1980
Application For OL	February, 1980
Startup Transformer Energized	August, 1981
First "Non-Safety-Related" Preoperational Test	August, 1981
First "Safety-Related" Preoperational Test	June, 1982
Primary System Hydro	May, 1983
Hot Functional Testing	September, 1983
ILRT	October, 1983
Fuel Load	December, 1983
Commercial Operations	May, 1984

MANAGEMENT RELATIONSHIP (POWER BLOCK)



SNUPPS STAFF

POSITION	DEGREE	YEARS EXPERIENCE: TOTAL PROFESSIONAL	TOTAL NUCLEAR	WOLF CREEK	OTHER COMM. NUCLEAR	NON-COMM. NUCLEAR
EXECUTIVE DIR. N. A. PETRICK	BS, MSME ORSORT, PE	38	33	9	7	17
TECHNICAL DIR. F. SCHWOERER	BS, MSME PE	38	25	8	10	7
MGR. NUCL. SAFETY J. O. CERMAK	BS, MSNE PHDNE, PE	21	21	2	18	1
MGR. TECH. SERV. E. F. BECKETT	CE, MSNE JD, PE	30	22	7	7	8
MGR. Q.A. S. J. SEIKEN	BS ENG. MSME	27	26	8	0	18
MGR. LICENSING R. L. STRIGHT	B.S. ENG MBA, PE	15	15	4	3	8
SITE REP R. D. BROWN	B.S. CHE PE	33	27	5	2	20
SITE REP W. R. REILLY	BS, MSNE MSMGMT, PE	30	17	6	0	11
MGR. ADMIN W. W. BALDWIN	BSME	32	8	8	0	0
ENGINEER R. P. WHITE	BSEE MSNE	13	13	7	5	1
ENGINEER J. H. RILEY	BS AERO MBA	10	6	1	0	5
ENGINEER D. J. KLEIN	BSME	5	5	2	3	0
12	TOTAL	292	218	67	55	96

SNUPPS DESIGN ASSURANCE PROGRAM

- CONTINUING SNUPPS STAFF & UTILITY REVIEW AND APPROVAL OF KEY DESIGN FEATURES AND DESIGN DOCUMENTS SELECTED BY SNUPPS:

- SNUPPS STAFF & UTILITY AUDITS OF A/E, NSSS AND SUBVENDOR DESIGN-RELATED ACTIVITIES AND PROCESSES; AND

- TECHNICAL VERIFICATION-TYPE AUDITS OF SELECTED DESIGN AREAS... PERFORMED BY SNUPPS STAFF & UTILITY SPECIALISTS, OR OFF-PROJECT DESIGNER SPECIALISTS, OR BOTH.

1. SNUPPS REVIEW AND APPROVAL OF LEVEL 1 & 2 A/E DESIGN DOCUMENTATION

In-line, series review by SNUPPS/Utility Staff of A/E design documentation selected (by SNUPPS) on basis of complexity, importance to safety and functional significance.

DOCUMENTS REQUIRING REVIEW AND APPROVAL INCLUDE:

- DESIGN CRITERIA...all disciplines
- SYSTEM DESCRIPTIONS
- P & I D's
- ARRANGEMENT DRAWINGS
- PROCESS FLOW DIAGRAMS
- CONTROL ROOM LAYOUTS
- SINGLE LINE & 3-LINE ELECTRICAL DIAGRAMS
- CLASS IE ELECTRICAL SCHEMATICS
- LOGIC DIAGRAMS
- EQUIPMENT SPECIFICATIONS...APPX 50% of TOTAL

Review process is an ongoing process utilizing collective experiences and technical capabilities of the SNUPPS/Utility Staffs.

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2. SNUPPS AUDITS OF A/E and NSSS DESIGN PROCESS

Elements of the A/E and NSSS design process have been subject to continuing SNUPPS/Utility audit including...

- DESIGN DEVELOPMENT
- DESIGN INTERFACE CONTROL
- DESIGN AND DESIGN DOCUMENT REVIEW
- DESIGN CHANGE CONTROL
- CONTROL OF DESIGN DEVIATIONS & NONCONFORMANCES
- COMPUTER PROGRAM VERIFICATION
- COMPLIANCE WITH LICENSING COMMITMENTS
- SPECIAL SCOPE DESIGN ACTIVITIES; i.e. fire protection, Category II/I

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3. SNUPPS AUDITS OF MAJOR EQUIPMENT SUPPLIERS

Continuing program of subvendor audits by SNUPPS and A/E designer teams focusing on critical areas of equipment design and fabrication. Audits performed are product-oriented focusing on the following activities...

- correct translation of design requirements into detailed drawings, procedures and manufacturing process specifications;
- design document control including subsupplier design document control;
- identification and control of deviations from design or process requirements;
- examination of analytical techniques and supporting computer programs;
- examination of inspection and test results to verify compliance with with design requirements.

The scope of subvendor audit program has included the following items of equipment...

- reactor pressure vessel
- reactor vessel internals
- reactor core/fuel
- steam generators
- post-tensioning systems
- reactor coolant and feedwater pumps
- control and relief valves
- process instrumentation & control panels

4. OFF-PROJECT TECHNICAL VERIFICATION AUDITS - COMPLETED TO DATE

Effort involves technical verification-type audits of selected design features or items performed by SNUPPS/Utility specialists or off-project designer personnel or both. Audits of this category completed to date include...

- Reactor core design review audit
- Class 1 piping stress analysis
- Hanger and pipe support design analysis
- BOP seismic stress analysis
- Class 1E equipment seismic qualification
- Design of seismic Category 1 pressure relieving devices
- Structural support design and analysis
- Seismic Category II/I hazards analysis
- Control systems design (in progress)

These verification audits are an integral part of the SNUPPS QA Program and are planned and coordinated by QA management with off-project, technical specialists utilized for the detailed evaluations.

5. TECHNICAL VERIFICATION AUDIT PROGRAM - 1982/83 PLANS

Preliminary plans for additional technical verification-type audits to be performed in 1982 and 1983 include the following....

- Examination of soils properties developed for seismic analyses
- Seismic and Seismic Spectra development and analysis
- IEEE-344 seismic equipment qualifications
- Piping stress analyses - Class 1 and nonClass 1 systems
- Pipe support, cable tray structural analysis
- Evaluation of loads, load combinations and damping values used for seismic structural design

SUMMARY OF
OPEN ITEMS, CONFIRMATORY ITEMS AND LICENSE CONDITIONS
FROM
SAFETY EVALUATION REPORT
FOR
WOLF CREEK GENERATING STATION, UNIT NO. 1

GENE P. RATHBUN, MANAGER LICENSING

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OPEN ITEMS FROM SER

	ITEM	NEXT ACTION	DATE
1.	High Energy Pipe Break Hazards Analysis	KG&E	5/82
2.	Pump & Valve Operability Assurance Program	NRC	----
3.	Seismic & Dynamic Qualifications of Mechanical and Electrical Equipment	NRC	----
4.	Environmental Qualification of Safety-Related Electrical Equipment	KG&E	10/82
5.	Fire Protection Programs - Alternate Shutdown Panel	NRC	----

OPEN ITEMS (CONT'D)

6. TMI ACTION PLAN ITEMS

I.A.1.1	SHIFT TECHNICAL ADVISOR	NRC	----
I.C.1	GUIDANCE FOR EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS	KG&E	7/82
I.C.8	PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL APPLICANTS	----	----
I.D.1	CONTROL ROOM DESIGN REVIEW	KG&E	6/82
II.B.2	PLANT SHIELDING FOR ACCESS TO VITAL AREAS AND TO PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION	KG&E	5/82
III.A.1.2	UPGRADE EMERGENCY SUPPORT FACILITIES	NRC	----

CONFIRMATORY ITEMS FROM SER

STATUS

* TOTAL OF 34 ITEMS + 10 TMI-RELATED ITEMS

NRC ACTION	17 ITEMS + 3 TMI
KG&E ACTION	16 ITEMS + 7 TMI
BOTH ACTION	1 ITEM

LICENSE CONDITIONS FROM SER

STATUS

* TOTAL OF 17 ITEMS + 1 TMI-RELATED ITEM

NRC ACTION	5 ITEMS
KG&E ACTION	12 ITEMS + 1 TMI ITEM

WOLF CREEK

KG&E MANAGEMENT ORGANIZATION

GLENN L. KOESTER - VICE PRESIDENT - NUCLEAR

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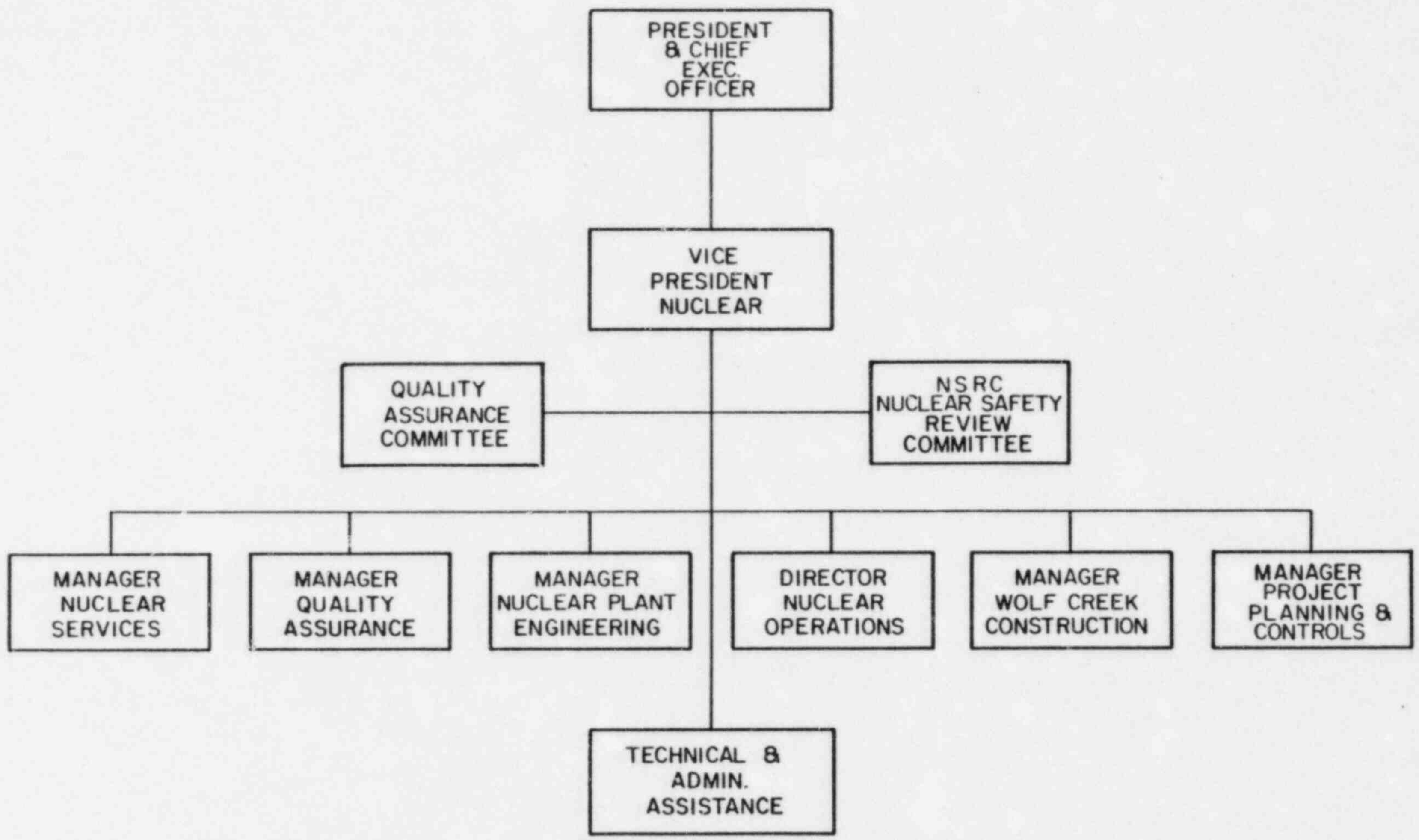
PRESIDENT & CHIEF EXEC. OFFICER
VICE PRESIDENT NUCLEAR
QUALITY ASSURANCE COMMITTEE
NSRC NUCLEAR SAFETY REVIEW COMMITTEE
MANAGER NUCLEAR SERVICES
MANAGER QUALITY ASSURANCE
MANAGER NUCLEAR PLANT ENGINEERING
DIRECTOR NUCLEAR OPERATIONS
MANAGER WOLF CREEK CONSTRUCTION
MANAGER PROJECT PLANNING & CONTROLS
TECHNICAL & ADMIN. ASSISTANCE

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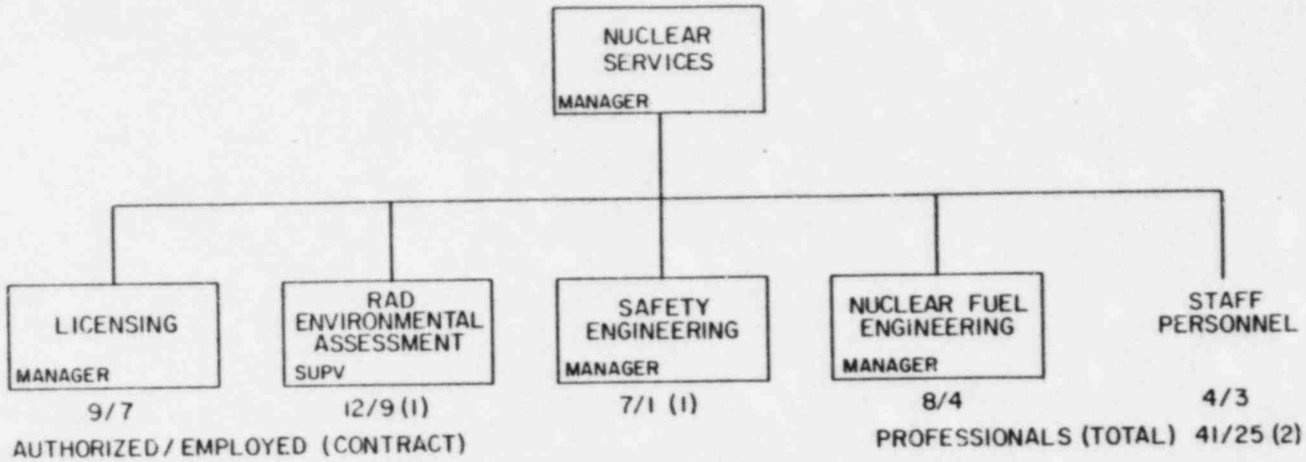
	Degree	Professional Job Related Experience	Total Nuclear Experience	Nuclear Experience: Commercial	Navy	Wolf Creek	Other Nuclear
Dir Nuclear Operations	BS Eng MS Mat Eng	20	20	11	7	2	
Mgr Nuclear Services	BS Eng Phy MS, PhD NE	20	18		4	7	7
Mgr Nuclear Plant Eng	BS, MS NE	28	26			9	17
Mgr Quality Assurance	BSEE	26	9			9	
Mgr Wolf Creek Const.	BSCE	16	16			3	13
Mgr Project Plng & Ctrls	BSCE MS Const Mgm	19	16			2	14
Tech & Admin Assistance		33	5			5	
Total (7 Individuals)		162	110	11	11	37	51

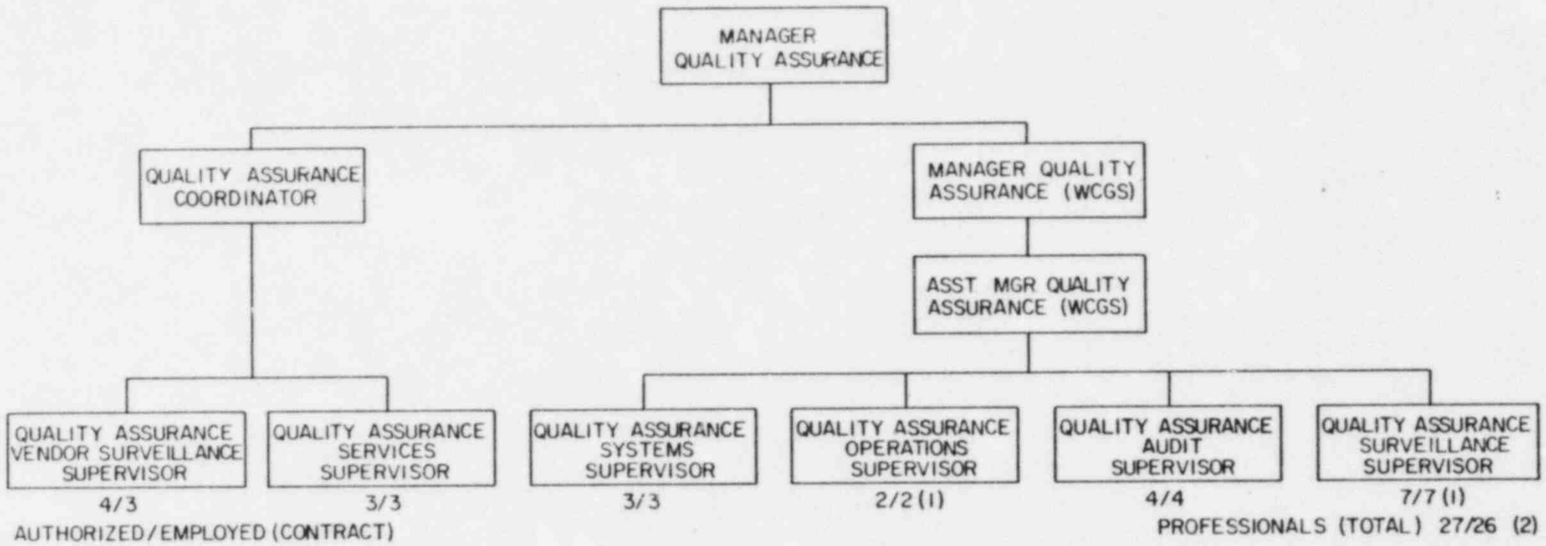


Position	Education	Professional Experience	Nuclear Experience: Total	Navy	Commercial Operating	WCGS	Other
Manager Nuclear Serv	BSE Phy, MS Ph D NE	20	19	4		8	7
Manager Licensing	BS & MS, NE	12	12			4	8
Manager Nuc. Fuel Eng	BSE Phy MS, ME	9	9			9	
Supervisor Safety Eng	BA Phy/Math MA Sec Ed MS, NE	11	8		5	1	2
Supervisor Rad/Env Asst	BA, BIO MS Rad Biophy	8	8		1	5	2
Uranium Prjts Administrator	BS Ceramic Eng	13	13				13
TOTAL (6 Individuals)		73	69	4	6	27	32

EXPERIENCE OF ALL

NUCLEAR SERVICES PROFESSIONAL PERSONNEL

			Nuclear Experience				
Section	# Personnel	Professional Experience	Total	Navy	Commercial Operating	WCGS	Other
Manager	1	20	19	4		8	7
Licensing	7	50	38	4	3	10	21
Rad/Environ Assessment	9	42	39		9	19	11
Safety Eng	1	11	8		5	1	2
Nuclear Fuel	4	27	23		1	13	9
Staff	3	17	17			4	13
TOTAL	25	167	144	8	18	55	63



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POSITION	DEGREE	PROFESSIONAL/ JOB RELATED EXPERIENCE	NUCLEAR EXPERIENCE TOTAL	COMMERCIAL	NAVY	WOLF CREEK	OTHER
Mgr. QA	BSEE	26	9			9	
QA Coordinator	BSEE	13	8			8	
QA Services Supv.		31	11				11
Mgr. QA WCGS	BS & MSNE	10	10	6		4	
Asst. Mgr. QA WCGS	BSEE	6	5			5	
QA Sys. Supv.		4	4	1		3	
QA Oper. Supv.		24	24	10		1	13
QA Audit Supv.	AA & BA Social SC	16	8	8			
QA Surv. Supv.		25	8	8			
TOTAL (9 Individuals)		155	87	33		30	24

QA DIVISION EDUCATION AND EXPERIENCE

NAME	POSITION	DEGREE	PROFESSIONAL/ JOB RELATED EXPERIENCE	NUCLEAR EXPERIENCE				
				TOTAL	COMMERCIAL	NAVY	WOLF CREEK	OTHER
E. Creel	Mgr. QA	BSEE	26	9			9	
W. Eales, Jr.	QA Coordinator	BSEE	13	8			8	
P. Nichols	QA Services Supv.		31	11				11
TOTAL			70	28			17	11
Home Office Staff (5 Personnel)			58	14	2		12	

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QA DIVISION EDUCATION AND EXPERIENCE (SITE)

NAME	POSITION	DEGREE	PROFESSIONAL/ JOB RELATED EXPERIENCE	TOTAL NUCLEAR	NUCLEAR EXPERIENCE COMMERCIAL	NAVY	WOLF CREEK	OTHER
D. PRIGEL	MGR QA WCGS	BS & MSNE	10	10	6		4	
G. REEVES	ASST MGR QA WCGS	BSEE	6	5			5	
C. PARRY	QA SYS SUPV		4	4	1		3	
R. YOUNGS	QA OPER SUPV		24	24	10		1	13
R. PEDERSON	QA AUDIT SUPV	AA & BA SOCIAL SC	16	8	8			
O. THERO	QA SURV SUPV		25	8	8			
TOTAL			85	59	33		13	13
SITE STAFF (12 PERSONNEL)			63	39	8	13	13	5

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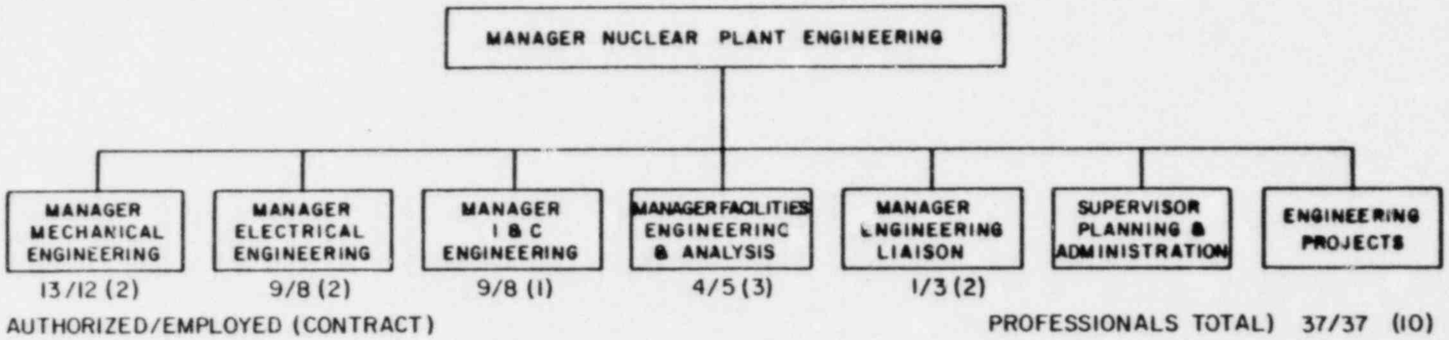
NUCLEAR PLANT ENGINEERING ORGANIZATION



	Degree	Professional/ Job Related Experience	Total Nuclear Experience	Nuclear Experience			
				Commercial	Navy	Wolf Creek	Other Nuclear
Manager Nuclear Plant Eng	BS, MS ME	28	26			9	17
Manager Mechanical Eng	BS NE	17	15	3	4	7	1
Manager Electrical Eng	BS EE	15	9			9	
Manager I & C Eng	BS EE	11	9			9	
*Manager Facilities Eng & Analysis	BS, MS ME	26	25			5	20
Total (5 Individuals)		97	84	3	4	39	38

*Information shown is for C.A. Gukeisen who is temporarily serving as Manager Engineering Liaison

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EXPERIENCE OF ALL
NUCLEAR PLANT ENGINEERING
PERMANENT PROFESSIONAL PERSONNEL

Section	# Personnel	Professional Experience	Nuclear Experience Total	Navy	Commercial Operating	WCGS	Other
Manager	1	28	26			9	17
Mechanical Eng.	10	49	40	4	7	20	9
Electrical Eng.	6	34	18			18	
I & C		37	20		3	17	
Facilities Eng. & Analysis	2	14	14			6	8
Eng. Liaison	1	26	25			5	20
Total	27	188	143	4	10	75	54

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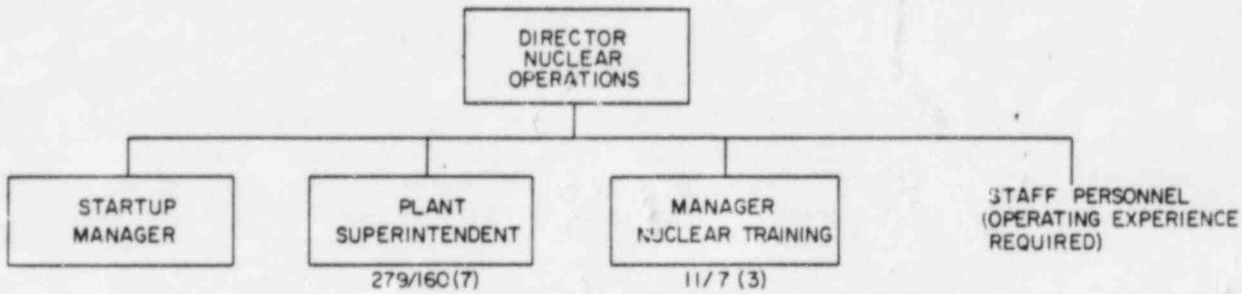
NUCLEAR OPERATIONS BRANCH
NUCLEAR DEPARTMENT
ORGANIZATIONAL STRUCTURE



STARTUP – AUTHORIZED / EMPLOYED (CONTRACT)

	AUTHORIZED	EMPLOYED	(CONTRACT)
KG&E	85	59	
CONTRACT	128	93	
TOTAL	213	152	(93)

	Degree	Professional/ Job Related Experience	Total Nuclear Experience	Nuclear Experience: Commercial	Navy	Wolf Creek	Other Nuclear
Director Nuclear Operations	BS Eng MS Mat Eng	20	20	11	7	2	
Startup Manager		24	24	8		1	15
Plant Superintendent	BS Eng	20	20	13	5	2	
Manager Nuclear Training	BS MS NE	16	16	5	2	1	8
Total (4 Individuals)		80	80	37	14	6	23

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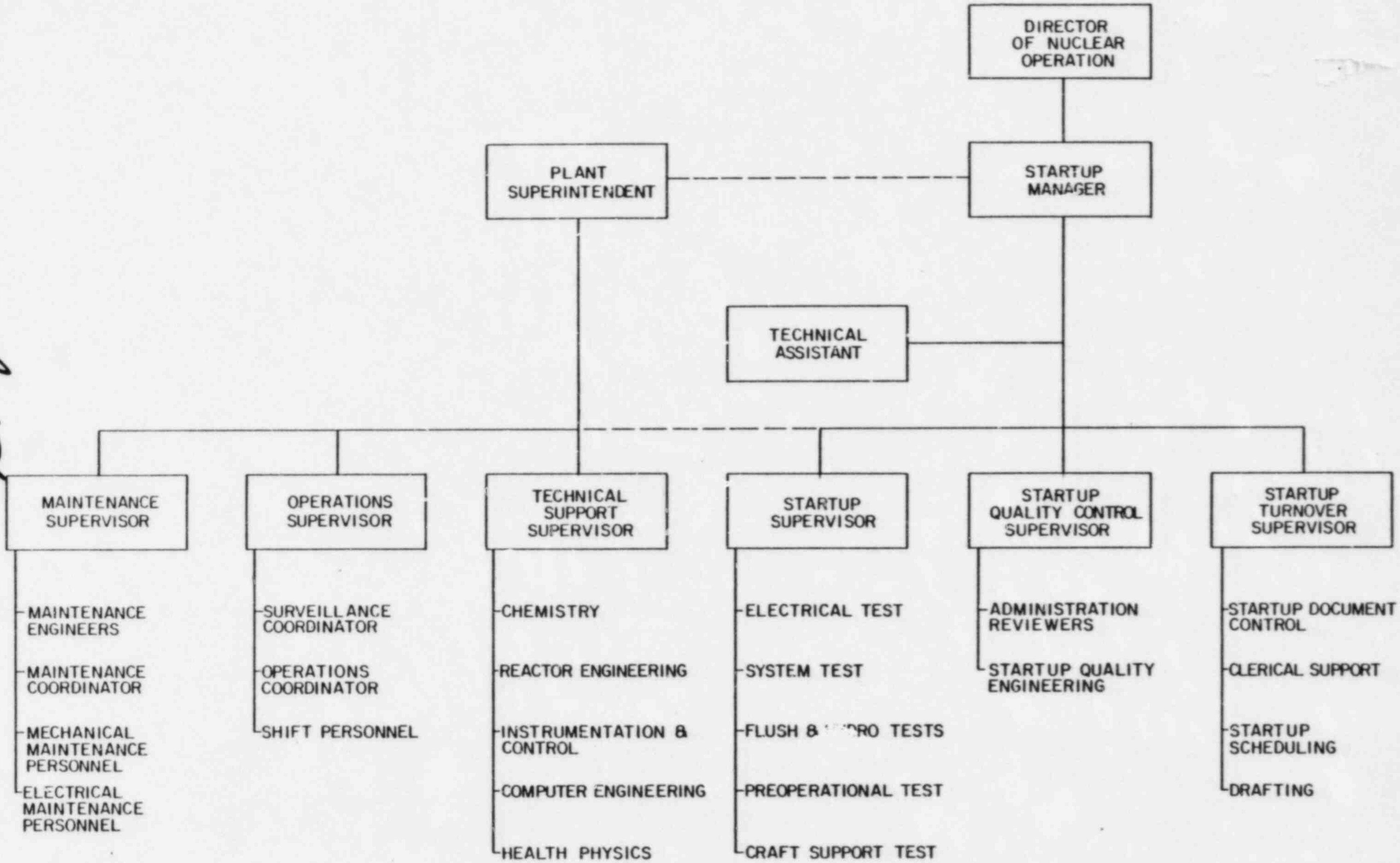
STARTUP ORGANIZATION MANAGEMENT



NSRC COMPOSITION AND EXPERIENCE OF MEMBERS

			Experience		
Title	Org	Name/Position	Professional	Nuclear	Education
Manager Nuclear Services	KG&E	Dr. R.C. Hagan (Chairman)	20	19	BS Eng Phy MS NE Ph.D. NE
Manager Nuclear Plant Engineering	KG&E	M.L. Johnson (Vice Chairman)	28	26	BS ME MS ME
Vice President - Engineering	KG&E	B.N. Ruddick	33	0	BS EE
Director Nuclear Operations	KG&E	T.D. Keenan	[illegible]	20	BS Eng MS Mat Eng
Quality Assurance Coordinator	KG&E	W.G. Eales	13	8	BS EE
Manager Licensing	KG&E	G.P. Rathbun	12	12	BS NE MS NE
Manager Engineering Services	NUS	T.R. Hencey	17	15	BS Math MS Math MS Eng Sci
Head, Department of Nuclear Engineering	KSU	Dr. N.D. Eckhoff	22	21	BS Ch E MS NE Ph.D. NE
Manager Safety	NPI	Dr. J.O. Cermak	21	21	BS ME MS NE Ph.D. NE
Manager Safety Engineering	KG&E	Vacant			
		TOTAL	186	142	

QUALITY ASSURANCE COMMITTEE COMPOSITION AND

EXPERIENCE OF MEMBERS

Title	Name/Position	Experience Professional	Experience Nuclear	Education
Vice Pres.-Nuclear	G. L. Koester (Chairman)	33	10	BS Math
Vice Pres.-Eng.	B. N. Ruddick (Vice Chairman & Secretary)	33	0	BSEE
Wolf Creek Const. Manager	G. L. Fouts	16	16	BSCE
Legal Counsel	R. D. Terrill	2	2	BSBA, JD
Manager Fossil Production	H. R. Macklin	21	2	BSEE
Director Nuclear Operations	T. D. Keenan	20	20	BS Eng MS Mat Eng

CORPORATE OPERATING

PHILOSOPHY

FOR THE

WOLF CREEK GENERATING STATION

UNIT NO. 1

THOMAS D. KEENAN, DIRECTOR NUCLEAR OPERATIONS

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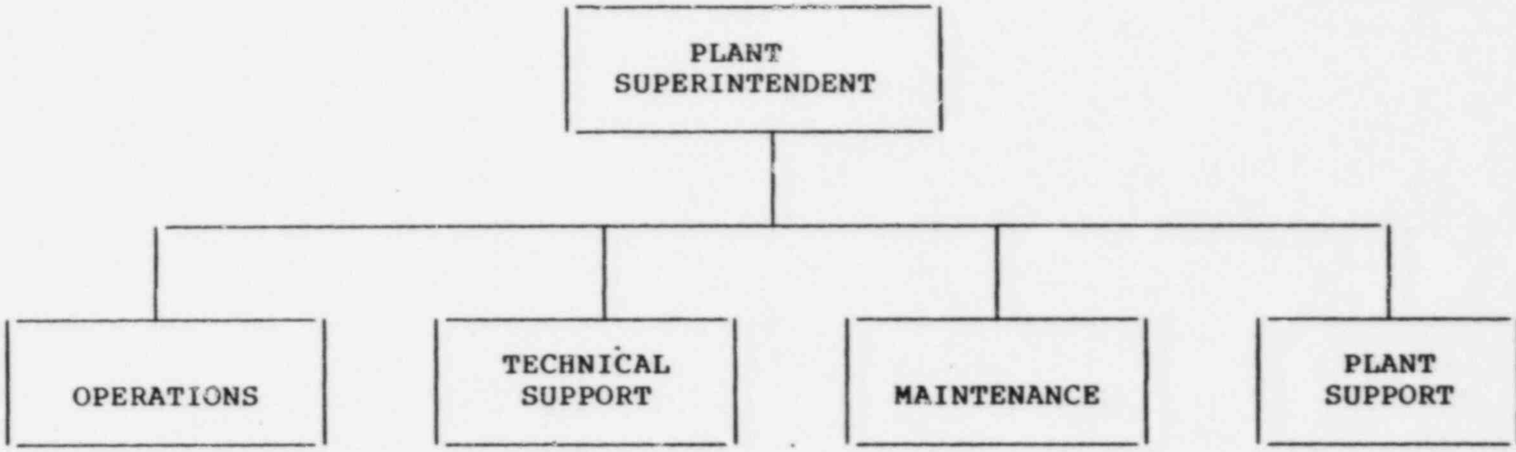
PLANT ORGANIZATION

FOR

WOLF CREEK GENERATING STATION,
UNIT NO. 1

FORREST T. RHODES, PLANT SUPERINTENDENT

A-140



	Education	Professional Job Related Experience	Total Nuclear Experience	Nuclear Experience		
				Commercial	Navy	Wolf Creek
OPERATIONS SUPERVISOR	BS NE	9	9	6		3
TECHNICAL SUPPORT SUPERVISOR	BS NE	15	9			9
MAINTENANCE SUPERVISOR	BS EE	18	3			3
PLANT SUPPORT SUPERVISOR	BS EE, MBA	22	16	6	9	1

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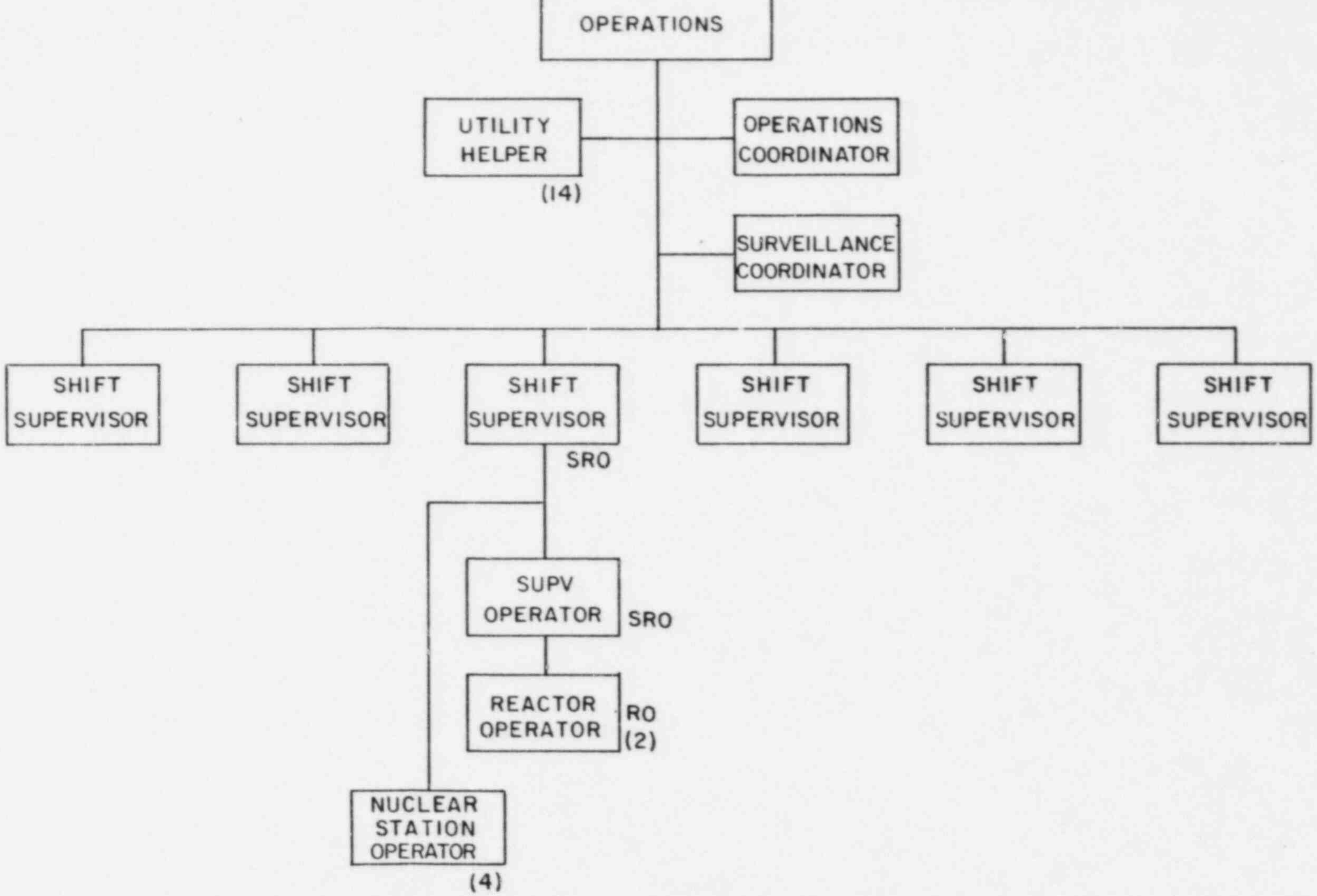
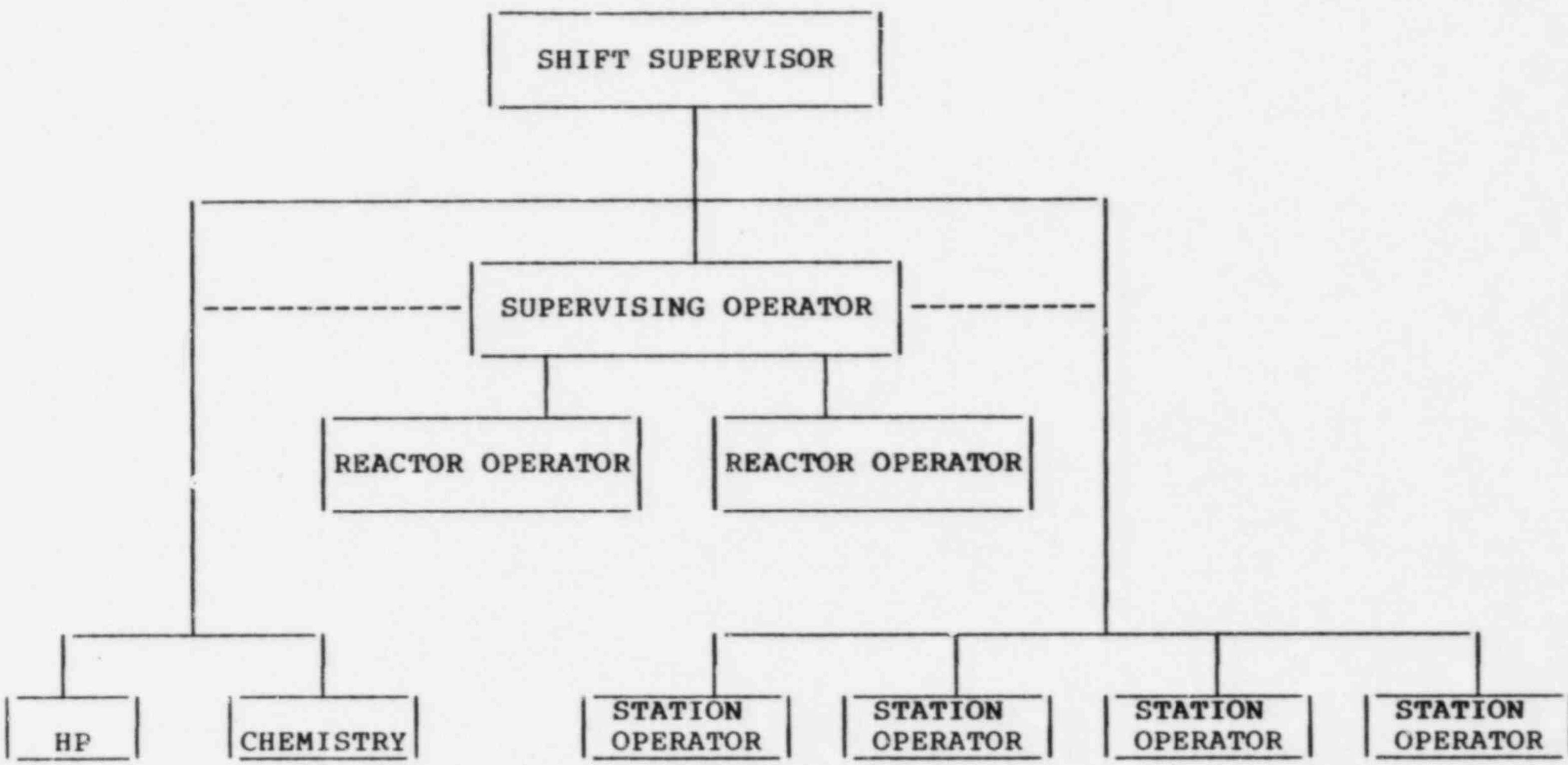
(TOTAL PROJECTED STAFF, 66)



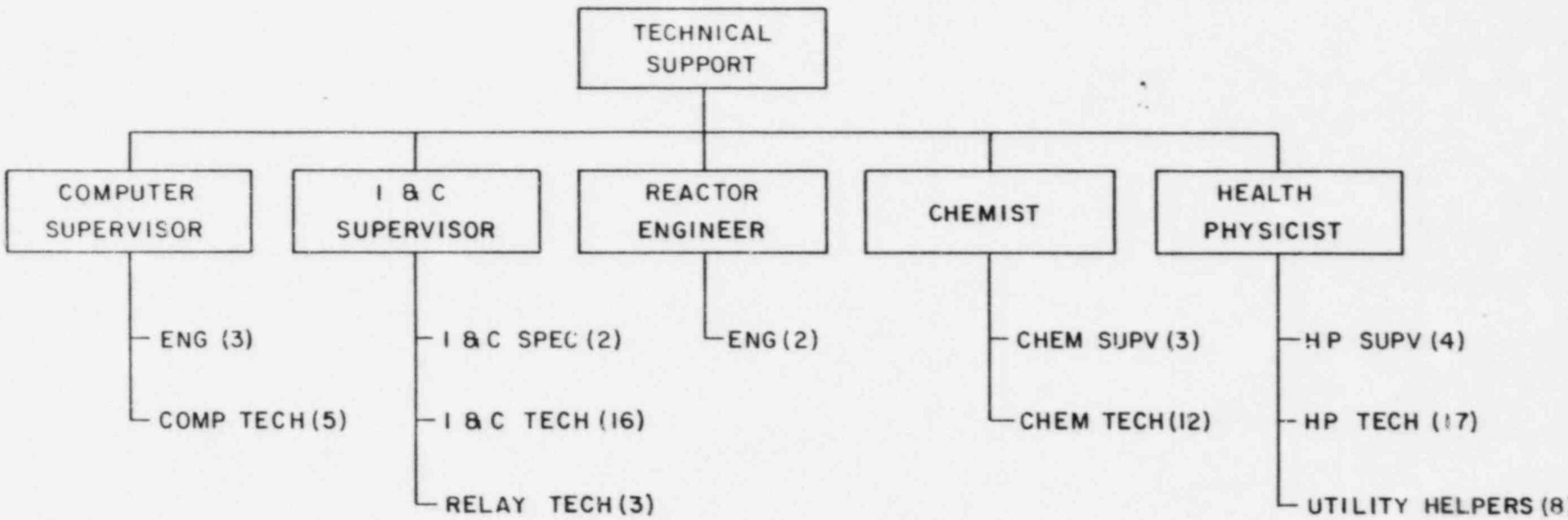
A-143


SHIFT SUPERVISOR
SUPERVISING OPERATOR
REACTOR OPERATOR
REACTOR OPERATOR
HP
CHEMISTRY
STATION OPERATOR
STATION OPERATOR
STATION OPERATOR
STATION OPERATOR

OPERATIONS EXPERIENCE BASE

	JOB TITLE	COMMERCIAL NUCLEAR EXPERIENCE	MILITARY NUCLEAR EXPERIENCE	JOB RELATED EXPERIENCE	WOLF CREEK EXPERIENCE
J. Zell	Ops Supv	6			3
J. McKinstray	Ops Coordn		6		4
J. Hansen	Sys Analyst		6		0
S. Austin	Shift Supv		6	5	5
L. Borders	Shift Supv			1	4
J. Houghton	Shift Supv		11	1	4
O. Korbelik	Shift Supv		18	1	3
D. Mosebey	Shift Supv	4		2	2
D. Naylor	Shift Supv				4
R. Middleton	Shift Supv	8	5		2
B. Erbe	Supv Operator		6		4
P. Martin	Supv Operator		5		5
R. Miller	Supv Operator		6		3
D. Neufeld	Supv Operator		8		4
S. Walgren	Supv Operator		10		2
J. Weeks	Supv Operator		5		3
Total of Reactor Operators			108	4	35
Total of Nuclear Operations Specialists			10		2
Total of Nuclear Station Operators			47	11	7
C. Beardon	Consultant	10	10		1
K. Bryon	Consultant	13			1
B. Jurrus	Consultant	10	7		1
B. Purdy	Consultant	10			1
R. Raykiewicz	Consultant	6	10		0
J. Walker	Consultant	12	8		0
TOTALS		79	292	25	100

(TOTAL PROJECTED STAFF, 81)



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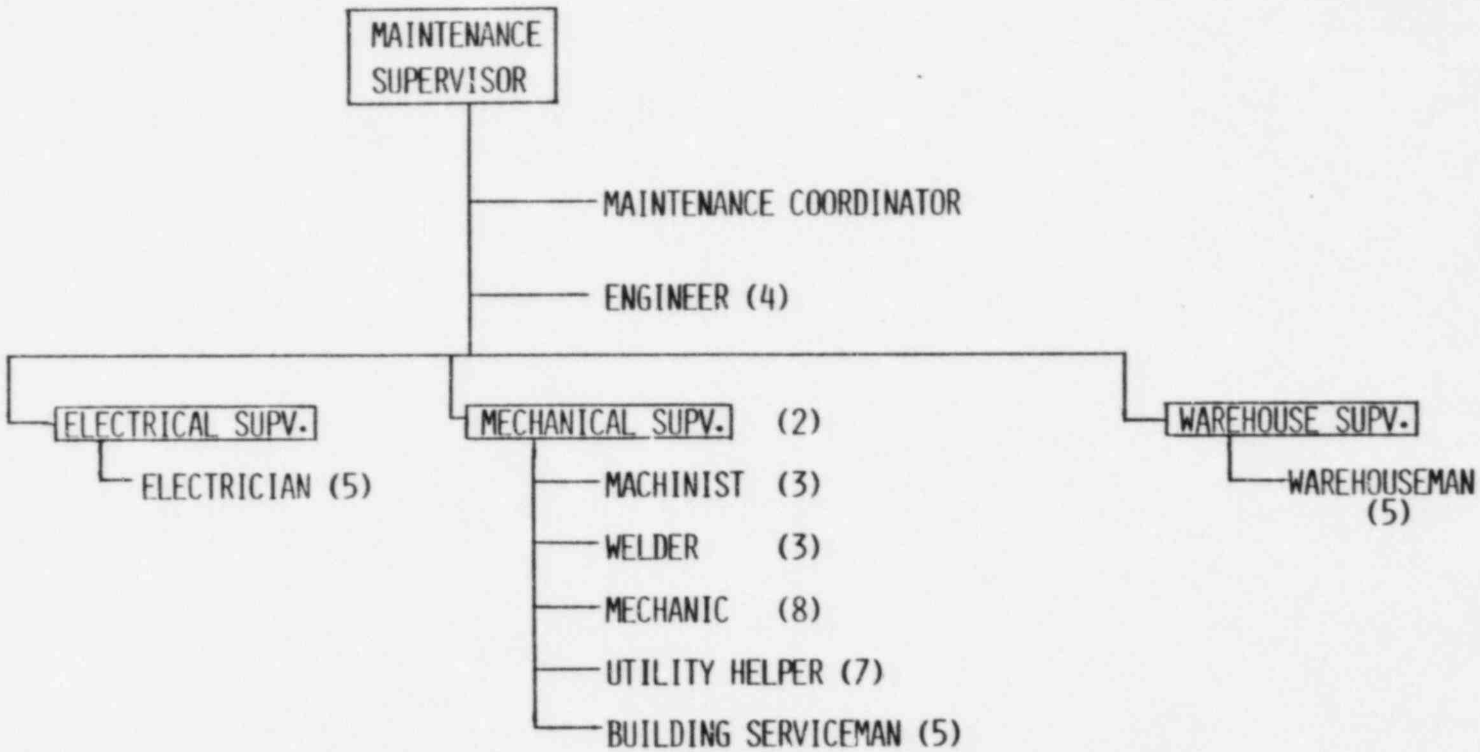
TECHNICAL SUPPORT EXPERIENCE BASE

	Job Title	Commercial Nuclear Experience	Navy Nuclear Experience	Job Related Experience	Wolf Creek Experience
G. Boyer	Tech Sup Sup			6	9
B. McKinney	I&C Supv	5		0	2
R. Klein	Sr Engr Spec	12	9	0	1
V. Tumbleson	Egr Spec III	5	6		2
Total of I&C Technicians		4	17	53	28
B. Norton	Reactor Engr	0			3
E. Lehman	Engineer II			4	1
M. Nichols	Hlth Physt	9		0	3
L. Breshears	H/P Supv	5		2	2
H. Davis	H/P Supv	2		3	2
J. Isom	H/P Supv	7	6	0	1
Total of Health Physics Technicians		2		17	11
B. Burke	Chem Supv	0		5	2
Total of Chemistry Technicians		1	12	13	7
M. Hawk	Ld Comp Egr			9	2
R. Parker	Egr Spec III		6	4	2
M. Shaffer	Engr Spec I				1
D. Breckenridge	Engr Spec I			2	1
Total of Computer Techs			0	30	7
TOTALS		52	56	148	87

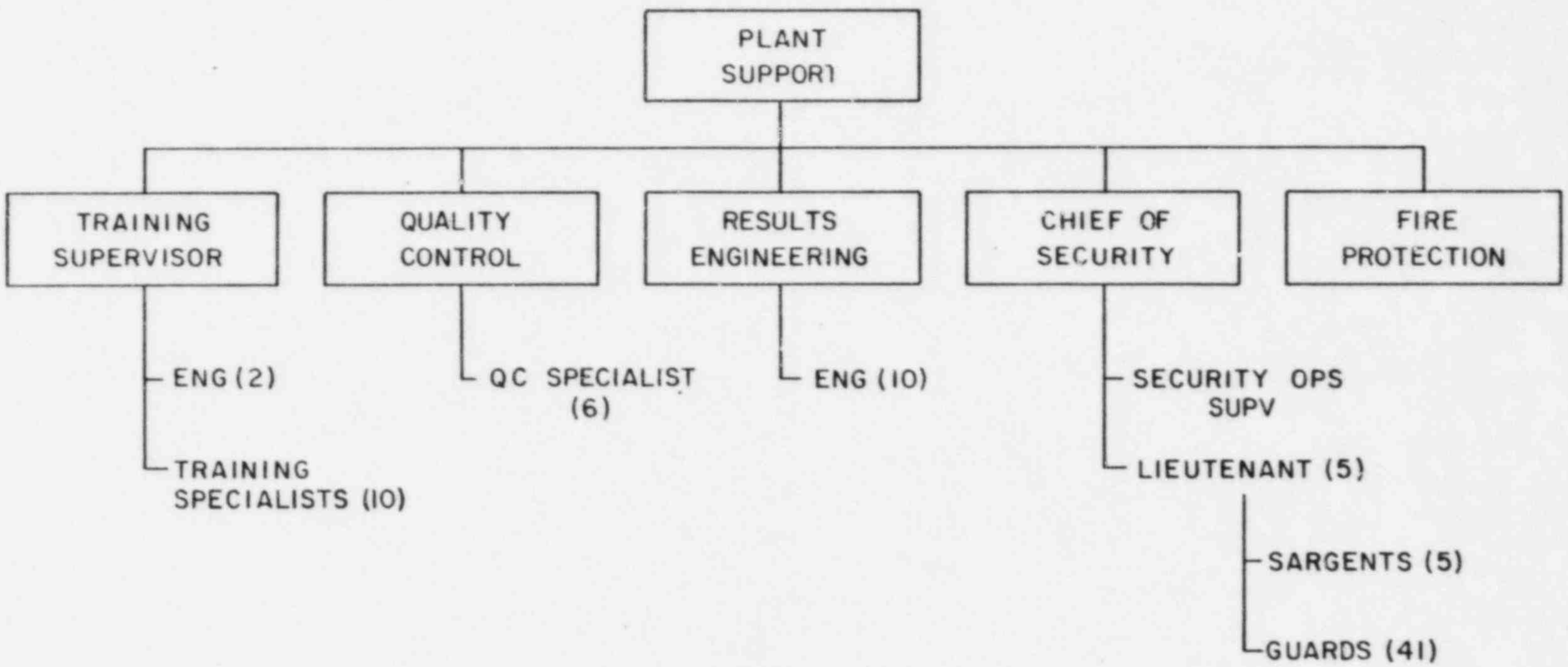
(TOTAL PROJECTED STAFF 46)



MAINTENANCE EXPERIENCE BASE

	JOB TITLE	COMMERCIAL NUCLEAR EXPERIENCE	MILITARY NUCLEAR EXPERIENCE	JOB RELATED EXPERIENCE	WOLF CREEK EXPERIENCE
D. Rich	Maint Supv			15	3
D. Walsh	Maint Coordr		5		2
J. Damet	Maint Engr			3	2
G. Lawson	Engr Spec			8	0
A. Montague	Engr Spec			24	1
D. Goodlove	Mech Supv		10		2
Total of Mechanics				70	2
C. Minor	Elect Supv			15	1
Total of Electricians			6	35	1
R. Stump	Whse Supv			15	2
TOTALS		0	21	185	16

(TOTAL PROJECTED STAFF, 86)



PLANT SUPPORT EXPERIENCE BASE

	JOB TITLE	COMMERCIAL NUCLEAR EXPERIENCE	MILITARY NUCLEAR EXPERIENCE	JOB RELATED EXPERIENCE	WOLF CREEK EXPERIENCE
D. Smith	Plnt Supp Supv	6	9	6	1
L. Blackwell	Fire Prot Spec			7	2
R. Hoyt	QC Supervisor	5	8	5	1
A. Mah	Training Supv		5		4
S. Hatch	Engr II, Trng	4	6	3	1
Total of Training Specialists			23	12	2
M. Estes	Results Supv			8	3
J. Stamm	Senior Engr			5	1
S. Fellers	Senior Engr			10	4
V. MacTaggart	Senior Engr	4	14		
A. Scott	Senior Engr	14			5
J. Mah	Engr III, Mech		5		3
B. Bumgarner	Engr III, Mech		5		2
R. Sims	Engr II			2	2
J. Johnson	Chief of Secur			2	3
D. Rice	Sec Ops Supv			11	2
Total of Security Shift Lieutenants				93	9
TOTALS		33	75	164	46

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ADMINISTRATIVE EXPERIENCE BASE

	JOB TITLE	COMMERCIAL NUCLEAR EXPERIENCE	MILITARY NUCLEAR EXPERIENCE	JOB RELATED EXPERIENCE	WOLF CREEK EXPERIENCE
F. Rhodes	Plant Supt	13	5		2
D. McDaniel	Admin Supv			7	4
A. Miller	RMS Supervisor			1	1
M. Williams	Consultant	13	8		1
TOTALS		26	13	8	8

SUMMARY OF EXPERIENCE BASE

	COMMERCIAL NUCLEAR EXPERIENCE	MILITARY NUCLEAR EXPERIENCE	JOB RELATED EXPERIENCE	WOLF CREEK EXPERIENCE
ADMINISTRATION	26	13	8	8
OPERATIONS	79	292	25	100
TECHNICAL SUPPORT	52	56	148	87
PLANT SUPPORT	33	75	164	46
MAINTENANCE		21	185	16
TOTALS	190	457	530	257

Plant Organization

- Plant Supt.
  - Maint. Supv.
    - Maint. Engr. (4)
    - Maint. Coord. (1)
    - Mech. Supv. (2)
      - Nuc. Mech-Mach. (3)
      - Nuc. Mech-Weld. (3)
      - Nuc. Mech. (8)
      - Utility Helper (7)
      - Bldg. Serviceman (5)
    - Elec. Supv.
      - Nuclear Elect. (5)
    - Warehouse Supv.
      - Warehouse Attendant (5)
  - Operations Supv.
    - Surveillance Coord.
    - Oper. Coord.
    - Shift Supv. (SRO) (7)
      - Supv. Oper. (SRO) (6)
      - Reactor Oper. (RO) (20)
      - Nuc. Station Oper. (16)
      - Utility Helper (14)
  - Tech. Supp. Supv.
    - Reactor Engr. Supv.
      - Reactor Engr. (2)
    - I&C Supv.
      - Engr./Specialist (2)
      - I&C Techs. (16)
      - Relay Tech. (3)
    - Lead Comp. Engr.
      - Computer Engr. (3)
      - Computer Tech. (5)
    - Chemist
      - Chem. Supv. (3)
      - Chem. Tech. (12)
    - Health Physicist *
      - Health Phy. Supv. (4)
      - Health Phy Tech. (17)
      - Utility Helper (8)
  - Plant Supp. Supv.
    - Fire Prot. Spec.*
    - Chief of Security *
      - Sec. Ops. Supv.
      - Lieutenant (5)
      - Sargeants (5)
      - Guards (41)
    - Qual. Cont. Supv.*
      - Qual. Cont. Spec. (6)
    - Training Supv.
      - Training Spec. (10)
      - Training Engr. (2)
    - Results Engineering Supv.
      - Results Engr. (M) (3)
      - Results Engr. (E) (2)
      - Additional Engr. (5)

*For Technical Matters of an immediate nature, the respective individual reports directly to the Plant Superintendent

Wolf Creek Generating Station Organizational Chart

PLANT SAFETY REVIEW COMMITTEE

CHAIRMAN - PLANT SUPERINTENDENT

MEMBERS - TECHNICAL SUPPORT SUPERVISOR
OPERATIONS SUPERVISOR
MAINTENANCE SUPERVISOR
PLANT SUPPORT SUPERVISOR
RESULTS ENGINEERING SUPERVISOR
CHEMISTRY SUPERVISOR
HEALTH PHYSICIST
INSTRUMENTATION AND CONTROL SUPERVISOR
REACTOR ENGINEERING SUPERVISOR
QUALITY ASSURANCE MANAGER (SITE)

STAFFING SCHEDULE

1. Operations

53/66

2. Technical Support

A) Computer - We are complete.

B) Reactor Engineers - 2/3

C) Health Physics - 15/30

D) Chemistry - 10/16

E) I&C - We are complete at this time.

3. Maintenance

20/46

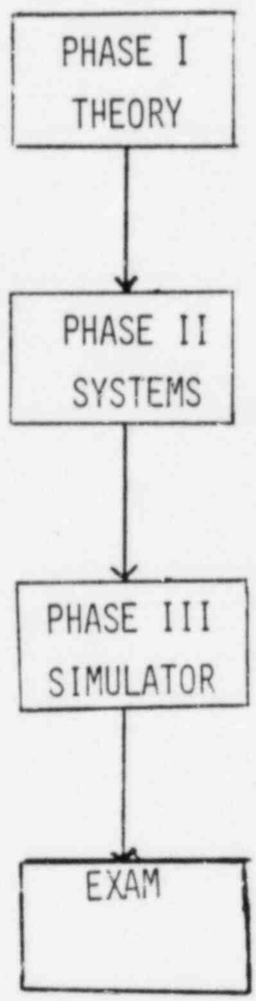
4. Plant Support

31/86

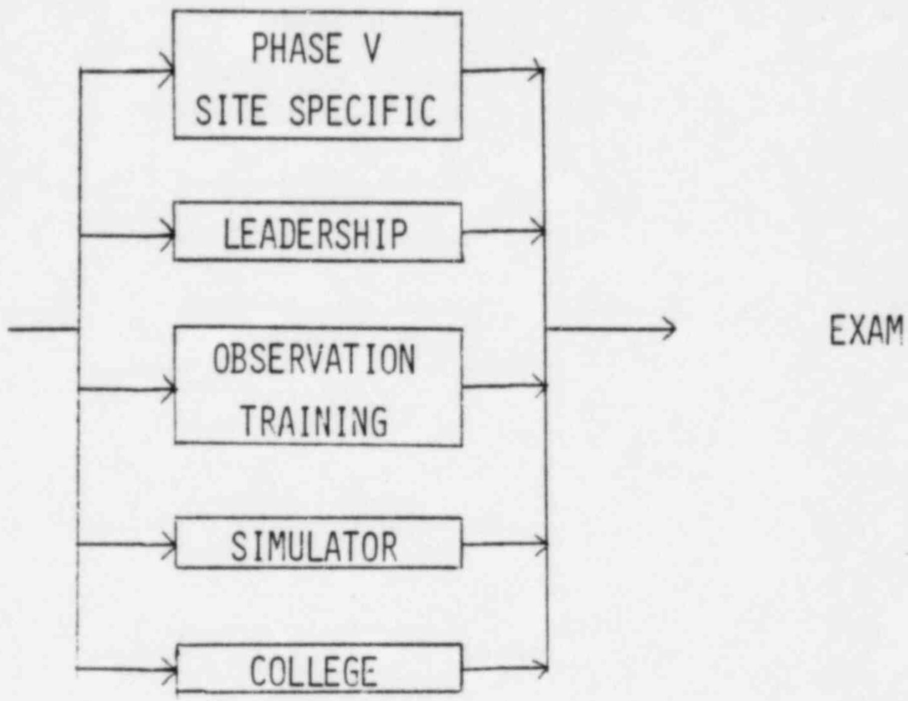
5. Administration

We are complete in this area.

WESTINGHOUSE
CERTIFICATION
PROGRAM



LICENSED OPERATOR
TRAINING AT WCGS



NON-LICENSED OPERATOR TRAINING

CLASSROOM INSTRUCTION

IN-PLANT EXPERIENCE

SIMULATOR DEMONSTRATION

GENERAL EMPLOYEE TRAINING

PLANT FACILITIES AND LAYOUT

RADIATION PROTECTION

PLANT SECURITY

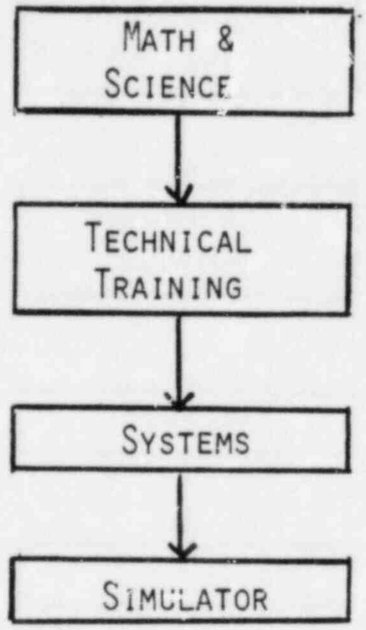
FIRE PROTECTION

INDUSTRIAL SAFETY

QUALITY ASSURANCE/CONTROL

EMERGENCY PLANS

MAINTENANCE PERSONNEL TRAINING



FIRE TRAINING

General Fire Training for Plant Personnel

WCGS Fire Protection Policies

Recognition and Response to Alarms

Reporting of Fires

Types of Fire Extinguishers and Use

Fire Brigade Members

Fire Fighting Plans

Fire Hazards

Use of Fire Fighting Equipment
and other Emergency Equipment

Fire Brigade Leaders

Fire Fighting Strategy

Affects of Plant Modifications on Fire Plans

SHIFT TECHNICAL ADVISOR

AND SIMULATOR TRAINING PROGRAMS

FOR THE

WOLF CREEK GENERATING STATION

UNIT NO. 1

RICHARD COULTHARD, MANAGER NUCLEAR TRAINING

A-162

CONSIDERATIONS FOR COLLEGE PROGRAM

1. DRAFT ANS 3.1 AND REGULATORY GUIDE 1.8 REQUIRED 60 HOURS OF COLLEGE IN SPECIFIC SUBJECTS
2. INSTITUTE OF NUCLEAR POWER OPERATIONS HAD SIMILAR S.T.A. ACADEMIC REQUIREMENTS
3. S.T.A. IS AN INTERIM POSITION, UNTIL SENIOR OPERATORS UPGRADED
4. WOLF CREEK 30 MONTHS FROM FUEL LOAD
5. S.R.O. CANDIDATES AVAILABLE FOR FULL TIME INSTRUCTION
6. KANSAS STATE & EMPORIA STATE WILLING TO TEACH AT PLANT SITE.

REQUIREMENTS

1. Per September 1980 draft of Regulatory Guide 1.8 for Senior Operators 60 semester hours in technical subjects <u>including</u>:

   Mathematics
   Reactor Physics
   Chemistry
   Materials
   Reactor Thermodynamics
   Fluid Mechanics
   Heat Transfer
   Electrical Theory
   Reactor Control Theory

   Accredited by A.B.E.T. or other nationally recognized agency

2. Addition INPO Shift Technical Advisor Academic Requirements

   Nuclear Materials
   Radiation Protection and Health Physics
   Radiation Detectors

SHIFT TECHNICAL ADVISOR ACADEMIC PROGRAM

Course No.	Title	Credit Hours
	Emporia State Mathematics	
MA-110	College Algebra	3
MA-112	Trigonometry	2
MA-315	Technical Calculus I	3
MA-316	Technical Calculus II	3
MA-317	Applied Differential Equations	3
MA-341	Description Statistics	3
	Emporia State College Level Sciences	
CH-123,124	Chemistry I with Lab	4
CH-126,127	Chemistry II with Lab	4
PH-140,141	Physics I with Lab	4
PH-143,144	Physics II with Lab	4
PH-315	Applied Statics	3
	Kansas State Basic Engineering	
540-410	Properties of Engineering Materials	2
540-514	Energy Conversion Technology	3
540-512	Fluid Mechanics	3
540-530	Electrical Circuit Theory with Lab	4
	Kansas State Nuclear Technology	
540-480	Materials of Nuclear Reactor Systems	2
540-484	Radiation Detection and Monitoring	3
540-481	Nuclear Reactor Technology I	3
540-483	Nuclear Reactor Technology II	3
540-584	Radiation Detection and Monitoring with Lab	3
540-585	Nuclear Reactor Thermal Technology	3
540-586	Radiation Protection Technology	2
	Total	64

Engineering Technology Program Accredited by
Accreditation Board for Engineering and Technology (ABET/ECPD)

COURSE DESCRIPTIONS
WOLF CREEK ACADEMIC PROGRAM
FOR SHIFT TECHNICAL ADVISORS

Course Descriptions

A. Mathematics - Emporia State University

1. College Algebra (MA-110) — 3 semester hours
Basic operations, products and factoring, linear and quadratic equations, graphs, ratio and proportion, inequalities, logarithms, mathematical induction, permutation, combinations, determinants.

2. Trigonometry (MA-112) — 2 semester hours
Trigonometric functions, identities, graphs, trigonometric equations, radian measure, complex numbers, polar coordinates, solving triangles, applications.

3. Technical Calculus I (MA-315) — 3 semester hours
A condensed course in analytic geometry and differential calculus with an emphasis on applications.

4. Technical Calculus II (MA-316) — 3 semester hours
A second course in calculus, including integral calculus with an emphasis on applications.

5. Applied Differential Equations (MA-317) — 3 semester hours
Methods of solution of elementary and linear differential equations, including LaPlace transforms, with applications to geometry and the physical sciences.

6. Descriptive Statistics (MA-341) — 3 semester hours
An introductory study of statist[illegible] student who wishes to apply statistics to his field [illegible] The course includes methods of presenting and interpreting data. Topics include mean, standard deviation, correlations, and Chi-Square tests.

B. Fundamental College Level Sciences - Emporia State University

1. Chemistry I (CH-123, 124) — 4 semester hours
Fundamental principles and concepts of chemistry, including atomic structure and chemical bonding, exemplary non-metals, chemical equations and their quantitative applications, phases of matter, solutions, and chemical kinetics. One credit hour of laboratory experiments is included.

2. Chemistry II (CH-126, 127) — 4 semester hours
A continuation of Chemistry I with emphasis on equilibria and properties of elements and compounds. One credit hour of laboratory experiments is included.

3. College Physics I (PH-140, 141) — 4 semester hours
General principles involved in mechanics, sound and heat with emphasis on energy and the relationship between various forms of energy as related to physics. One credit hour of laboratory experiments is included.

4. College Physics II (PH-143, 144) 4 semester hours
General principles involved in electricity, magnetism, light, and modern physics with emphasis on energy and the relationship between various forms of energy as related to physics. One credit hour of laboratory experiments is included.

5. Applied Statics (PH-315) 3 semester hours
A course for engineering technology students, dealing with rigid bodies and structure and the forces acting on them while at rest. Topics include: free body diagrams: dot and cross products; friction, centroids and center of gravity; area moments of inertia; shear and bending moments of beams; and virtual work.

C. Basic Engineering Courses - Kansas State University

1. Properties of Engineering Materials (540-410) 2 semester hours
Engineering requirements of materials; arrangements of atoms in materials; metallic and ceramic phases and their properties; polymers; multiphase equilibrium and non-equlibrium relationships; modification of properties through changes in microstructure; thermal behavior in service; corrosion; effect of radiation on materials.

2. Energy Conversion Technology (540-514) 3 semester hours
Introduction to energy and power, thermodynamics, power cycles and refrigeration. Topics include heat balance, development and use of steam tables and Molier charts, concepts of efficiency and steam cycles.

3. Fluid Mechanics (540-512) 3 semester hours
Fluid properties, fluid statics, fluid dynamics of high and low viscosity fluids, including pipe flow, open channel flow, flow about immersed objects, fluid machinery and fluid flow measurement.

4. Electrical Circuit Theory (540-530) 4 semester hours
D-C and A-C steady state circuit analysis. Study of resistance, capacitance and inductance. Basic magnetic circuits. Polyphase steady state circuits. Study of A-C machinery with emphasis on applications. One credit hour of laboratory experiments is included.

D. Nuclear Engineering Technology Courses - Kansas State University

1. Materials of Nuclear Reactor Systems (540-480) 2 semester hours
A course on the properties and behavior of structural materials; fuels and components in the radiation environment. Selected nuclear fuel cycle topics are covered.

2. Radiation Detection and Monitoring (540-584) 3 semester hours
Operation principles and characteristics of devices used in the detection and measurement of ionizing radiation. Applications in radiation monitoring and surveillance. One credit hour of laboratory work is included.

3. Nuclear Reacto• Technology I (540-481) 3 semester hours
Introduction to atomic and nuclear physics, including interaction of neutrons and other radiation with matter, production of neutrons, statistics, basic nuclear reactor core neutron balances, and nuclear fuel cycle.

4. Nuclear Reactor Technology II (540-483) 3 semester hours
Theory of diffusion and slowing down of neutrons with application to critical and subcritical reactors. Multigroup diffusion theory, fuel depletion calculations and nuclear reactor kinetics and control.

5. Nuclear Reactor Thermal Technology (540-585) 3 semester hours
Introduction to conduction, convection and radiation heat transfer as applied to reactor cores and systems. Discussion fuel element removal mechanisms and processes. Consideration of nuclear reactor safety and power reactor systems.

6. Radiation Protection Technology (540-586) 2 semester hours
Principles of radiation protection. A study of the biological effects on nuclear radiation, radiation measurement and shielding techniques, and the state and federal regulations concerning radiation safety.

KANSAS GAS AND ELECTRIC COMPANY

ENDORSEMENTS OF SHIFT TECHNICAL ADVISOR

ACADEMIC TRAINING PROGRAM

1. Letter from INPO of February 2, 1981 endorsing program

2. NUREG-0737 endorsement of INPO standard for STA's (top paragraph of page I.A.1.1-2)

INPO INSTITUTE OF NUCLEAR POWER OPERATIONS
1820 Water Place
Atlanta, GA 30339
(404) 953-3600

February 2, 1981

Mr. Richard Coulthard
Manager, Nuclear Training
Kansas Gas and Electric Company
Box 208
Wichita, Kansas 67201

Dear Mr. Coulthard:

Per your request we have reviewed your College Program on Engineering Fundamentals for Shift Technical Advisors.

This series of courses taught by Emporia State University and Kansas State University appear to meet the intent of the requirements specified in our April 30, 1980, "Recommendations for Position Description, Qualifications, Education and Training."

Sincerely,

E. L. Thomas

E. L. Thomas
Director
Training and Education Division

ELT/SCB/dgh

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I.A.1.1 SHIFT TECHNICAL ADVISOR

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements resulting from NUREG-0660 and the October 30, 1979 letter from H. R. Denton to all operating nuclear power plants.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. This paper confirms these requirements and requests additional information.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue.

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979 letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980 letter (as revised by item I.A.1.3 of this enclosure), an STA be available for duty on each operating shift when a plant is being operated in Modes 1-4 for a PWR and Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the October 30, 1979 letter was issued, several efforts have been made to establish, for the longer term, the minimum level of experience, education, and training for STAs. These efforts include work on the revision to ANS-3.1, work by the Institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO recently made available a document entitled "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training." A copy of Revision 0 of this document, dated April 30, 1980, is attached as Appendix C. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. (Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter).)

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their STA training program and their plans for requalification training. This description shall indicate the level of training attained by STAs by January 1, 1981 and demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their long-term STA program, including qualification, selection criteria, training plans, and plans, if any, for the eventual phaseout of the STA program. (Note: The description shall include a comparison of the licensee/applicant program with the above-mentioned INPO document. This request solicits industry views to assist NRC in establishing long-term improvements in the STA program. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.)

Applicability

This requirement applies to all licensees of operating reactors and applicants for operating licenses.

Implementation

(1) Training that meets the lessons-learned requirements shall be completed by January 1, 1981 or by the time the fuel-loading license is issued, whichever is later.

(2) A description of the current training program and demonstration of conformance with the October 30, 1979 letter shall be submitted

SHIFT TECHNICAL ADVISOR TRAINING PROGRAM

1. ACADEMIC QUALIFICATIONS OR TRAINING
2. NUCLEAR POWER PLANT CLASSROOM TOPICS
   A. APPLIED SPECIFICS IN PLANT FUNDAMENTALS
   B. WOLF CREEK PLANT SYSTEMS
   C. GENERAL OP. PROCEDURES, TECHNICAL SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
   D. MANAGEMENT/SUPERVISORY SKILLS
   E. MITIGATING CORE DAMAGE
   F. OPERATIONAL TRANSIENT AND ACCIDENT ANALYSIS
   G. STA ACCIDENT ANALYSIS RESPONSE COURSE
3. SIMULATOR TRAINING
   A. OPERATIONAL AND ACCIDENT TRAINING COURSE
   B. STA ACCIDENT RESPONSE AND RECOVERY COURSE
4. ANNUAL REQUALIFICATION
   A. ADDITIONAL REQUIREMENTS BEYOND SRO PROGRAM

WOLF CREEK SIMULATOR CAPABILITIES

Replicates Wolf Creek Main Control Board

Simulates over 40 SNUPPS Systems

Models Wolf Creek Specific Systems

Over 200 Different Multi-Variable Malfunctions

Can fail any Meter, Component, Valve

Backtrack, Freeze, Replay and Slow Time Capabilities

Major Modification and Upgrading Program Planned

SIMULATOR TRAINING PROGRAMS

Cold License and Hot License Operator Training

Hot License Startup Certification Program

Licensed Operator Requalification

Reactor Operator to Senior Reactor Operator Upgrade Training

Shift Technical Advisor Training

Professional Staff Training

SHIFT TECHNICAL ADVISOR

PROGRAM

FOR THE

WOLF CREEK GENERATING STATION

UNIT NO. 1

THOMAS D. KEENAN, DIRECTOR NUCLEAR OPERATIONS

A-175

NUREG 0578, July 1979

TMI-2 Lessons Learned Task Force

Status Report and Short-Term Recommendations

Section 2.2 Operations

2.2.1 Improved Reactor Operations Command Function

" . . . Improvements in operator qualifications, training and licensing; technical qualifications of overall reactor operations . . . will be recommended by NRR and others in the coming months. In the interim, the Task Force recommends prompt implementation of the following administrative changes and controls to significantly improve existing operational capabilities.

b. Shift Technical Advisor

Provide on shift at each nuclear power plant a qualified person (the shift technical advisor) with a bachelor's degree or equivalent in a science or engineering discipline and with specific training in the plant response to off-normal events and in accident analysis of the plant.

Shift technical advisors shall serve in an advisory capacity to shift supervisors. The licensee shall assign normal duties to the shift technical advisor that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience."

ACRS LETTER TO DR. HENDRIE, AUGUST 13, 1979

Page 2, Section 2.2.1.b Shift Technical Advisor

"The Committee agrees completely with the two closely related objectives of this recommendation. One relates to the presence in the control room during off-normal events of an individual having technical and analytical capability and dedicated to concern for safety of the plant. The other relates to the need for an on-site, and perhaps dedicated, engineering staff to review and evaluate safety-related aspects of plant design and operation. The achievement of these objectives will contribute significantly to the safe operation of a plant.

"The Committee believes that there may be difficulty in finding a sufficient number of people with the required qualifications and interest in shift work to fill the te. nical advisor positions. The Committee therefore believes the solution proposed by the Staff should not be mandatory but that alternate solutions also should be considered."

NUREG 0660, August 1980
NRC Action Plan Developed as a Result
of TMI-2 Accident

"Task I.A.1

1. Shift technical advisor.

a. Description: Technical advisors with engineering expertise and special training in plant dynamic response are required by NRC to accomplish two functions: 1) on-shift advice and assistance to the control room supervisor in the event of an accident, and 2) evaluation of operating experience. In the past, the staff has accepted the assignment of these two functions to two separate groups at the prerogative of the individual licensee. With the implementation of Item I.B.1.1, the staff will require that the operating experience evaluation function be assigned to the onsite safety engineering group. The long-term need for a shift technical advisor to provide advice to the control room supervisor may be eliminated when upgraded qualifications for the control room supervisor (Item I.A.2.6) and improved control rooms (Task I.D.1) have been attained."

NUREG 0731, SEPTEMBER 1980

Guidelines for Utility Management Structure and Technical Resources

"c. Shift Technical Advisor

A Shift Technical Advisor (STA) shall be available onsite to each operating shift. There shall be at least one STA assigned full time at each site from which one or more reactors is operating; the STA shall be available to report to the control room to act in an advisory capacity to the shift supervisor in a matter of minutes. The STA shall be qualified to ; ovide technical support to the shift supervisor in the ar.:as of thermal hydraulics, reactor engineering, and plant analysis. <u>The long-term need for a shift technical advisor to provide advice to the control room supervisor may be elininated when upgraded qualifications for the control room supervisor and improved control rooms have been attained</u>. Minimum qualification requirements for the STA are as described in Section 4 of ANSI/ANS 3.1."

EXCERPTS FROM NUREG 0737, NOVEMBER 1980

Clarification of TMI Action Plan Requirements

"The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until those long-term improvements are attained, the need for an STA program will continue."

"INPO recently made available a document entitled "Nuclear Power Plant Shift Technical Advisor - Recommendations for Position Description, Qualifications, Education and Training." A copy of Revision 0 of this document, dated April 30, 1980, is attached as Appendix C. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. (Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter)."

"No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their long-term STA program, including qualification, selection criteria, training plans, and plans, if any, for the eventual phaseout of the STA program. (Note: The description shall include a comparison of the licensee/applicant program with the above-mentioned INPO document. This request solicits industry views to assist NRC in establishing long-term improvements in the STA program. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.)"

EXCERPTS FROM ANS 3.1 DRAFT REVISION 4/10/81

Standard for Selection, Qualification and Training of Personnel for Nuclear Power Plants

"The Shift Technical Advisor's position of this standard is indicated as being a temporary operating staff position in this standard since the ANS-3 committee believes that <u>for the long term the line organization of shift management must be trained to fulfill this need</u>. The STA is therefore considered to have a narrow scope of responsibility concentrating on plant transient analysis and response, recognition of degradation of safety system and core cooling parameters and advising the shift supervisor regarding corrective actions that should be taken to maintain core cooling and keep the plant in a safe condition."

"4.3 SUPERVISORS

4.3.1 SUPERVISORS REQUIRING NRC LICENSE

4.3.1.1 SHIFT SUPERVISOR

(Person in charge of operations on shift at the station).

a. EDUCATION: High school diploma, plus the equivalence of sixty (60) semester hours of college level education (900 classroom or instructor conducted hours) in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, electrical and reactor control theory.

<u>If the shift supervisor does not meet these educational requirements, a shift technical advisor (4.4.8) shall be present during this supervisor's shift.</u>"

The membership of ANS-3 at the time of its approval of this standard was:

J. E. Smith, Chairman, Duke Power Company
G. Carl Andognini, Boston Edison Company
Samuel E. Bryan, U.S. Nuclear Regulatory Commission
W. W. Crouch, Power Authority of the State of New York
Frank W. Dougherty, EDS Nuclear Inc.
Norm Elliott, The Babcock & Wilcox Company
H. Falter, Power Systems - A Morrison - Knudsen Division
Harry J. Green, Tennessee Valley Authority
Frank L. Kelly, Personnel Qualification Services
Hans L. Ottoson, Southern California Edison Company
Frank A. Palmer, Commonwealth Edison Company
W. J. Ritsch, EDS Nuclear, Inc.
R. J. Rodriguez, Sacramento Municipal Utility District
Donald J. Skovholt, U.S. Nuclear Regulatory Commission
Jim Shiffer, Pacific Gas & Electric
Phillip Snyder, American Nuclear Insurers
E. L. Thomas, Institute of Nuclear Power Operations
W. T. Ullrich, Peach Bottom Atomic Power Station
G. K. Whitham, Argonne National Laboratory
Peter Walzer, Combustion Engineering, Inc.

Note: F. A. Palmer served as director for the effort to produce this revision.

APPENDIX 3 TO SECOND PROPOSED REVISION 2
TO REGULATORY GUIDE 1.8
PERSONNEL QUALIFICATION AND TRAINING

Criteria for Shift Technical Advisor
(An Excerpt from NRR September 13, 1979 letter
to all operating power plants)

"4. Detachment from Operations

The plant response assessment function requires a measure of detachment from the manipulation of controls or immediate supervision of operators. This is intended to provide the perspective and the time for assessing plant conditions and advising on appropriate operator actions. It has been called a safety monitor characteristic. Currently only three operators would normally be in the control room at the time an unusual event occurred, and it is allowed that at times there would be fewer. This number is only enough to satisfy the demands for prompt control and supervisory actions under off normal conditions. The time necessary to make a considered assessment and permit independent monitoring of plant safety requires one more person in the form of the Shift Technical Advisor or some alternative in the control room.

5. Independence from Operations

In order to provide both perspective in assessment of plant conditions and dedication to the safety of the plant, this function should have a clear measure of independence from duties associated with the commercial operation of the plant. In an accident situation where command authority should not be diluted, complete independence is not desirable and is not necessary to the safety assessment function.

6. Availability

This capability should be readily available in the control room, preferably immediately at all times, but at most within ten minutes. Having this capability on duty for each shift is the best approach."

CONCLUSION

The KG&E program for incorporating the STA function within the operating shift line management - the shift supervisor - is an appropriate application for the following reasons -

1. It meets the NRC issued program recommendations as delineated in -
   a. NUREG 0578 July 1979
   b. NUREG 0660 August 1980
   c. NUREG 0731 Sept 1980
   d. NUREG 0737 Nov 1980
   e. ANS 3.1, 4/10/81 - endorsed by Regulatory Guide 1.8

2. The Academic Instruction Program is fully accredited and has been approved by INPO.

3. Experienced shift consultants will be used during first year of operation.

4. Site specific simulator is available for training.

5. Shift manning level meets or exceeds NRC staff recommendations for total size and licenses.

6. Human Factors Upgrade of Control Room accomplished, including addition of Safety Parameter Display System.

7. Similar concept is already in use at several operating plants.

8. Operating experience assessments - an original function of the STA - is now assigned to the ISEG on site.

9. Effective management rules support this approach. Even though the roles of shift supervisor and STA can be carefully delineated by procedure and training, industrial and military experience indicate that a direct-line organization wherein authority and responsibility are interdependent is required to effectively operate in a crisis environment.

Table B-1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift	Additions Within 30 minutes
Plant Operations and Assessment of Operational Aspects		Shift Supervisor (SRO)[1/]	1	--
		Shift Foreman (SRO)[1/]	1	--
		Control Room Operators	2	--
		Auxiliary Operators	2	
Emergency Direction and Control (Emergency Coordinator)		Shift Technical Advisor	1**	--
		Shift Supervisor or designated facility manager		
Notification/ Communication	Notify licensee, State local and Federal personnel & maintain communication		1	3
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency Operations Facility (EOF) Director	Senior Manager	--	1
	EOF Offsite Dose Assessment	Senior Health Physics (HP) Expertise		1
	Offsite Surveys		--	4
	Onsite (out-of-plant)		--	2
	In-plant surveys	HP Technicians	1	2
	Chemistry/Radio-chemistry	Rad/Chem Technicians	1	1
Plant System Engineering, Repair and Corrective Actions	Technical Support	Shift Technical Advisor[2/]	1	--
		Core	--	1
		Electrical	--	1
		Mechanical	--	1
	Repair and Corrective Actions	Mechanical Maintenance/	1**	1
		Rad Waste Operator		1
		Electrical Maintenance/	1**	2
		Instrument and Control (I&C) Technician		1

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Table B-1 (contd)

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Additions Within 30 Minutes
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	4
Firefighting	--	--	Fire Brigade per Technical Specifications	Local Support
Rescue Operations and First-Aid	--	--	2**	Local Support
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	All per Security plan	
		Total	10	26

4-18-86

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator. This means that a single unit will require a minimum shift complement of 10, a two-unit complex 13, and a three-unit complex 16.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

1/ At least one of these must be a Senior Reactor Operator (SRO).

2/ For a multi-unit site this function may be filled by a Shift Supervisor or Foreman, provided all other qualification requirements are met.

Tai -1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES (See B.5.)

Major Functional Area	Location	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Plant Operations and Assessment of Operational Aspects			Shift Supervisor (SRO)	1	--	--
			Shift Foreman (SRO)	1	--	--
			Control Room Operators	2	--	--
			Auxiliary Operators	2	--	--
Emergency Direction and Control (Emergency Coordinator)***			Shift Technical Advisor, Shift Supervisor or designated facility manager	1**	--	--
Notification/ Communication****		Notify licensee, State local and Federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment		Emergency Operations Facility (EOF) Director	Senior Manager	--	--	1
		Offsite Dose Assessment	Senior Health Physics (HP) Expertise		1	--
		Offsite Surveys		--	2	2
		Onsite (out-of-plant)		--	1	1
		In-plant surveys	HP Technicians	1	1	1
		Chemistr dio-chemist	Rad/Chem Technicians	1	--	1
Plant System Engineering, Repair and Corrective Actions		Technical Support	Shift Technical Advisor	1	--	--
			Core/Thermal Hydraulics	--	1	
			Electrical	--	--	1
			Mechanical	--	--	1
		Repair and Corrective Actions	Mechanical Maintenance/	1**	--	1
			Rad Waste Operator			1
			Electrical Maintenance/	1**	1	1
			Instrument and Control (I&C) Technician	--	1	--

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Table B-1 (contd)

Major Functional Area	Major Tasks	Position Title or Expertise	On Shift*	Capability for Additions 30 min	60 min
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting c. Personnel monitoring d. Dosimetry	HP Technicians	2**	2	2
Firefighting	--	--	Fire Brigade per Technical Specifications	Local Support	
Rescue Operations and First-Aid	--	--	2**	Local Support	
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	All per Security plan		
		Total	10	11	15

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

**** May be performed by engineering aide to shift supervisor.

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NUREG 0654, Rev 1, November 1980
STAFFING REQUIREMENTS

	Major Functional Area	Title	NUREG 0654	Wolf Creek
1.	Plant Operations and Assessment of Operational Aspects	Shift Supervisor (SRO)	1	1
		Shift Foreman (SRO)	1	1
		Control Room Operators (RO)	2	2
		Auxiliary Operators	2	4
2.	Emergency Direction & Control	Shift Technical Advisor	1**	1**
3.	Notification Communication	(no title)	1	1**
4.	Radiological Accident Assessment and Support	H/P Technicians	1	1
		Rad/Chem Technicians	1	1
5.	Plant System Engineering, Repair, and Corrective Actions	Shift Technical Advisor	1	assigned to TSC staff
6.	Site Access and Control	Security Personnel	per Security plan	per Security plan
	TOTAL		10	10
	Shift Consultant*			1*
			10	11***

*First year of operation at Wolf Creek
**May be provided by shift personnel assigned other functions
***First year of operation; thereafter 10

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SEISMIC DESIGN OF THE PLANT
AND EQUIPMENT

FOR

WOLF CREEK GENERATING STATION,
UNIT NO. 1

CANDACE J. M. SPROUT, TECHNICAL STAFF ENGINEER,
FACILITIES ENGINEERING AND ANALYSIS

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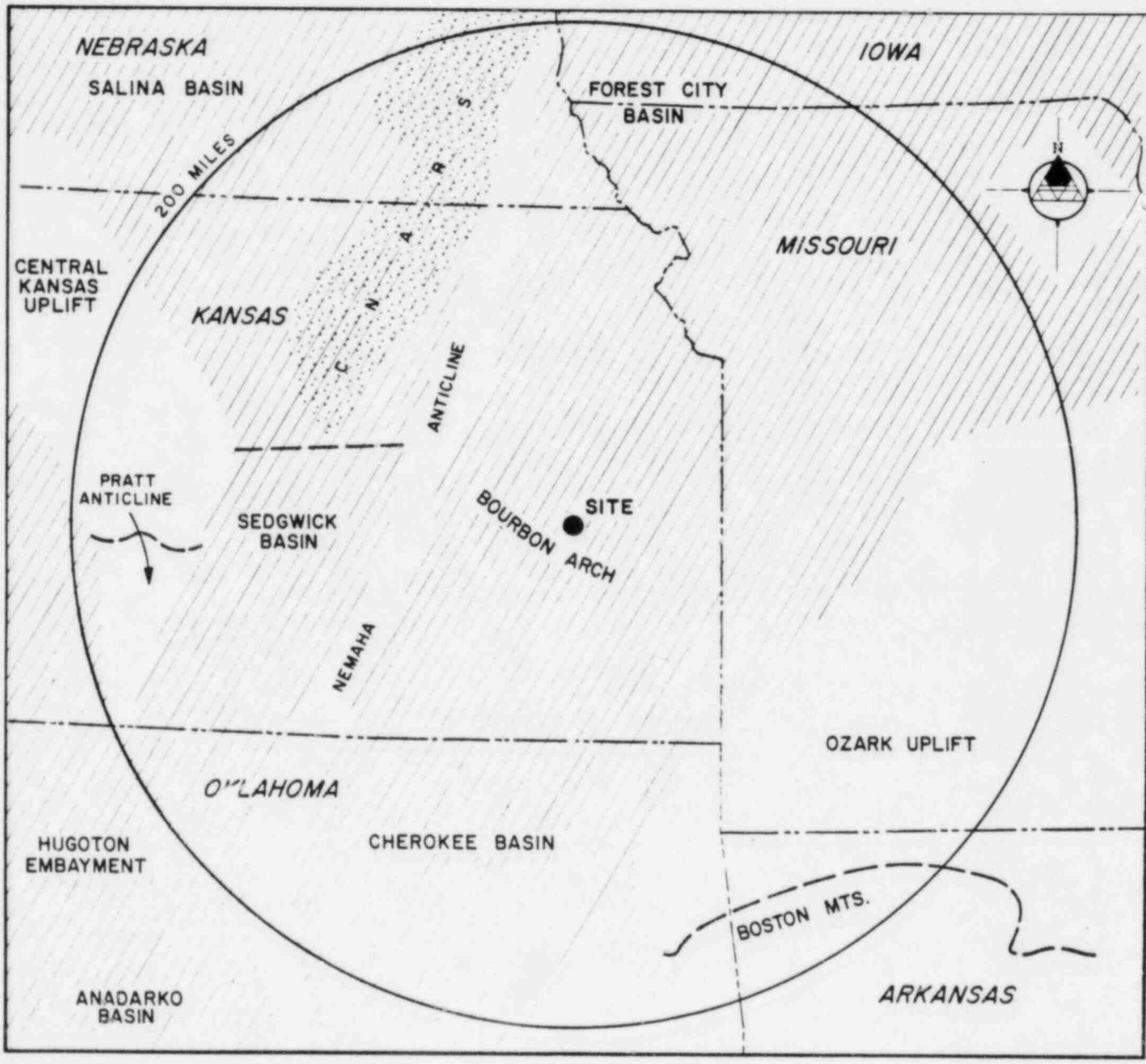
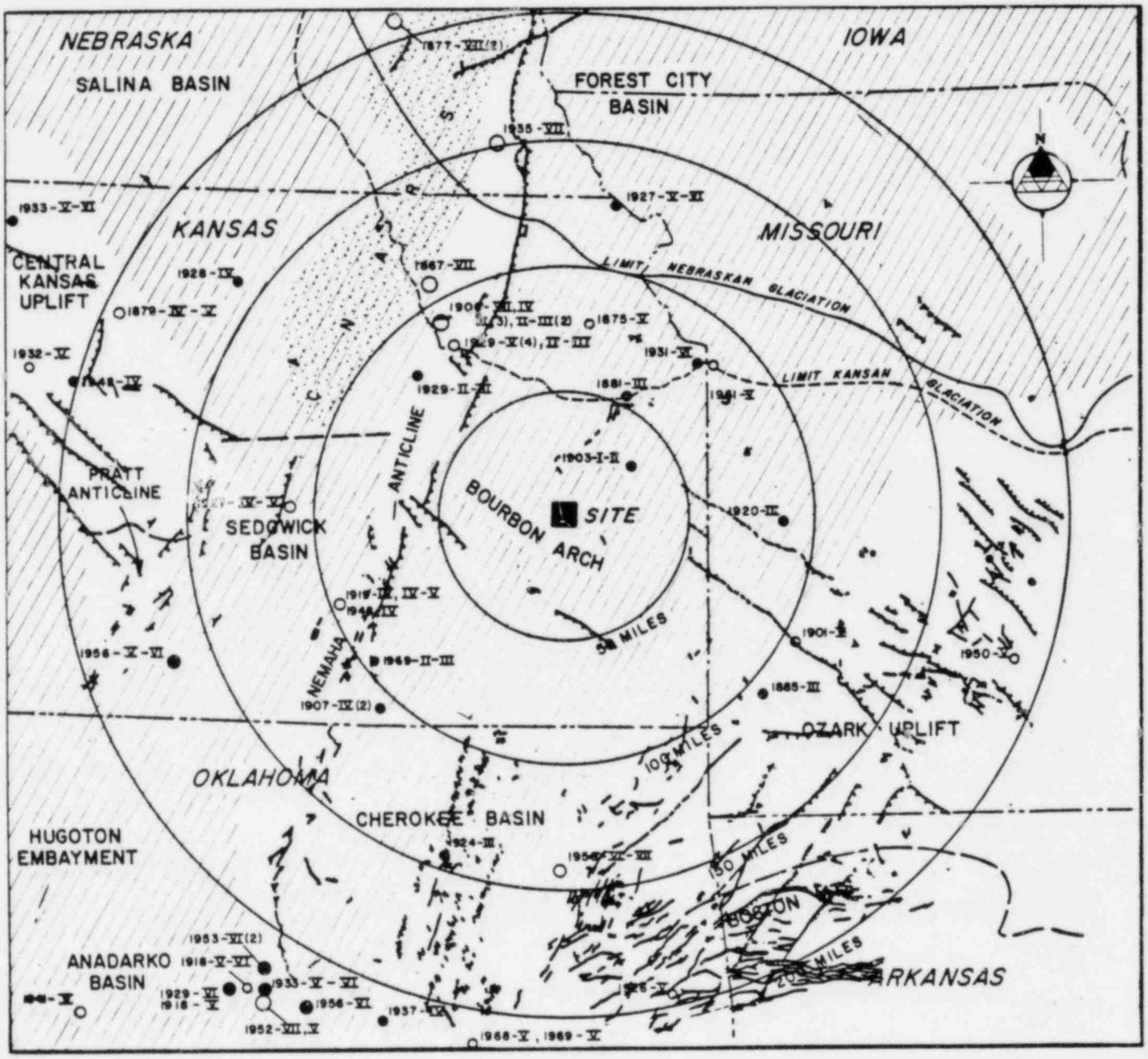

NEBRASKA
SALINA BASIN
IOWA
FOREST CITY
BASIN
200 MILES
N
CENTRAL
KANSAS
UPLIFT
KANSAS
MISSOURI
ANTICLINE
PRATT
ANTICLINE
SEDGWICK
BASIN
SITE
BOURBON ARCH
NEMAHA
OZARK UPLIFT
O'LAHOMA
HUGOTON
EMBAYMENT
CHEROKEE BASIN
BOSTON MTS.
ANADARKO
BASIN
ARKANSAS


NEBRASKA
IOWA
SALINA BASIN
FOREST CITY BASIN
KANSAS
MISSOURI
CENTRAL KANSAS UPLIFT
LIMIT NEBRASKAN GLACIATION
LIMIT KANSAN GLACIATION
PRATT ANTICLINE
SEDGWICK BASIN
NEMAHA ANTICLINE
BOURBON ARCH
SITE
50 MILES
100 MILES
150 MILES
200 MILES
OZARK UPLIFT
OKLAHOMA
CHEROKEE BASIN
HUGOTON EMBAYMENT
ANADARKO BASIN
BOSTON
ARKANSAS
N

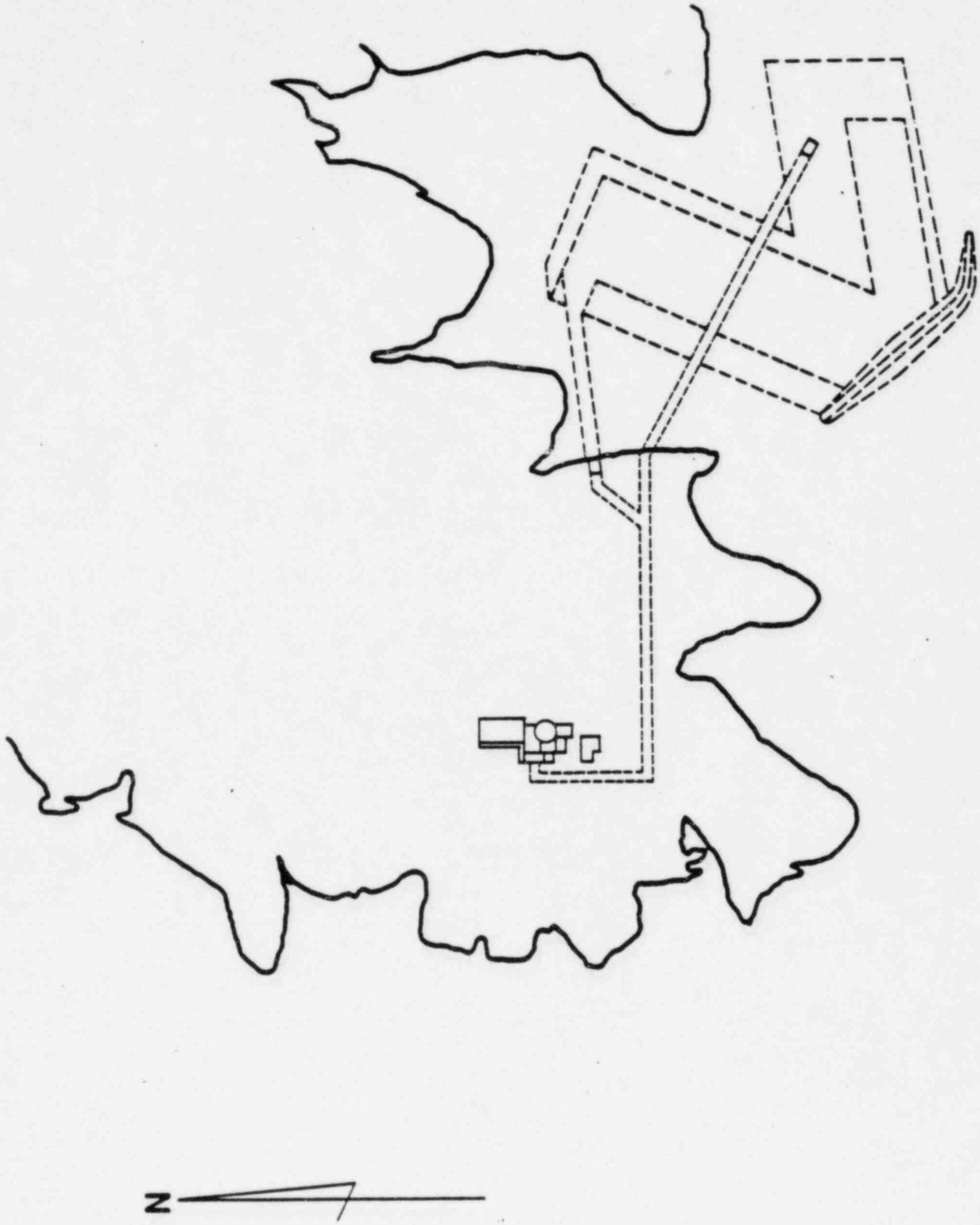
TECTONIC PROVINCE METHOD

PROBABLE MAXIMUM EVENT	INTENSITY TO MAGNITUDE EQUATION (NUTTLI AND HERRMAN, 1978)		RECENT ATTENUATION EQUATIONS	MAXIMUM SITE GROUND ACCELERATION
1. MAXIMUM AT NEMAHA (50 MILES)	INTENSITY VIII = m_b OF 5.75	→	ATTENTUATION	= .05g < DESIGN = .12g
2. MAXIMUM RANDOM (WITHIN 25 KM)	INTENSITY VII = m_b OF 5.25	→	ATTENTUATION	= .10g < DESIGN = .12g

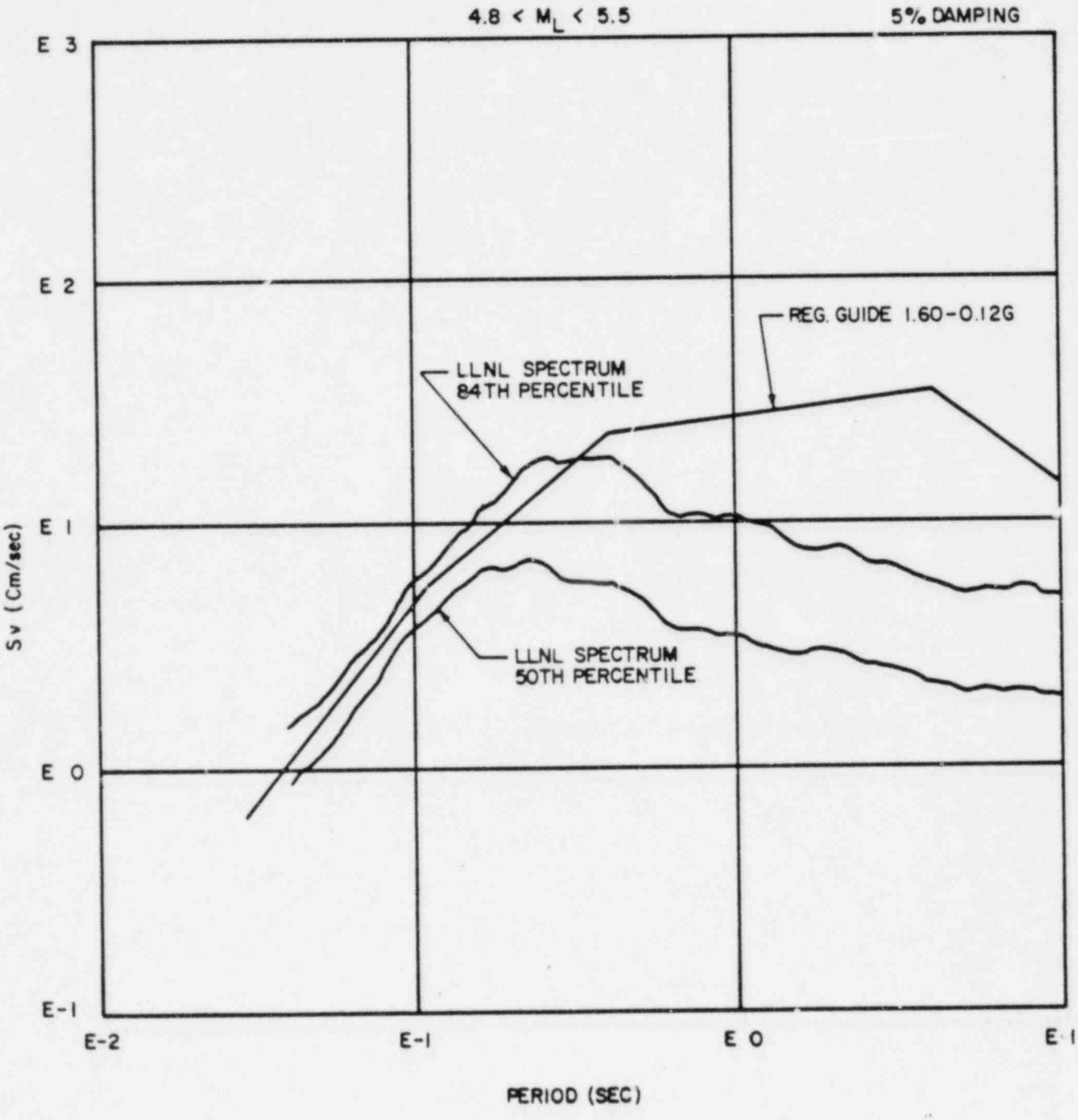
SYSTEMATIC EVALUATION PROGRAM METHOD (LLNL REPORT, APP.E, NUREG-0881)

1. MAXIMUM AT NEMAHA (80 KM)	REAL RECORDS $5.5 \leq M_L \leq 6.5$ $3\text{KM} \leq R \leq 25$ KM	→	SCALE TO SITE CONDITIONS	= SCALED SPECTRA	< DESIGN = .12g
2. MAXIMUM RANDOM (WITHIN 20 KM)	m_b OF 5.2 = M_L OF 5.3	→	REAL SPECTRA ROCK SITES $R \leq 20$ KM $4.8 \leq M_L \leq 5.6$	= 84TH PERCENTILE SPECTRA	> DESIGN = .12g

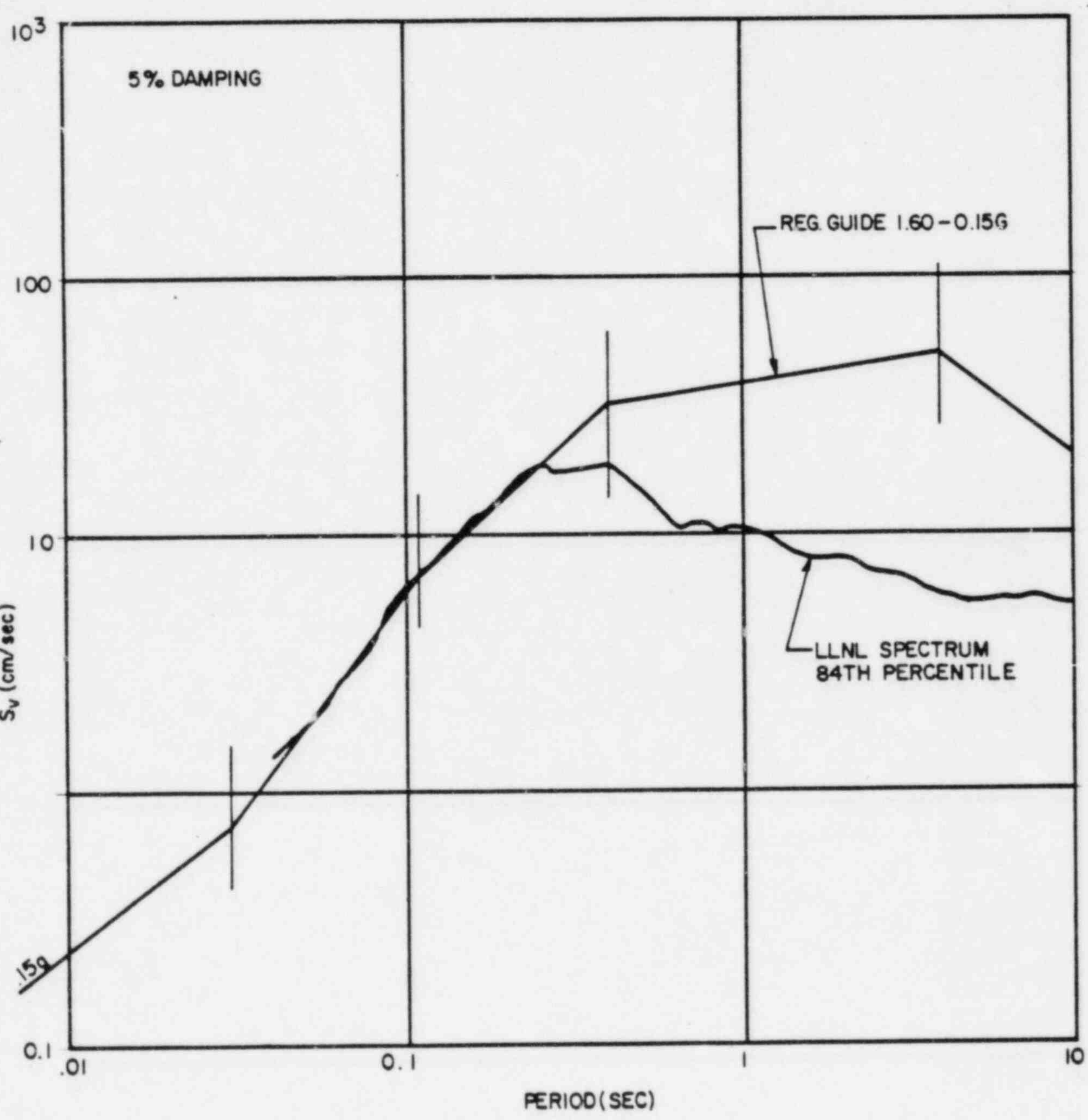
A-19 3

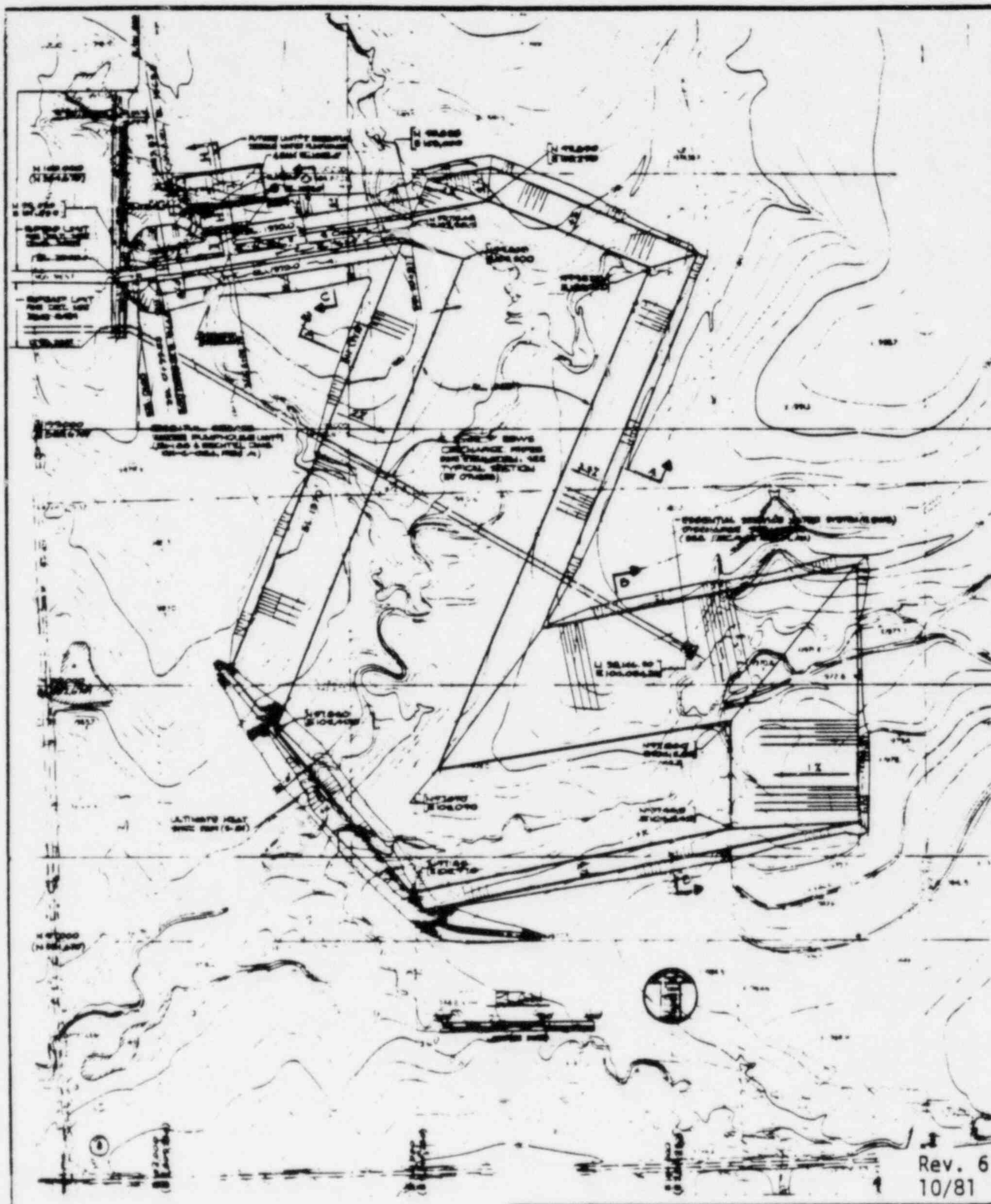


COMPARISON OF R.G. 1.60 SPECTRUM (0.12g) TO 50TH & 84 PERCENTILE SPECTRA FOR ROCK SITES.



0.15g REGULATORY GUIDE 1.60 DESIGN SPECTRA





Rev. 6
10/81

WOLF CREEK GENERATING STATION
UNIT NO. 1
FINAL SAFETY ANALYSIS REPORT

FIGURE 2.5-108

ULTIMATE HEAT SINK

A-197

ESSENTIAL SERVICE WATER SYSTEM

Electrical Manholes

Valvehouse

Discharge Structure

Piping Encasement

OBE Load Case $U=1.4D + 1.7L + 1.9E_{OBE}$

SSE Load Case $U=1.0D + 1.0L + 1.0E_{SSE} + (T_o+R_o)$

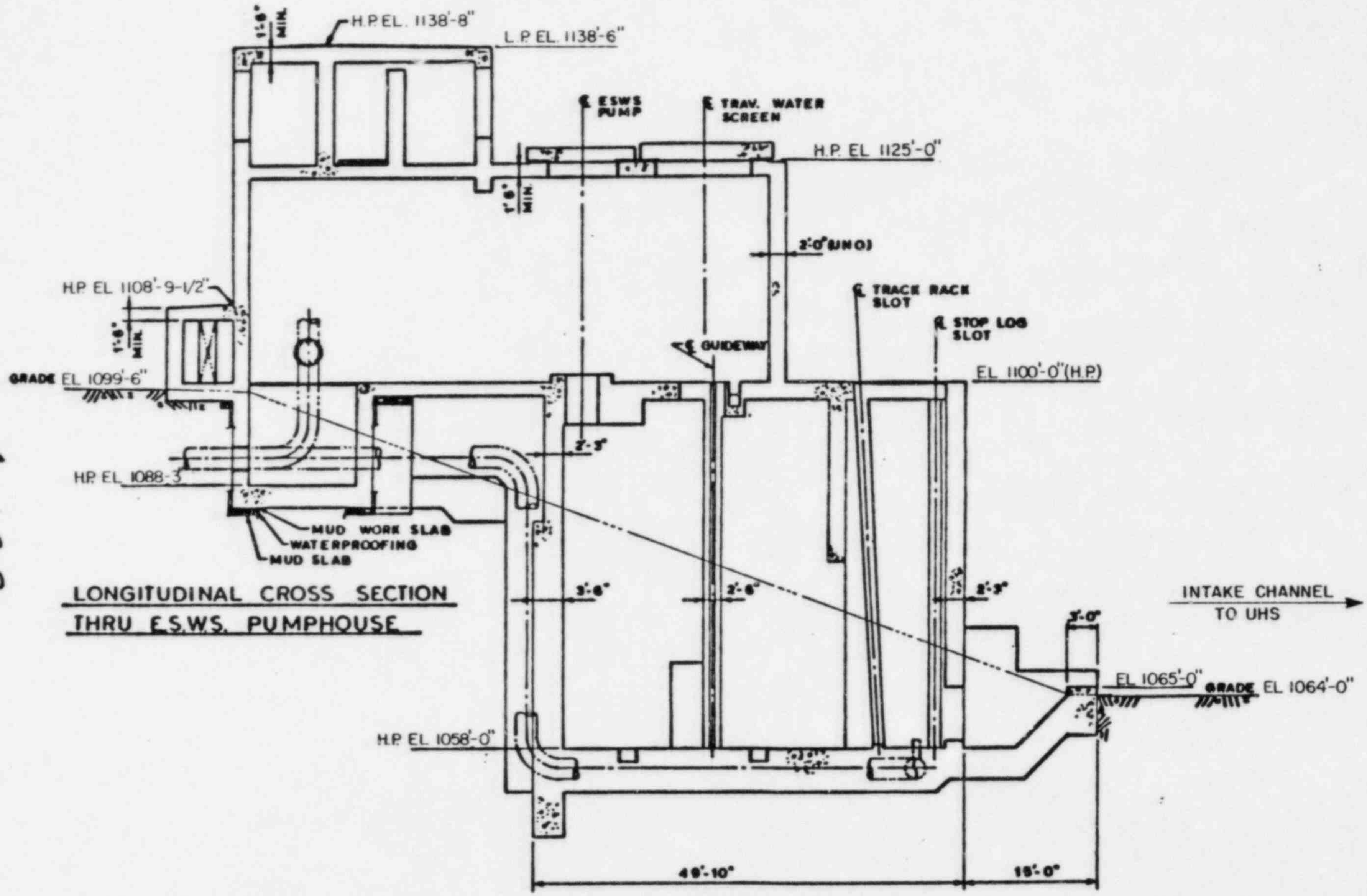
Where U=Required Section Strength

D=Dead Load

L=Live Load

E_{OBE}, E_{SSE}=Seismic Loads

(T_o & R_o) are thermal and pipe reaction loads which are either negligible or do not affect design for these structures)



LONGITUDINAL CROSS SECTION THRU E.S.W.S. PUMPHOUSE

E-W SECTION - E.S.W.S. PUMPHOUSE

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CONCLUSIONS

1. THE WOLF CREEK GEOLOGICAL AND SEISMOLOGICAL ANALYSIS COMPLIES WITH APPENDIX A TO 10CFR100.

2. THE SITE-SPECIFIC DESIGN RESPONSE SPECTRUM COMPLIES WITH THE REGULATORY GUIDE 1.60 SPECTRA ANCHORED AT .12g.

3. BASED UPON THE TECTONIC PROVINCE APPROACH, THE CALCULATED MAXIMUM CREDIBLE GROUND ACCELERATION AT THE WOLF CREEK SITE IS .10g.

4. THE 84TH PERCENTILE, REAL ROCK SPECTRUM OF THE LAWRENCE LIVERMORE ANALYSIS EXCEEDS THE WOLF CREEK DESIGN SPECTRUM (REGULATORY GUIDE 1.60 ANCHORED AT .12g).

5. THE EFFECTS OF THE EXCEEDANCE ON SITE-SPECIFIC CATEGORY I STRUCTURES HAVE BEEN EVALUATED. THE CATEGORY I STRUCTURES AT WOLF CREEK REMAIN WITHIN ALL LICENSING STRESS LIMITS IMPOSED ON THE ORIGINAL DESIGN.

CONTROL ROOM DESIGN

FOR

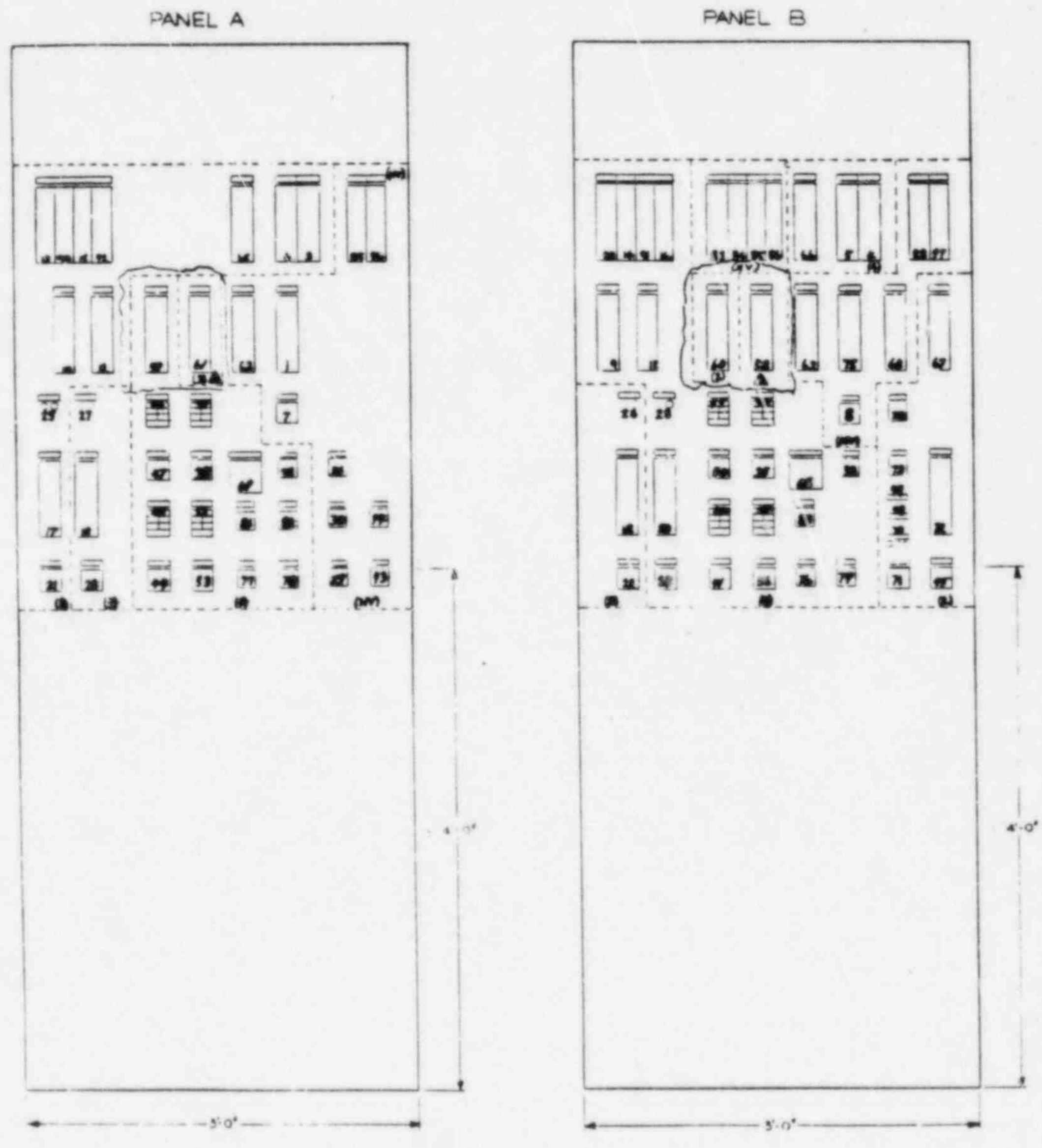
WOLF CREEK GENERATING STATION,
UNIT NO. 1

JACK M. PIPPIN, MANAGER INSTRUMENTATION AND CONTROL

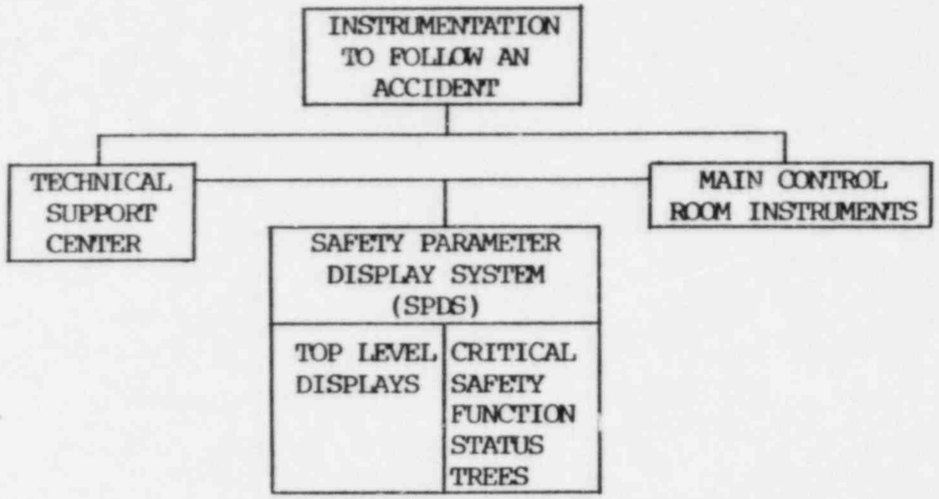
A-201

RP 118

AUXILIARY SHUTDOWN PANEL



A-202



PLANT INSTRUMENTATION SUPPLYING DATA					
BALANCE OF PLANT INSTRUMENTATION	RADIATION MONITORING AND RELEASE TRACKING SYSTEMS	NUCLEAR INSTRUMENT SYSTEM	SUBCOOLING MONITORING SYSTEM		REACTOR COOLANT INSTRUMENTATION
STEAM GENERATOR LEVEL PRESSURE FW FLOW STEAM LINE PRESSURE AVAILABILITY STEAM DUMP CONDENSER CONTAINMENT PRESSURE SUMP LEVEL % HYDROGEN HUMIDITY	METEOROLOGY TOWERS PROCESS RADIATION MONITORS POST ACCIDENT SAMPLING SYSTEM AREA RADIATION MONITORS	SOURCE INTERMEDIATE POWER RANGE	REACTOR COOLANT TEMPS T/Cs HOT&COLD RTDs RCS WIDE RANGE TEMP/PRESS	REACTOR VESSEL LEVEL	REACTOR COOLANT PUMPS PRESSURE PRESSURIZER LEVEL PRESSURIZER PRESSURE RHR SYSTEM

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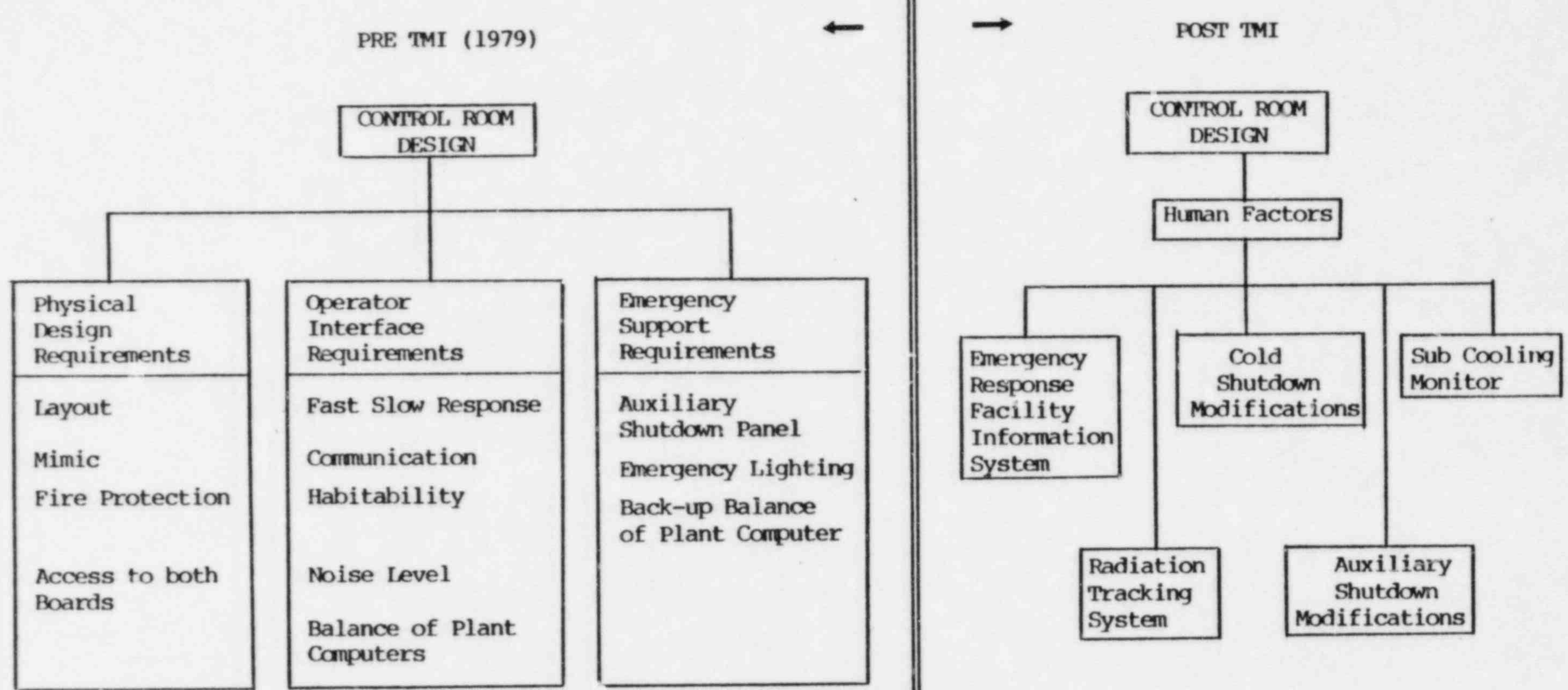

PRE TMI (1979)
CONTROL ROOM DESIGN
Physical Design Requirements
Layout
Mimic
Fire Protection
Access to both Boards
Operator Interface Requirements
Fast Slow Response
Communication
Habitability
Noise Level
Balance of Plant Computers
Emergency Support Requirements
Auxiliary Shutdown Panel
Emergency Lighting
Back-up Balance of Plant Computer
POST TMI
CONTROL ROOM DESIGN
Human Factors
Emergency Response Facility Information System
Radiation Tracking System
Cold Shutdown Modifications
Auxiliary Shutdown Modifications
Sub Cooling Monitor

CONTROL ROOM
HUMAN FACTORS REVIEW

FOR

WOLF CREEK GENERATING STATION,
UNIT NO. 1

JAMES M. McKINSTRAY, OPERATIONS COORDINATOR

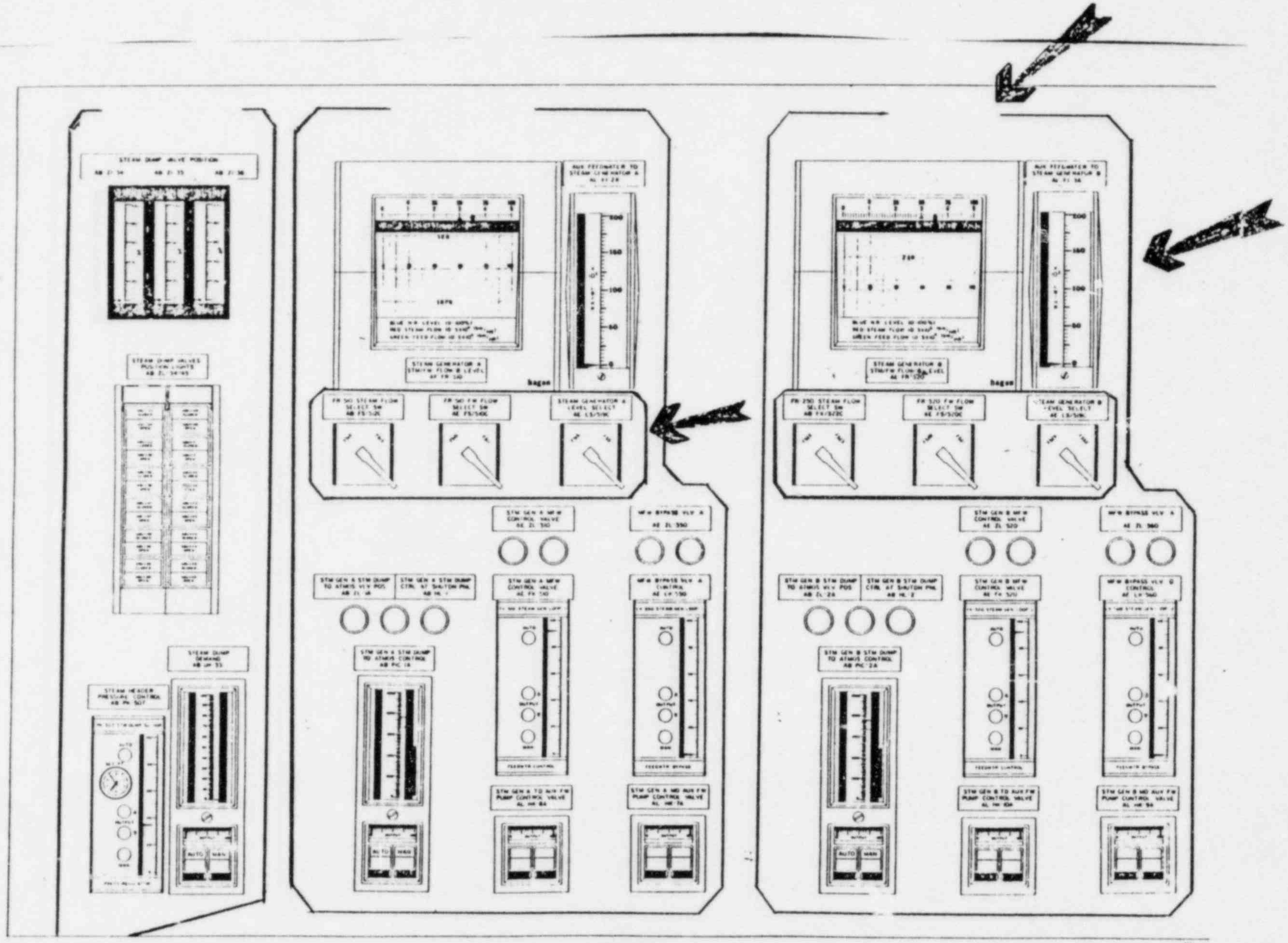
A-205

COMPARISON OF
ANNUNCIATOR ENGRAVING
BEFORE AND AFTER
HUMAN FACTORS REVIEW

BEFORE	AFTER
TURBINE VACUUM TRIP	VAC LO TURB TRIP

USE OF DEMARCATION

ON BALANCE OF PLANT PANEL RL006



PREPARATION OF EMERGENCY OPERATING PROCEDURES

FOR

WOLF CREEK GENERATING STATION, UNIT NO. 1

JAMES A. ZELL, OPERATIONS SUPERVISOR

A-208

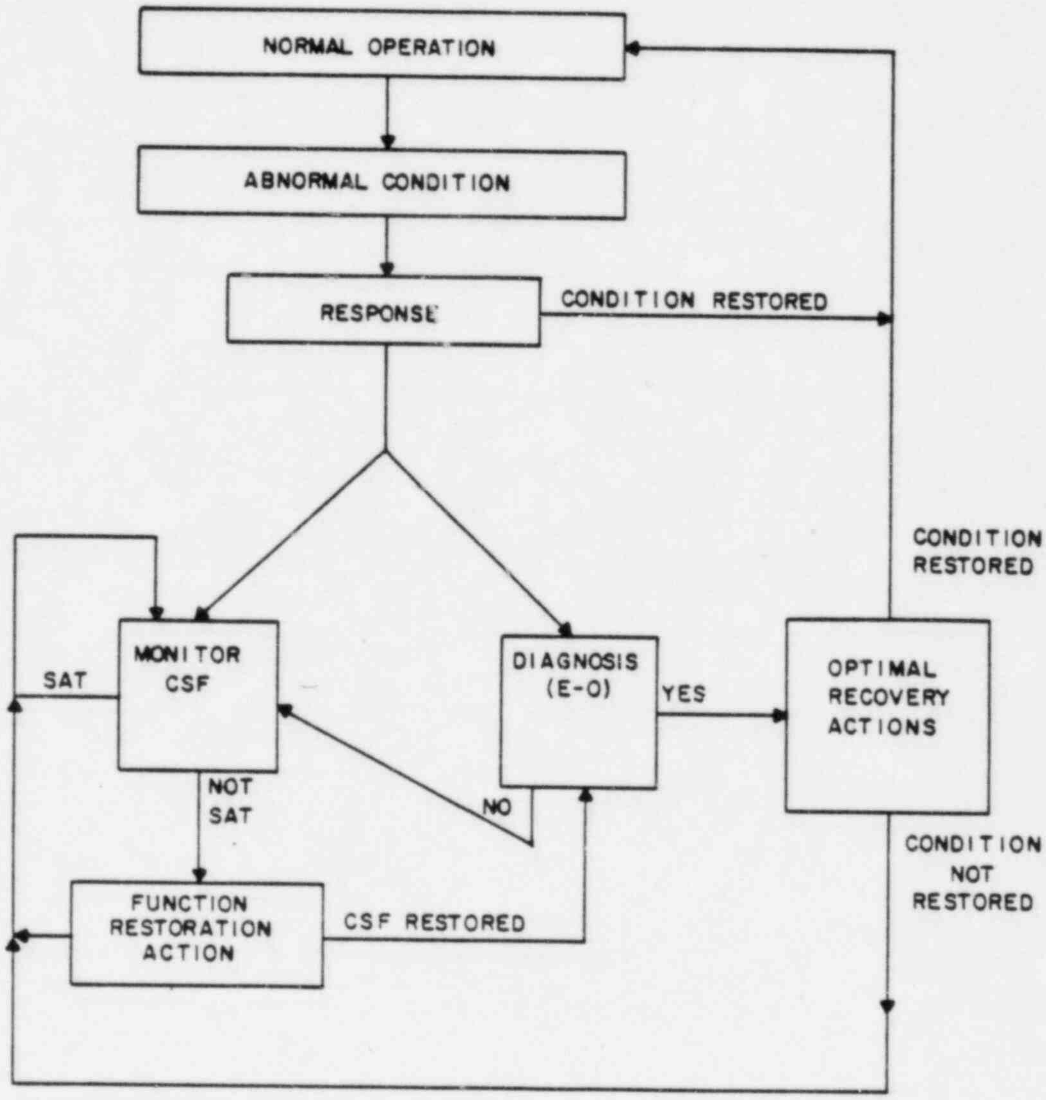
EMERGENCY PROCEDURE SYSTEM

A. EVENT

1. Diagnostic Emergency Procedures
2. Optimal Recovery Guidelines

B. FUNCTION

1. Critical Safety Function Status Trees
2. Functional Recovery Guidelines



COORDINATED USE OF EMERGENCY RESPONSE GUIDELINES

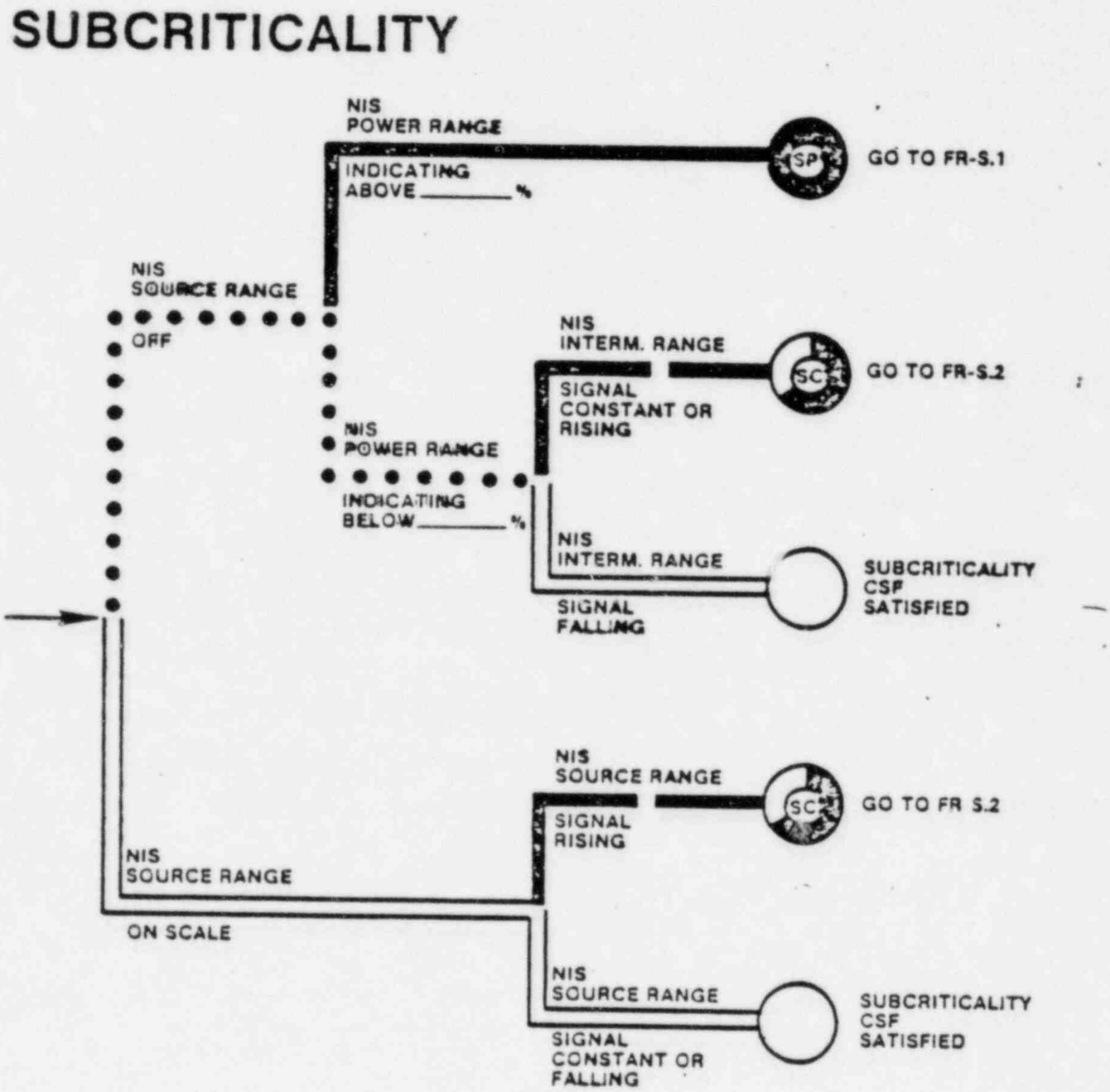
FIGURE 1

FIGURE 2

DEFENSE IN DEPTH PROTECTION

BARRIER	CRITICAL SAFETY FUNCTION
FUEL MATRIX AND CLADDING	MAINTENANCE OF SUBCRITICALLY
	MAINTENANCE OF CORE COOLING
	CONTROL OF REACTOR COOLANT INVENTORY
RCS BOUNDARY	MAINTENANCE OF A HEAT SINK
	MAINTENANCE OF REACTOR COOLANT SYSTEM INTEGRITY
	CONTROL OF REACTOR COOLANT INVENTORY
CONTAINMENT VESSEL	MAINTENANCE OF CONTAINMENT INTEGRITY

Figure 3


SUBCRITICALITY
NIS
POWER RANGE
INDICATING
ABOVE ______ %
GO TO FR-S.1
SP
NIS
SOURCE RANGE
OFF
NIS
INTERM. RANGE
SIGNAL
CONSTANT OR
RISING
SC
GO TO FR-S.2
NIS
POWER RANGE
INDICATING
BELOW ______ %
NIS
INTERM. RANGE
SIGNAL
FALLING
SUBCRITICALITY
CSF
SATISFIED
NIS
SOURCE RANGE
SIGNAL
RISING
SC
GO TO FR S.2
NIS
SOURCE RANGE
ON SCALE
NIS
SOURCE RANGE
SIGNAL
CONSTANT OR
FALLING
SUBCRITICALITY
CSF
SATISFIED

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23	**Check Steam Generator Blowdown Radiation:**	
	a. Radiation – NORMAL	a. IF high, THEN go to step 27.
24	**Check Condenser Air Ejector Radiation:**	
	a. Radiation – NORMAL	a. IF high, THEN go to step 27.
25	**Check If SI Can Be Terminated:**	
	a. RCS pressure – GREATER THAN 2000 PSIG AND INCREASING	a. DO NOT TERMINATE SI. Go to step 27.
	b. Pressurizer level – GREATER THAN (1) %	b. DO NOT TERMINATE SI. Go to step 27.
	c. RCS subcooling – GREATER THAN (2) °F	c. DO NOT TERMINATE SI. Go to step 27.
	d. Secondary heat sink:	
	1) Total AFW flow to non-faulted steam generators – GREATER THAN (3) GPM -OR- 2) Wide range level in at least one non-faulted steam generator – – GREATER THAN (4) %	d. IF neither condition is satisfied, THEN DO NOT TERMINATE SI. Go to step 27.
26	**Terminate SI:**	
	a. Go to ES-0.3, SI TERMINATION FOLLOWING SPURIOUS SI	

(1) Enter plant specific no-load value.

(2) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

(3) Enter plant specific value derived from background document.

(4) Enter plant specific value which is above top of steam generator U-tubes.

RADIATION PROTECTION PROGRAM

FOR

WOLF CREEK GENERATING STATION,
UNIT NO. 1

RAYMOND F. LEWIS, SUPERVISOR RAD/ENV ASSESSMENT

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MAJOR DESIGN AND PROGRAMMATIC EFFORTS IMPLEMENTED

TO MAINTAIN OCCUPATIONAL RADIATION EXPOSURES ALARA

I. USE OF THE BECHTEL MODEL

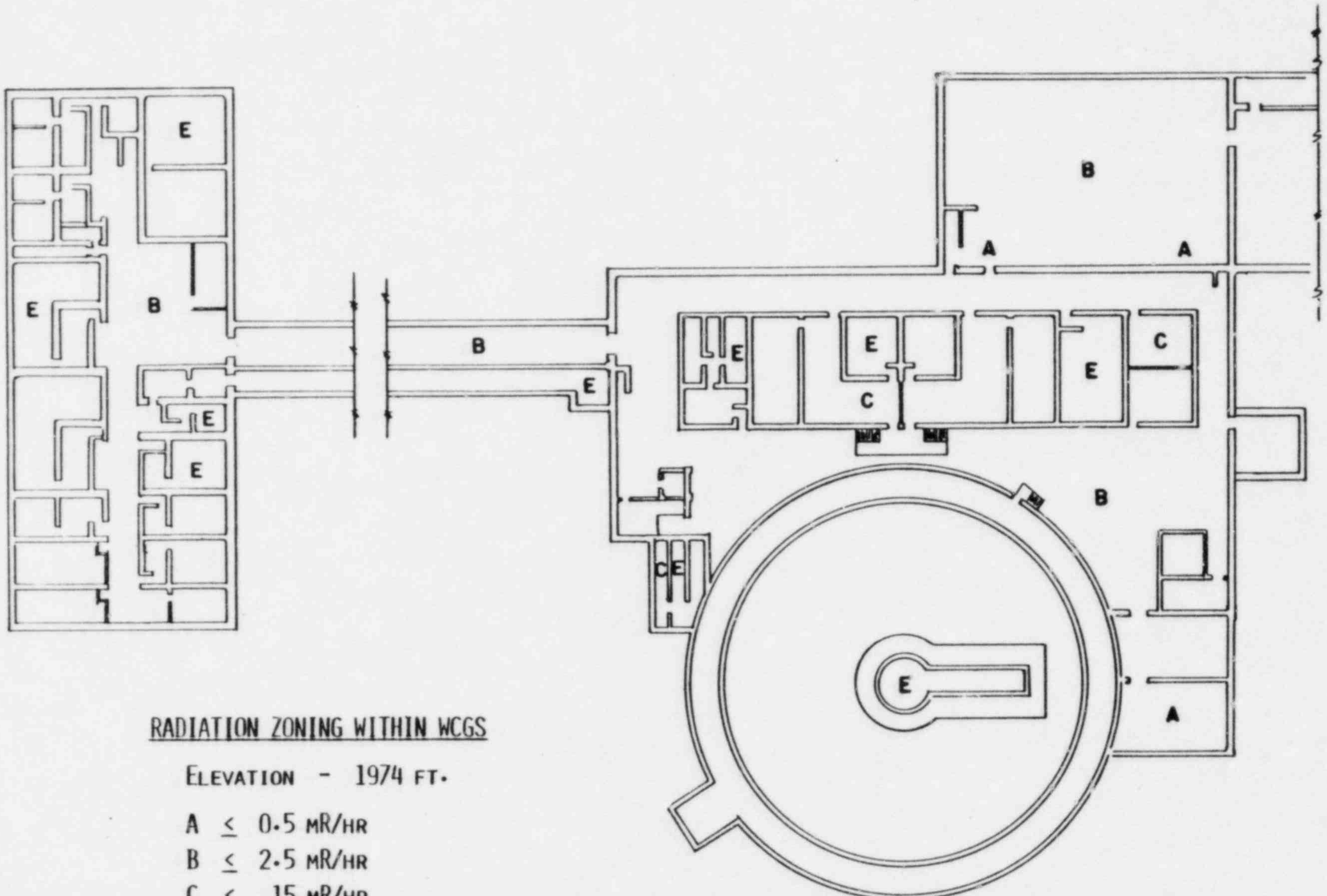
II. RADIATION ZONING WITHIN THE PLANT

III. DESIGNED LAYOUT OF THE PLANT

IV. SYSTEM AND EQUIPMENT DESIGNS

V. CHEMISTRY CONTROL

VI. RADIATION PROTECTION - ALARA PROGRAM



RADIATION ZONING WITHIN WCGS

ELEVATION - 1974 FT.

A ≤ 0.5 mR/HR
B ≤ 2.5 mR/HR
C ≤ 15 mR/HR
D ≤ 100 mR/HR
E > 100 mR/HR

EXAMPLES OF PLANT FACILITIES DESIGNED FOR ALARA

REMOTE AND SHIELDED VALVE GALLERIES

PIPING	- STRATEGICALLY BURIED OR REROUTED
	- BUTT VERSUS SOCKET WELDS
	- LOW POINT DRAINS
	- LARGE RADIUS BENDS
	- ANTI-STREAMING PENETRATIONS
CONTAMINATION CONTROL	- TRAFFIC PATTERN STUDY
	- EQUIPMENT AND TANK VENTS TO RADWASTE
	- VENTILATION (LOW TO HIGH)
ROOM ISOLATION	- PHYSICAL BOUNDARIES
	- LABYRINTH SHIELDING

EXAMPLES OF EQUIPMENT DESIGNS FOR ALARA

SEMI-REMOTE AND SHIELDED FILTER HANDLING SYSTEM

REMOTE DEMINERALIZER RESIN HANDLING SYSTEM

PUMPS
- Mechanical Seals
- Accessible
- Modular Design

LIGHTING
- Long Life Bulbs
- Redundant Fixture Concept

EVAPORATORS
- Separation of Most Radioactive Parts
- Remote Instruments and Controls

EMERGENCY PLANNING

FOR

WOLF CREEK GENERATING STATION,
UNIT NO. 1

RAYMOND F. LEWIS, SUPERVISOR RAD/ENV ASSESSMENT

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CHRONOLOGY & PRESENT STATUS OF THE WCGS

RADIOLOGICAL EMERGENCY RESPONSE PLAN

EARLY YEARS - 1974 INITIAL LETTERS OF AGREEMENT
1977 INITIAL DEVELOPMENT OF EMERGENCY PLAN
1978 REVISION 1 OF PLAN COMPLETE

1978 KANSAS IS 1 OF 14 STATES WITH AN EMERGENCY PLAN UNDER PL 93-288 THAT HAS BEEN APPROVED BY NRC

WINTER 1980
- NUREG 0654 DRAFT
- KG&E BEGINS REVISION 2 OF THE WCGS PLAN

FALL 1980
- INDUSTRY COMMENTS ON NUREG 0654
- NUREG 0654 FINALIZED

SPRING 1981
- REVIEW OF WCGS REV. 2 WITH COUNTY & STATE AGENCIES
- FSAR/EMERGENCY PLAN SUBMITTED TO NRC

SUMMER 1981
- TECHNICAL ASSISTANCE GIVEN TO COUNTY & STATE
- COUNTY PROCURES CONSULTANT AND BEGINS PLAN DEVELOPMENT

FALL 1981
- PREPARATION AND SUBMITTAL OF WCGS SECTIONS TO STATE
- COUNTY COMMISSSION ACCEPTS COUNTY PLAN
- SCOPING OF WCGS EPIPS

WINTER 1981
- WCGS EPIP DRAFTING BEGINS
- KG&E SITE STAFF ASSIGNED TO TECHNICALLY ASSIST AND COORDINATE COUNTY EMERGENCY PLANNING ACTIVITIES

SPRING 1982
- REVISION 3 TO WCGS EMERGENCY PLAN INCORPORATING NRC COMMENTS
- FIRST ROUGH DRAFT OF EPIPS
- TECHNICAL SUPPORT TO STATE PLAN

PRESENT STATUS

COUNTY AND STATE EMERGENCY PLANS

COFFEY COUNTY PLAN

- NOVEMBER 1981 FORMAL ACCEPTANCE BY COUNTY COMMISSION
- DESIGNATION OF FIRST TIER KEY PERSONNEL NEAR COMPLETE
- DEVELOPMENT OF DETAILED SOP'S UNDERWAY - SUMMER COMPLETION ANTICIPATED
- FEMA SUBMITTAL SCHEDULED FOR JUNE

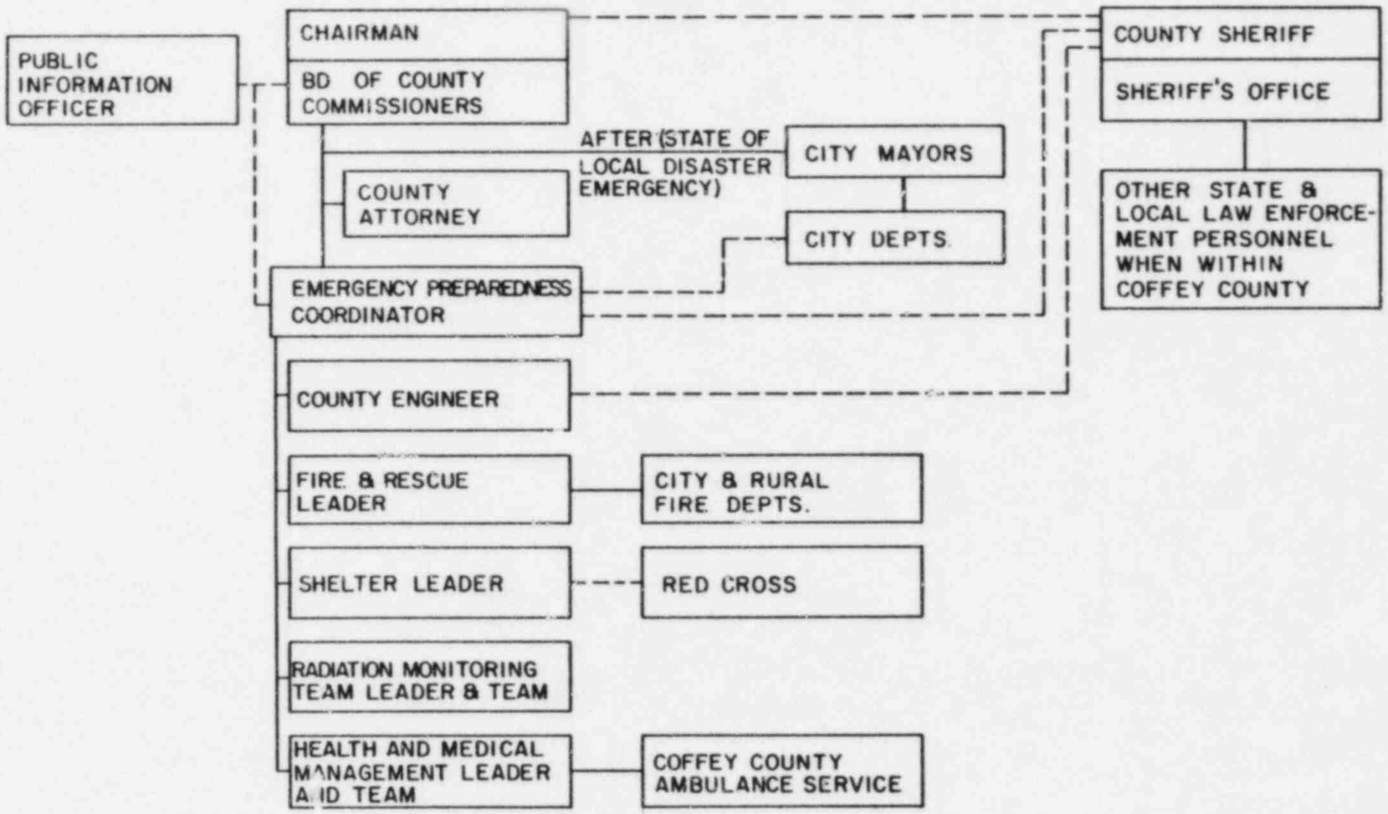
STATE PLAN

- KG&E SECTIONS COMPLETE AND RECEIVED BY STATE
- KG&E TECHNICAL ASSISTANCE AND REVIEW UNDERWAY
- FEMA CONCURRANCE WITH NPPD COOPER STATION FUNDAMENTALS GIVEN
- FEMA INFORMAL REVIEW OF WCGS SECTIONS, SUBMITTED WITH COOPER SECTIONS GOOD
- FEMA SUBMITTAL SCHEDULED FOR JUNE

Future Milestones for WCGS Emergency Planning Activities

Spring 1982	- Assist with County SOP Development - Technically support the State Plan - Scope Emergency Training Activities
Summer 1982	- Finalize WCGS EPIPS - Complete County SOP assistance and develop ASLB Testimony - Complete Phase II Alert and Notification studies - Develop Emergency Training outlines
Fall 1982	- FEMA Interactions - EOF and TSC construction complete - Finalize ASLB Testimony - Begin installation of Alert and Notification System
Winter 1982	- ASLB Hearings - Procure equipment for EOF & TSC - Phase I Emergency Training - Fundementals - Emergency Exercise scenario development
Spring 1983	- Finalize State SOP's - Revise and update WCGS, County and State Plans - Complete Alert and Notification System
Summer 1983	- Phase II Training - Role Enactment - Final equipment checks - NRC appraisal
Fall 1983	- Exercise and critique
Winter 1983	- Fuel load

COFFEY COUNTY EMERGENCY RESPONSE ORGANIZATION



STATE OF KANSAS

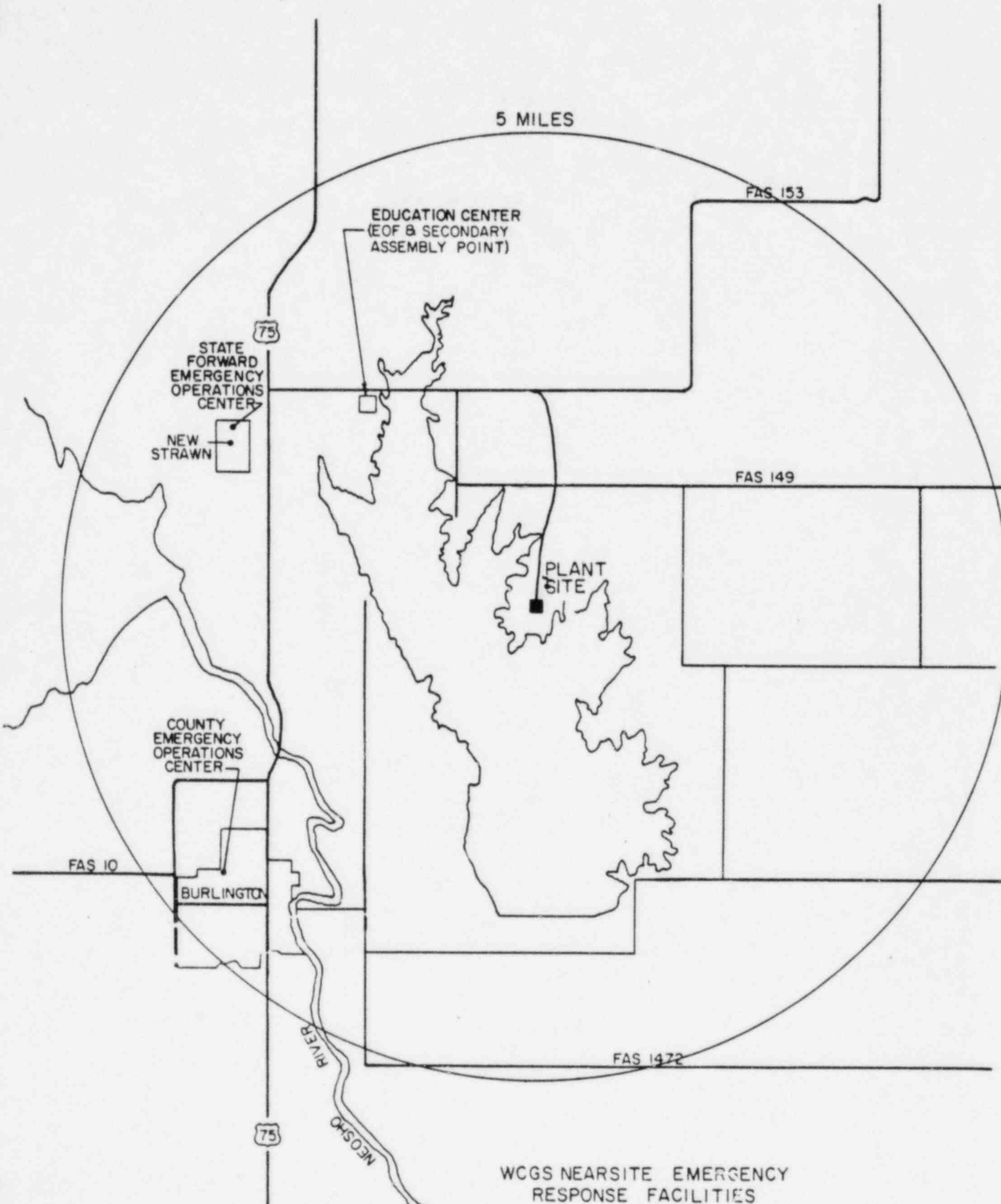
EMERGENCY RESPONSE ORGANIZATION

GOVERNOR

- DEPT. OF HEALTH & ENVIRONMENT
  - DIVISION OF ENVIRONMENT
    - BUREAU OF RADIATION CONTROLS
- DEPT. OF ADJUTANT GENERAL ← (1) COFFEY COUNTY COMMISSION
  - DIVISION OF EMERGENCY PREPAREDNESS
    - RADIOLOGICAL ASSISTANCE MANAGEMENT
      - STATE AGENCIES & RESOURCES

COFFEY COUNTY COMMISSION
- COUNTY EMERGENCY PREPAREDNESS COORDINATOR
  - COUNTY AGENCIES & RESOURCES

(1) FOLLOWING DECLARATION BY THE GOVERNOR OF A "STATE OF DISASTER" EMERGENCY, OTHERWISE COUNTY ACTS INDEPENDENTLY



WCGS NEARSITE EMERGENCY
RESPONSE FACILITIES
&
PRIMARY EVACUATION ROUTES

FIGURE 4.1-1

REV. 2
4/81

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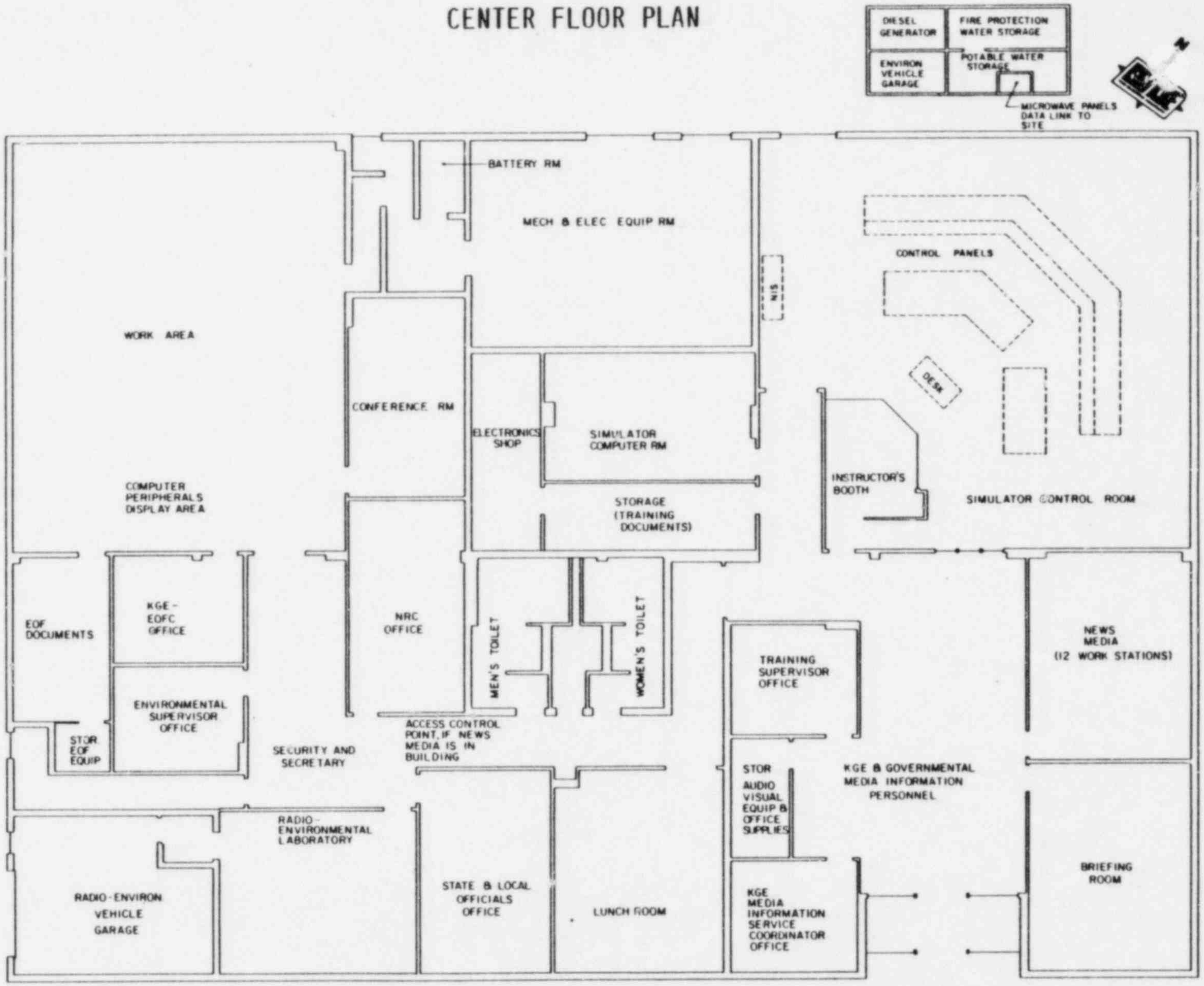
WCGS EDUCATION CENTER FLOOR PLAN



COUNTY DEMOGRAPHY

PRESENT AND PROJECTED MAXIMUM

POPULATION DISTRIBUTIONS

RADIAL DISTANCE (MILES FROM PLANT)	1980 POPULATION VALUES [A]	2000 POPULATION PROJECTIONS
0-2	33	60
0-5	3924	4810
0-10	6658	6120

POPULATION DENSITIES

RADIAL DISTANCE (MILES FROM PLANT)	1980 POPULATION VALUES [A]	2000 POPULATION PROJECTIONS
0-2	2.6	4.8
0-5	50.0	61.2
0-10	21.2	19.5

A 1980 CENSUS ESTIMATES, MODIFIED BY HOUSE COUNT DATA AND CONSERVATIVE VALUES FOR BURLINGTON AND NEW STRAWN

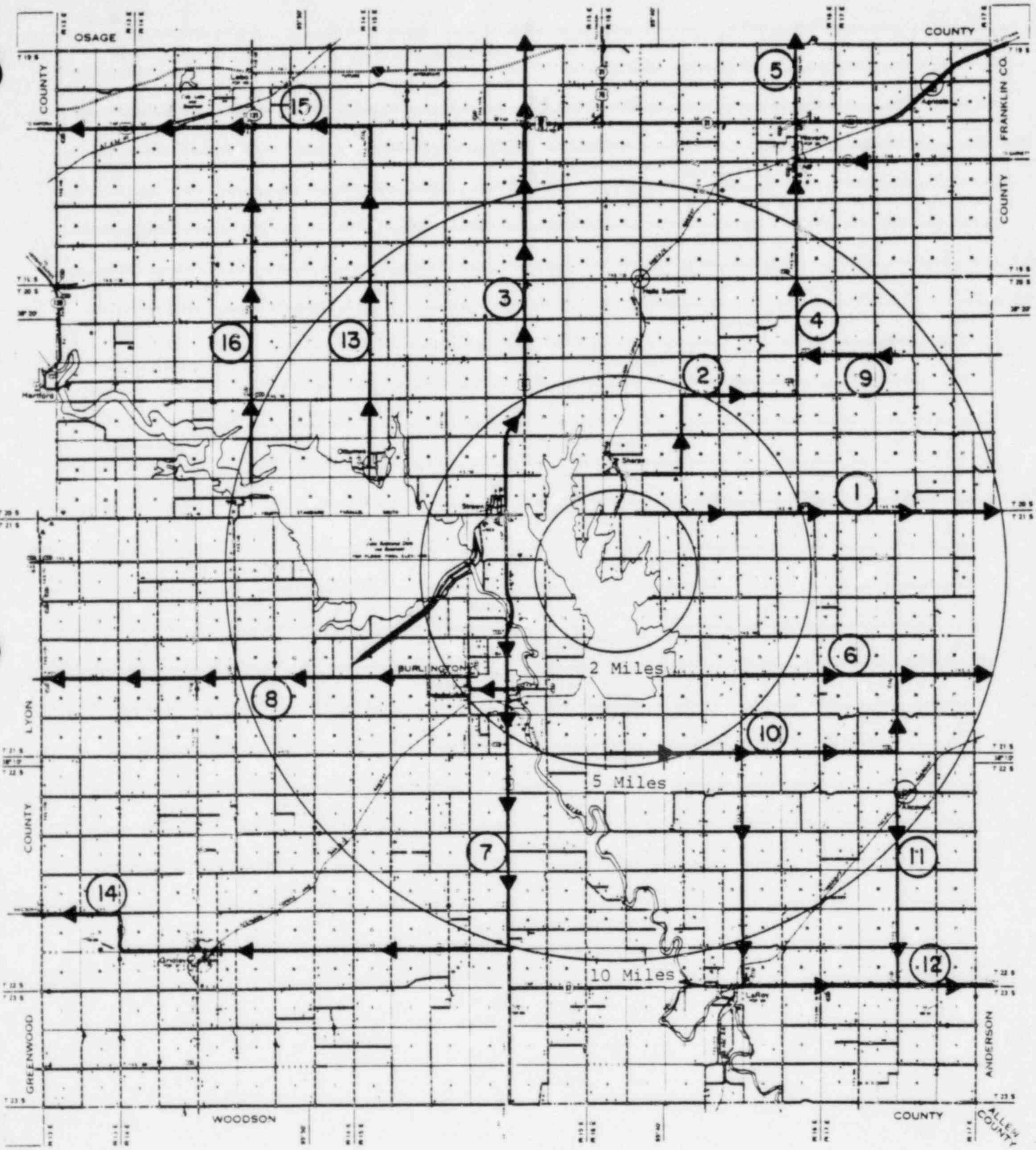


FIGURE DD-3

EVACUATION ROUTES

REV. 2
4/81

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PRESENTATION TO ACRS FULL COMMITTEE

FOR WOLF CREEK,

UNIT 1

MAY 6, 1982

PREPARED BY
JON B. HOPKINS
LICENSING PROJECT MANAGER

WOLF CREEK REVIEW BASIS

- WOLF CREEK UNIT 1 DESIGN (SNUPPS PORTION) IS A DUPLICATE OF CALLAWAY UNIT 1
- ADDITIONAL INFORMATION HAS BEEN PROVIDED, SINCE CALLAWAY SER ISSUANCE (10/81)
- UNIQUE PLANT DESIGN FEATURES OUTSIDE SCOPE OF CALLAWAY (SNUPPS) DUPLICATE DESIGN

PORTIONS OF WOLF CREEK OUTSIDE THE
SCOPE OF THE DUPLICATE PLANT DESIGN

. CIRCULATING WATER SYSTEM SCREEN HOUSE

. ULTIMATE HEAT SINK DAM AND COOLING LAKE

. ESSENTIAL SERVICE WATER SYSTEM PUMPHOUSE

. ADMINISTRATIVE AND SUPPORT BUILDINGS

. TECHNICAL SUPPORT CENTER

. EMERGENCY OPERATIONS FACILITY

. SWITCHYARD AND OFFSITE POWER SOURCES

. STORAGE TANKS

. SECURITY FACILITIES

OPEN ITEMS FROM SER

ITEM		NEXT ACTION	EXPECTED CLOSE
PART A - WOLF CREEK SITE-SPECIFIC ITEMS			
1.	SEISMIC & DYNAMIC QUALIFICATIONS OF MECHANICAL & ELECTRICAL EQUIPMENT*	NRC	EARLY 1983
2.	ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT*	KG&E	LATE 1982
3.	TMI ACTION PLAN ITEMS		
1.A.1.1	SHIFT TECHNICAL ADVISOR (I.A.1.3 & III.A.1.2)	NRC	MID 1982
I.D.1	CONTROL ROOM DESIGN REVIEW*	BOTH KG&E & NRC	EARLY 1983
III.A.1.2	UPGRADE EMERGENCY SUPPORT FACILITIES	NRC	EARLY 1983

*ALSO, INCLUDES DUPLICATE PLANT DESIGN FEATURES.

OPEN ITEMS FROM SER

ITEM			NEXT ACTION	EXPECTED CLOSE
PART B - DUPLICATE PLANT ITEMS				
1.	HIGH-ENERGY PIPE BREAK HAZARDS ANALYSIS		KG&E	MID 1982
2.	PUMP & VALVE OPERABILITY ASSURANCE PROGRAM		NRC	EARLY 1983
3.	FIRE PROTECTION PROGRAM-ALTERNATE SHUTDOWN PANEL		NRC	LATE 1982
4.	TMI ACTION PLAN ITEMS			
	I.C.1	GUIDANCE FOR EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS & ACCIDENTS	KG&E	MID 1982
	I.C.8	PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR TERM OL APPLICANTS	KG&E	LATE 1982
	II.B.2	PLANT SHIELDING FOR ACCESS TO VITAL AREAS AND TO PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION	KG&E	MID 1982

CONFIRMATORY ITEMS FROM SER

STATUS

PART A - 7 ITEMS + 2 TMI-RELATED ITEMS

PART B - 27 ITEMS + 8 TMI-RELATED ITEMS

- o TOTAL OF 34 ITEMS + 10 TMI-RELATED ITEMS
- o MOST ITEMS WILL BE CONFIRMED IN 1982-1983

LICENSE CONDITIONS FROM SER

STATUS

PART A - 1 LICENSE CONDITION

PART B - 17 LICENSE CONDITIONS

- o TOTAL OF 18 LICENSE CONDITIONS
- o EXPECT MOST TO BE IMPLEMENTED PRIOR TO LICENSING AND THEREFORE WILL NOT BECOME LICENSE CONDITIONS

APPENDIX XI
COLLEGE LEVEL TRAINING PROGRAM FOR WOLF CREEK

THE ENGINEERING TECHNOLOGY COLLEGE LEVEL TRAINING PROGRAM FOR THE WOLF CREEK GENERATING STATION OPERATORS

presented to

THE TENTH BIENNIAL TOPICAL CONFERENCE
ON REACTOR OPERATING EXPERIENCE
August 16-19, 1981
Cleveland, Ohio

Sponsored by American Nuclear Society
Reactor Operations Division

Authors

Richard C. Coulthard, Manager Nuclear Training
Kansas Gas and Electric Company

J. Lance Kramer, Ph.D., Assistant Vice President
Kansas State University

David F. Cropp, Ed.D., Associate Dean
Emporia State University

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THE ENGINEERING TECHNOLOGY COLLEGE LEVEL TRAINING PROGRAM FOR WOLF CREEK GENERATING STATION OPERATORS

The Wolf Creek Generating Station (1150 Mwe PWR) is scheduled for initial fuel load in 1983 and is faced with the challenge to provide its operators with college level training in nuclear sciences. This paper will describe the requirements and how Kansas State University, Emporia State University, and Kansas Gas and Electric Company have developed an Accreditation Board for Engineering and Technology (ABET) approved program to meet these requirements.

Although the Three Mile Island accident occurred over two years ago, a precise definition of required college level training requirements is still not available. The ANSI 3.1 draft of April 10, 1981 specifies 30 credit hours for the SRO candidate and 60 hours for the shift supervisor. A draft of Regulatory Guide 1.8 of September 1980 raised three requirements to 60 hours and a Bachelor of Science degree respectively, while defining college level education as being " . . . conducted by a college or university with curricula accredited by a nationally recognized agency such as the Accreditation Board for Engineering and Technology (ABET/ECPD)." It thus appears that until the USNRC develops guidance for accreditation of training institutions that accreditation by regional organizations such as the North Central Association may not be sufficient. In November

1980, SECY-490 which contained the draft of proposed changes to 10CFR55 was issued and made all the course topics listed after the words "such as" mandatory by using the word "including" in front of the requirement for courses in reactor physics, reactor thermodynamics, heat transfer, and reactor control theory. Courses such as these are generally taught during the senior year in a Nuclear Engineering curriculum.

After the Three Mile Island incident, the position of Shift Technical Advisor (STA) was created to provide an academically trained person on shift to assist in analyzing plant transients. The Institute of Nuclear Power Operations (INPO) issued a standard on April 30, 1980 which outlined the qualifications, including academic training requirements, for the STA position. This was subsequently endorsed by the NRC as an acceptable method of qualifying STAs. These academic requirements for the STA do not require a Bachelor of Science degree and actually do not significantly exceed the course requirements for the SRO. If the STA is viewed as an interim position until the academic portion of operator training is upgraded, then a program that meets both SRO and STA requirements can eliminate the need for an STA, separate from the two SROs on shift. It should be noted that there is a body of opinion within the NRC staff that would like to retain the STA position "independent from the pressures of operations" and thus institutionalize what was originally an interim position.

Regulatory siting criteria had not previously required that nuclear power plants be sited close to one of the about 45 schools of nuclear engineering so the plant's operators could participate in nuclear engineering courses part time. The closest school with a nuclear engineering program is Kansas State University, which is over a two hour drive (115 miles) away. Given the problems of distance and the rigors of the proposed curriculum, Kansas State and KG&E entered into discussions.

The statement, "you don't have to be an aeronautical engineer to be an airline pilot, so why does a reactor operator have to be a nuclear engineer?" has been made many times in the last year. Fortunately, at Kansas State an alternative to the more theoretically rigorous nuclear engineering program exists. It is the ABET-accredited curriculum leading to a Bachelor of Science in Engineering Technology degree. This program consists of 65 hours of core courses in liberal arts and basic engineering technology sciences plus a 55 hour option program in areas such as environmental engineering, computers, and production management. The Engineering Technology program is designed for people seeking to engage in routine design development and liaison and supervision of crafts. The emphasis of the engineering technology program is less theoretical than engineering, with emphasis on hardware and applications.

The next problem was how to overcome the distance between Wolf Creek and Kansas State in order to deliver the courses. Fortunately, located 40 miles from the Wolf Creek site is Emporia State University (enrollment 5600) which has for years been teaching basic science fundamentals for students who transfer to Kansas State to continue an engineering education. Emporia State did not have the basic program for Engineering Technology, so it had to take on the assignment of teaching two new courses in Technical Calculus as well as developing a new follow-on course in Applied Differential Equations. This course includes an introduction to LaPlace Transforms. Kansas State is developing four new courses to support this program and its new Nuclear Reactor Technology option program. The first table summarizes the operator training courses and the remaining requirements for a Bachelor of Science in Engineering Technology degree.

Emporia State has added two faculty members to conduct these courses at Wolf Creek and Kansas State will be sending several faculty members with expertise in the various courses. This program is being taught to about twenty SRO candidates in seven eight-week sessions between June 1981 and the end of 1982. The curriculum has six laboratory courses, five of which will be conducted by transporting students to the Emporia State campus. The radiation detection lab will be conducted at Wolf Creek.

There will be a continuing requirement for this program, since all Senior Reactor Operator applicants after 1984 will be required to have this college level training. It is not economical or practical to bring in faculty members to teach the three or four SRO candidates a single unit station might have in a year. All of the original courses will be videotaped and additional funds have been allocated to develop these courses for remote teaching. This will be achieved by editing videotapes and developing course notes and problem books to supplement the videotape lectures and textbooks. Thus, the cost of instruction for each small group of students will be significantly decreased due to the use of self-based instructional materials. Specified on-site instructor time plus telephone counseling will increase the cost effectiveness of course delivery without reducing the quality or effectiveness of instruction. The goal of this program is to significantly reduce instructor time so that a 45 contact hour course (3 credit hours) may only require about 12 hours of instructor time with no more than half of this at the Wolf Creek site.

The first session of College Algebra, Trigonometry and Chemistry I was completed on July 24. About 70% of the grades awarded were either A or B and only one D was earned. The nine credit hour in an eight week schedule has placed a significant demand on the time of the operators outside the normal eight hour day for homework problems and laboratory reports. The two members of the Emporia State faculty involved in teaching the first session remarked that the reason for the above average grades was that the operators were more mature and motivated than the average campus college student.

The establishment of this program has required much cooperation between the utility and the universities. Utility concerns of minimizing time requirements and suitability of the topics for the operators have had to be offset by the universities' concern for the academic quality of courses and problems of remote teaching. Continued cooperation will be required not only to conduct this program, but to employ some of the basic courses in the training of chemistry, health physics, and instrumentation technicians. A method to allow the completion of all the degree requirements will also have to be established.

Conclusions:

1. It is noticeably easier for a utility two or more years from an operating license to provide college level training for its operators than one in the final stages of startup or one that has a plant operating. The real challenge will be to develop a program that will allow new Senior Reactor Operator candidates to obtain this college level training in addition to normal training requirements and the need to keep the plant manned with two licensed reactor operators and four equipment operators. Two of the attachments show how college level training might be incorporated into a program for operator development from entry level to the senior reactor operator position. If this progression is over a four to five year period, it will be impossible to meet all the college requirements on a part-time basis. Some of the college requirements will have to be met by full-time devotion to the college program. The NRC has

recently realized the importance of "grandfathering" existing SROs when this program is phased in, so the major challenge for utilities that have had nuclear plants operating for some time is to develop a program that will provide a source of replacement SRO candidates.

2. The discussion of whether to require 30, 45 or 60 credit hours of college level training is somewhat academic as long as the curriculum must include courses in reactor physics, reactor thermodynamics and reactor control theory. Any course of study is probably going to require about 50 credit hours to progress from college algebra to this level of course. Most rigorous nuclear engineering curriculums would probably require over 60 credit hours to progress to this point.

3. The subject of accreditation should be addressed as soon as possible by the NRC rather than deferring it until 1983. The question of accreditation goes beyond the question of who besides the ABET/ECPD can provide satisfactory approval of a course of study. The other questions include maximum number of classroom contact hours per week (generally 15 to 20 hours for a full-time student) and the necessity of formally administered laboratory experiments. In order for utilities to establish college level training programs, they need to know the accreditation requirements as well as the course content requirements.

4. If the resources are available, there is a definite advantage to using nearby colleges and universities to conduct this training. It results in a pool of manpower being available, without travel problems or demands of other utility clients, to reinstate programs, provide tutoring, assist in conduct of laboratory experiments, and accelerate programs due to rapid turnover. It is the charter of state-supported educational institutions to meet the educational needs of its citizens and industries and they should be more responsive.

REQUIREMENTS

60 semester hours in technical subjects <u>including</u>:

Mathematics

Reactor Physics

Chemistry

Materials

Reactor Thermodynamics

Fluid Mechanics

Heat Transfer

Electrical Theory

Reactor Control Theory

Accredited by A.B.E.T. or other nationally recognized agency

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THE ENGINEERING TECHNOLOGY SEQUENCE
NUCLEAR REACTOR TECHNOLOGY CURRICULUM

	Credit Hours	Univ.	Notes
1. Core Courses Taught In Operator Program			
College Algebra	3	ESU	
Plane Trigonometry	2	ESU	
Technical Calculus I	3	ESU	
Technical Calculus II	3	ESU	
Descriptive Statistics	3	ESU	
Chemistry I	4	ESU	Lab
Physics I and II	8	ESU	Lab
Electrical Circuit Technology I	4	KSU	Lab
TOTAL	30		
2. Option Courses Taught In Operator Program			
Applied Differential Equations	3	ESU	New
Chemistry II	4	ESU	Lab
Properties of Engineering Materials	2	KSU	
Applied Statics	3	ESU	
Energy Conversion Technology	3	KSU	
Mechanics of Fluids	3	KSU	
Materials of Reactor Systems	2	KSU	
Radiation Detection and Monitoring	3	KSU	Lab
Nuclear Reactor Technology I	3	KSU	New
Nuclear Reactor Technology II	3	KSU	New
Nuclear Reactor Thermal Technology	3	KSU	New
Radiation Protection	2	KSU	New
TOTAL	34		
3. Remaining Courses in Core Program			
English and Communications	11	ESU	
Graphics and Computer Courses	4	ESU or KSU	
Free Elective and 1 hour Physical Ed.	5	ESU	
Economics	3	ESU	
Humanities/Social Science Electives	12	ESU	
TOTAL	35		
4. Remaining Courses In Nuclear Reactor Technology Option			
	21	KSU	
GRAND TOTAL	120		

Note: ESU is Emporia State University
KSU is Kansas State University

NUCLEAR ENGINEERING PROGRAM		ENGINEERING TECHNOLOGY PROGRAM	
Basic Program		**Basic Program**	
Calculus & Differential Eq.	16	College Algebra & Trig.	5
Numerical Analysis	3	Technical Calculus	6
Chemistry I & II	8	Chemistry I	4
Physics I & II	8	Physics I & II	8
Basic Engineering Courses (Non-Nuclear)	17	Basic Engineering Courses	15
		Statistics	3
TOTAL	52 hours		41 hours
Nuclear Engineering Courses		**Nuclear Technology Courses**	
Intro. Nuclear Courses	6	Chemistry II	4
Radiation Detection & Lab	4	Applied Diff. Equations	3
Nuclear Engineering Materials	2	Nuclear Engineering Materials	2
Radiation Protection Engr.	2	Radiation Protection Engr.	2
Specific Other Nuclear Engr. Courses	21	Nuclear Reactor Tech I & II	6
		Nuclear Reactor Thermal Tech	3
		Radiation Detection Lab	3
TOTAL	35 hours		23 hours
Remaining Courses		**Remaining Courses**	
Technical Electives	20	Technical Courses Remaining	21
English & Communications	9	English & Communications	11
Human & Social Sciences, etc.	12	Human & Social Sciences, etc.	17
Economics	3	Economics	3
		Graphics & Computers	4
TOTAL	44 hours		56 hours

	NRC Prereq.	NRC Require.	INPO-STA Only Prereq.	INPO-STA Require.
College Algebra	3			
College Trigonometry	2			
Technical Calculus I		3		
Technical Calculus II	3			
Descriptive Statistics			3	
Chemistry I		4		
Chemistry II	4			
Differential Equations	3			
Physics I & II	8			
Electrical Circuit Theory		4		
Properties of Engr. Materials		2		
Applied Statics	3			
Energy Conversion Tech.		3		
Mechanics of Fluids		3		
Nuclear Engineering Materials				2
Nuclear Reactor Tech I		3		
Nuclear Reactor Tech II		3		
Nuclear Reactor Thermal Tech		3		
Radiation Detection				3
Radiation Protection				2
TOTAL HOURS	26	28	3	[illegible]

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Type of Training	Training Required to Become				
	Class C Operator	Class B Operator	Class A Operator	Reactor Operator	Senior Reactor Operator
Educational Topics (4)	0	12 days	8 days	20 days	0
Power Plant Fundamentals and Intro. to Nuclear Plant	22..5 days	13 days	14 days	0	0
Plant Systems Classes	20 days	35 days	40 days	20 days	0
In Plant System Checkouts (1)	10 systems	15 systems	15 systems	5 systems	0
Leadership/Supervisory/ Communication	7.5 days	0	0	0	10 days
License Exam Prep. Training (2)	0	0	0	60 days	60 days
Prescribed Watch Standing	0	0	0	65 days	65 days
License Retraining (3)	0	0	0	0	25 days
College Level Program-Part-time	0	9 cr. hours	9 cr. hours	12 cr. hours	12 cr. hours
Full time	0	0	0	0	22 hours (100 days)
Orientation/Plant Administration	5 days	0	3 days	4 days	2 days
DAYS IN CLASS	55 days	60 days	65 days	104 days	197 days
MINIMUM TIME IN PLANT TO COMPLETE TRAINING REQUIREMENTS	2 months	6 months	6 months	5 months	3 months
MINIMUM POWER PLANT EXPERIENCE	None	None	3 months as B	36 months	12 months as RO
NEW HIRE TO POSITION	5-6 months	15-18 months	27-33 months	36-45 months	54-69 months

(1) Accomplished while on shift with concurrent operational duties
(2) Includes simulator training time
(3) This is an annual requirement
(4) These numbers are significantly lower than INPO guidelines, since college program will take care of much of this educational training

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Type of Training	Training Required to Become				
	Class C Operator	Class B Operator	Class A Operator	Reactor Operator	Senior Reactor Operator
Training in Classroom (2)	55 days	60 days	65 days	104 days	197 days
Plant Checkouts (1)	10 systems	15 systems	15 systems	5 systems	0
Prescribed Watch Standing	0	0	0	65 days	65 days
License Retraining	0	0	0	0	25 days
College Level Program-Part-time	0	9 cr. hours	9 cr. hours	12 cr. hours	12 cr. hours
Full-time	0	0	0	0	22 cr. hours
Minimum Power Plant Experience	0	0	3 months as B	36 months	12 months as RO
New Hire to Position	5-6 months	15-18 months	27-33 months	36-45 months	54-69 months

(1) Accomplished on shift with concurrent operational duties
(2) Excludes part-time college

NRC CONSULTANT'S

COMMENTS ON NUREG-0820

Contact:
W. Russell
x-29794

OVERALL ASSESSMENT OF PLANT

R. BUDNITZ - SHOULD INCLUDE USI AND TMI ISSUES. THEIR SAFETY SIGNIFICANCE IS PROBABLY FAR GREATER THAN A MAJORITY OF ISSUES IN NUREG-0820.

- ISSUES INVOLVING MANAGEMENT OF PLANT NEED TO BE ADDRESSED.

S. BUSH - ROLE OF TOP MANAGEMENT IN PLANT OPERATION NOT SUFFICIENTLY STRESSED.

J. HENDRIE - WITH REGARD TO THE FTOL, USI, TMI AND GENERIC ITEMS (ASSOCIATED CIRCUITS AND MAIN FEEDWATER ISOLATION) WILL EITHER HAVE TO BE IN HAND, OR THE COMMISSION WILL HAVE TO EXPLICITLY EXCLUDE THEM FROM SUCH PROCEEDINGS.

H. ISBIN - OVERALL ASSESSMENT OF THE PLANT SHOULD UTILIZE RESULTS OF TMI AND USI TASKS, IE BULLETINS AND GENERIC LETTERS.

- RESOLUTION OF USI'S REMAINS A CONTINUING ACTIVITY ALONG WITH MANDATED ANNUAL REPORTS TO CONGRESS CONCERNING IDENTIFICATION OF ANY NEW ISSUES. OVERALL ASSESSMENT OF PLANT MUST UTILIZE ALL THESE INPUTS AND, THEREFORE, THE "FINAL" SAFETY ASSESSMENT MAY BE ASYMPTOTIC - RECOMMEND DELETION OF FINAL.

Z. ZUDANS - THE FINAL INTEGRATED ASSESSMENT REPORT SHOULD BE A JOINT EFFORT OF SEP, USI, TMI AND OTHERS.

APPLICABILITY OF TOPICS

S. BUSH - SOME OF THE STRONG POSITIONS TAKEN AT THE INCEPTION OF THE PROGRAM (1976-77) HAVE WEAKENED IN THE PAST 4-5 YEARS. SOME OF THE ISSUES, IF WRITTEN IN 1982 WOULD DIFFER SUBSTANTIALLY FROM THE WORDS GENERATED IN 1977.

- PROBABILISTIC APPROACH COULD LEAD TO DROPPING LESS SAFETY SIGNIFICANT TOPICS OR THE "LESSER SAFETY SIGNIFICANCE APPROACH."

J. HENDRIE - ALL OF THE IMPORTANT SAFETY MATTERS ARE BEING COVERED UNDER THE SEP FOR THESE OLDER PLANTS, AND THAT IS WHAT THE COMMISSION WANTED.

- IN SPITE OF THEIR AGE, IT STRIKES ME THAT THE 137 SAFETY TOPICS STILL FORM AN APPROPRIATE LIST OF AREAS FOR REVIEW OF THESE OLDER PLANTS.

H. ISBIN - THE PROCESS OF REDUCING THE TOPIC LIST TO 90 APPLICABLE TOPICS HAS AN ACCEPTABLE RATIONALE PROVIDING ALL CURRENT ITEMS INVOLVING USI, TMI AND OTHER GENERIC MATTERS ARE TO BE INCLUDED.

BACKFIT RATIONALE

R. BUDNITZ - AGREE WITH RATIONALE THAT HARDWARE FIXES SHOULD BE REQUIRED ONLY IF NO OTHER TYPE OF BACKFIT IS AVAILABLE.

- USE OF PRA APPROPRIATE AS BACKUP TO ENGINEERING JUDGMENT.

- INTEGRATED ASSESSMENT APPROPRIATE.

- STAFF'S THOROUGHNESS, ISSUE BY ISSUE, IS COMMENDABLE.

J. HENDRIE - STAFF RECOMMENDATIONS FOR BACKFITTING AND NOT BACKFITTING ARE REASONABLE AND APPROPRIATE AND BASES UPON WHICH RECOMMENDATIONS ARE MADE ARE ADEQUATE.

H. ISBIN - THE OVERALL PLANNING OF THE SEP REVIEW TO ACHIEVE A "BALANCED AND INTEGRATED..." DECISION ON EACH TOPIC APPEARS TO BE GOOD.

USE OF PRA

R. BUDNITZ - THE WAY PRA HAS BEEN USED IS JUST ABOUT RIGHT.

H. HENDRIE - THE RESULTING ASSESSMENTS FROM THE PRA ON SAFETY IMPORTANCE AND OF BENEFIT IN RISK REDUCTION FROM BACKFITTING ARE NECESSARILY ROUGH, BUT ARE STILL USEFUL INPUTS TO BE CONSIDERED IN THE OVERALL ASSESSMENT OF THE TOPIC.

H. ISBIN - PRA APPLICATION, LIMITED, BUT USEFUL. CONSIDERING NRC RESEARCH IN THIS AREA MORE FEEDBACK SHOULD BE GOING INTO THE SEP ACTIVITIES.

Z. ZUDANS - PROVIDED USEFUL INSIGHT IN RELATIVE VALUE OF BACKFITS; I.E., IT PROVIDED LOGICAL SUPPORT FOR ENGINEERING JUDGMENT IN COMPLICATED SITUATIONS.

OPERATING EXPERIENCE

R. BUDNITZ - MANAGEMENT AND ENGINEERING COMPETENCE OF THE PLANT SHOULD BE DEVELOPED FROM OPERATING EXPERIENCE.

S. BUSH - THE EVENTS IN THE OPERATING HISTORY EXTEND OVER A PERIOD OF TIME THAT IS INDICATIVE OF TOP MANAGEMENT TO TAKE APPROPRIATE ACTION. UNLESS THERE IS POSITIVE EVIDENCE OF AN IMPROVEMENT IN OPERATOR ACTIONS I QUESTION APPROVING A FTOL.

- THE OPERATING HISTORY POINTS OUT THE HIGH INCIDENCE OF LOSS OF POWER. THIS COMBINED WITH SOME OF THE OPERATOR ERRORS LISTED COULD YIELD A DEFINITE DEGRADATION IN SAFETY MARGINS.

H. ISBIN - OPERATING HISTORY SHOULD INCLUDE A THOROUGH EVALUATION OF THE LICENSEE'S RESPONSE IN TERMS OF CORPORATE POLICIES, MANAGEMENT AND STAFF CONTROL MEASURES, TRAINING AND REQUALIFICATION PROGRAMS, PROCEDURES AND QUALITY ASSURANCE.

OVERALL ASSESSMENT OF REPORT

R. BUDNITZ - AGREE WITH POLICY TYPE DECISIONS IN REPORT. REPORT REFLECTS A "GENERAL FEELING" THAT PALISADES IS INDEED "ADEQUATELY SAFE."

- FIRST SEP REPORT HAS BEEN QUITE SUCCESSFUL.

H. HENDRIE - BELIEVE THAT NUREG-0820 HAS BEEN A THOROUGH AND CAREFUL JOB THAT FULFILLS THE INTENT OF THE COMMISSION WHEN IT AUTHORIZED PHASE II OF THE SEP IN 1977.

Z. ZUDANS - CONSIDERABLY MORE SOUND ENGINEERING EFFORT HAS BEEN PUT IN PALISADES SEP REVIEW, IN PARTICULAR IN TERMS OF PROPER UNDERSTANDING OF DESIGN, PROCESSES AND CONSEQUENCES INVOLVED, THAN MAYBE NORMALLY DONE DURING REGULAR LICENSING REVIEW PROCESS, (SEP TOPIC LIST COVERS ESSENTIALLY ALL SAFETY RELATED DESIGN ASPECT OF A NUCLEAR POWER PLANT).

- IN GENERAL, NUREG-0820 PROVIDES A COMPREHENSIVE DISCUSSION OF DEFINITION, SAFETY OBJECTIVES AND STATUS OF ALL SEP TOPICS.

REVISION 1 (4-28-82)

PRELIMINARY AGENDA - 265th ACRS MEETING
INTEGRATED PLANT SAFETY ASSESSMENT
SYSTEMATIC EVALUATION PROGRAM = PALISADES PLANT
May 7, 1982

1:30 p.m.	I.	Subcommittee Report - C. Siess
1:40 p.m.	II.	Overview of SEP - W. Russell, Staff
		a. General
		b. Palisades
2:00 p.m.	III.	General Plant Description and History CPCO - R. Vincent
2:10 p.m.	IV.	Met Current Criteria or Were Acceptable on Another Defined Basis W. Russell/Staff
		A. Examples
		. Seismic Review of Palisades Plant
		1. Seismic Hazards Study by L.L.L. Current Status
		2. Seismic Analysis for Staff by L.L.L. & SMA
2:25 p.m.		. Residual Heat Removal System Reliability
		****** BREAK ******
2:40 p.m.		B. Acceptability Based upon Modification
2:50 p.m.	V.	Limited Risk Analysis - A. Thadani/M. Rubin
3:00 p.m.	VI.	Integrated Assessment Topics - T. Michaels
4:00 p.m.	VII.	Consultants review of Palisades SEP - W. Russell/Staff (Hendrie, Bush, Zudans, Isbin, Budnitz)
4:20 p.m.		I&E Report on Utility Management Experience - James Keppler, - I&E Region III
4:45 p.m.	VIII.	Licensee Comments - CPCO, R. Vincent
5:00 p.m.		ADJOURN

SEP PHASE II REVIEW

PALISADES

- INTRODUCTION AND PROPOSED SCHEDULE
- SEP OVERVIEW
  - PHASE I
  - PHASE II
  - PHASE III/NREP
- TENTATIVE ACRS REVIEW SCHEDULE FOR PHASE II PLANTS
- LICENSE CONVERSION (POL TO FTOL)
- PALISADES IPSAR OVERVIEW
- INTEGRATED ASSESSMENT PROCESS

CONTACT: WILLIAM T. RUSSELL
X29794

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SEP OVERVIEW

PURPOSE (SECY-76-545)

- ASSESS SAFETY OF DESIGN AND OPERATION
- DOCUMENT COMPARISON WITH CURRENT CRITERIA
- CAPABILITY TO MAKE BALANCED AND INTEGRATED BACKFITTING DECISIONS
- EARLY IDENTIFICATION OF SIGNIFICANT ISSUES
- EFFICIENTLY USE RESOURCES

PHASE I

- IDENTIFICATION OF 137 TOPICS

PHASE II

- ORIGINALLY 11 PLANTS (NOW 10) Dresden 1 indefinitely shutdown
- POL TO FTOL CONVERSION (7 PLANTS)

PHASE III

- CONCEPTUAL APPROVAL BY COMMISSION
- IMPLEMENTATION PLAN - JUNE 30, 1982
  - IDENTIFY REDUCED SET OF TOPICS
  - IDENTIFY PLANTS FOR FY 83/84 REVIEW

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NRR OPERATING PLAN

SEP PHASE II SCHEDULE

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

LICENSE CONVERSION

POL TO FTOL (SECY-77-539)

PLANTS	POL
SAN ONOFRE 1	MARCH 27, 1967
LACROSSE*	JULY 3, 1967
OYSTER CREEK	APRIL 9, 1969
GINNA*	SEPTEMBER 19, 1969
DRESDEN 2	DECEMBER 22, 1969
MILLSTONE 1	OCTOBER 7, 1970
PALISADES	MARCH 24, 1971

* HEARING

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, 23 were deleted because the topics did not apply to Palisades (see Appendix C).

FINAL NUMBER OF TOPICS REVIEWED FOR PALISADES
90 (see Section 3.1 and Appendix E)

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS
57 (see Section 3.2)

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS AFTER MODIFICATIONS MADE DURING TOPIC REVIEW
2 (see Sections 3.2, 3.3.3, and 3.3.4)

TOPICS CONSIDERED FOR BACKFIT IN THE INTEGRATED ASSESSMENT
31 (see Table 4.1 and Sections 4.1-4.31)

INTEGRATED ASSESSMENT

FACTORS CONSIDERED

- SAFETY SIGNIFICANCE
- ALTERNATIVES (SECY-77-561)
  - DIFFERENCE CAN BE JUSTIFIED AS NOT SIGNIFICANT
  - USE OF NON-SAFETY SYSTEMS
  - ADMINISTRATIVE OR PROCEDURAL CHANGES
  - AUGMENTED SURVEILLANCE PROGRAMS
  - SELECTED BACKFITTING
- USED LIMITED PRA
  - SYSTEM IMPORTANCE
  - RELATIVE IMPROVEMENT TO BE GAINED
- COORDINATION WITH OTHER NRC REQUIREMENTS
  - TMI
  - USI
- LICENSEE INPUT TO "OPTIMIZE" AND IDENTIFY "COMMON FIXES"
  - MEETINGS
  - SITE VISITS
  - FORMAL SUBMITTALS OF PROPOSED ACTIONS

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NRC RESOURCES

COSTS OF SEP FOR PALISADES

- STAFF EFFORT 10.7 PSY
- CONTRACTOR EFFORT $710K (APPROX.)

NRR EFFORT ON PHASE II

	PRIOR	FY81	FY82	TOTAL
STAFF (PSY)	49	21	24	94
$ (1,000)	3382	2424	1,318	$7,124

PSY = 1800 HOURS

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CONSUMERS POWER COMPANY
PALISADES PLANT

NSSS: COMBUSTION ENGINEERING PWR

ARCHITECT-ENGIN CONSTRUCTOR: BECHTEL SAN FRANCISCO

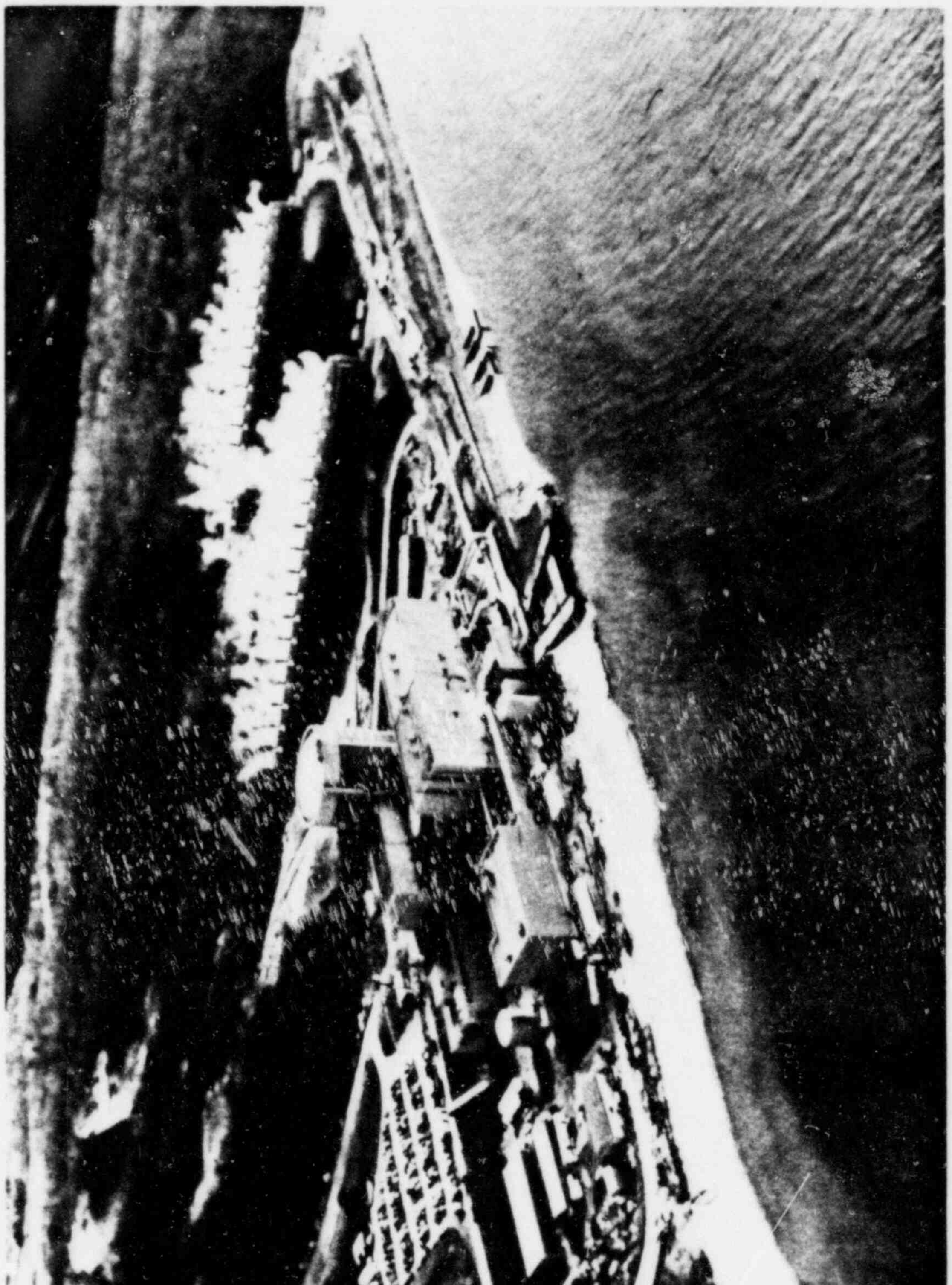
TURBINE-GENERATOR: WESTINGHOUSE

COOLING MODE: ONCE-THROUGH LAKE MICHIGAN UNTIL 1974-75 CONSTRUCTION OF MECHANICAL DRAFT COOLING TOWERS

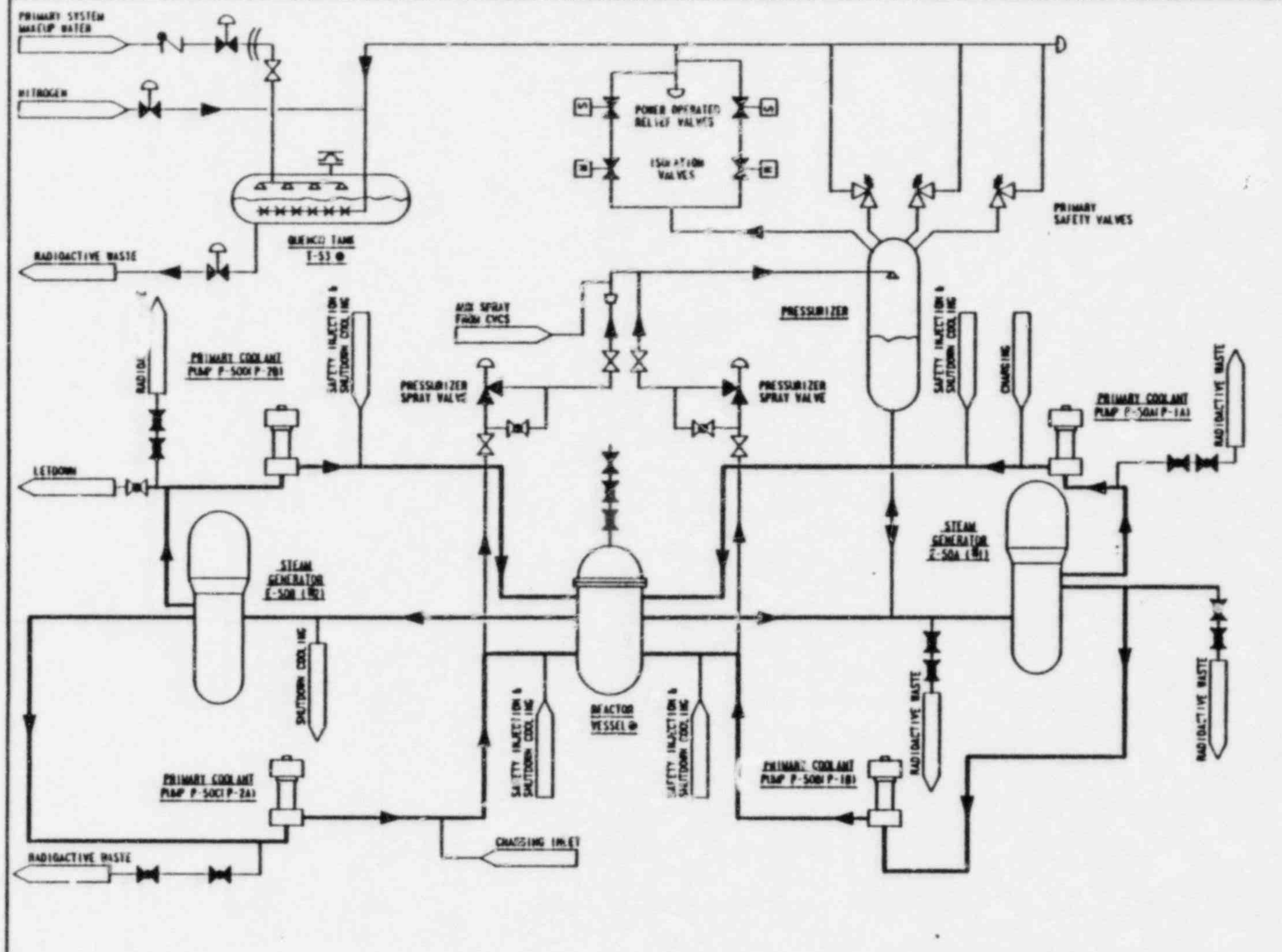
DESIGN/LICENSED THERMAL POWER: 2650/2530 Mw_t

RATED ELECTRICAL OUTPUT: 740 Mw_e

CONSTRUCTION/STARTUP:	START SITE CLEARING	AUGUST, 1966
	START CONSTRUCTION	MARCH, 1967
	LOW POWER TESTING LICENSE (POL)	MARCH, 1971
	INITIAL CRITICALITY	MAY, 1971
	LICENSED TO 20% POWER	NOVEMBER, 1971
	COMMERCIAL OPERATION	DECEMBER 31, 1971
	LICENSED TO 60% POWER	MARCH, 1972
	FULL POWER LICENSE (60% LIMIT)	OCTOBER, 1972
	POWER LIMIT 85% POWER	DECEMBER, 1972
	100% POWER (2200 Mw_t) AUTH. BY AEC	MARCH, 1973
	APPLICATION FOR POL TO FTOL CONVERSION	JANUARY, 1974
	2530 Mw_t POWER LEVEL AUTH. BY NRC	NOVEMBER, 1977



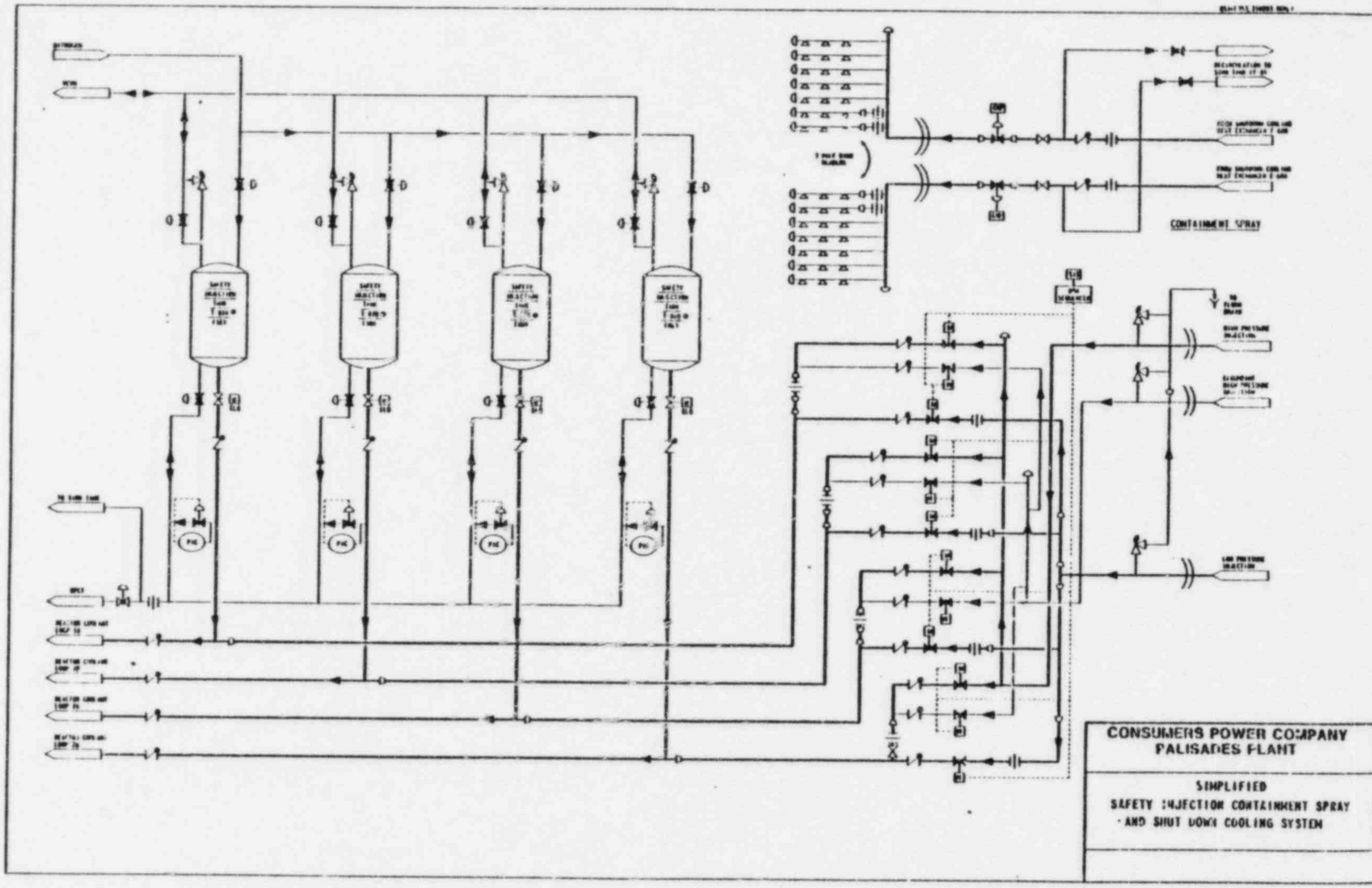
A-268



CONSUMERS POWER COMPANY
PALISADES PLANT

PRIMARY COOLANT SYSTEM

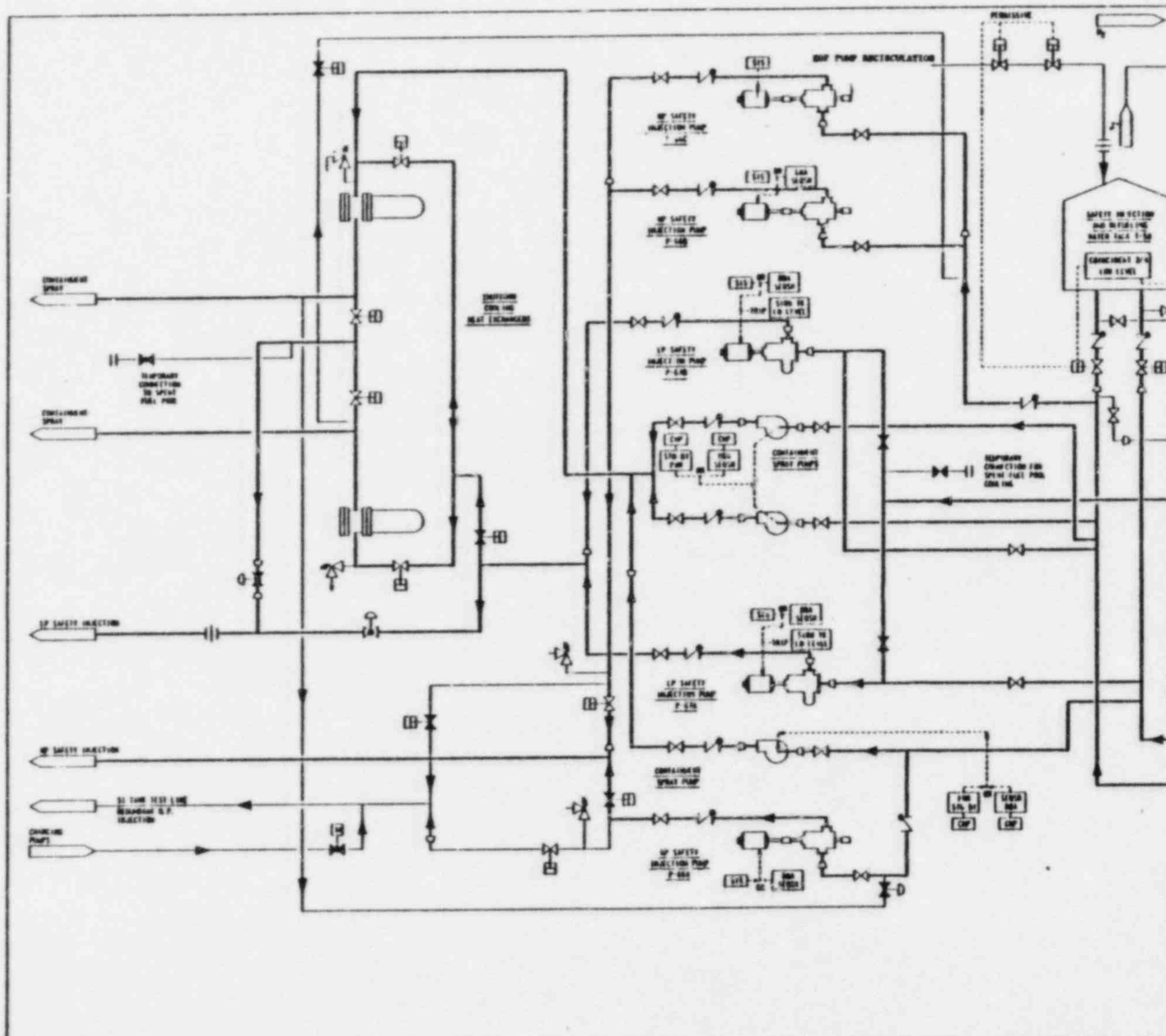
A-269



CONSUMERS POWER COMPANY
PALISADES PLANT

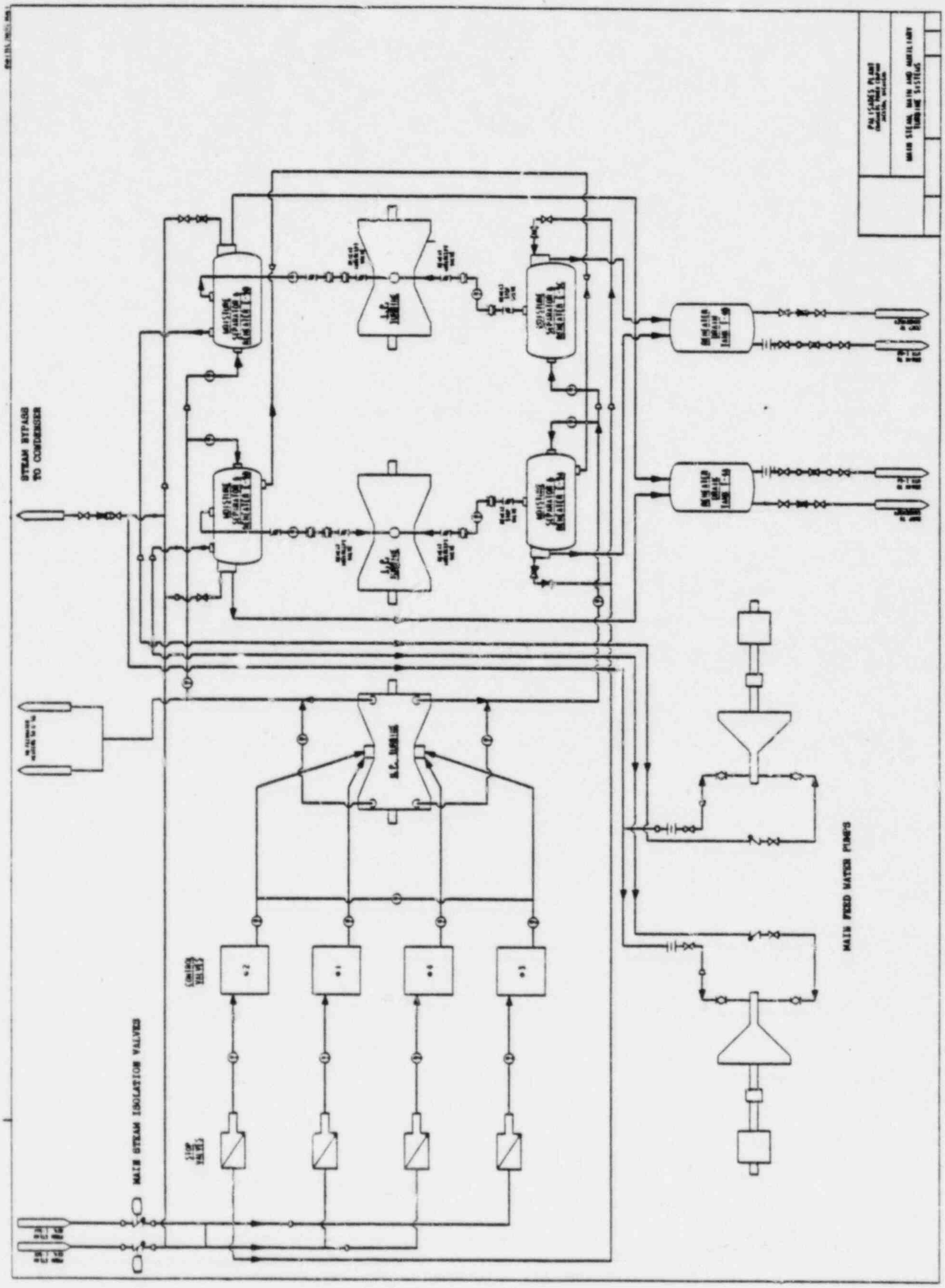
SIMPLIFIED
SAFETY INJECTION CONTAINMENT SPRAY
AND SHUT DOWN COOLING SYSTEM

A-270

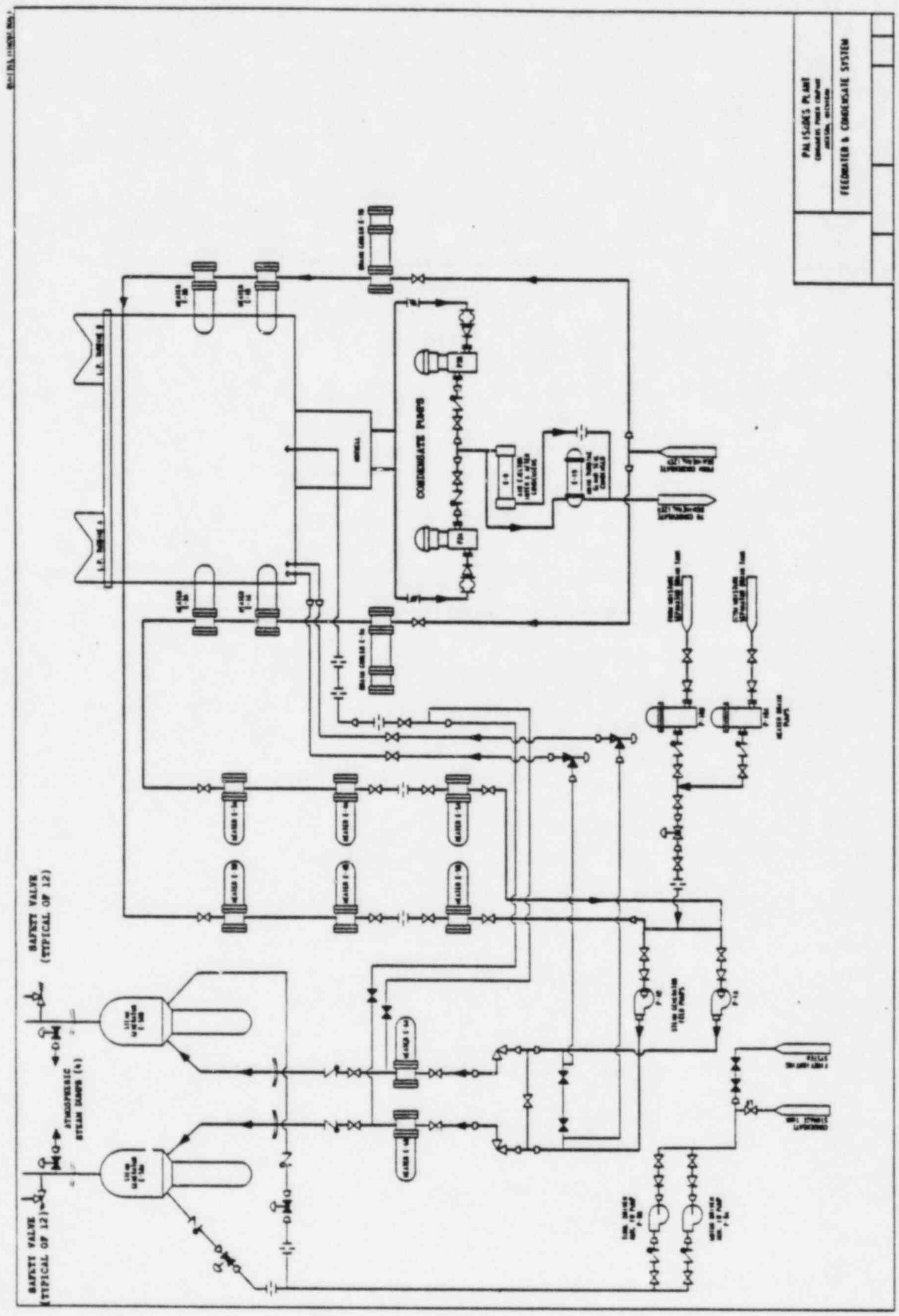


CONSUMERS POWER COMPANY
PALISADES PLANT

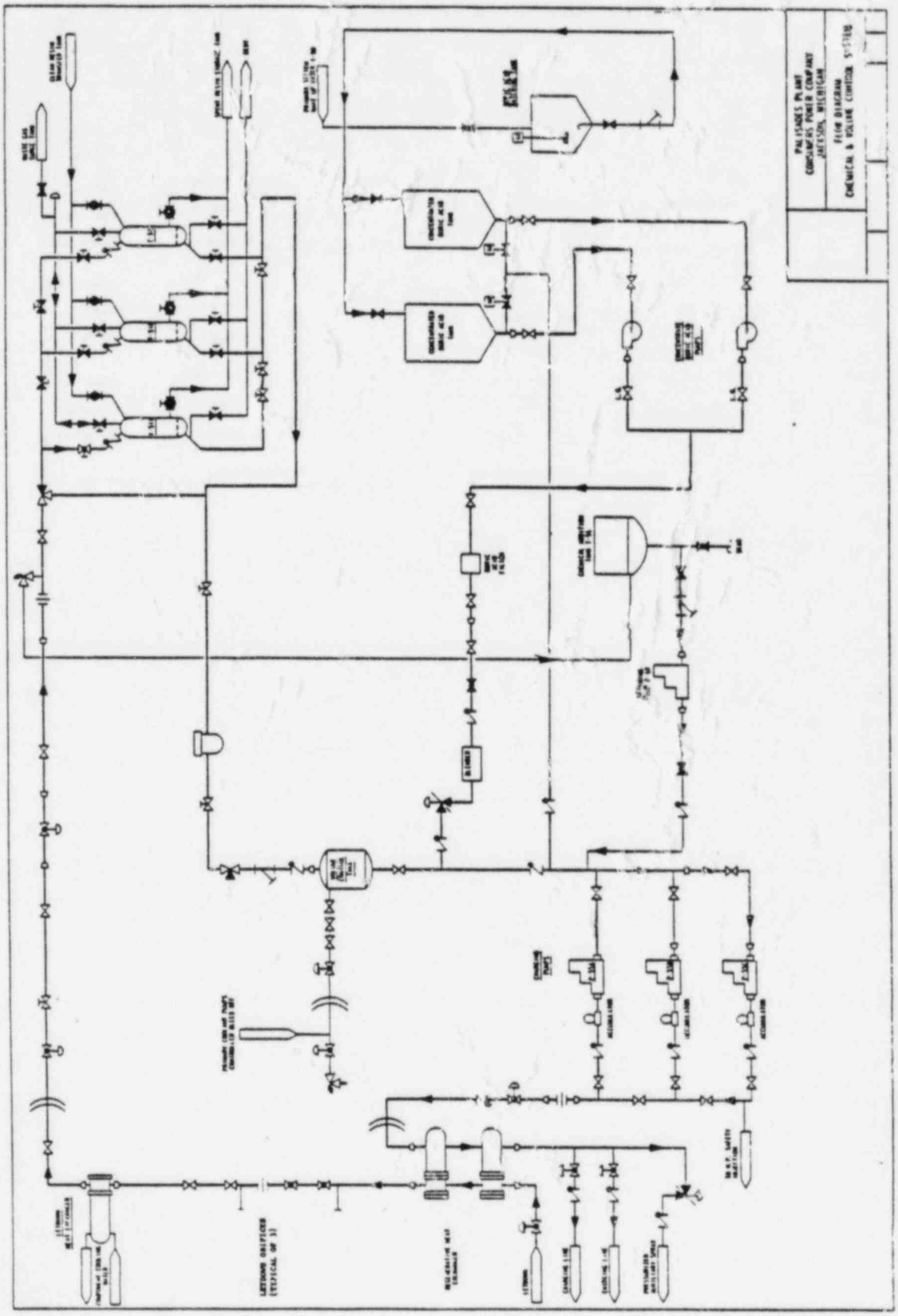
SIMPLIFIED
SAFETY INJECTION CONTAINMENT SPRAY
AND SHUT DOWN COOLING SYSTEM


STEAM BYPASS TO CONDENSER
MAIN STEAM ISOLATION VALVES
MAIN FEED WATER PUMPS

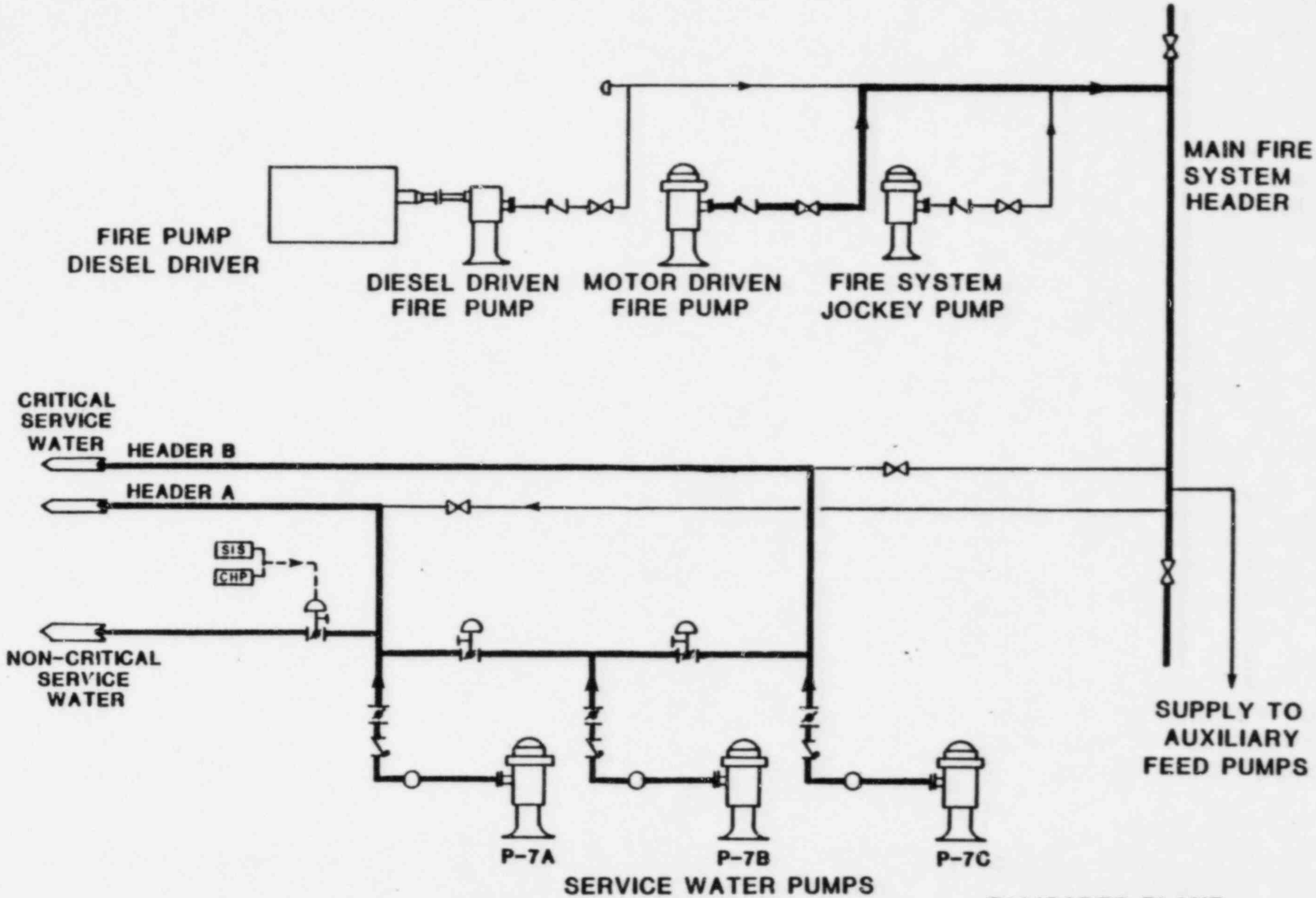
A-272


PALISADES PLANT
FEEDWATER & CONDENSATE SYSTEM
SAFETY VALVE (TYPICAL OF 12)
SAFETY VALVE (TYPICAL OF 12)
ATMOSPHERIC STEAM DUMPS (4)
CONDENSATE PUMPS

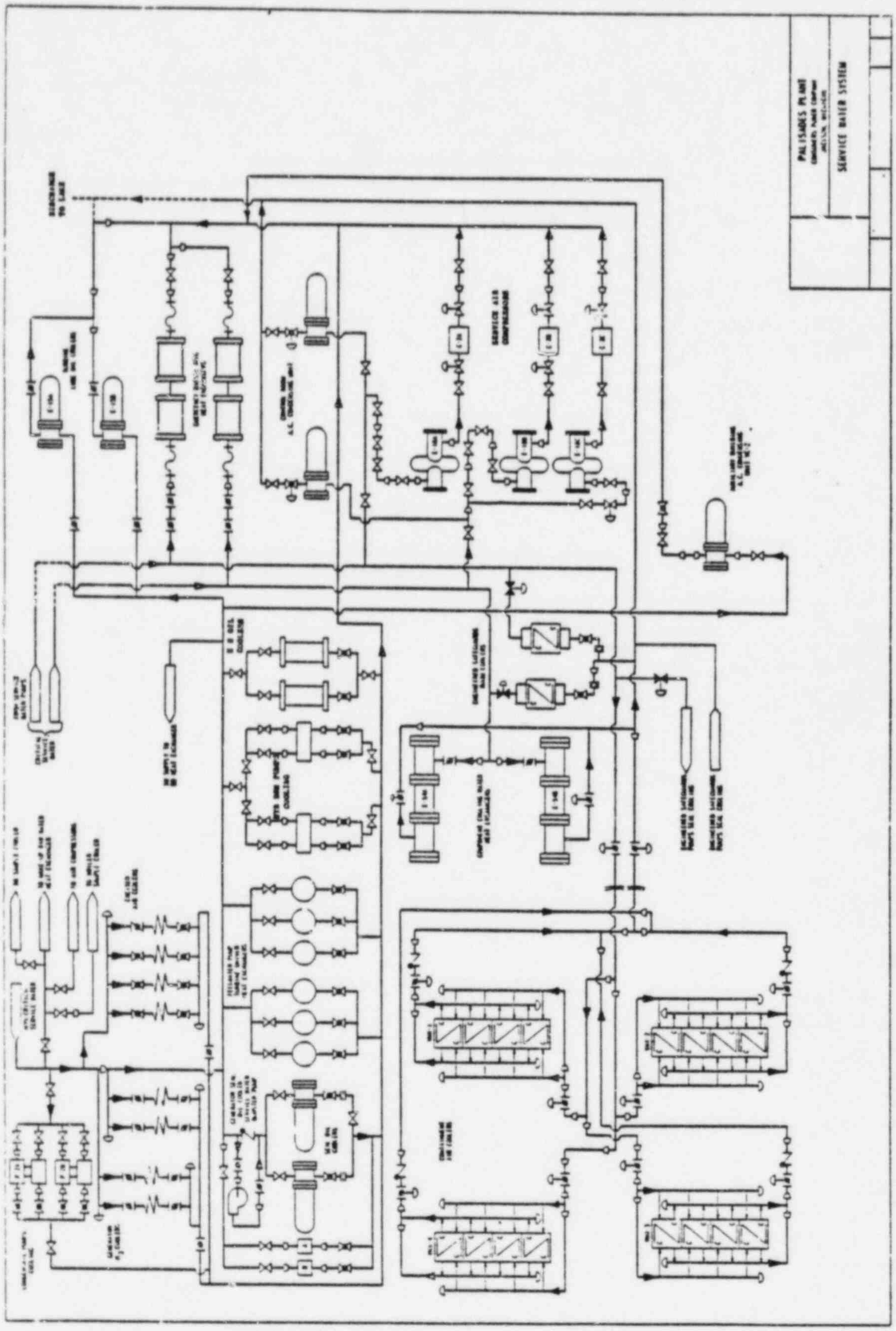
A-273

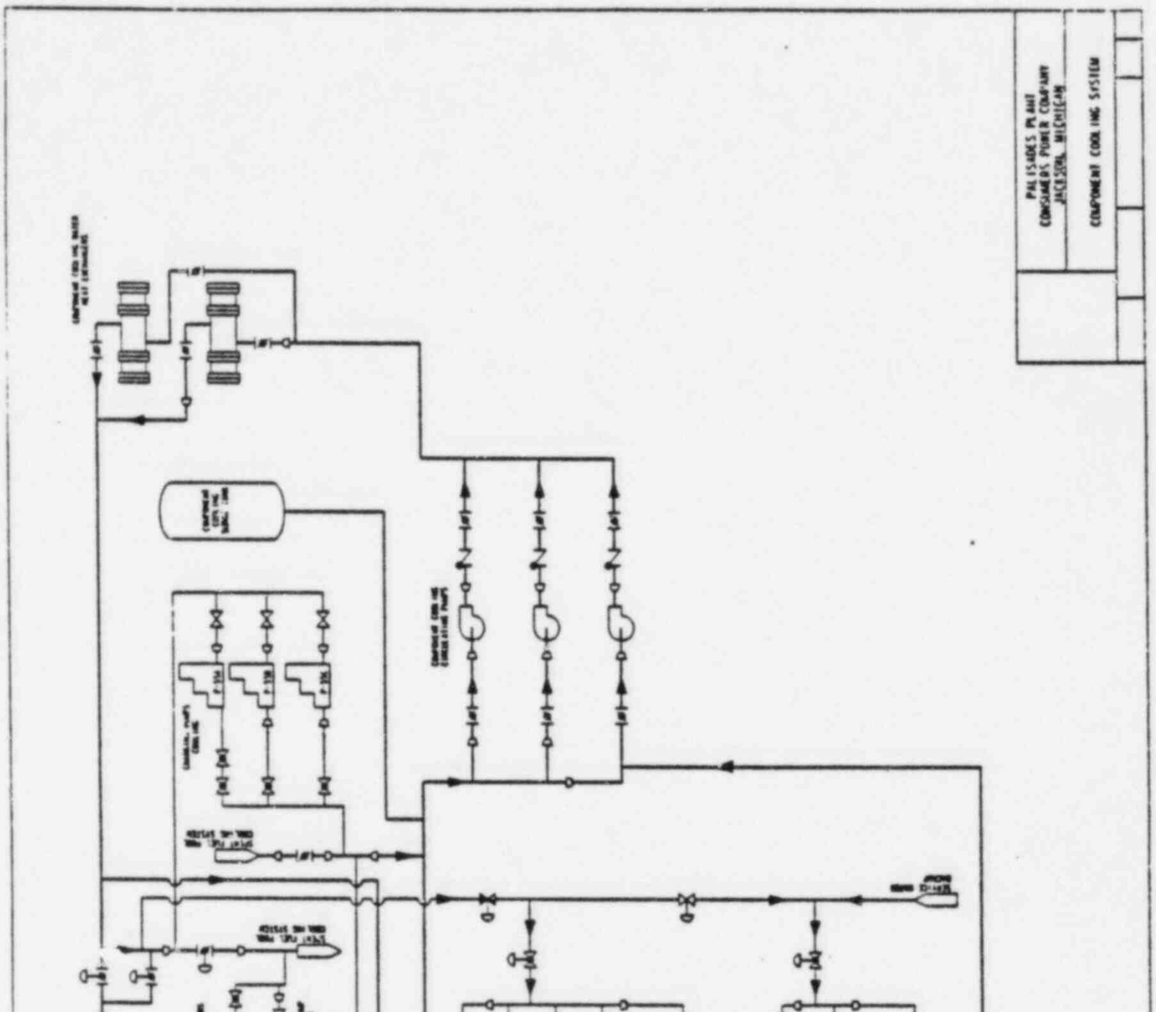

PALISADES PLANT
CONSUMERS POWER COMPANY
JACKSON, MICHIGAN
FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM

A-275



PALISADES PLANT
CONSUMERS POWER COMPANY
SIMPLIFIED SERVICE WATER AND
FIRE PUMP ARRANGEMENT


PALISADES PLANT
SERVICE WATER SYSTEM
SERVICE AIR COMPRESSORS


PALISADES PLANT
CONSUMERS POWER COMPANY
JACKSON, MICHIGAN
COMPONENT COOLING SYSTEM

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PALISADES PLANT
MODIFICATIONS PLANNED BY CPCO
TO RESOLVE SEP FINDINGS

TOPIC	
VI-2.D/VI-3 XV-2	Perform appropriate modifications to make MSIV/MSS configuration single failure proof with respect to concerns for two steam generator blowdowns inside containment.
VI-4	Modify the 3" pressurizing line on penetration 19 to remove threaded pipe joints between the outer air lock wall and isolation valve P5A.
	Provide a second remotely operated valve in series with the existing air-operated valve for penetration 44.
VII-1.A	Install suitable isolation devices or channel separation which meets IEEE 279-1971 for Primary System flow and Steam Generator A and B pressure inputs to the Fischer & Porter Plant Computer.
VII-3	Provide a second channel of CCW expansion tank level indication.
IX-3	Verify or add drainage capability to the intake structure to ensure postulated leaks will not flood service water pump motors, preferably without operator action. If coincident operator action is needed, verify that sufficient response time is available to prevent loss of service water.
	Provide spray protection over the service water pump ventilation louvers.
	Provide a control room alarm to warn of intake structure flooding.

PALISADES PLANT
ADMINISTRATIVE ACTIONS PLANNED BY CPCO
TO RESOLVE SEP FINDINGS

TOPIC	
II.3.A II.3.B.1 II.3.C	Submit final report on storm surge study (Completed 03/23/82)
III-2 III-4.A VII-3	Review protected water sources available for PCS and SG makeup and verify that procedures are sufficient for operators to use those sources in a timely manner.
III.3.C	Formalize the inspection program for the intake crib, pipe and intake structure by incorporation into the plant management system for either preventive maintenance or surveillance testing activities.
III.7.A	Submit proposed change to Technical Specifications to modify tendon ISI acceptance criteria.
	Reinspect tendons BF-65 and D 1-38 during the next scheduled containment tendon surveillance ISI.
III.7.C	Perform one-time delamination inspection similar to previous inspection in 1970.
	Submit p. posed Technical Specification change to ncorporate requirement for an additional elamination inspection in the event that a corrective retensioning program is required for 5% or more of the total number of dome tendons installed.
V-10.B/V-11.A	Submit proposed Technical Specification change to require the Low Temperature Overpressure Protection System to be in service whenever the Shutdown Cooling System is in service.
	Review exiting procedures and modify if appropriate to verify that operators are provided with sufficient guidance to direct use of safety grade systems in the event of failures in non safety grade systems.

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PALISADES PLANT
ADMINISTRATIVE ACTIONS (Cont'd)

V-11.A (Elec)	Revise operating procedure to verify LPSI check valve closure prior to criticality after each use of the LPSI system for shutdown cooling.
VII-3	Review procedures and modify as necessary to ensure that operators have guidance for removing non essential DC loads to extend battery life if conditions warrant.
VIII-3.A	Implement station battery capacity and service testing program.

PALISADES PLANT
FURTHER ANALYSES PLANNED BY CPCO
TO RESOLVE SEP FINDINGS

TOPIC	ACTION
III-1	Complete review and verify, on a sampling basis, the adequacy of the open items listed in NRC letter dated 12/28/81.
III.5.A V-5	Complete evaluation considering guidance provided in NRC letter of 12/04/81. Provide schedule for any modifications determined to be necessary, including consideration of Topic V-5 conclusions.
III-6	Complete evaluation of integrity of one electrical panel; verify adequacy of mountings of large internal components in safety-significant panels.
III-7.B	Review the 22 specific code changes identified in NRC letter of 11/16/81 for applicability and determine a method to evaluate the effects of applicable changes as they pertain to adequacy of plant design.
VIII-4	Complete circuit evaluations and determine need for upgrading secondary overload protection for circuits penetrating containment.
IX-3	Perform detailed analysis to verify CCW temperature limits are not exceeded under the postulated accident conditions. If the analysis indicates the need, modify procedures to direct isolation of non-essential service water loads under the postulated conditions.

TOPIC	ACTION
IX-5	Verify by test or analysis that the auxiliary feedwater pump room would not heat up excessively and compromise operation of both auxiliary feed pumps due to a loss of offsite power to the ventilation fans.
	Verify by test or analysis that the cable spreading and switchgear rooms would not heat up excessively and compromise operation of vital equipment due to a loss of offsite power to the ventilation fans.

PALISADES PLANT
SYSTEMATIC EVALUATION PROGRAM - RESOURCES EXPENDED AS OF 03/01/82

	Palisades
CPCo RESOURCES (EQUIVALENT PERSONS) EXPENDED OVER 4 YEARS	APPROX 39000 mh (APPROX 20 MANYEARS)
AVERAGE EFFICIENCY (90 TOPICS)	433 mh/TOPIC
CONTRACTOR COSTS TO DATE	APPROX $2,3000,000
TOTALS	~ $4,000,000
ESTIMATED FUTURE COSTS RESULTING FROM SEP (MODIFICATIONS, ANALYSES, PROCEDURE REVIEWS)	~ $3-4,000,000
MAJOR ANALYSES PERFORMED OR UPGRADED AS A RESULT OF SEP	• SEISMIC QUALIFICATION OF SELECTED EQUIPMENT TO CONFIRM EXISTING PLANT DESIGN ADEQUACY • SEISMIC QUALIFICATION OF ELECTRICAL SWITCH-GEAR (SEPOG) • HELB INSIDE CONTAINMENT • CABLE TRAY QUALIFICATION (SEPOG) • PROTECTION OF CONTAIN-MENT ELECTRICAL PENE-TRATIONS

NOTES:
1. EEQ EXCLUDED AFTER 6/80
2. ONLY MAJOR MODIFICATION RESULTING DIRECTLY FROM SEP WAS UPGRADE OF ELECTRICAL EQUIPMENT ANCHORAGES.
3. NO OTHER MAJOR MODIFICATIONS OR ADMINISTRATIVE CHANGES HAVE BEEN MADE TO DATE SOLELY AS A RESULT OF SEP. SEVERAL WERE RELATED TO SEP BUT MANDATED BY OTHER CONCERNS. THIS INCLUDES IEB 79-14 and 79-02 SEISMIC ANALYSES AND MODIFICATIONS (PALISADES > $20,000,000), DC BUS STATUS ANNUNICATION, CONDENSATE PUMP PIT FLOODING ALARMS AND CONTROL ROOM VENTILATION MODIFICATIONS.

SEISMIC REVIEW OF PALISADES PLANT

- GENERAL PHILOSOPHY AND SCOPE
- OVERVIEW OF REVIEW APPROACHES
- DETERMINATION OF SEISMIC HAZARD
- BASES FOR REEVALUATION
- CONCLUSIONS

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X29794

GENERAL PHILOSOPHY

- CONSIDER SAFE SHUTDOWN EARTHQUAKE ONLY
- SAMPLING APPROACH WITH CONSERVATIVE SEISMIC INPUT
- CONSIDER THE CONSERVATISMS ASSOCIATED WITH ORIGINAL ANALYSIS METHODS AND DESIGN CRITERIA (PHASE I)
- CONFIRM ADEQUACY OF ORIGINAL SEISMIC DESIGN (PHASE II)
- WHERE ORIGINAL SEISMIC DESIGN IS NOT ADEQUATE IMPLEMENT APPROPRIATE MODIFICATIONS

REVIEW SCOPE

THE SEP SEISMIC REEVALUATION OF PALISADES FACILITY WAS A LIMITED REVIEW CENTERING ON:

- ASSESSMENT OF THE GENERAL INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY

- EVALUATION OF THE CAPABILITY OF ESSENTIAL STRUCTURES, SYSTEMS AND COMPONENTS REQUIRED TO SHUTDOWN THE REACTOR SAFELY AND TO MAINTAIN IT IN A SAFE SHUTDOWN CONDITION (INCLUDING THE CAPABILITY FOR REMOVAL OF RESIDUAL HEAT) DURING AND AFTER A POSTULATED SEISMIC EVENT

REVIEW APPROACH

- DEVELOP REVIEW CRITERIA (NUREG/CR-0098 AND SSI GUIDELINES)
- ORGANIZE SEISMIC REVIEW TEAM (NRC STAFF AND CONSULTANTS)
- DOCKET REVIEW
- SITE VISIT
  - OBSERVE AS-BUILT PLANT SPECIFIC FEATURES
  - OBTAIN INFORMATION NOT AVAILABLE IN THE DOCKET
  - DISCUSS STAFF'S DOCKET REVIEW FINDINGS WITH LICENSEE
  - IDENTIFY SAMPLES FOR CONFIRMATORY ANALYSES
- REVIEW ADDITIONAL INFORMATION FROM LICENSEE, A/E, NSSS VENDOR FILES
- CONDUCT CONFIRMATORY ANALYSES/EVALUATION USING CONSERVATIVE SPECTRA (R.G. 1.60)
  - STRUCTURAL RESPONSES
  - IN-STRUCTURE SPECTRA
  - EVALUATION OF SAMPLED PIPING/EQUIPMENT
- COMPARE WITH PERFORMANCE CRITERIA
- RESOLVE OPEN ISSUES
- DOCUMENT RESULTS (CONSULTANT REPORTS AND SER)

DETERMINATION OF SEISMIC HAZARD

- A SEISMIC HAZARD ANALYSIS WAS CONDUCTED BY LLL, NRC CONTRACTOR, AND RESULTS WERE DOCUMENTED IN NUREG/CR-1582, VOL. 2-4

- SITE SPECIFIC SPECTRA WERE DEVELOPED BASED ON:

  1) LLL ANALYSIS (NUREG/CR-1582)

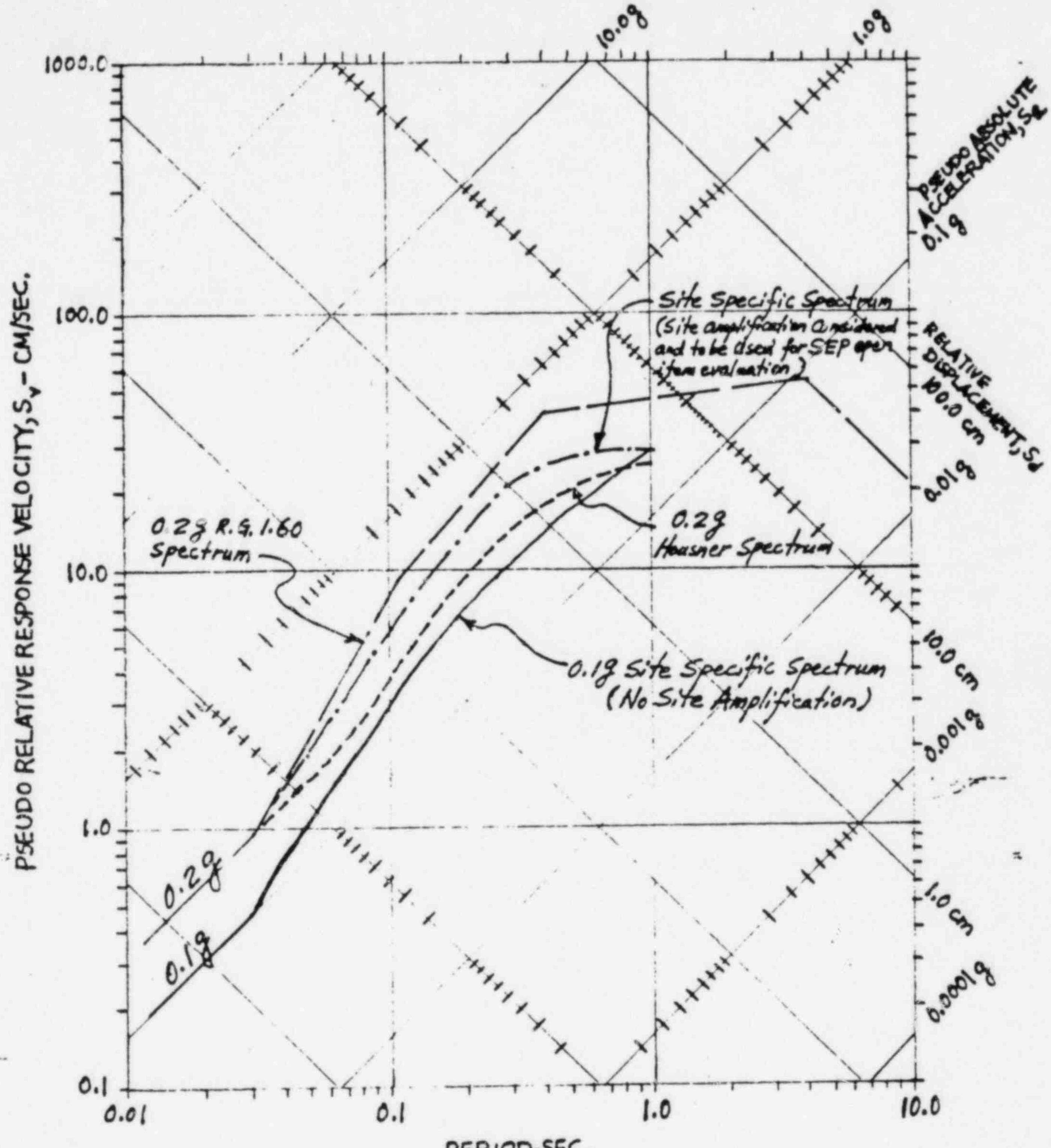
  2) LICENSEE ANALYSIS AND STUDIES

  3) PREDICTIONS BY ALTERNATIVE METHODS (I.E., DETERMINISTIC METHODS)

  4) SENSITIVITY STUDIES

  5) EXPERT OPINION AND FEEDBACK OF RESULTS TO EXPERTS

- COMPARISON OF ALL SPECTRA



COMPARISON OF GROUND RESPONSE SPECTRA AT PALISADES SITE

Figure 1

CONCLUSIONS

STRUCTURES

ALL SAFETY RELATED STRUCTURES AND STRUCTURAL ELEMENTS WERE FOUND TO BE ADEQUATELY DESIGNED TO RESIST THE POSTULATED SSE.

PIPING SYSTEMS

- EVALUATION CRITERIA - ASME CODE REQUIREMENT FOR CLASS 2 PIPING SYSTEMS AT APPROPRIATE SERVICE CONDITIONS

- THE NRC CONFIRMATORY ANALYSES IDENTIFIED THAT 5 OUT OF 5 SAMPLED PIPING SYSTEMS WERE FOUND TO BE OVERSTRESSED AND/ OR TO HAVE LARGE DISPLACEMENTS UNDER THE POSTULATED SSE

- THE LICENSEE HAD IMPLEMENTED A SEISMIC UPGRADING PROGRAM FOR SAFETY-RELATED PIPING GREATER THAN 2½" IN DIAMETER (IE BULLETIN 79-14)

- DURING THE SEP REVIEW, ANALYSIS AND MODIFICATION OF PIPING AND SUPPORTS TO CORRECT "AS BUILT" DIFFERENCES WERE COMPLETED

- THE SUBSEQUENT CONFIRMATORY ANALYSES BY NRC DEMONSTRATED THAT THE DESIGN OF SAMPLED PIPING SYSTEMS WAS ACCEPTABLE

EQUIPMENT

- SEISMIC QUALIFICATION OF EQUIPMENT CENTERED ON TWO AREAS - STRUCTURAL INTEGRITY AND FUNCTIONABILITY
- REPRESENTATIVE SAMPLES WERE SELECTED FROM EACH GROUP OF EQUIPMENT (E.G., CONTROL ROOM PANELS, MOTOR OPERATED VALVES, ETC.)
- A TOTAL OF 21 EQUIPMENT ITEMS WERE SAMPLED (15 MECHANICAL ITEMS AND 6 ELECTRICAL ITEMS)
- FOUR OF THESE ITEMS REMAIN OPEN DUE TO LACK OF DESIGN INFORMATION OR ONGOING SEP OWNERS GROUP GENERIC PROGRAMS:

  1) SMALL PIPING (LESS THAN 2½" IN DIAMETER) WITH LARGE VALVE OPERATORS AND FUNCTIONABILITY OF THESE VALVES

  2) ELECTRICAL EQUIPMENT - STRUCTURAL ADEQUACY OF THE LOAD PATH BETWEEN INTERNAL COMPONENTS OR DEVICES THROUGH PANEL FRAME AND BRACING TO THE ANCHORAGE AND SUPPORT SYSTEM

  3) FUNCTIONABILITY OF ALL ELECTRICAL EQUIPMENT IS BEING RESOLVED THROUGH SEP OWNERS GROUP PROGRAM

  4) QUALIFICATION OF ELECTRICAL CABLE TRAYS IS BEING CONDUCTED THROUGH SEP OWNERS GROUP PROGRAM

- TOPIC V-10.B, RHR RELIABILITY

REVIEW CRITERIA

GDC 1 TO 5	OVERALL REQUIREMENTS
GDC 34	RESIDUAL HEAT REMOVAL

REVIEW GUIDELINES

SRP 5.4.7, RESIDUAL HEAT REMOVAL (RHR) SYSTEM

REGULATORY GUIDE 1.139, GUIDANCE FOR RESIDUAL HEAT REMOVAL

BTP RSB 5-1, DESIGN REQUIREMENTS FOR THE RESIDUAL HEAT REMOVAL SYSTEM

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- TOPIC V-10.B, RHR RELIABILITY

SAFETY OBJECTIVE

TO INSURE RELIABLE PLANT SHUTDOWN CAPABILITY USING SAFETY-GRADE EQUIPMENT

- TOPIC VII-3, SYSTEMS NEEDED FOR SAFE SHUTDOWN

SAFETY OBJECTIVES:

1) TO ASSURE ADEQUACY OF SYSTEM TO INITIATE OPERATIONS NEEDED FOR SHUTDOWN

2) TO ASSURE THAT NEEDED SYSTEMS TO MAINTAIN HOT SHUTDOWN ARE LOCATED OUTSIDE THE CONTROL ROOM WITH POTENTIAL CAPABILITY FOR COLD SHUTDOWN

3) TO ASSURE THAT ONLY SAFETY-GRADE EQUIPMENT IS REQUIRED TO BRING REACTOR COOLANT SYSTEM TO A LOW PRESSURE COOLING CONDITIONS

EVALUATION

- SITE VISIT BY TEAM
- EVALUATION OF SYSTEMS CAPABILITY TO PERFORM NEEDED FUNCTIONS FOR SHUTDOWN
- ASSUMPTIONS
  - LOSS OF OFFSITE POWER
  - SINGLE ACTIVE FAILURE
  - 4 HOUR WAIT PRIOR TO COOLDOWN INITIATION
- DEVELOPMENT OF MINIMUM LIST OF SYSTEMS NEEDED TO SHUTDOWN AS INPUT TO RELATED TOPICS
- ASSESSMENT OF CONFORMANCE TO GDC 1 TO 5 DONE UNDER THESE RELATED TOPICS
  - MISSILES/PIPE BREAK
  - SEISMIC DESIGN
  - QUALITY GROUP CLASSIFICATION
- ELECTRICAL, INSTRUMENTATION AND CONTROL ASPECTS REVIEWED SEPARATELY
- ASSESSMENT OF ADEQUACY OF AUXILIARY FEEDWATER SUPPLY (TMI TAP INTERFACE)

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CONCLUSIONS

- SHUTDOWN COOLING SYSTEM IS SUSCEPTIBLE TO SINGLE FAILURES
  - PLANT CAN STILL ATTAIN "COOL" DEPRESSURIZED CONDITION
  - TIME AVAILABLE FOR RESTORATION OF SCS
  - ALTERNATE HEAT REMOVAL PATHS
    STEAM GENERATORS
    LOW PRESSURE FEED AND BLEED
- PLANT SYSTEMS EXIST TO PERFORM THE SHUTDOWN UNDER REVIEW ASSUMPTIONS
- PROCEDURES MAY NOT BE ADEQUATE
  - THEY RELY ON NON-SAFETY GRADE EQUIPMENT
  - THEY DO NOT SPECIFY HOW SAFETY-GRADE SYSTEMS WOULD BE USED IF NON-SAFETY SYSTEMS WERE NOT AVAILABLE
- CONDENSATE STORAGE TANK INVENTORY NOT ADEQUATE
  - TANK TOO SMALL FOR COOLDOWN TO SCS INITIATION UNDER REVIEW ASSUMPTIONS
  - ALTERNATE SOURCES ARE AVAILABLE

TOPICS FOUND ACCEPTABLE DURING TOPIC REVIEW BASED

ON EQUIVALENCY TO CURRENT CRITERIA

Topic No.	Title
II-2.A	Severe Weather Phenomena
II-4	Geology & 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.D	Stability of Slopes
II-4.F	Settlement of Foundations and Buried Equipment
III-3.A	Effects of High Water Level on Structures
III-4.B	Turbine Missiles
III-4.D	Site Proximity Missiles (Including Aircraft)
III-5.B	Pipe Break Outside Containment
III-6	Seismic Design Consideration
III-7.D	Containment Structural Integrity Tests
III-10.B	Pump Flywheel Integrity
V-6	Reactor Vessel Integrity
V-7	Reactor Coolant Pump Overspeed
V-11.B	RHR Interlock Requirements
VI-1	Organic Materials and Post Accident Chemistry
VI-7.F	Accumulator Isolation Valves Power and Control System Design
VIII-2	Onsite Emergency Power Systems - Diesel Generator
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation

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Topic No.	Title
IX-1	Fuel Storage
XV-4	Loss of Non-Emergency AC Power to the Station Auxiliaries
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
XV-15	Inadvertent Opening of a PWR Pressurizer Safety-Relief Valve or a BWR Safety/Relief Valve

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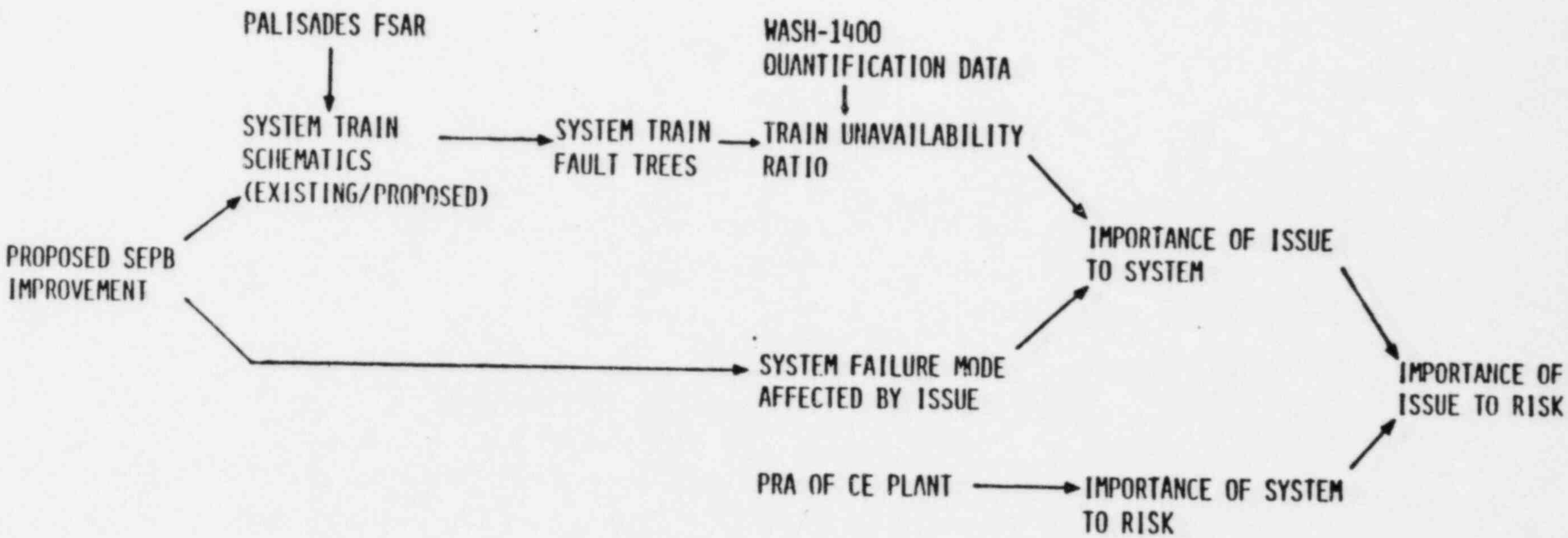

Figure 1. Study Methodology

TABLE 5

IMPORTANCE OF SYSTEMS TO RISK

	SYSTEM	RELATIVE CONSTRIBUTION TO RISK
H	AUXILIARY FEEDWATER SYSTEM	1.0
	MAIN FEEDWATER SYSTEM	$4x10^{-1}$
	OFFSITE POWER	$4x10^{-1}$
	RECOVERY OF MAIN FEEDWATER	$3x10^{-1}$
	REACTOR PROTECTION SYSTEM	$2x10^{-1}$
	DIESEL GENERATORS	$8x10^{-2}$
	BATTERIES	$5x10^{-2}$
	SERVICE WATER SYSTEM	$5x10^{-2}$
	RELIEF VALVE (STICKS OPEN)	$5x10^{-2}$
M	TRANSIENTS (OTHER THAN LOP, MFW)	$4x10^{-2}$
	RECOVERY OF STUCK OPEN RELIEF VALVE	$4x10^{-2}$
	RECOVERY OF OFFSITE POWER	$2x10^{-2}$
	COMPONENT COOLING WATER SYSTEM	$1x10^{-2}$
	RECOVERY OF DIESEL GENERATORS	$1x10^{-2}$
	HIGH PRESSURE INJECTION SYSTEM	$9x10^{-3}$
	ROOM COOLERS	$6x10^{-3}$
	SMALL LOCA, S_2	$5x10^{-3}$
L	SUMP VALVES	$3x10^{-3}$
	RECIRCULATION ACTUATION SYSTEM	$2x10^{-3}$
	CONTAINMENT LEAKAGE	$1x10^{-3}$
	SAFETY INJECTION ACTUATION SYSTEM	$8x10^{-5}$

TABLE 6

ISSUE CLASSIFICATION

HIGH IMPORTANCE TO RISK

VII-3 A):	BATTERY REQUIREMENTS ON LOSS OF TRANSMISSION LINE FROM SWITCHYARD TO PLANT
VIII-3.A:	BATTERY TESTING
IX-5:	VENTILATION OF AUXILIARY FEEDWATER PUMP ROOM

MEDIUM IMPORTANCE TO RISK

V-11.A:	CHECK VALVE FAILURE BETWEEN LPSI AND HPSI
VII-1.A:	ISOLATION OF THE REACTOR PROTECTION SYSTEM
VII-3 B):	WATER SUPPLY FOR THE AFWS
VII-3 E):	COMPONENT COOLING WATER SURGE TANK INSTRUMENTATION
IX-3:	SERVICE WATER REQUIREMENTS WITH ONLY ONW PUMP OPERABLE
XV-2:	BLOWDOWN OF BOTH STEAM GENERATORS

LOW IMPORTANCE TO RISK

V-5:	REACTOR COOLANT LEAKAGE DETECTION
VI-2.D VI-3	EFFECTS ON CONTAINMENT OF BLOWDOWN OF BOTH STEAM GENERAGORS
VI-10.A:	RESPONSE TIME TESTING OF ACTUATION SYSTEMS

DEMONSTRABLY LOW IMPORTANCE TO RISK

II-1.A:	DEFECTS IN LAND TITLES
III-8.A:	LOOSE PARTS MONITORING
V-10.B A):	OVERPRESSURE PROTECTION DURING OPERATION OF SDCS

TABLE 6 (CONT'D)

V-10.B B):	COLD SHUTDOWN, WITH SAFETY-GRADE EQUIPMENT
VII-3 C):	BORIC ACID TEMPERATURE
VII-3 D):	COMPONENT COOLING WATER PRESSURE INSTRUMENTATION
VII-3 F):	CHARGING PUMP FLOW INSTRUMENTATION
XV-12:	CONSEQUENCES OF ROD EJECTION ACCIDENTS

INTEGRATED ASSESSMENT OF 31 TOPICS

CONSIDERED FOR BACKFIT

APPENDIX XX

INTEGRATED ASSESSMENT TOPICS

- TOPICS NOT REQUIRING BACKFIT
- TOPICS WITH PROCEDURAL BACKFITS
- TOPICS WITH HARDWARE BACKFITS
- TOPICS WITH ANALYSIS AND POTENTIAL HARDWARE BACKFITS
- TOPICS WITH DIFFERENCES BETWEEN CPCo AND STAFF

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TOPICS NOT REQUIRING BACKFIT

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

DIFFERENCE

POTENTIAL DEFICIENCIES IN TITLES TO SOME LANDS WITHIN EXCLUSION AREA BOUNDARY

RESOLUTION

LICENSEE'S AUTHORITY OVER EXCLUSION AREA IS ADEQUATE

• TOPIC III-2, WIND AND TORNADO LOADINGS
(SECTION 4.6.2, SUPPLY AND EXHAUST PIPING FOR EMERGENCY DIESEL GENERATORS)

DIFFERENCE

SUPPLY AND EXHAUST PIPING FOR EMERGENCY DIESEL GENERATORS HAS NOT BEEN SHOWN TO WITHSTAND WIND AND TORNADO LOADINGS

RESOLUTION

- PIPING SURROUNDED BY REINFORCED CONCRETE WALLS ON ALL BUT ONE SIDE

- EACH DIESEL's PIPING IS IN ITS OWN ENCLOSURE

- TOPIC III-2, WIND AND TORNADO LOADINGS - (CONTINUED) (SECTION 4.6.3, STEEL FRAME ENCLOSURE OVER SPENT FUEL POOL)

DIFFERENCE

STEEL FRAME ENCLOSURE OVER SPENT FUEL HAS NOT BEEN SHOWN TO WITHSTAND WIND AND TORNADO LOADINGS

RESOLUTION

- PLANT SHUTDOWN NOT AFFECTED
- SHEET METAL SIDING IS EXPECTED TO FAIL BEFORE STEEL STRUCTURE THEREBY REDUCING POSTULATED WIND AND TORNADO LOAD
- SIDING IMPINGMENT ON FUEL POOL ACCEPTABLE
- STEEL FRAME WITHOUT SIDING IS ACCEPTABLE

- TOPIC III-4.A, TORNADO MISSILES (SECTION 4.8.2, SUPPLY AND EXHAUST PIPING FOR EMERGENCY DIESEL GENERATORS)

DIFFERENCE

SUPPLY AND EXHAUST PIPING FOR EMERGENCY DIESEL GENERATORS WAS FOUND NOT TO BE PROTECTED FROM TORNADO MISSILES

RESOLUTION

- PIPING SURROUNDED BY REINFORCED CONCRETE WALLS ON ALL BUT ONE SIDE

- EACH DIESEL'S PIPING IS IN IT'S OWN ENCLOSURE

- TOPIC III-4.A, TORNADO MISSILES (CONTINUED)
(SECTION 4.8.3, ATMOSPHERIC RELIEF STACKS OF STEAM
RELIEF VALVES)

DIFFERENCE

ATMOSPHERIC STACKS OF STEAM RELIEF AND DUMP VALVES
WERE FOUND NOT TO BE PROTECTED FROM TORNADO MISSILES

RESOLUTION

- ONLY 1 RELIEF OR DUMP VALVE NEEDED TO BRING REACTOR
TO SAFE SHUTDOWN

- STACKS SHELTERED BY OTHER STRUCTURES

• TOPIC III-4.A, TORNADO MISSILES (CONTINUED)

(SECTION 4.8.4, COMPRESSED AIR SYSTEM)

DIFFERENCE

COMPRESSED AIR SYSTEM (SERVICE AND INSTRUMENT AIR) WAS FOUND NOT TO BE PROTECTED FROM TORNADO MISSILES

RESOLUTION

- COMPRESSED AIR SYSTEM NOT REQUIRED FOR SAFE SHUTDOWN (MANUAL BACKUP AVAILABLE)

- INDEPENDENT AIR SYSTEM FOR ENGINEERED SAFEGUARDS EQUIPMENT

- BACKUP NITROGEN SYSTEM AVAILABLE FOR AUXILIARY FEED-WATER VALVES

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

DIFFERENCE

NO LOOSE PARTS MONITORING PROGRAM

RESOLUTION

- LOOSE PARTS BACKFITTING IS BEING CONSIDERED IN REVISION 1 TO REGULATORY GUIDE 1.133 BY THE COMMITTEE FOR REVIEW OF GENERIC REQUIREMENTS

- LOOSE PARTS CAN BE DETECTED DURING REFUELING

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

  DIFFERENCE

  - NO AIRBORNE PARTICULATE RADIOACTIVITY MONITOR
  - LOW SENSITIVITY OF SUMP LEVEL MONITOR
  - SENSITIVITY OF GASEOUS RADIATION MONITOR BELOW 1 PERCENT FAILED FUEL NOT APPARENT
  - ONLY GASEOUS RADIOACTIVITY MONITORING SYSTEM IS TESTABLE
  - SOME SYSTEMS NOT SEISMICALLY QUALIFIED

  RESOLUTION

  - SUFFICIENT SYSTEMS ARE AVAILABLE TO DETECT A 1-GPM LEAK FROM RCPB TO CONTAINMENT WITHIN 24 HOURS AND LARGER LEAKS IN LESS TIME
  - RELATIVE IMPORTANCE OF SMALL LOCA TO RISK IS LOW
  - LEAK FREQUENCY WAS DOMINATED BY REACTOR COOLANT PUMP SEAL FAILURE WHICH WOULD NOT BE PREVENTED BY ENHANCED LEAKAGE DETECTION
  - WILL BE REVIEWED IN CONJUNCTION WITH HIGH ENERGY PIPE BREAKS INSIDE CONTAINMENT

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

(SECTION 4.15.3, MONITORING OF REACTOR COOLANT INTERSYSTEM LEAKAGE)

<u>DIFFERENCE</u>

- CONTROL ROD DRIVE MECHANISM SEAL LEAKOFF NOT TESTED DURING NORMAL OPERATION NOR IS IT SEISMICALLY QUALIFIED

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

<u>RESOLUTION</u>

- SEAL LEAKOFF CAN BE DETECTED BY CONTAINMENT SUMP LEVEL MONITOR

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

  A) HAVE TWO DIVERSE SYSTEMS

  B) SAMPLING FOR SECONDARY ACTIVITY CAN BE PERFORMED IF THE MONITORS BOTH FAIL

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

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

(SECTION 4.15.4, CHEMICAL AND VOLUME CONTROL SYSTEM MAKEUP FLOW-RATE INFORMATION)

DIFFERENCE

CVCS MAKEUP FLOW RATE AS A MEANS OF LEAK DETECTION COULD NOT BE QUANTIFIED DURING TOPIC REVIEW

RESOLUTION

- MAKEUP AND LETDOWN FLOW RATES IN CONJUNCTION WITH PROGRAMMED PRESSURIZER LEVEL ARE SENSITIVE AND ARE USED BY THE OPERATORS TO IDENTIFY LEAKAGE. SENSITIVITY COULD NOT BE QUANTIFIED

- AUGMENTS OTHER LEAK DETECTION SYSTEMS

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

(SECTION 4.15.5, 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 III PERSONNEL AND FOUND ACCEPTABLE WITH RESPECT TO TECHNICAL SPECIFICATION REQUIREMENTS OF 1-GPM UNIDENTIFIED LEAKAGE FROM THE RCPB

- TOPIC VI-2.D, MASS AND ENERGY RELEASE FOR POSTULATED PIPE BREAKS INSIDE CONTAINMENT

- TOPIC VI-3, CONTAINMENT PRESSURE AND HEAT REMOVAL CAPABILITY

DIFFERENCE

CONTAINMENT CAN BE OVERSTRESSED TO 1.53 DESIGN PRESSURE FOR A STEAM LINE BREAK INSIDE CONTAINMENT AND FAILURE OF THE MSIV IN THE UNFAULTED STEAM LINE

RESOLUTION

- PROBABILITY OF CONTAINMENT FAILURE RESULTING FROM OVERPRESSURE EXCEEDING DESIGN IS LOW

- SINGLE FAILURE DEPENDENCY OF TWO STEAM GENERATOR BLOW-DOWN TO CONTAINMENT BEING ELIMINATED (TOPIC XV-2)

- TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM (SECTION 4.20.1, VALVE LOCATION)

DIFFERENCE

- TWO ISOLATION VALVES OUTSIDE CONTAINMENT INSTEAD OF ONE INSIDE AND ONE OUTSIDE

RESOLUTION

- PIPING BETWEEN CONTAINMENT AND CONTAINMENT ISOLATION VALVE IS RATED FOR AT LEAST TWICE CONTAINMENT DESIGN PRESSURE

- PIPING RUNS SHORT AS POSSIBLE

- ALL PIPING IS SUPPORTED AND DESIGNED FOR PIPE BREAK LOADS

- ALL PIPING PENETRATIONS IN AREAS PROTECTED FROM TORNADO MISSILES

- PRA FOUND IMPACT OF RESOLUTION TO BE LOW

- TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM (CONTINUED)

(SECTION 4.20.2, PIPE CAPS AND BLIND FLANGES)

DIFFERENCE

PIPE CAPS OR BLIND FLANGES ARE USED AS CONTAINMENT ISOLATION BARRIERS

RESOLUTION

- BLIND FLANGES ARE ACCEPTABLE AS ISOLATION BARRIERS IF TESTED IN ACCORDANCE WITH APPENDIX J

- 1977 ASME BPV CODE, SECTION III, ARTICLE NE-3367 STATES PENETRATIONS OF 2-IN PIPE SIZE OR LESS CAN BE MADE WITH PIPE CAPS

- TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING RESPONSE-TIME TESTING

  DIFFERENCE

  - OVERALL RESPONSE TIME TESTING IS NOT BEING PERFORMED

  RESOLUTION

  - CRITICAL COMPONENTS THAT CONTRIBUTE TO TIME CONSTRAINTS ARE TESTED I.E., DIESEL GENERATOR LOAD-SEQUENCER TIMING, DIESEL GENERATOR START TIMES, ROD INSERTION TIMES, AND STORAGE TIME OF IMPORTANT VALVES

  - PRA ANALYSIS - RATED LOW

- TOPIC VII-1.A, ISOLATION OF REACTOR PROTECTION SYSTEM FROM NONSAFETY SYSTEMS, INCLUDING QUALIFICATIONS OF ISOLATION DEVICES
(SECTION 4.23.2, THERMISTOR/ZENER DIODE ISOLATION)

<u>DIFFERENCE</u>

THERMISTOR/ZENER DIODE ISOLATION OF RPS ANALOG SIGNALS DOES NOT MEET IEEE STD 279-1971 PARAGRAPH 4.7.2

<u>RESOLUTION</u>

- 2 OUT 4 LOGIC PERMITS OPERATION WITH ONE FAILURE

- ALL ANALOG SIGNALS ARE ON CHANNEL A - INTERCHANNEL FAILURES MINIMIZED

- EACH CHANNEL'S POWER SUPPLY BUS IS ALARMED AND CAN BE ISOLATED

- TOPIC VII-1.A, ISOLATION OF REACTOR PROTECTION SYSTEM FROM NONSAFETY SYSTEMS, INCLUDING QUALIFICATIONS OF ISOLATION DEVICES (CONTINUED)

(SECTION 4.23.3, A709C OPERATIONAL AMPLIFIERS)

DIFFERENCE

A709C OPERATIONAL AMPLIFIERS NOT EVALUATED AS CLASS 1E EQUIPMENT AND HAVE RELATIVELY LOW INPUT IMPEDANCE

RESOLUTION

- INPUT IMPEDANCE IS 1 MEGOHM WHICH IS ADEQUATE

- OUTPUT IS FUSED TO PROTECT AGAINST HIGH VOLTAGE OR EXCESSIVE LOAD ON OUTPUT

- AMPLIFIERS ARE LOCATED IN CONTROL ROOM

- TOPIC VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN (4.24.3, BORIC ACID HEAT TRACING)

<u>DIFFERENCE</u>

- BORIC ACID HEAT TRACING AND BORIC ACID CONCENTRATED TANK HEATERS ARE SUPPLIED FROM NON-CLASS 1E POWER SOURCES
- HEAT TRACING NOT CLASS 1E

<u>RESOLUTION</u>

- ANOTHER SOURCE OF BORATED WATER SIRW, CAN BE USED TO MAINTAIN A HOT SHUTDOWN
- BORON IS NOT REQUIRED FOR SHUTDOWN MARGIN FOR SEVERAL HOURS, ASSUMING A STUCK ROD, BECAUSE XENON BUILDUP WOULD ADD NEGATIVE REACTIVITY
- PROCEDURES ARE AVAILABLE TO CONNECT TO CLASS 1E POWER
- SYSTEM IS INSPECTED EVERY SHIFT
- BORIC ACID TANKS ARE IN A ROOM WITH ELEVATED TEMPERATURES
- TEMPERATURE MONITORING INSTRUMENTATION IS REDUNDANT

• TOPIC VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN (CONTINUED)

(SECTION 4.24.4, PRESSURE SENSOR ON COMPONENT COOLING CIRCULATING PUMPS)

DIFFERENCE

ONLY ONE PRESSURE SENSOR ON THE OUTPUT OF CCW PUMPS

RESOLUTION

OTHER MEANS TO DETECT INADEQUATE COOLING WATER FLOW SUCH AS TEMPERATURE MONITORS AND ALARMS ON CCW HEAT EXCHANGERS, CONTROL ROD DRIVE MECHANISMS AND PRIMARY COOLANT PUMP SEALS

- TOPIC XV-12, SPECTRUM OR ROD EJECTION ACCIDENTS (PWR)

DIFFERENCE

FUEL FAILURE ANALYSIS BASED ON CRITERION FOR CLADDING DAMAGE OF 200 CALORIES/GRAM RATHER THAN A DEPARTURE FROM NUCLEATE BOILING

RESOLUTION

- ASSUMPTION OF 10 PERCENT FUEL FAILURE SHOWED ACCEPTABLE RESULTS

- 10% FUEL FAILURE HAS BEEN SHOWN BY GENERIC ANALYSIS TO BE CONSERVATIVE FOR RWA

TOPICS WITH PROCEDURAL BACKFITS

TOPICS RELATED TO PRIMARY AND SECONDARY WATER SUPPLIES AND SYSTEMS

- TOPIC III-2, WIND AND TORNADO LOADINGS
  (SECTION 4.6.1, SAFETY INJECTION AND REFUELING WATER (SIRW) AND CONDENSATE STORAGE TANKS (CST))
- TOPIC III-4.A, TORNADO MISSILES
  (SECTION 4.8.1, SIRW AND CST)
- TOPIC V-10.B, RESIDUAL HEAT REMOVAL SYSTEM RELIABILITY
  (SECTION 4.16.2, USE OF SAFETY-GRADE SYSTEMS FOR SAFE SHUTDOWN)
- TOPIC VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN
  (SECTION 4.24.5, ADEQUATE SEISMIC CATEGORY I WATER SUPPLY FOR THE AFW SYSTEM)

DIFFERENCES

THE SIRW AND CST HAVE NOT BEEN SHOWN TO BE ABLE TO WITHSTAND TORNADO WIND AND PRESSURE LOADINGS

THE SIRW AND CST WERE FOUND NOT TO BE PROTECTED FROM TORNADO MISSILES

PROCEDURES FOR USE OF SAFETY-GRADE SYSTEMS USED TO BRING THE PLANT TO COLD SHUTDOWN

SUFFICIENT SAFETY-GRADE WATER IS NOT MAINTAINED IN A SEISMICALLY QUALIFIED TANK(S) TO BEGIN SHUTDOWN COOLING

RESOLUTION

ADDITIONAL TORNADO OR TORNADO MISSILE PROTECTION FOR THE SIRW AND CONDENSATE STORAGE TANKS, OR ADDITIONAL SAFETY-GRADE PRIMARY OR SECONDARY TANKS WERE NOT REQUIRED BECAUSE ALTERNATE SOURCES OF SAFETY-GRADE PRIMARY AND SECONDARY WATER ARE AVAILABLE TO ACHIEVE COLD SHUTDOWN.

THE LICENSEE SHALL REVIEW AND UPGRADE PROCEDURES TO ASSURE THAT COLD SHUTDOWN CAN BE ACHIEVED USING ALTERNATE SOURCES OF WATER AND SAFETY-GRADE EQUIPMENT.

- TOPIC III-7.A, INSERVICE INSPECTION, INCLUDING PRESTRESSED CONCRETE CONTAINMENT WITH EITHER GROUTED OR UNGROUTED TENDONS (SECTION 4.11.1, TENDON FORCE ACCEPTANCE CRITERIA)

DIFFERENCE

MEASURED TENDON FORCES ARE COMPARED WITH A CONSTANT ACCEPTANCE CRITERIA EQUAL TO THE AVERAGE DESIGN PRESTRESS

RESOLUTION

ACCEPTANCE CRITERIA SHALL BE DEVELOPED FOR EACH TENDON THAT VARY WITH TIME

- TOPIC V-5, REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION
(SECTION 4.15.2, TECHNICAL SPECIFICATIONS REGARDING OPERABILITY OF LEAKAGE DETECTION SYSTEMS)

DIFFERENCE

NO TECHNICAL SPECIFICATIONS ON OPERABILITY OF LEAKAGE DETECTION SYSTEMS TO MONTIOR LEAKAGE TO CONTAINMENT

RESOLUTION

TECHNICAL SPECIFICATIONS ON OPERABILITY OF LEAKAGE DETECTION SYSTEMS WILL BE INTEGRATED WITH POSSIBLE LEAK DETECTION REQUIREMENTS OF TOPIC III-5.A

- TOPIC V-10.B, RESIDUAL HEAT REMOVAL SYSTEM RELIABILITY (SECTION 4.16.1, OVERPRESSURIZATION PROTECTION OF SHUTDOWN COOLING SYSTEM)

DIFFERENCE

SCS CAN BE PLACED IN SERVICE BEFORE OPS

RESOLUTION

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

- TOPIC V-11.A, REQUIREMENTS FOR ISOLATION OF HIGH AND LOW-PRESSURE SYSTEMS

DIFFERENCE

LPSI SYSTEM COULD BE OVERPRESSURIZED AND RESULT IN COMMON MODE FAILURE IF ONE CHECK VALVE FAILED TO OPERATE

RESOLUTION

PROCEDURE TO VERIFY CHECK VALVE CLOSURE BEFORE CRITICALITY AFTER EACH USE OF THE LPSI SYSTEM FOR SHUTDOWN COOLING

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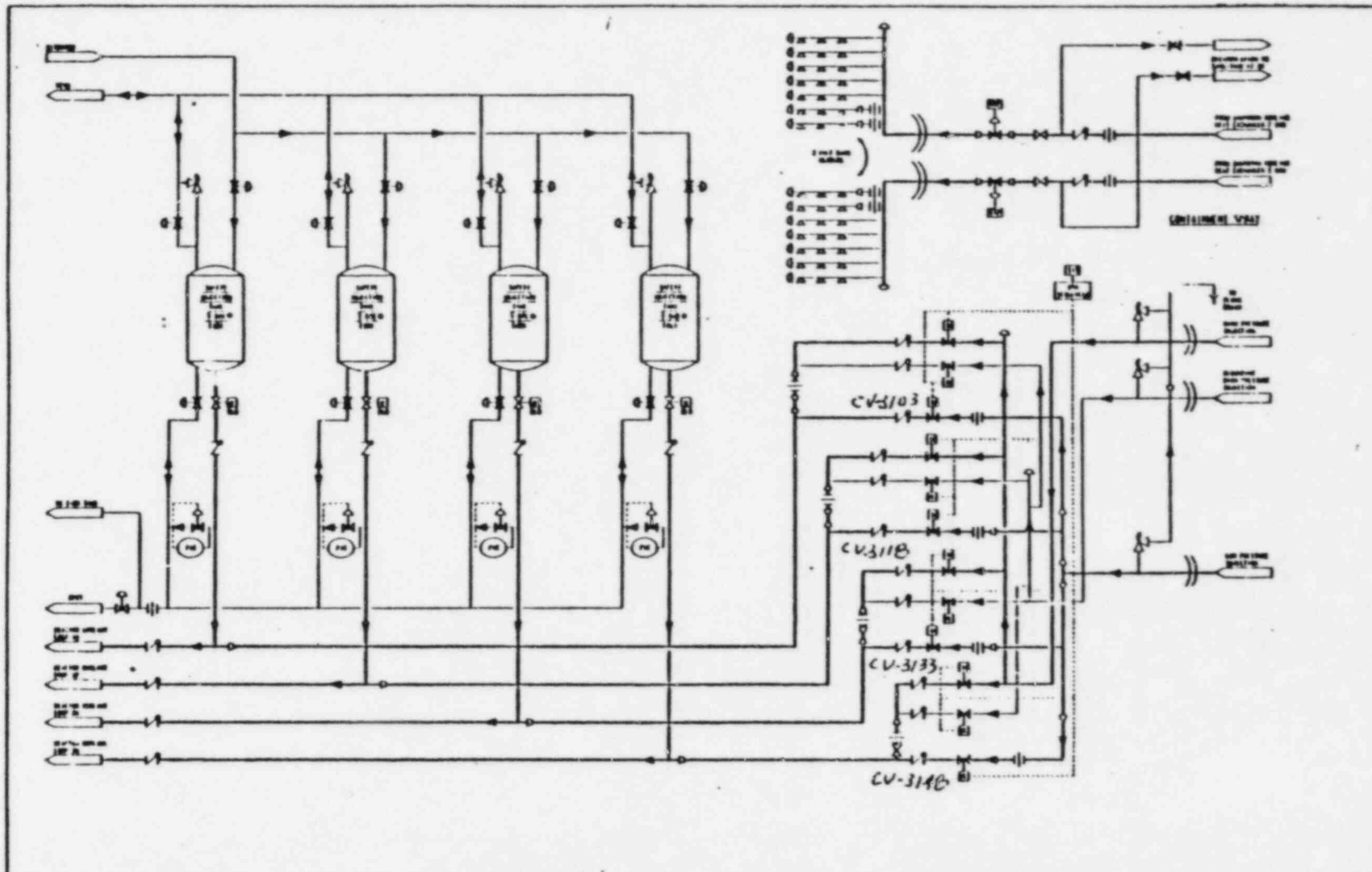

PALISADES SAFETY INJECTION, CONTAINMENT SPRAY
AND SHUTDOWN COOLING SYSTEM

TOPIC VI-6, CONTAINMENT LEAK TESTING

DIFFERENCE

LICENSEE HAS REQUESTED AN EXEMPTION TO THE REQUIREMENTS OF AIRLOCK LEAK TESTING IF THE AIRLOCK IS OPENED DURING THE INTERVAL BETWEEN THE 6-MONTH TYPE B TESTS

RESOLUTION

REQUEST FOR EXEMPTION DENIED. AIRLOCK DOOR SEAL INTEGRITY MUST BE VERIFIED WITHIN 72 HOURS OF EACH OPENING OR THE FIRST OF A SERIES OF OPENINGS DURING THE INTERIM BETWEEN THE 6-MONTH TESTS.

- TOPIC VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN (SECTION 4.24.1, REMOVING OF NONESSENTIAL LOADS AS AN ALTERNATIVE TO GDC 17)

DIFFERENCE

ABILITY TO SUSTAIN VITAL INSTRUMENTATION TO ACCOMPLISH SAFE SHUTDOWN IN CASE OF LOSS OF OFFSITE AND ONSITE POWER

RESOLUTION

DEVELOP PROCEDURES TO REMOVE NONESSENTIAL LOADS FROM THE BATTERY IF THE IMMEDIATE SOURCES OF OFFSITE AND ONSITE POWER ARE NOT AVAILABLE

TOPIC VIII-3.A, STATION BATTERY CAPACITY TEST REQUIREMENTS

DIFFERENCE

BATTERIES ARE NOT TESTED SUFFICIENTLY

RESOLUTION

A BATTERY SERVICE TEST AND A BATTERY DISCHARGE TEST SHALL BE PERFORMED

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TOPICS WITH HARDWARE BACKFITS

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- TOPIC VI-4, CONTAINMENT ISOLATION SYSTEM
(SECTION 4.20.3, MANUAL ISOLATION VALVE)

DIFFERENCE

MANUAL VALVE ON PENETRATION 44 NOT ACCEPTABLE

RESOLUTION

VALVE WILL BE CHANGED TO POWER OPERATING VALVE

(SECTION 4.20.4, THREADED PIPE CONNECTION)

DIFFERENCE

THREADED PIPE CONNECTION BETWEEN CONTAINMENT AND OUTERMOST ISOLATION VALVE

RESOLUTION

PIPE JOINTS IN THIS PORTION OF LINE WILL BE WELDED

- TOPIC VII-1.A, ISOLATION OF REACTOR PROTECTION SYSTEM FROM NONSAFETY SYSTEMS, INCLUDING QUALIFICATIONS OF ISOLATION DEVICES
(SECTION 4.23.1, INADEQUATE ISOLATION)

DIFFERENCE

NO ISOLATION FROM RPS TO PLANT COMPUTER ON 3 CHANNELS

RESOLUTION

QUALIFIED ISOLATION DEVICES WILL BE INSTALLED ON THE STEAM GENERATOR A AND B PRESSURE CHANNELS AND ON THE REACTOR COOLANT FLOW CHANNEL

- TOPIC VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN (SECTION 4.24.2, COMPONENT COOLING WATER SURGE TANK LEVEL)

DIFFERENCE

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

RESOLUTION

SECOND SENSOR AND INDICATOR WILL BE ADDED

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- TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.27.2, FLOODING OF SAFETY SYSTEMS IN INTAKE STRUCTURE)

DIFFERENCE

INTAKE STRUCTURE IS SUSCEPTIBLE TO INTERNAL FLOODING POTENTIALLY CAUSING THE LOSS OF SERVICE WATER PUMPS AND THE ULTIMATE HEAT SINK

RESOLUTION

- PROVIDE DRAINAGE IN INTAKE STRUCTURE FOR POSTULATED LEAKS OR BREAKS
- ALARMS IN CONTROL ROOM TO INDICATE OCCURRENCE OF FLOODING IN INTAKE STRUCTURE
- ASSURANCE THAT THE OPERATOR CAN REACT IN SUFFICIENT TIME TO PREVENT INUNDATION OF SERVICE WATER PUMPS
- SPRAY PROTECTION FOR THE SERVICE WATER PUMPS

- TOPIC XV-2, SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT (PWR) (SECTION 4.30.1, MAIN STEAMLINE BREAK INSIDE CONTAINMENT)

  DIFFERENCE

  MAINSTEAM LINE BREAK INSIDE CONTAINMENT AND SINGLE FAILURE OF MSIV ON INTACT LINE RESULTS IN TWO STEAM GENERATOR BLOWDOWN TO CONTAINMENT

  RESOLUTION

  APPROPRIATE MODIFICATIONS TO MAKE MSIV/MAIN STEAM CONFIGURATION SINGLE-FAILURE PROOF WITH RESPECT TO TWO STEAM-GENERATOR BLOWDOWN INSIDE CONTAINMENT

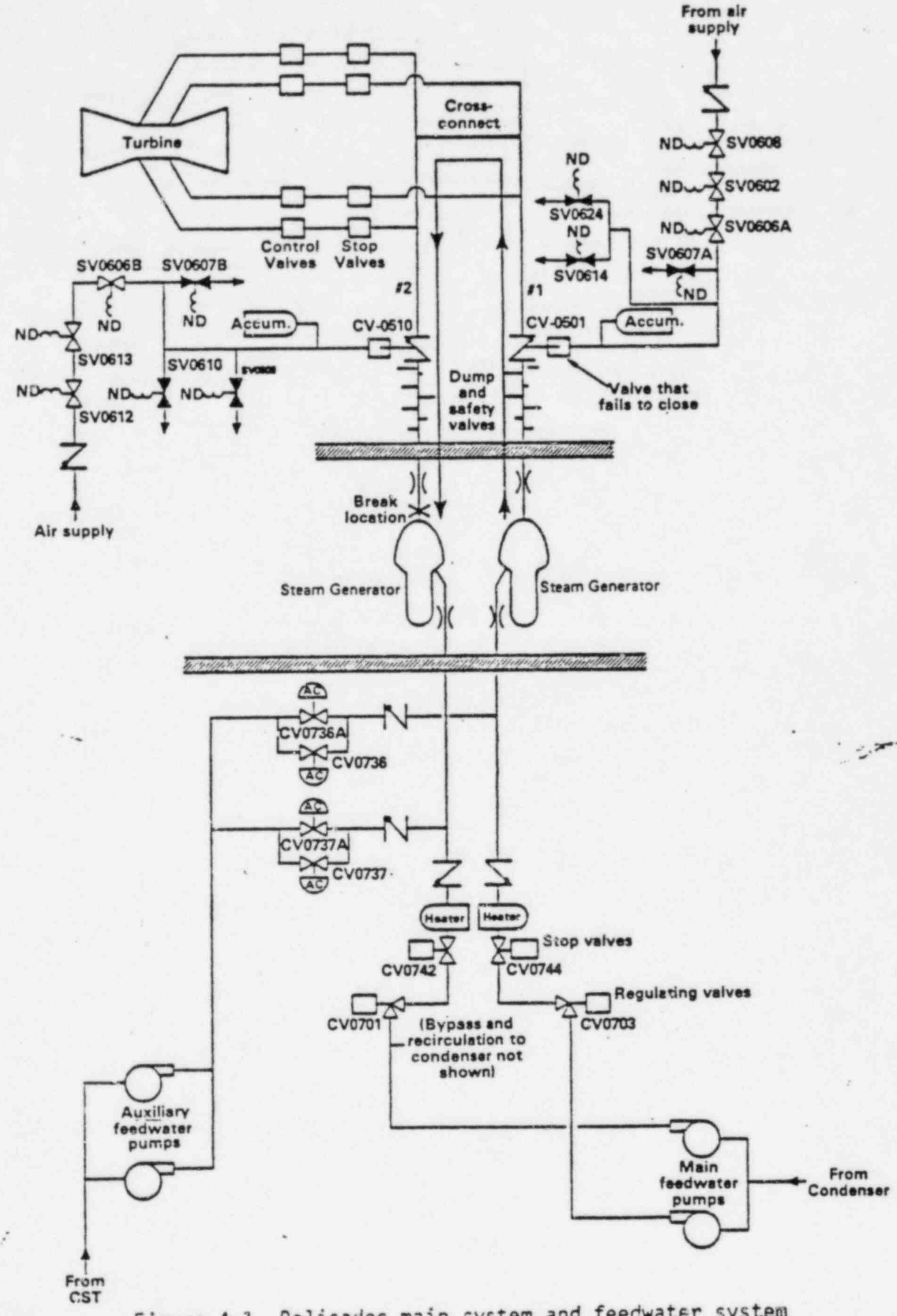

Figure 4.1 Palisades main system and feedwater system

TOPICS WITH ANALYSIS AND POTENTIAL

HARDWARE BACKFITS

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- TOPIC II-3.B - FLOODING POTENTIAL AND PROTECTION REQUIREMENTS (SECTION 4.2)

- TOPIC II-3.B.1 - CAPABILITY OF OPERATING PLANTS TO COPE WITH DESIGN BASIS FLOODING CONDITIONS (SECTION 4.3)

- TOPIC II-3.C - SAFETY-RELATED WATER SUPPLY (ULTIMATE HEAT SINK (UHS)) (SECTION 4.4)

STATUS

- LICENSEE HAS RECENTLY COMPLETED FINITE ELEMENT ANALYSIS THAT ESTIMATES SEICHE FLOODING LEVEL OF 588.4 FT. MSL.

- STAFF ANALYSIS ESTIMATES SEICHE FLOODING LEVEL OF 597.1 FT. MSL.

- SAFETY-RELATED EQUIPMENT IS PROTECTED TO 594.7 FT. MSL.

ANALYSIS SUBMITTED MARCH 23, 1982

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

STATUS

LICENSEE IS PERFORMING ANALYSIS OF:

1) RADIOGRAPHY REQUIREMENT ON CATEGORY A, B AND C WELD JOINTS

2) FRACTURE TOUGHNESS - TO IDENTIFY FRACTURE TOUGHNESS REQUIREMENTS FOR COMPONENTS NOT EXEMPTED FROM FRACTURE TOUGHNESS REQUIREMENTS, I.E., CARBON STEEL COMPONENTS

3) CLASS 1 VALVES TO VERIFY THAT STRESS LIMITS MEET CURRENT CRITERIA FOR BODY SHAPE AND SERVICE LEVEL C CONDITIONS

   CLASS 2 AND 3 VALVES TO VERIFY THAT PRESSURE RATINGS ARE COMPARABLE TO PRESENT STANDARDS

4) PUMPS AND STORAGE TANKS TO EVALUATE QUALITY STANDARDS USED FOR PUMPS AND VALVES NOT DESIGNED TO ASME CODE

5) STORAGE TANKS TO VERIFY THAT COMPRESSIVE AND TENSILE STRESSES ARE WITHIN CURRENT ALLOWABLE LIMITS

ANALYSIS COMPLETION AND SCHEDULE FOR IMPLEMENTATION OF MODIFICATIONS - MAY 31, 1982

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

STATUS

LICENSEE IS ANALYZING APPROXIMATELY 200 PIPE BREAK LOCATIONS IN ACCORDANCE WITH NRC GUIDANCE

ANALYSIS COMPLETION AND SCHEDULE FOR IMPLEMENTATION OF MODIFICATIONS - JULY 30, 1982

• TOPIC III-6, SEISMIC DESIGN CONSIDERATIONS

STATUS

LICENSEE IS PERFORMING ANALYSIS OF:

1) SAFETY RELATED PIPING LESS THAN 2½ IN. TO VERIFY THAT ECCENTRIC LOADS FROM LARGE VALVE OPERATORS WOULD NOT CAUSE PIPING FAILURES AND THAT VALVES WILL REMAIN FUNCTIONAL UNDER SEISMIC LOAD

2) SAFETY RELATED ELECTRICAL COMPONENTS

ANALYSIS COMPLETION AND SCHEDULE FOR IMPLEMENTATION OF MODIFICATIONS - JULY 30, 1982

- TOPIC III-7.B, DESIGN CODES, DESIGN CRITERIA, LOAD COMBINATIONS AND REACTOR CAVITY DESIGN CRITERIA

STATUS

LICENSEE IS EVALUATING THE CODE, LOAD AND LOAD COMBINATION CHANGES ON EXISTING "AS BUILT" STRUCTURES TO ASSESS THE LEVEL OF CONSERVATISM IN DESIGN

ANALYSIS COMPLETION AND SCHEDULE FOR IMPLEMENTATION OF MODIFICATIONS - MAY 31, 1982

- TOPIC VIII-4, ELECTRICAL PENETRATIONS OF REACTOR CONTAINMENT

STATUS

LICENSEE IS PERFORMING ANALYSIS OF:

- INTERRUPTING CAPACITY FOR ALL POWER CIRCUIT PENETRATIONS
- INTERRUPTING CAPACITY FOR SAMPLED CONTROL AND INSTRUMENT CIRCUIT PENETRATIONS
- SURVEILLANCE TESTING FOR CIRCUIT PROTECTIVE DEVICES
- MODIFICATIONS NEEDED TO CONFORM TO CURRENT LICENSING CRITERIA

ANALYSIS COMPLETION AND SCHEDULE FOR IMPLEMENTATION OF MODIFICATIONS - SEPTEMBER 1, 1982

- TOPIC IX-3, STATION SERVICE AND COOLING WATER SYSTEMS (SECTION 4.2.7.1, COOLING OF CCW HEAT EXCHANGER)

STATUS

LICENSEE IS PERFORMING AN ANALYSIS OF SERVICE WATER FLOW REQUIREMENTS, AND IF NECESSARY, SHALL DEVELOP PROCEDURES TO ENSURE ADEQUATE SERVICE WATER TO COMPONENT COOLING HEAT EXCHANGERS IN CASE OF LOSS OF OFFSITE POWER AND FAILURE OF DIESEL 1-2

SCHEDULED COMPLETION - MAY 31, 1982

- TOPIC IX-5, VENTILATION SYSTEMS
(SECTION 4.2.8.1, VENTILATION OF AUXILIARY FEEDWATER PUMP ROOM)

STATUS

LICENSEE WILL DEMONSTRATE, BY TEST OR ANALYSIS, THE OPERABILITY OF THE AUXILIARY FEEDWATER PUMPS WITH LOSS OF VENTILATION

SCHEDULED COMPLETION - SEPTEMBER 30, 1982

(SECTION 4.2.8.2, VENTILATION OF CABLE SPREADING, SWITCHGEAR AND BATTERY ROOMS)

STATUS

LICENSEE WILL DEMONSTRATE THAT EQUIPMENT SERVICED IN THESE ROOMS WOULD NOT BE ADVERSELY AFFECTED BY LACK OF VENTILATION

SCHEDULED COMPLETION - SEPTEMBER 30, 1982

* TOPICS WITH DIFFERENCES BETWEEN CPCo AND STAFF

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TOPIC III-3.C - INSERVICE INSPECTION OF WATER CONTROL STRUCTURES

SECTION 4.7.1 - COOLING WATER SYSTEM STRUCTURES INSPECTION

ISSUE	STAFF POSITION	LICENSEE POSITION
INSPECTION OF - STEEL GRILL STRUCTURE - INTAKE BELL - MOUNDED STRUCTURE OF RIPRAP AND CEMENT FILLED SACKS OVER 11FT. DIAMETER INTAKE PIPE	ANNUALLY-AFTER EACH ICE SEASON	EVERY 5 YEARS

BASIS FOR STAFF POSITION

- ICE FLOES HAVE DAMAGED PREVIOUS STRUCTURE
- INTAKE CRIB ONLY SEISMICALLY QUALIFIED SOURCE FOR ULTIMATE HEAT SINK
- FAILURE OR BLOCKAGE OF ULTIMATE HEAT SINK COULD AFFECT SAFE SHUTDOWN

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

SECTION 4.11.3 - INSPECT TENDON-END ANCHORAGES

<u>ISSUE</u>	<u>STAFF POSITION</u>	<u>LICENSEE POSITION</u>
INSPECTION OF CONCRETE SURROUNDING TENDON-END ANCHORAGES	INSPECT CONCRETE SURROUNDING TENDON-END ANCHORAGES DURING INTEGRATED LEAK RATE TEST (ILRT) IF NEW CRACKS NOTED DURING TENDON INSERVICE INSPECTION (ISI)	MAP CRACKS DURING ISI AND NOTE DIFFERENCES FROM PREVIOUS ISI - TAKE APPROPRIATE ACTION IF GREATER THAN NORMAL CRACKING OR MOVEMENTS NOTED

<u>BASIS FOR STAFF POSITION</u>

- THE REGULATORY POSITION OF R.G. 1.35 SAYS CONCRETE SURROUNDING TENDON-END ANCHORAGES SHOULD BE EXAMINED DURING <u>EVERY</u> ILRT

- STAFF POSITION SAYS EXAMINE DURING ILRT <u>ONLY</u> IF NEW CRACKS ARE NOTED DURING ISI

- STAFF BELIEVES THAT SEVERITY OF CRACKS CAN BE MORE ACCURATELY DETERMINED WHEN CONTAINMENT IS UNDER PRESSURE

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TOPIC III-7.C - DELAMINATION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (SECTION 4.13)

ISSUE	STAFF POSITION	LICENSEE POSITION
INSPECTION OF CONTAINMENT FOR DELAMINATION	INSPECT DOME FOR DELAMINATION EVERY 5 YEARS	PERFORM ONE MORE INSPECTION IN JULY 1982 AND WHENEVER CORRECTIVE RETENSIONING IS REQUIRED ON 5% OR MORE OF TOTAL DOME TENDONS

BASIS FOR STAFF POSITION

- RADIAL RESTEEL NOT IN DOME
- DELAMINATION OF THE DOME AFFECTS DESIGN PRESSURE RATING
- DELAMINATION MAY OCCUR WITHOUT RETENSIONING
- MUST BE COGNIZANT OF CONTAINMENT INTEGRITY
- COST TO PERFORM MINIMAL

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TOPIC IX-6, FIRE PROTECTION
(SECTION 4.29.1, INSTRUMENTATION FOR REACTIVITY CONTROL AND REACTOR COOLANT MAINTENANCE)

<u>ISSUE</u>

ALTERNATE SHUTDOWN CAPABILITY IN CASE OF FIRE

<u>STAFF POSITION</u>

INSTRUMENTATION AS FOLLOWS:

1) SOURCE RANGE FLUX MONITOR
2) STEAM GENERATOR PRESSURES
3) PUMP FLOW RATES FOR
   - A) CHARGING PUMPS
   - B) SERVICE WATER PUMPS
   - C) COMPONENT COOLING PUMPS
4) TANK LEVELS FOR
   - A) BORON ADDITION TANK
   - B) SIRW TANK
   - C) COMPONENT COOLING SYSTEM SURGE TANK

BASIS FOR STAFF POSITION
SECTION III.L.2.d OF APPENDIX R (10 CFR 50)

<u>LICENSEE POSITION</u>

A-35-5

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

APR 30 1982

Docket No. 50-255
LS05-82

APPENDIX XXI
NRC STAFF CONSULTANTS' COMMENTS

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

Dear Dr. Shewmon:

SUBJECT: NRC STAFF CONSULTANTS' REVIEW OF PALISADES DRAFT INTEGRATED PLANT SAFETY ASSESSMENT REPORT

PALISADES PLANT

Enclosed for the information of the Committee are twenty (20) copies of NRC staff consultants' comments on their reviews of Draft NUREG-(0820), Integrated Plant Safety Assessment for Consumers Power Company's Palisades Plant. The staff's consultants for this review are:

Dr. Robert J. Budnitz
Dr. Stephen A. Bush
Dr. Joseph M. Hendrie
Dr. Herbert S. Isbin
Dr. Zenon Zudans

Sincerely,

Gus C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
As stated

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BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

Upton, Long Island, New York 11973

Department of Nuclear Energy

(516) 282-2443
FTS 666-

April 27, 1982

Mr. William T. Russell, Chief
Systematic Evaluation Program Branch
Mail Stop 516
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

REF: INTEGRATED PLANT SAFETY ASSESSMENT, PALISADES PLANT, SYSTEMATIC EVALUATION PROGRAM

Dear Bill:

This letter is my technical evaluation report on the Palisades Integrated Plant Safety Assessment as set down in NUREG-0820 (the draft report). It fulfills the requirements of Task 1 of the project "Consultant Services to Review SEP Integrated Plant Safety Assessment Reports," FIN A-3367, B&R No. 20-19-20-21-1.

CONCLUSIONS

I believe the Systematic Evaluation Program, as represented by the Draft Integrated Plant Safety Assessment Report on the Palisades Plant, is fulfilling the intent of the commission when it authorized Phase II of the program in late 1977. I consider the staff recommendations for backfitting (and in other areas for no backfitting) for the Palisades Plant to be reasonable and appropriate and the bases upon which those recommendations are made to be adequate.

At this stage of the Palisades evaluation, several of the staff recommendations (requirements, really) are for further analysis, evaluation, and testing by the licensee. When the results of these efforts are in hand, decisions will have to be made about possible equipment or procedural backfitting. These decisions should be made on the same integrated assessment basis as those reported in the draft report. These "further evaluation" topics will need to be resolved before any proceedings on the full term operating license.

A number of topics that had been listed among the 137 safety topics to

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be considered in the Systematic Evaluation Program reviews are currently being treated generically as Unresolved Safety Issues or as Three Mile Island Action Plan items and are, therefore, not included in the Palisades-specific review work reported in NUREG-0820. Also, there are two topics being treated generically, and thus outside the SEP Program, under other programs. One of these is the fire protection of associated circuits, being treated under implementation of Appendix R, 10CFR50, and the other is failure of main feedwater isolation, being treated under IE Bulletin 80-04. Palisades-specific resolutions for each of these topics will be needed eventually. With regard to the full term operating license, those Palisades-specific resolutions will either have to be in hand before any proceeding on the full term operating license, or the Commission will have to explicitly exclude them from such proceedings.

DISCUSSION

THE OVERALL PROGRAM

The Systematic Evaluation Program as it now functions was established late 1977, soon after I joined the Commission. Earlier work by the staff, in 1976, had resulted in Commission approval of a program to evaluate operating power reactors with respect to then-current licensing criteria, and to document the results of those evaluations and the need for any plant changes. The staff was told to prepare a list of safety topics to be considered under the program and to report back to the Commission. The staff did this in late 1977 and proposed a specific group of eleven older operating plants to be reviewed in what was called Phase II of the program. The objectives of the program were, and are, to (1) assess the safety adequacy of operating plants, (2) establish documentation to show compatibility with current requirements or justification for deviations, (3) make "integrated and balanced" decisions on backfitting, (4) give early identification and resolution of significant deficiencies, and (5) use resources efficiently and minimize impacts on staff and industry.

The need for some sort of safety review of the older plants in particular had been obvious for some time. The ACRS had been recommending a systematic review of operating plants for many years. There were always questions arising, particularly with Congressional staff and committees, about whether the older plants, designed and constructed to an earlier set of safety standards, still met the Commission's current regulations. Along with the need for some assessment of the safety adequacy of the older operating plants, it was also clear that it would be most useful to have current documentation which would show the compatibility of the design of those plants with current criteria or the basis for permitting deviations to exsist.

The assessement of safety adequacy by the staff for the Palisades plant, as evidenced in the draft report NUREG-0820, is, in my view, a competent and sufficient job on the items covered thus far. There remain, of course, the Unresolved Safety Issue topics and the Three Mile Island Action Plan topics for Palisades, as well as the two other generic items being pursued outside the SEP. Assuming that these will be treated for Palisades in the same fashion as the topics that have been reviewed and reported upon thus far, the safety assessment work has been a thorough and careful job that meets the Commission's intent in this area.

Not every item conceivably related to safety at a nuclear plant is encompassed in the SEP, of course. The original culling of more then 800 possible items for consideration down to the final Phase I list of 137 topics indicates that very clearly. But, in my view, all of the important safety matters are being covered under the SEP for these older plants, and that is what the Commission wanted.

The documentation of the assessment, which is mainly in the safety evaluation report letters, one for each of the SEP topics dealt with in the Palisades review, seems to me to be sufficient for the purpose. NUREG-0820 summarizes the assessment, deals with each of the 31 safety topics on which there were deviations and for which questions of backfitting arose, and includes in its appendices the Sandia report on probabilistic risk assessments of some topics and an Oak Ridge report on the operating history at Palisades. I presume that a supplement or supplements to NUREG-0820 will be issued to cover the outcome of the Unresolved Safety Issue and Three Mile Island Action Plan items and also the results of current analyses and evaluations being carried out by the licensee.

A major element in my own approval of the Systematic Evaluation Program in 1977 was the proposition that these plant reviews would be done on a integrated and balanced basis in recognition of the fact that they were dealing with plants that had been operating more or less successfully for some time. I would not have agreed to an SEP in which the review was to be done as if it were a new license, item by item, with all of the i's dotted and t's crossed. What was needed in my view, if the work was to be done at all, was an overall safety assessment of the plant as an entity, looking for places where safety upgrading was clearly needed. After reviewing the Palisades documents, I conclude that the SEP staff has done a good job in performing that sort of "integrated and balanced" assessment.

Of the other two program objectives, early identification of significant deficiencies and efficient utilization of resources for both staff and industry, I note that no urgent safety deficiencies were found at Palisades, so there was no need to exercise that objective. As for the last objective, I am inclined to think the utilization of resources has been done reasonably efficiently, although the job has taken a lot longer then originally projected. Three Mile Island bears a substantial responsibility for this, of course.

I conclude that the staff, in carrying out the Systematic Evaluation Program assessment of the Palisades Plant, has fulfilled the Commission's intent as reflected in the major program objectives laid down in the staff papers that are the basis for the SEP.

POL-FTL CONVERSION

One of the Commission's aims in establishing the SEP was that the safety assessment work and its documentation would serve as a primary basis for the conversion of provisional operating licenses held by five of the plants in the Phase II program to full term operating licenses. I recall, in fact, that this was a major consideration for me in approving the program. Those provisional operating licenses, automatically renewed every 18 months, had long been an embarrassemnt. Conversion to full term operating licenses was going to be necessary at some point, and the sooner the better. The SEP effort offered precisely the kind of safety review that was needed for the conversion, that is, one which took an integrated view of the whole plant and its operations from a safety standpoint.

The material at hand from the Palisades SEP review will be the primary documentation of the staff work and the plant status in going forward with conversion of the Palisades Provisional Operation License. The material developed thus far will serve the purpose, I think. This objective of the SEP, then, is also being achieved. There are various parts of the review that are still to come, of course. The results of licensee evaluations, analyses, and tests now being done will have to be considered and any possible backfitting matters settled and documented in a supplement to NUREG-0820. There are two matters being treated generically under other programs that are, nevertheless, listed among the Palisades SEP topics. These are the fire protection of associated circuits and main feed Water isolation. I presume that both of these will be resolved for Palisades specifically and the results of those resolutions included in the Palisades POL-FTL conversion documents. Then there are the major outstanding items, the Unresolved Safety Issue topics and the Three Mile Island Action Plan topics that are being treated generically under those two programs and outside the SEP. These include important safety topics for Palisades.

The POL-FTL conversion proceedings for Palisades cannot go forward until the results of the USI and TMI resolutions for Palisades are documented, unless the Commission specifically removes these matters from consideration in the POL-FTL conversion proceedings. This latter course is a possible one and would be justifiable on the basis that when the generic resolutions of the USI and TMI topics are achieved, the operating license (possibly a full-term license by that time) for Palisades would be amended to include those resolutions.

Although it is a possible course and could be justified as noted, I am inclined against it if there is any hope of achieving Palisades-specific resolutions of the outstanding USI and TMI topics. The reason is that setting them aside for later treatment as license amendments exposes the process to a second possible hearing when the USI and TMI amendments to the license are imposed. I expect that on most occasions these days when the opportunity for a hearing is offered, there will be a hearing. So, it would be handy all around if the USI and TMI outstanding topics could be resolved for Palisades on a schedule that would allow their inclusion in the proceeding on the POL-FTL conversion.

THE STAFF SAFETY REVIEW

The Palisades Plant was reviewed against the 137 SEP safety topics. These are listed in Appendix A of the draft report. These 137 topics were sorted out in early 1977, following Commission approval of the initial SEP proposal and in preparation for the October 1977 paper to the Commission. In spite of their age, it strikes me that the 137 safety topics still form an appropriate list of areas for review of these older plants.

Of the 137 topics, 23 are not applicable to Palisades and were deleted from the review. In addition, 24 topics were deleted from the Palisades review because they are being covered generically under the Unresolved Safety Issues or the Three Mile Island Action Plan programs. The remaining 90 topics are reported upon in the draft report NUREG-0820.

The 137 SEP topics are heavily oriented toward design matters. Only three are specific to the operation of the plant per se. These are XIII-1, Conduct of Operations; XIII-2, Safeguards/Industrial Security; and XVII, Operational Quality Assurance Program. The first, Conduct of Operations, is a TMI issue and is not treated in the draft report. (There is, however, a report commissioned by the SEP from Oak Ridge National Laboratory on the Palisades operations: This is given in Appendix F of the draft report.) The other two topics are covered in the Palisades review and both were found to be satisfactory.

The 90 topic reviews came out in one of three ways: (1) Palisades either is consistent with, or equivalent to current licensing criteria. 57 of the topic reviews came out this way. (2) Palisades is not consistent with current licensing criteria, but the licensee has implemented or committed to implement equipment or procedural changes that make it consistent with or equivalent to current criteria. Two topics came out this way. (3) Palisades is not consistent with current licensing criteria and the topic was turned over to a staff team for a integrated assessment and possible backfitting recommendations. 31 of the topics fell into this category. No urgent safety problems were identified of a nature that required immediate action. Current licensing criteria are taken from the current Standard Review Plan (July, 1981).

In 14 of the 31 topics for which backfitting was a possibility, a probabilistic risk assessment was found to be possible either on the whole topic or on some subsection of it. The risk assessment was done on a relative basis by Sandia Laboratories. Sandia compared the Palisades as-is system with a backfitted system to obtain a measure of the reduction in risk (primarily in the probability of occurrence) that might follow from backfitting. The Sandia report is included in NUREG-0820 as Appendix D. Since there is no complete probabilistic risk assessment for Palisades, or even for a Combustion Engineering plant, the Sandia work had to depend on an unpublished risk assessment for Calvert Cliffs as a baseline. The resulting assessments of safety importance and of benefit in risk reduction from backfitting are necessarily rough but are still useful inputs to be considered in the overall assessment of the topic.

The results of the integrated assessment of the 31 safety topics in which Palisades had significant deviations from current licensing criteria may be tallied as follows. The 31 topics include a number of topics which have several sections that had to be treated essentially as separate reviews. If one counts all of these separable issues, the 31 topics become 58 subtopics or issues. Of the 58 issues considered:

- 2 are being treated generically outside the SEP,
- 23 were found to require no backfitting measures, and
- 33 were found to require some backfitting measures.

Of the 33 issues that were found to require some backfitting measures:

- 7 required equipment changes or additions,
- 14 required procedural changes or additions, and
- 12 required further analysis, evaluation, or testing, which could lead in turn to requirements for equipment or procedural changes or additions.

Of the 21 equipment or procedural changes and additions, 12 led to new Technical Specifications being required.

In addition, during the review, the licensee made or committed to various equipment changes and modifications under 5 topics. These would have added to the 31 topics or the 58 issues if they had not been fixed during the review.

So, the integrated assessment team has, thus far, required equipment changes or additions in only 7 out of 58 safety issues before it. In addition, in 14 cases, issues were settled by procedural changes or additions. Nevertheless, it seems clear that the integrated assessment team has not come down blindly for backfitting no matter what the cost or safety benefit.

The October, 1977 staff paper, which was the basis of Commission approval of Phase II of the SEP, noted that when deviations from current licensing criteria were identified there were a number of alteratives or combinations of the same that would be considered as a basis for acceptability. These included acceptance of the deviation as not significantly decreasing the safety level, use of non-safety grade systems to perform safety functions, administrative or procedural changes to enhance safety system reliability, augmented surveillance programs for the same purpose, and selected backfitting. Deviations from current criteria were to be acceptable if the staff evaluation showed that the plant would respond satisfactorily to the various design basis events and the probability of those or the consequences were not significantly higher than for plants licensed in accordance with current criteria.

In reviewing the safety evaluations of the 31 topics where significant deviations have been identified, I conclude that the SEP staff has followed that direction faithfully. They have looked carefully at the risk reduction and safety benefit to be achieved by any changes and have utilized all of the available alternatives in arriving at these final judgements. I think the staff bases for requiring equipment changes or modifications in the few cases where that has been done, for requiring procedural changes in other areas, and for concluding that no backfitting is required in yet other areas are adequate and reasonable and are consistent with the Commission's directives of long ago. I am particularly pleased to see the staff willing to declare that there is no need for backfitting in those cases where it offers little or no reduction in risk and would have substantial impact on the plant if required. That has not always been a characteristic of staff reviews.

It is going on 5 years since the SEP Phase II came before the Commission for approval. I voted for it with a certain amount of trepidation. I had some concern then over the staff's ability to do a balanced assessment on an older operating plant and to come up with results that were meaningful from a safety standpoint and did not simply end up requiring total conformance with current criteria regardless of the safety benefits. That concern did not abate much in the years following and I used to confront the bright-eyed proposers of an SEP Phase III with the direction to go back and produce something from Phase II and then we would see. Now we have the first product from Phase II. I think it is a good job. My compliments to the staff.

Sincerely,

Joe

Joseph M. Hendrie

A-36Y

PALISADES PLANT

A CRITIQUE OF THE INTEGRATED PLANT SAFETY ASSESSMENT
SYSTEMATIC EVALUATION PROGRAM

S. H. Bush

Since Palisades is the first plant reviewed under the Systematic Evaluation Program, the approach taken and the criteria used to establish the acceptability of assessment are somewhat tentative, particularly because there has been no opportunity to interface with authors and other reviewers. Two suggested benchmarks are:

- Does the report meet the original AEC/NRC Commission Charter for SEPs.
- Are the items identified as problems adequately described, including justification of their resolution.

An examination of documents SECY-76-545 and SECY-77-561 provided some insight into the approach used to handle SEP plants. The five program objectives can be used as criteria for measuring compliance. The suggested approach for handling deviations can permit an assessment of the resolutions suggested in the Palisades report. These criteria follow.

The following five objectives of the program were established by the Task Force:

1. The review program must assess the adequacy of the design and operation of all currently licensed nuclear power plants.
2. The program should establish documentation which shows how each operating plant compares with current criteria on significant safety iss s, and provide a rationale for acceptable departures from these criteria.

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3. The program should provide for the capability to make integrated and balanced decisions with respect to any required backfitting.

4. The program should be structured for early identification and resolution of significant deficiencies.

5. The program should efficiently utilize available resources and minimize requirements for additional resources by NRC or industry.

The planned systematic evaluation would establish the adequacy of all operating power reactors with respect to safety and provide clear written documentation bases for this conclusion.

When deviations from current licensing criteria are identified, the following alternatives (or combinations of alternatives) will be considered as a basis for establishing acceptability:

1. The deviation can be justified as not significantly decreasing the level of safety.

2. Use of non-safety systems to perform safety functions.

3. Administrative or procedural changes to enhance system reliability.

4. Augmented surveillance programs.

5. Selected backfitting to enhance system reliability.

Presumably one critical evaluation of Appendix A will be sufficient on the assumption that these items will remain unchanged in the future. While Appendix B covering generic issues may change somewhat, one review as to adequacy should be sufficient. Obviously, Appendix C will change because of plant and site specificity. Appendices E and F will need review on a case-by-case basis.

Examination of Appendices A, B, and C unearthed some problems. The wording, references and approach used with the items in Appendix A reveal the "mind set" of the 1976-77 period. Personally, I feel that some of the strong positions taken then have weakened in the past 4-5 years. An example might be valve lockout. As predicted some of the locked out valves have been found to be in the wrong position so the effects of an accident would be exaggerated.

I suspect a probabilistic approach could lead to dropping others; however, the option appears to exist in the so-called "lesser safety significance" approach.

With regard to Appendix B as related to A, I am at a loss as to why some of the unresolved safety issues were ignored. Specifically, issues A-11, A-31, A-45 and A-49 were not cited. If these were included, some other items would shift to the generic packet. While I understand the words regarding folding in the USI and TMI issues, it is not immediately obvious how this will be accomplished.

I suspect that the issues in Appendix A, if written in 1982-82, would differ substantially from the words generated in 1977; however, those words can be accepted.

SECTION 1

An item of major concern becomes apparent in the listings on page 1-7 and in Appendix F. While the number of LERs arising from personnel or procedural errors is not large, the safety significance of some of the events is substantial, particularly with regard to loss of containment integrity and improper positioning of safety-related valves. These events extend over a sufficiently long period that is indicative of an indifference on the part of top management to take appropriate action. In my opinion the document does not stress this area sufficiently. Unless there is positive evidence of an improvement in operator actions, I question approving a full-term operating license.

SECTION 2

Explanatory only--no comments.

SECTION 3

The positive actions taken to resolve issues III-6, VII-3, VIII-2, VIII-3B and VI-6 are considered appropriate. My personal opinion is that some of the changes under III-6 may not have contributed much to plant safety.

SECTION 4

In essence, this section represents the actions and bases for the actions taken including a factoring in of the PRA in Appendix B.

II-1-A no comment; no problem.

II-3B, B1.C Pending; probable backfit.

III-1 Positive actions that should provide missing information and enable decision as to acceptability of various items.

III-2 A good example of accepting alternate approaches when deviation occurs. Instead of backfitting, it is recognized that sources of water can be made available. Emphasis is on clearly defined procedures covering use of alternate water sources than on upgrading or backfitting.

III-3-C The positions of staff and utility are apparent. I would have thought this to be an economic problem that would become apparent during operation rather than under accident conditions. I agree with staff.

III-4-A I applaud the decision not to backfit. It's appropriate.

III-5-A, III-6 I disagree on philosophic grounds with this item. In ten years of review I have yet to find a case where piping failed from seismic loads and no breaks result from an unrealistic application of the design load cycles. Current analytic technique yield a false picture of piping response that seemingly is not recognized.

III-7-A No disagreement--okay.

III-7-B Primarily a bookkeeping activity to provide analytic answers.

III-7-C I understand the need to do another examination for delamination. I do not understand an arbitrary five-year repeat. We don't require that on embedded flaws in vessels.

III-8-A May shift to generic.

V-5 A realistic approach. I agree with staff analyses.

V-10-B Action taken resolves issue.

V-11-A This had potential to overpressurize and fail piping. The action only resolves it partially since case of re eased flapper is not covered.

VI-2-D, VI-3 I agree with decision and PRA value. No action required.

VI-4 Removal of threaded piping is appropriate. Other decisions acceptable.

VI-6 Forced action taken--no issue.

VI-10-A No action.

VII-1-A A good example of use of PRA to require revision or accept status quo.

VII-3 DC power obviously is important. Basically handled as generic problem. Other actions based on a realistic assessment of tradeoffs.

VIII-3-A Important issue. Must assume loss of diesel generator plus offsite power.

VIII-4 Action taken.

IX-3 Presumably fix will be procedural in nature. Not clear. Second item procedural plus modification.

IX-5 Analytic only--not complete.

IX-6 In essence a generic backfit item.

XV-2 I am not surprised regarding the uncertainty in failure rates. Basically, this will be handled generically.

XV-12 A realistic approach to the problem.

With regard to equipment and design items, the authors addressed to a major degree the SEP task force objectives as well as applying the tiered criteria to resolve deviations. Generally, the approach is even-handed, not requiring backfit arbitrarily. I am less satisfied with the handling of operating history.

Appendix F points out the high incidence of loss of power. This combined with some of the operator errors listed could yield a definite degradation in safety margins.

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Room 418

15 April 1982

RECEIVED
APR 19 1982

Mr. William T. Russell
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear ~~Mr. Russell~~ Bill:

This letter comprises my report to you under Purchase Order # DR-82-0961, in which my assignment has been to review Draft Report NUREG-0820, "Integrated Plant Safety Assessment: Systematic Evaluation Program, Palisades Plant". As you know, I received a copy of this report personally on April 2nd, when I was visiting Bethesda on other business. I arrived back here in Berkeley on April 7th, and after verbal authorization from Ms. Arlene McNulty of NRC Division of Contracts on April 9th, I began the review process in earnest. Unfortunately, I am departing on April 16th (tomorrow) for a two-week business trip, so my review has had to be squeezed into the few days between April 9th and 15th, plus the time I put into it after I got the draft report on April 2nd.

Regarding the mission of the SEP, I have used as primary references a pair of Commission papers (SECY-76-545 and SECY-77-561) that you furnished. I understand that these together comprise the 'charter' for the SEP effort. Of course, during my two years at NRC (1978-80) I learned a lot about the SEP and therefore have considerable additional background as to its goals, methodology, and constraints.

While I have nothing in writing telling me my own scope of work, I have read the scopes of work for two other reviewers (Drs. Bush and Hendrie), and I have assumed that my own scope is identical. I understand that the objective is "to provide an evaluation of the adequacy of the rationale used by the staff in identifying and making recommendations for backfit requirements." I have interpreted this charter slightly more broadly, and my comments will reflect my broader interpretation. To be specific, I will provide discussion touching on each of the following questions:

1) Is the Palisades SEP report asking and answering the right questions ?
2) What implicit policy-type decisions do I detect in the report ? Is their rationale appropriate ?
3) Is the review methodology appropriate ?

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4) Are important issues left out of the review ?

5) What has been included in the review that might have been omitted without significant compromise ?

6) How adequate is the rationale used to identify and recommend backfit requirements ? (This question is the specific objective of the review.)

Because I have been short on time, I have not provided herein any comments on specific safety topics. I do have several specific comments, with varying degrees of importance, which I will assemble into a coherent package during my trip over the next two weeks. If it seems useful later, I can provide additional material to you upon my return, after May 2nd.

A. Is the Palisades SEP Draft Report asking and answering the right questions ?

In a narrow sense, I believe that the answer to this question is affirmative: that is, the original charter seems to emphasize reviewing the older plants against modern review criteria (the modern Standard Review Plan, modern regulatory guides and standards, etc.), with the subsidiary goal that for plants with Provisional Operating Licenses the SEP review would form part of the basis for conversion to Full Term Operating Licenses. To the extent that these objectives have guided the SEP effort, they have been quite successfully fulfilled, in my view. I find that there has been a systematic analysis of the areas where the Palisades plant review, were it being conducted by the NRC staff today, would have been different: in some areas the review procedure would have been different, and in others the plant or its operating procedures would have been different.

When I used the word 'narrow' in the first sentence of the paragraph above, I meant it in its purest form: that is, I realize that it is extremely important that every item on the 'list' be discussed properly, and its resolution documented. I find that the draft report has accomplished this effectively. As one who has generally been uncomfortable with the (apparently ubiquitous) need to get papers into the file covering every gnat's eyebrow, I find the detail contained in some of the explanations and resolutions to be a little extreme .... but I do recognize the legitimate reason for this, namely that bringing Palisades into line and up-to-date with the large number of newer operating units is important in its own right.

My discomfort arises from the following perception: I personally believe, and have believed for some time, that plants such as Palisades have been built and operated in a manner that assures "adequate protection of the public health and safety", in the sense that the NRC Commissioners and staff have used that phrase or its generalized counterparts like 'no undue risk'. However, this belief does not rest upon specific, well-founded grounds except for the strongly affirmative safety record of the industry to date; therefore, analyses that tend to confirm it are always important. In this regard, one philosophical rationale for undertaking the Systematic Evaluation Program has been to look at older plants like Palisades, attempting to uncover any safety concerns that might cast doubt on the "adequate protection" determination. In this regard I have reached two conclusions after reading the draft SEP report. The first is that none of the safety issues treated seem in my view to have turned out to be highly important to safety, after analysis .... and this conclusion comforts me a good deal: I can almost hear myself breathing more easily. Second, and in some ways more significant, is my conclusion that a few quite important safety issues are absent from the analysis in this draft report ! I elaborate as follows: if somebody asked me for my personal opinion about what safety issues might compromise the judgment that Palisades poses no undue risk, I would list several items that are broadly encompassed by the USI (Unresolved Safety Issues) and TMI (Three Mile Island Action Plan) categories. Thus, I continue to be worried about things like systems interactions (USI A-17), station blackout (USI A-44), control systems issues (USI A-47), the full range of human factors concerns, and the dependency of safety systems on crucial support functions like instrument air, service water, and electrical distribution buses of uncertain reliability.

The fact that NRC is systematically addressing these USI and TMI issues gives me comfort. In my view it is very likely that all of them will be resolved sooner or later, that all of our plants will somehow be safer because of it, and that the safety imporvements will be highly cost-effective. Nevertheless, I believe that the draft report I have in front of me is somehow inadequate or insufficient to the extent that it does not highlight this key point. I would feel better if the report had something like the following, up front somewhere, to guide the reader:

> "The regulatory staff recognizes that several of the most important safety issues have not been addressed or resolved in the course of this SEP effort, in each case because they are being coped with through other regulatory efforts: in particular, the Unresolved Safety Issues list and the Three Mile Island Action Plan list contain some issues whose safety significance is probably far greater than a majority of the issues dealt with and resolved herein."

In summary, my answer to the question posed above ("Is the report asking and answering the right questions ?") is that while it asks most of the right questions, it finds itself unable to answer a reasonable fraction of them.

Phrasing my concern another way, I find a lot of what is in this draft report to be operating in the strange make-believe land of the traditional NRC approach to regulation, an approach where regulation per se survives as important separate from safety. One concrete example of this is the (luckily few) places where, instead of assuring by other means that the licensee carries out a certain procedure, the staff wants a change to the Technical Specifications. I had the impression that the staff was moving toward a philosophy of having less specificity in tech specs .... and if it isn't so moving, I believe it ought to ! Yet here is the SEP effort sticking more little stuff into tech specs. Isn't there some other, better way ?

B. What implicit policy-type decisions do I detect in the report ? Is their rationale appropriate ?

I detect several policy-type decisions that I agree with. Perhaps the most important is the general feeling that I get in reading the report that Palisades is, indeed, 'adequately safe'. This feeling pervades the text of the report as I read it, and it apparently has played a part is some of the decisions on whether urgency is required for various backfits.

Another important policy decision is the strong presence of the concepts of PRA (probabilistic risk assessment) as a valuable tool in safety decision-making. I endorse this with delight: I feel that the way PRA has been used is just about right. It has been used for its insights into relative safety importance, but for not much in the way of quantitative information. (Of course, this is partly because there has been no PRA done on Palisades itself; the Sandia-written appendix only references a comparison between systems at Palisades and at a mysteriously-unnamed different plant of CE design.) The Sandia write-up on the PRA analysis is lucid, and explicitly recognizes the major uncertainties in any quantitative conclusions.

Another key policy-level decision seems to have been that hardware fixes should be required only if no other type of backfit or procedural arrangement is available. I applaud this decision. Conscious efforts to avoid unnecessary backfits are, in my view, an important element in NRC's regaining credibility with the licensees.

C. Is the review methodology appropriate ?

I am pleased to report my finding that the methodology used in the report is appropriate and adequate for the purpose. As mentioned above, I am especially pleased to note that PRA methods have been used to rank the safety significance of several of the issues, and that PRA insights have assisted the staff in deciding on the importance or urgency of required changes. (I could quote the Lewis Report here, but I will restrain myself.)

The methodology gives different depth of treatment to issues of differing safety significance, and this is fully appropriate. I especially applaud the concept of an integrated assessment in which a large number of issues are viewed in sum rather than one-by-one. This integration affords the analyst broad insights into urgency and cost-effectiveness; and affords the reviewer or critic the chance to grasp the whole SEP analysis more fully. I think that whoever has brought about this conceptualization of the SEP effort should be congratulated for clarity of thought.

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I also find the approach used by the Oak Ridge group in analyzing operating experience at Palisades to be a good one. Their logic in identifying those events with real safety significance seems to be fully satisfactory, and there are some excellent discussions of specific topics (especially about control rod drive mechanism problems and partial/full loss of offsite and onsite power.)

The one part of the methodology that leaves me a little uncomfortable is the linkage of the analysis, at least in a structural sense, to the outmoded issues list compiled in about 1977. The list itself (the definitions in Appendix A of the report, for example) contains some examples of thinking about safety/regulation/retrofits/NRC-licensee interactions that are today outdated, or at least overtaken by the events following Three Mile Island. However, the implementation of the methodology overcomes much of the difficulty imposed by the use of the outmoded list and definitions: there are several examples of more up-to-date thinking about issues.

D. Are important issues left out of the review ?

I have already discussed my discomfort that several important issues are not analyzed in this draft report because they are being coped with through a different regulatory mechanism (USI, TMI, etc.). I understand the rationale for this, and accept it prima facie.

I also have discomfort about the omission of a collection of issues involving management. Specifically, I know that various utility managements are viewed in different ways within the NRC staff: some are thought to be more competent than others, without necessarily implying that any one or more of them are insufficiently competent. What struck me as I read this draft report is that I cannot, for the life of me, figure out from it how Consumers Power's management is viewed ! (The discussion on page 1-6, penultimate paragraph, is the only clue I found as to what NRC thinks about Palisades management.) For all I know, they are thought to be 'the best utility around', or 'the worst', or whatever. Since everybody now appreciates how crucial good management is to safety, some specific treatment of this issue would seem to be called for.

The same comment applies to utility in-house engineering competence. Some utilities have very fine engineering staffs, while others are weaker, relying instead on outside assistance. Again, I cannot figure out where Consumers Power fits into this spectrum, yet engineering competence, like management competence, looms large as a key element in safety.

Finally, the entire SEP seems to give insufficient treatment to the human factors and control systems side of safety. Even considering that many human-factors and control systems issues are bound up in the TMI Action Plan list, I would have felt better had there been more discussion of them, especially in the integrated assessment part of the report.

E. What has been included in the review that might have been omitted without significant compromise ?

In one phrase, "not much" .... except for the apparently ubiquitous need to cross all the t's and dot all the i's.

F. How adequate is the rationale used to identify and recommend backfit requirements ?

This is the question that was asked directly of me as a reviewer.

I already mentioned, and will repeat here, my finding that the rationale for decision-making is fully satisfactory. I find the staff's thoroughness, issue by issue, to be commendable. I find that the use of PRA as an aid to engineering insight is at just about the right level. I am pleased with the apparent decision not to seek hardware changes except in those few areas of high safety significance where no other remedy could be identified.

Summary

I will summarize by stating that I believe this first SEP report has been quite successful: the metaphor of the laundry list that has been cleaned up is appropriate. Maybe Palisades can get a regular operating license now, for one thing; and maybe the utility staff and the regulatory staff can go on from this mop-up activity to think hard about the real issues of safe operation of Palisades, issues hardly dealt with in the analyses within this draft report.

Finally, I do think it is important to state my view that it is only in retrospect, after the analysis, that one is at liberty to characterize the SEP list for Palisades as a 'laundry list': beforehand, we didn't really know what would crop up. So in that regard the activity has been successful indeed.

Robert Budnitz 15 April 1982
Robert J. Budnitz

W1Russell11

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April 23, 1982

To: Project Officer W. Russell
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From: H. S. Isbin H. S. Isbin

Draft
Review of Integrated Plant Safety Assessment
Systematic Evaluation Program
Palisades Plant
Draft NUREG-0820

Enclosed please find my draft review in response to your request to "...provide an evaluation of the adequacy of the rationale used by the staff in identifying and making recommendations for backfit requirements." In addition to Draft NUREG-0820, I received SECY 77-561 (October 26, 1977), SECY 76-545 (November 12, 1976), a November 15, 1977 memorandum from S. J. Chilk to L. V. Gossick, and a draft Statement of Work. I have not had the benefit of any discussions with the SEP staff nor with any reviewers. Please let me know if you desire any changes in the focus of my review.

The highlights of my review are as follows:

- The planning used for SEP is outstanding from the point of view of identifying safety items.
- The objectives have been well conceived; however one major objective may have been inadvertently omitted in the NUREG report.
- The review of operating experiences needs to be updated and augmented.
- Limited assistance was provided by the probabilistic risk assessment for this plant.
- The reporting of the Topics and the ensuing approach to the decision making, in general, are well done.
- Too many events and changes have occurred in the past three years to be able to evaluate whether the SEP program is efficiently and economically using NRC and Industry resources.
- An important finding is that no SEP Topic was considered to be of sufficient importance to require a prompt resolution.

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- Attention has been focussed on achieving an "...integrated and balanced..." decision, considering that the SEP program is being carried out in conjunction with major NRC and Industry efforts for implementing TMI Action Plan Items, and responding to IEBulletins and Generic Letters. Resolution of Unresolved Safety Issues remains a continuing activity along with mandated annual reports to the Congress concerning identification of any new issues. Overall assessment of safety of the plant must utilize all these inputs.

Additional SEP supplements have been planned for the Palisades plants. The Status of the SEP Topics, presented in Appendix A, is for the date April 1977, with some having been updated to May 1981 for inclusion of TMI tasks, USI, and several IE Bulletins. I understand that no changes were made in Topic definitions. Has consideration been given to updating all the SEP Topics regarding status and References?

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DRAFT REVIEW

by

H. S. Isbin

COMMENTS ON SECTIONS OF NUREG-0820

1.2 Systematic Evaluation Program Objectives

Only three objectives are presented. I believe that the fourth objective is essential and from the referenced material I extract

> "and (4) an overall evaluation of all safety topics evaluated in the SEP and other ongoing programs..."

I would emphasize overall, all, other ongoing programs.

I assume that the presentation of the original five SEP objectives is to augment the present objectives. Any program that seeks "...to make integrated and balanced decisions with respect to any required backfitting" and to "...efficiently use available resources and minimize requirements for additional resources by NRC or industry "merits our standing ovation.

A variety of actions taken during the last three years on generic matters, including the TMI related items, have considerably altered priorities. In my review, I have chosen to focus on what elements need to be included to achieve the "...integrated and balanced..." backfitting decisions.

1.4 Summary of Operating History and Experience

This section is inadequate because it is not updated.

The ORNL detailed review is a worthy study up to and including the year 1979 and represents an "external" appraisal. More emphasis needs to be given on implementation of corrective actions.

The summary of Escalated Enforcement Actions presents significant events through March 1981. I suggest including more "internal" reviews, including the Region III "Systematic Assessment of Licensee Performance" (SALP), and the periodic Inspection Reports by the resident inspector(s) and by the special teams. The updating might include the Licensee's Annual Report of Changes, Tests and Experiments.

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. The listing of the civil penalties and orders is not as important as the thorough evaluation of the Licensee's response in terms of corporate policies, management and staff control measures, training and requalification programs, procedures, and quality assurance. What are the lessons learned and how are the improvements being implemented? Are these actions contributing to overall plant safety?

Corrective features are embodied in the TMI Action Plan. Analysis and feedback of operating experiences must be achieved in a realistic program.

The SEP report does not reflect any improvements in training nor even the considerable augmentation of staff (which I assume must be taking place).

In my opinion, it is too early for the NRC to reference INPO reports and documents; however, if the Licensee has made or is making improvements as a consequence of INPO evaluations, such information assists in making "...balanced and integrated..." decisions.

Programs involving SEE-IN and NOTEPAD should be checked for updating operating experiences and reference should be made to the NRC Generic Letter 82-04. Have the SEP reviewers made any use of the improvements underway on handling and managing the collation of LERs? Is the Sequence Coding and Search System operational?

A feature of the SEP program that I had expected to find, but did not, is concerned with "aging" of components and systems. Have there been any discernible trends? Not all events are reportable, and thus some important trends might be missed if recourse is made to just LERs. The cooperation of the maintenance and inspection groups is needed. Are the IE Information Notices helpful in ascertaining whether any special aging effects could have an impact on safety?

2. Review Method

2.2 Selection of Topic List

The identification of the more than 800 candidate items took place in 1976 and 1977, and the methods used are impressive. The process of reducing this number to 90 topics applicable for the Palisades SEP review has an acceptable rationale, providing all current items involving Unresolved Safety Issues, TMI Action Plan Items, and other generic matters are to be included. The draft SEP report indicates that a supplement is to be issued which will designate the status of the USIs and TMI Action Plan Items. I consider this supplement to be a key factor in balancing overall decisions to be made on the SEP items. The magnitude of this task should not be underestimated.

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2.3 Topic Evaluation Procedures

The two methods used for the preparation of the final Safety Evaluation Reports for the 90 topics have involved Licensee participation to ascertain that correct information was used. I have not seen any of the SERs, but I think that the approach used is good.

The finding by the NRC Staff that no topic identified in the SEP review required immediate action is significant.

Topics were grouped into categories regarding no further action, action initiated by the Licensee which is acceptable to the NRC, and finally those which require decisions on whether backfitting is needed. This approach is logical.

2.4 Integrated Plant Safety Assessment

The overall planning of the SEP review to achieve a "...balanced and integrated..." decision on each topic appears to be good. The approach used by the NRC Staff appears to be consistent with the designated objectives.

I was not able to judge how the SEP work loads have impacted on the Licensee's resources. Nor have I been able to judge how the NRC plans to mold the backfitting decisions with decisions made and to be made on the USIs and TMI Related Safety Items. Perhaps this subject will be treated in a supplement.

COMMENTS ON SELECTED TOPICS

3.3 Topics for Which Plant Design Meets Current Criteria

Based on Modifications Implemented by the Licensee

3.3.1 Topic III-6 Seismic Design Considerations

The Licensee must be responding to IE Bulletins, dealing for example with structural integrity of masonry walls and safety related piping systems. The report would be improved if the relationships of the programs involved for the IE Bulletins to the SEP concerns were clearly presented.

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Integrated Assessment

4.6 Topic III-2 Wind and Tornado Loadings

4.6.1 Safety Injection and Refueling Water (SIRW) and Condensate Storage Tanks

The approach taken by the Staff is good.

"...if the SIRW tank or condensate storage tank is lost..."

Rather than just "or", don't you mean either or both? Further, wouldn't you need to comment that the failure(s), in themselves, do not produce any undue flooding effects?

This is one of the few items where the TMI Action Plan, Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents," is mentioned. This SEP topic is a part of a much broader and more complete task, with priority on Licensee's resources to be given to the TMI Item.

4.10 Topic III-6 Seismic Design Considerations

Once again, for an "...integrated and balanced..." approach, SEP concerns need to be factored into the broader areas being addressed through responses to IE Bulletins.

4.15 Topic V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The rationale used by the Staff in arriving at conclusions fits the objective set forth for SEP. Is there any impact on radiation exposure to workers?

4.16.2 Use of Safety-Grade Systems for Safe Shutdown

On page A-42, modification of the Branch Technical Position RSB5-1 is being suggested. What is the current status?

4.19 Topic VI-3 Containment Pressure and Heat Removal Capability

The emphasis of this resolution is on the capability of the containment to withstand the increased pressure resulting from a two-steam generator blowdown. A "...balanced and integrated..." approach should avoid exacerbating other possible issues. For example, would there by any impact on resolving the concerns presented in IE Bulletin 79-01B, dealing with the environmental conditions for the qualification of safety-related electrical equipment?

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4.23 - Topic VII-1.A. Isolation of Reactor Protection System from Nonsafety Systems, Including Qualification of Isolation Devices.

No reference is given to the completion of Technical assignment Control No. 6696, nor whether there have been any continuing studies.

4.24 Topic VII-3, Systems Required for Safe Shutdown

I had expected to find a discussion on what can be done outside the control room "...to achieve and maintain a safe shutdown condition of the plant..." (See A-66).

The SERs started with the general topic and then determined specific items. Using this Topic as an example, I suggest that both the specific and overall conclusion be stated.

4.27 Topic IX-3, Station Service and Cooling Water Systems

Again, as noted in 4.24, not all the safety objectives given for this topic (see pages A-77 and -78) are addressed. Further, under the heading of Status, reference is made to proposed generic reviews and technical activities. Were these proposals carried out? Additionally, are there any current probabilistic studies of flood hazards and flooding effects which should be noted?

4.29 Topic IX-6, Fire Protection

The application of 10CFR50, Appendix R, concerning fire protection and safe shutdown analysis and compliance, is a major undertaking. All that is noted is that associated circuits will be reviewed generically and outside the context of the SEP.

Additional Comments on Achieving An

"...integrated and balanced..." Decision

From the descriptions given of the methods used to identify topics of safety significance in Phase I of SEP, I conclude that the identification process was thorough, at least for the conditions known up to and including 1977. I appreciate the need for restricting expansion of the topics so that designated goals can be assigned. The developers of SEP recognized that resolution of generic items would proceed independently of SEP, but that somehow there would be an integrationof emerging NRC positions into an effective and efficient molding for the "...final safety assessment of the plant." Since

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the approach to the "final" safety assessment may be asymptotic, I recommend deletion of "final".

Supplements are to be issued to present the status of the generic items. Additionally, the NRC, in special reports to the Congress, prepares the "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants." For example, NUREG-0705, March 1981, identified four new USIs and listed a number of candidate issues for consideration as USIs. (I have not seen the 1982 report.) Along with the moving target of USIs, SEP reviewers need to include the TMI Action Plan Items, a variety of IE Bulletins, and Generic Letters. Attention has to be given to the management and incorporation of the information being generated into overall safety assessments.

The "...integrated and balanced approach..." should recognize the concerns of the Licensees regarding any possible unwarranted diversion of engineering staff from needed tasks. For example, see NUREG-0839, "A Survey by Senior NRC Management to Obtain Viewpoints on the Safety Impact of Regulatory Activities from Representative Utilities Operating and Constructing Nuclear Power Plants," August 1981. On the other hand, a dissenting viewpoint within the NRC, such as given by Demetrious L. Basdekas (published in the Minneapolis Star and Tribune, April 9, 1982) cannot go unchallenged, particularly when he writes that "...the government and industry are unable or unwilling to deal honestly and urgently with far reaching nuclear-safety problems." I know in the past that dissenting views were acknowledged and answered. Perhaps this has already been done for the present case.

Peripheral issues may need to be included. For example, proposed changes for Technical Specifications purport to reduce the number and level of detail in the technical portion and permit the use of a supplementary category. The criteria being developed for these changes should be consulted before implementing resolution of SEP Topics through added Tech Specs.

Only a limited, but useful, application could be made of the probabilistic risk assessment study for this plant. Considering the large NRC budgeted research in this area for the past several years, the emphasis on development of methodologies, and the various current applications, I expect more feedback into the SEP activities.

H. S. Isbin
April 23, 1982

Franklin Research Center
A Division of The Franklin Institute

Z. ZUDANS, PH.D.
Senior Vice President and Chief Operating Officer

April 28, 1982

Mr. W. Russell
SEP Project Manager/Technical Coordinator
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Review of SEP Integrated Plant Safety Assessment Report

References: 1) SECY 76-545
2) SECY 77-561
3) NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program - Palisades Plant, Consumers Power Company, Docket No. 50-255

Dear Mr. Russell:

In accordance with your request I have reviewed the Palisades SEP program and offer the following comments.

At present, the SEP Program appears to be well organized and well managed. The referenced documents summarizing Palisades SEP review provide a comprehensive historical review of the entire SEP program since its inception in February, 1978 by the NRC.

Significant amount of thought and effort has been put in development of the procedure for SEP. The procedure as shown on Figures 1 to 3 was constructed from the Referenced Documents and from various personal conversations with the SEP Program Staff. As it can be seen from Figures 1 to 3, the procedure is generally well defined and at the completion should lead to the satisfaction of the Commission's goals for the SEP program.

In order to make the Integrated Assessment process more responsive to reviewers related human factors, procedure blocks following item (1) (circled) should be provided with more specific guidelines as to the interfacing between SEP, USI, TMI Action Plan, and other personnel involved in the remaining steps of the procedure. This implies that the Final Integrated Assessment Report (FIAR) should be a joint effort of SEP, USI, TMI Action Plan, and others. Details in NUREG-0820 show that such interaction took place for topics of concern to various programs (supplement to NUREG-0820 for resolution of USI, TMI Action Plan, and other items, for example).

SEP review for Palisades, NUREG-0820, is comprehensive and engineering arguments are sound. Following this procedure, Figure 1, 24 topics exit at (1) identified as generic items related to USI and TMI Action Plan, 23 topics exit at (2) because these are not applicable to Palisades and 90 topics reach (3) where the actual review of SEP topics for Palisades begins. It is not identified in NUREG-0820 how many topics were handled by Method 1 and how many by Method 2. However, at the step A (Disposition of Topics), all but 31 were left for backfit candidacy, the remaining 59 having been put in one of the categories 1 to 3. None of the topics fell in the category 4 (i.e., safety significant departure), requiring prompt action. I find that technical arguments leading to distribution of topics to various categories are acceptable.

With respect to Topics in the group of Integrated Assessment, Sections 4.1 to 4.31 of NUREG-0820, represent the Draft Integrated Assessment Report (DIAR). Since Licensee response and resolution for most Integrated Assessment Topics is already contained in NUREG-0820, it also represents Final Integrated Assessment Report (FIAR). However, there is no integration of 24 USI and TMI Action Plan SEP Topics until supplement to NUREG-0820 is issued. This Supplement will also form the basis for conversion from POL to FTOL. In other words, it appears that for Palisades the procedure shown in Figures 1 to 3 was not followed strictly, or stated otherwise, actual Palisades review procedure indicates that plant and licensee specific circumstances may require flexibility in the procedure itself.

With respect to the specific topics reviewed in NUREG-0820, I offer the following additional comments.

For 14 of 31 topics slated for Integrated Assessment, risk assessment by Sandia (SAI) (using Calvert Cliffs unreleased PRA) provided useful insight in relative value of backfits, i.e., it provided logical support for engineering judgment in complicate situations.

Similarly, an extensive use of the plant operating experience in support of engineering judgment is probably the best decision made by the SEP staff. This practice should be followed in the review of other SEP plants.

My overall impression of Palisades SEP Review is that considerably more sound engineering effort has been put in Palisades SEP review, in particular in terms of proper understanding of design, processes and consequences involved, than maybe normally done during regular licensing review process, (SEP Topic list covers essentially all safety related design aspect of a nuclear power plant). The process however, is not complete until all open items are resolved in an integrated manner.

Topic V-11.B is listed in Section 3.1 (final list of 90 topics for Palisades), but it is not referred to in Sections 3.2 (topics meeting current design criteria), 3.3 (meets current criteria because of modifications implemented by the licensee), or Section 4.1 (Integrated Assessment Topics). I believe the substance of Topic V-11.B is addressed in Section 4.16.

I am particularly impressed with the discussion of high and low pressure fluid systems interaction of Sections 4.16 and 4.17. Of the proposed corrective alternatives, Staff's alternative 2 is the best choice in my opinion.

Section 4.7, Topic III-3.C discusses inspection program for water control system structures. I find the five year inspection frequency proposed by the licensee not technically sound (icing season comes once a year).

With respect to Topic III-5.4 (Section 4.9) I am not a strong believer of rigid piping systems. Accordingly, I suggest that the number of pipe whip supports (for postulated 200 pipe break location) should be kept at the minimum.

Section 4.11.1, Topic III-7.A, brings up an important point on need for monitoring forces in individual tendons rather than the average of all tendon forces. Relative to the concrete crack inspection at tendon anchorages, one must note that tendons are always under prestress load which is the bulk of the load ever seen by the tendon. If the tendons are lifted (for tendon force verification), load on anchorages may exceed the load applied due to pressure used for leak rate testing. Accordingly, anchorage concrete inspection should be done at the time when tendons are lifted for force testing (if they are lifted).

Relative to Topic VI-4, I like to point out that the internal pressure load on piping is less significant than the structural loads imposed by geometric constraints of attachment points, I agree with the staff that no backfitting is required.

Topic IX-3, flooding of intake structures, licensee proposes alarms in control room to indicate occurrence of flooding, presumably to give operator time to prevent inundation of service water pumps. If the flooding comes in a form of a 13 ft wave driven by seiche, alarm will not help to prevent water intake structure flooding nor can the operator do much to stop the wave.

For Topic IX-5, the Licensee has proposed (by test or by analysis) to demonstrate the operability of AFW with loss of ventilation in pump room. I doubt that anlaysis can predict realistically the time AFW will remain operable under such conditions. It might be possible to run a relatively short duration test, observe ΔT in room and equipment and make an extrapolation from the test results.

.Main steam line break in Containment (Topic XV-2) with postulated single failure is clearly an issue important to risk.

In general, NUREG-0820 provides comprehensive discussion of definition, safety objectives and status of all SEP topics. Adequate list of references is provided for each topic for detail study and for a complete documentation of the decision process.

I am well impressed with the work done by the staff on Palisades SEP review.

Very truly yours,

Zenons Zudans
Senior Vice President

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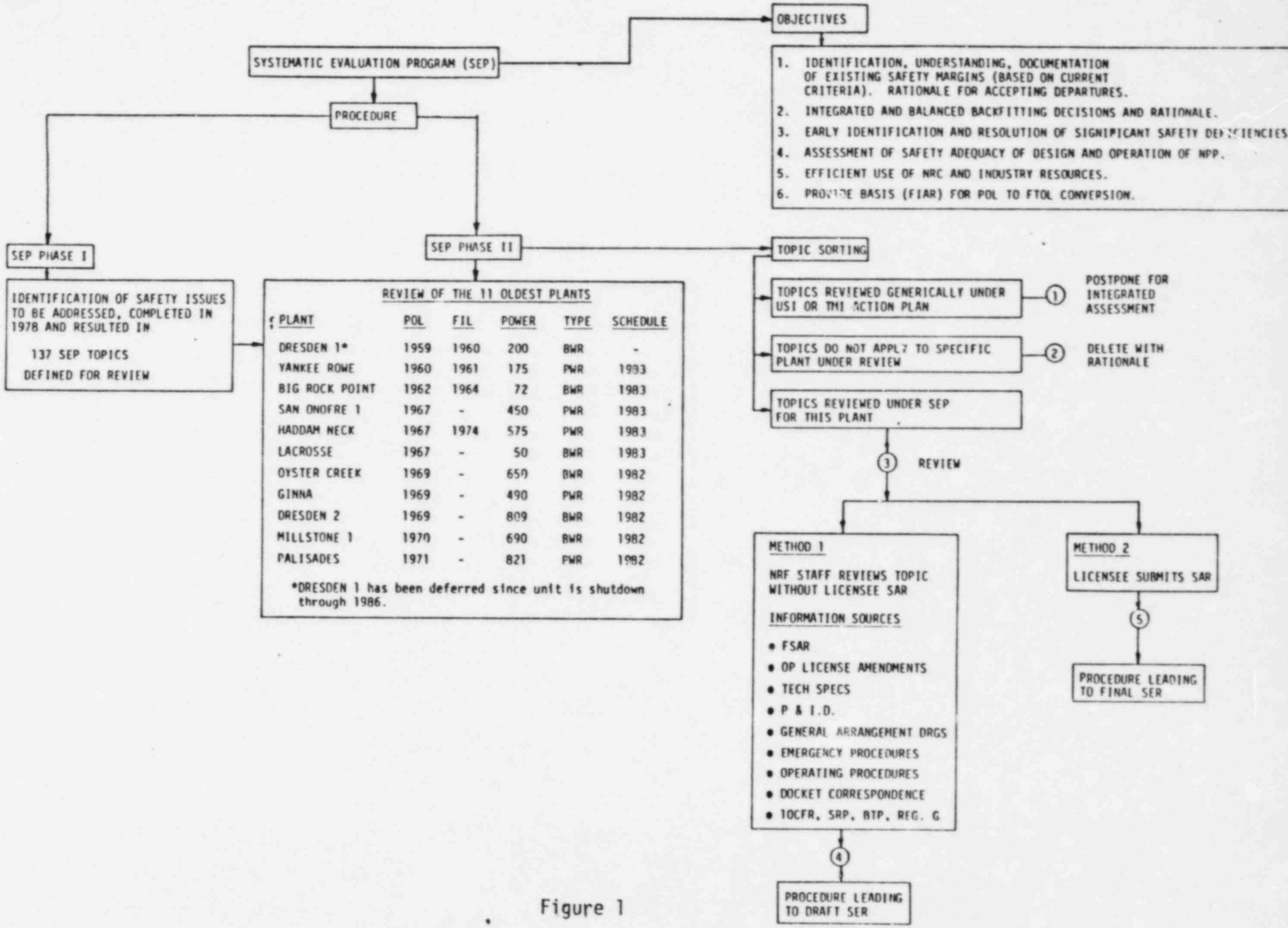
encls.
Figs. 1 to 3



REVIEW OF THE 11 OLDEST PLANTS

PLANT	POL	FIL	POWER	TYPE	SCHEDULE
DRESDEN 1*	1959	1960	200	BWR	-
YANKEE ROWE	1960	1961	175	PWR	1983
BIG ROCK POINT	1962	1964	72	BWR	1983
SAN ONOFRE 1	1967	-	450	PWR	1983
HADDAM NECK	1967	1974	575	PWR	1983
LACROSSE	1967	-	50	BWR	1983
OYSTER CREEK	1969	-	650	BWR	1982
GINNA	1969	-	490	PWR	1982
DRESDEN 2	1969	-	809	BWR	1982
MILLSTONE 1	1970	-	690	BWR	1982
PALISADES	1971	-	821	PWR	1982

*DRESDEN 1 has been deferred since unit is shutdown through 1986.

Figure 1

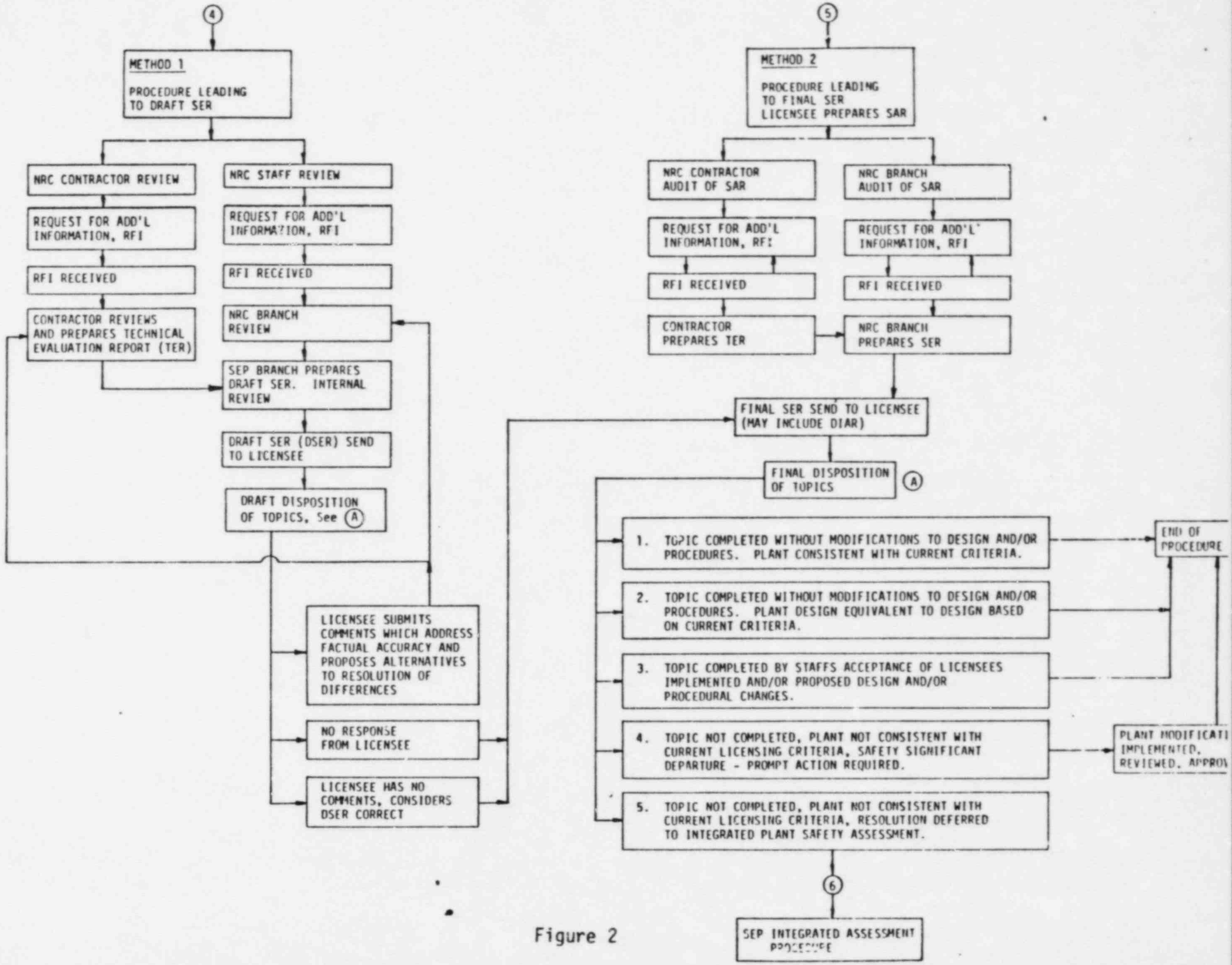


Figure 2

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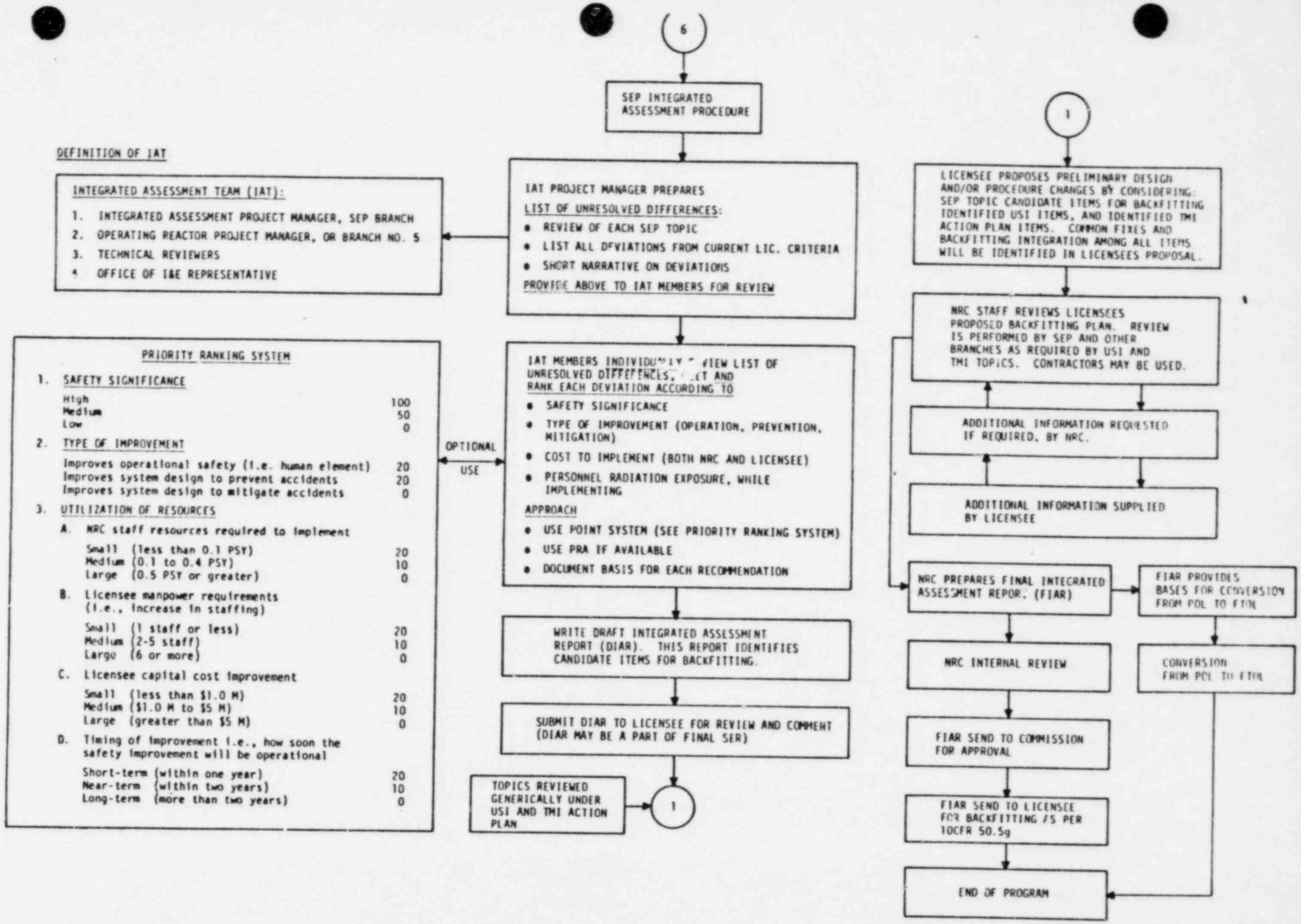

Figure 3

Reference: May 15, [illegible]

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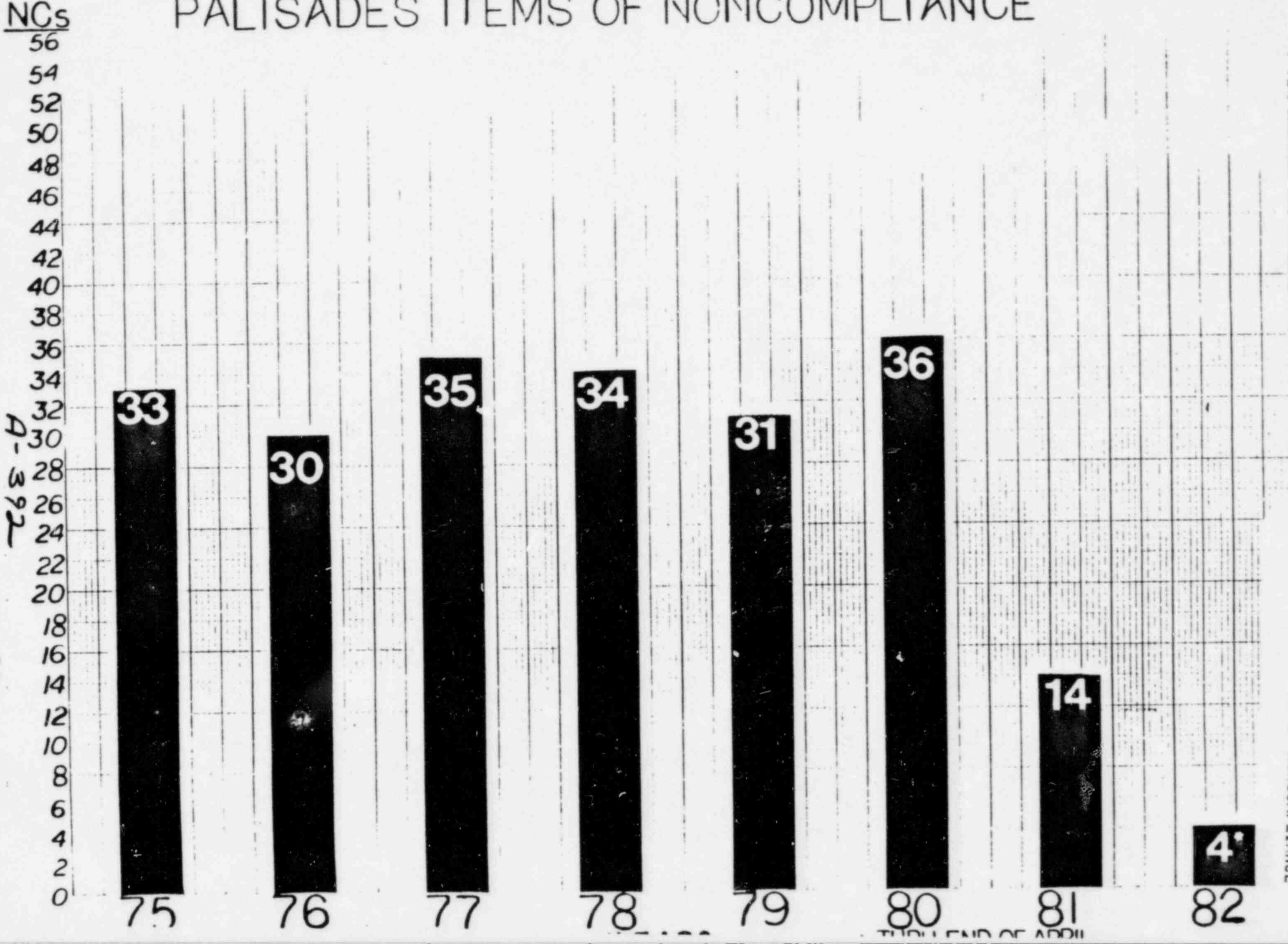
PALISADES ITEMS OF NONCOMPLIANCE



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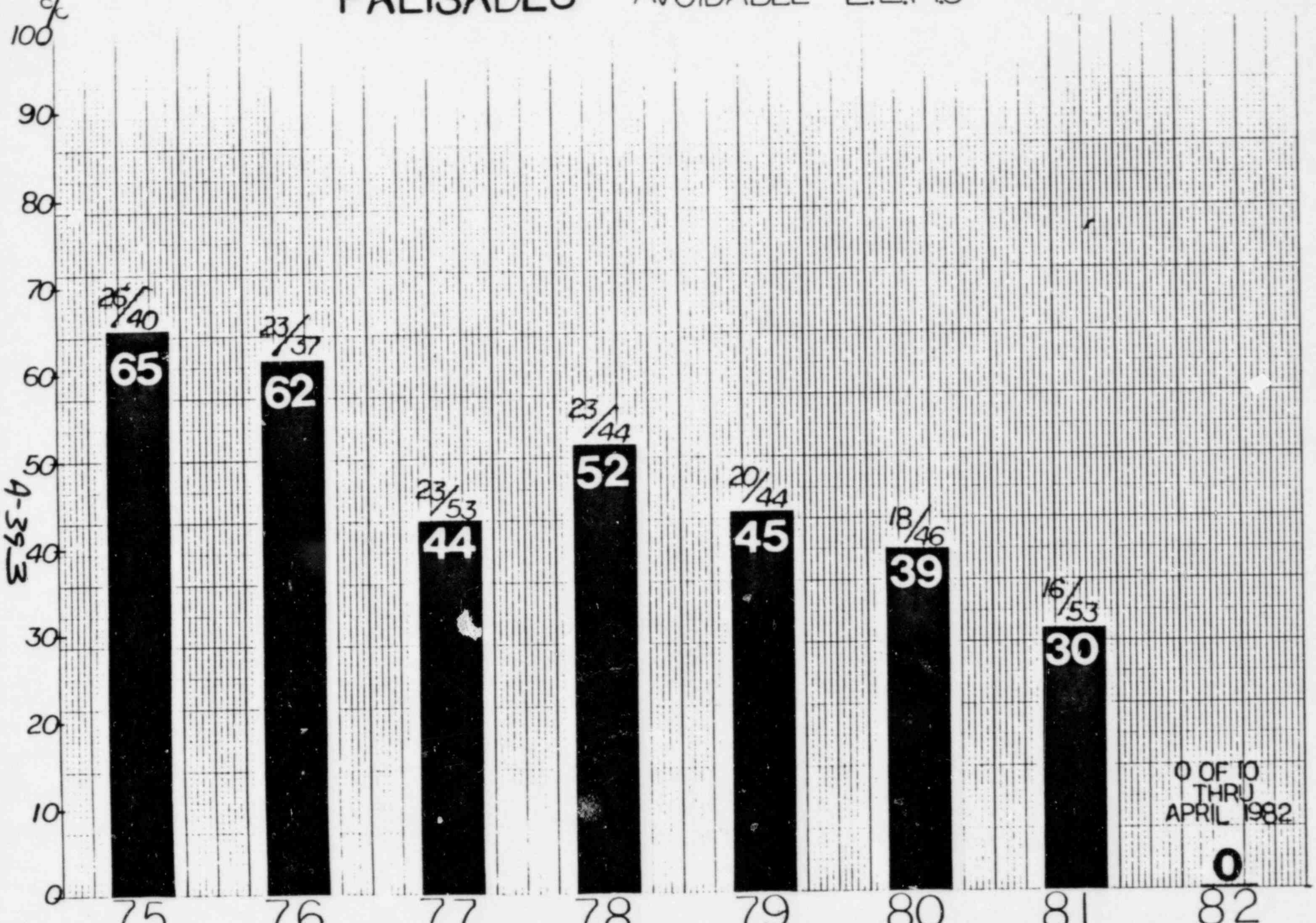
APPENDIX XXII
PALISADES ITEMS OF NONCOMPLIANCE

PALISADES AVOIDABLE L.E.R.'S



4-39-3

PALISADES ESCALATED ENFORCEMENT

- ORDER AND CIVIL PENALTY 11/9/79

  CONTAINMENT INTEGRITY VIOLATION ($225,000)

- CIVIL PENALTY 9/16/80

  PERSONNEL ERRORS - VALVE MISPOSITIONINGS ($16,000)

- CONFIRMATORY ORDER 3/9/81

  PERSONNEL ERROR - BATTERY BANK DISCONNECTION

- CIVIL PENALTY PENDING

  CONTAINMENT INTEGRITY VIOLATION
  PROCEDURE VIOLATION - MAINTENANCE

III I.E. Actions on Evaluated Weaknesses

A. Order 11-10-79

1. Examine and correct procedures for activity control
2. Verify operability monthly
3. Civil penalty

B. IAL 7-31-80

1. Operations personnel retraining
2. Modify shift turnover process
3. Civil penalty

C. IAL 1-9-81

1. Daily auditing of operations activities
2. Testing and maintenance procedural controls review
3. Instruct personnel emphasizing "disciplined" performance
4. Independent verification of proper "Manipulation"

D. Order 3-10-81

1. Control licensed operator overtime
2. Corporate review and recommendation on significant events
3. Management evaluation by independent consultant
4. Evaluate/modify procedure development process/control
5. Evaluate/modify training programs
6. Operations staff adequacy evaluation
7. Establish personnel management/motivation to adhere to procedures
8. Management auditing on implementation of 3-6 above
9. Other elevated enforcement action being considered

LICENSEE CORPORATE ORGANIZATION & MANAGEMENT ACTIONS SINCE 1979

- NUCLEAR OPERATIONS DEPARTMENT (NOD)
  - FORMED WITH MANAGEMENT AT V.P. LEVEL
  - INCORPORATED TRAINING & Q.A.
  - STUDIED NEEDS - NOD TASK FORCE

- STAFFING LEVELS INCREASED

- MANAGEMENT CHANGES - PERSONNEL

- MANAGEMENT CHANGES - ORGANIZATIONAL
  - TASK FORCE IDENTIFIED PROBLEMS
  - MANAGEMENT CONSULTANT HIRED
  - CONSULTANT RECOMMENDATIONS IMPLEMENTED

- AUGMENT CORPORATE/PLANT INTERFACES

NOD TASK FORCE
OBJECTIVES

- Organize & Implement Improved Management Systems.
- Develop & Implement NOD Training
- Upgrade Recruiting - Strategy/Performance
- Draft Manpower Plans
- Improve NOD Career Planning Process
- Minimize Salary Structure Defects
- Improve Relocation Plan
- Improve Personnel Requisition Authorization System

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NOD TASK FORCE
JANUARY 1981

FINDINGS

SUBSTANTIAL EVIDENCE OF INEFFECTIVE PROBLEM SOLVING

- REACTION TO PROBLEMS VS ANTICIPATION
- BAND-AID SOLUTIONS PREVALENT
- LACK OF ORGANIZATIONAL CLARITY
- CONTINUED FAILURE TO RESOLVE SPECIFIC PROBLEMS

PROPOSED ACTION

- EVALUATE DEPARTMENT STRUCTURE AND MANAGEMENT SYSTEMS - USING A QUALIFIED MANAGEMENT CONSULTANT - MAKE APPROPRIATE CHANGES.

NRC ORDER - 3/9/81

- REACHED SAME CONCLUSION CONCERNING CONSULTANT EVALUATION.

A-398

MANAGEMENT ANALYSIS
COMPANY RECOMMENDATIONS

1. COMBINE NUCLEAR ACTIVITIES & NUCLEAR SERVICES

2. ESTABLISH RESPONSIBILITY FOR DAY-TO-DAY REVIEW OF PLANT OPERATIONS.

3. EVALUATE NUCLEAR ACTIVITIES DEPARTMENT MANAGEMENT LEVEL. CONSIDER ELEVATION OF REPORTING DEPARTMENT HEADS ON CASE-BY-CASE BASIS.

4. ESTABLISH STAFF ORGANIZATION TO PROVIDE FOR MIDLAND TRANSITION, PLANNING AND OTHER ADMINISTRATIVE ACTIVITIES.

5. ESTABLISH A REACTOR ENGINEERING DEPARTMENT IN NUCLEAR ACTIVITIES.

6. INTERNALIZE WITHIN NOD THE ENGINEERING EXPERTISE FOR ELECTRICAL, MECHANICAL, CIVIL/STRUCTURAL, INSTRUMENTATION AND CONTROL AND IN-SERVICE INSPECTION.

THESE RECOMMENDATIONS WERE INCORPORATED IN THE JULY 1, 1981 CPCO REORGANIZATION.

IMPROVING CORPORATE/PLANT INTERFACES

1. Daily Audit of Plant Activities by Corporate Staff

2. Special Corporate Reviews

   - Personnel Error LER's

   - Safety-Sifnificant LER's

3. N.O.D. Engineers Assigned at Site

4. Standardization of Policies and Procedures

   - N.O.D. Standards

   - QA Plan Rewrite

   - Corporate Radiation Safety Standard

   - Training Department Procedures

PALISADES SITE

ORGANIZATION & MANAGEMENT
ACTIONS SINCE 1980

1. STAFFING

   --NEW PERSONNEL

   - GENERAL MANAGER
   - OPS & MAINTENANCE SUPERINTENDENT
   - OPS SUPERINTENDENT (SRO)
   - CHEMICAL & HEALTH PHYSICS SUPERINTENDENT

   --INCREASE STAFF (50%)

   --PERSONNEL MOTIVATION

2. SPACE AND FACILITY ORGANIZATION

   --CENTRALIZED DOCUMENT CONTROL

   --MANAGEMENT "MODULE"

   --SITE FACILITIES STUDY

A-401

SITE ACTIONS
(CONTINUED)

1. ORGANIZATIONAL CHANGES

   --QC GROUP MOVED FROM TECHNICAL DEPARTMENT TO QA

   --TECHNICAL DEPARTMENT REORGANIZED ON SYSTEMS BASIS

   --PLANNING AND SCHEDULING GROUP CREATED

2. STRONGER PERFORMANCE CONTROL AND VERIFICATION

   --PROCEDURES - MASSIVE REVIEW/REWRITE, ESPECIALLY OPERATIONS AND SURVEILLANCE

   --VERIFICATION

   - QC HOLD POINTS NO LONGER OPTIONAL ON AVAILABILITY OF QC INSPECTOR
   - DIRECT SUPERVISION REQUIREMENTS - CERTAIN PROCEDURES
   - INDEPENDENT VERIFICATION FOR SAFETY-SYSTEM MANIPULATIONS - EQUAL SKILL
   - MONTHLY CHECK FOR PROPER SAFETY-SYSTEM STATUS OUTSIDE CONTROL ROOM
   - SHIFT TURNOVER CONTROL ROOM CHECKS
   - UPGRADED ACTIVITY DOCUMENTATION AND REVIEW REQUIREMENTS

A-402

March 11, 1982

SECY-82-111

POLICY ISSUE
(Notation Vote)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY

Purpose: To request Commission approval of a set of basic requirements for emergency response capability and approval for the staff to work with licensees to develop plant-specific implementation schedules.

Discussion: One of the first issues reviewed by the Committee to Review Generic Requirements (CRGR) was the broad area of emergency response facilities and capabilities at nuclear plants. The Committee found that implementation schedules were not being coordinated within the NRC. In addition, existing NRC documents published as guidance to licensees were sometimes being used as firm requirements. Discussions with industry representatives and the staff indicated that some licensees had slowed down on work in this area pending NRC clarification of its requirements. Some utilities have virtually stopped work on some of the items, while others have proceeded and, in some cases, completed some of the items. The Committee recommended that steps be taken by the Office Directors involved to clarify the requirements and implementation schedules for the Safety Parameter Display System (SPDS), Control Room Design Review, upgraded Emergency Operating Procedures, Regulatory Guide 1.97, Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). In my memo to the Commission dated December 31, 1981, I noted that the DEDROGR staff would work with the program offices to clarify the basic requirements in this area and establish a revised implementation plan.

Enclosed are the staff's recommendations for the requirements in the broad area of emergency response facilities and capabilities outlined above. The requirements were developed by the program offices

Contact:
V. Stello, Jr., DEDROGR
49-29704

and are supported by CRGR. The enclosure represents a distillation of fundamental requirements from the broad range of guidance documents that NRC has issued (principally NUREG reports and Regulatory Guides). The staff intends that the guidance documents referred to in the enclosure not be used to impose requirements on licensees, but rather that they be used as sources of guidance for NRC reviewers and licensees regarding acceptable means for meeting the fundamental requirements proposed.

In discussions with owners' groups and individual licensees, the staff has learned that the Commission approved schedule of October 1, 1982, for implementation of the TSC and EOF probably cannot be met. In recognition of this fact and the difficulty of implementing generic deadlines, the staff is proposing that plant-specific schedules be established which take into account the unique status of each plant. Each licensee would be requested to submit a proposed schedule for completing the actions to comply with the fundamental requirements. The NRC Project Manager for each plant should be knowledgeable of the overall work effort going on at a plant and, based on guidance received from NRC management, could reach agreement with licensees on schedules which optimize use of utility and NRC resources. The agreed upon completion dates would be formalized in an order. By this approach, future staff coordination problems regarding implementation schedules will be avoided.

Resource Estimates: The costs to licensees to implement the requirements proposed in the enclosure were included in the estimates set out in NUREG-0660.

Recommendation: That the Commission:

1. Approve the fundamental requirements described in the enclosure.

2. Approve the issuance of the requirements in the enclosure by 50.54f letters as a revision to NUREG 0737.

3. Approve the method for establishing plant-specific implementation schedules described in the enclosure.

4. Approve the implementation of these requirements through plant-specific orders.

5. Note that the staff intends to use the previously issued NUREG reports and Regulatory Guides as guidance documents only.

Scheduling: Licensees are currently required to establish a TSC and EOF by October 1. Prompt action on this paper is required in order to provide guidance to licensees.

William J. Dircks
Executive Director for Operations

Enclosure:
NRC Staff Recommendation on the Requirements for Emergency Response Capability

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Monday, March 29, 1982.

Commission Staff-Office comments, if any, should be submitted to the Commissioners NLT Monday, March 22, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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A-405

ENCLOSURE

A-406

NRC STAFF RECOMMENDATIONS

ON THE

REQUIREMENTS FOR

EMERGENCY RESPONSE CAPABILITY

March 10, 1982

A-407

CONTENTS

A-408

CONTENTS (Continued)

EMERGENCY RESPONSE CAPABILITY

1. INTRODUCTION

This report was prepared as a result of a review by the Committee to Review Generic Requirements (CRGR). The recommendations herein have been developed by the program offices and are supported by CRGR. The report represents the staff's attempt to distill the fundamental requirements for nuclear plant Emergency Response Capability from the wide range of guidance documents that NRC has issued. It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be ignored; they are still useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document.

These fundamental requirements are further specification of the general guidance specified previously by the Commission in its regulations, orders and policy statements on emergency planning and TMI issues. It is intended that these fundamental requirements would be applicable to licensees of operating nuclear power plants and holders of construction permits for nuclear power plants. For applicants for a construction permit (CP) or manufacturing license (ML), the requirements described in this document must be supplemented with the specific provisions in the rule specifying licensing requirements for pending CP and ML applications. In this regard, it is expected that the staff would review CP and ML applications against the guidance in the current Standard Review Plan, and this might lead to more detailed requirements than prescribed in this document.

Based on discussions with licensees, the staff has learned that many of the Commission approved schedules for emergency response facilities probably will not be met. In recognition of this fact and the difficulty of implementing generic deadlines, the staff proposes that plant-specific schedules be established which take into account the unique status of each plant. The following sequence for developing implementation schedules is proposed.

When the basic requirements for emergency response capabilities and facilities are finalized, they should be transmitted to licensees by a generic letter from NRR, promulgated to NRC staff, and incorporated as regulatory requirements (e.g., in the Standard Review Plan or by regulation or Order, as appropriate). The letter to licensees should request that licensees submit a proposed schedule for completing actions to comply with the basic requirements. Each licensee's proposed schedules would then be reviewed by the assigned NRC Project Manager, who would discuss the subject with the licensee and mutually agree on schedules and completion dates. The implementation dates would then be formalized into an enforceable document.

The basic requirements in this document do not alter previously issued guidance, which remains in effect. This document does attempt to place that guidance in perspective by identifying the elements that the NRC staff believes to be essential to upgraded emergency response capabilities. The proposal to formalize implementation dates in an enforceable document reflects the level of importance which the NRC staff attributes to these basic requirements. The NRC staff does not recommend that existing guidance be imposed in this manner, but rather that it be used as guidance to be considered in upgrading emergency response capabilities. This indicates the distinction which the staff believes should be made between the basic requirements and guidance.

The following sections describe NRC staff recommendations on basic requirements, their interrelationships, and NRC actions to improve management of emergency response regulation. Reference documents are cited with a description of content as it relates to specific initiatives.

2. USE OF EXISTING DOCUMENTATION

The NRC staff recommends that the following NUREG documents are intended to be used as sources of guidance and information, and the Regulatory Guides are to be considered as guidance or as an acceptable approach to meeting formal requirements. The items by virtue of their inclusion in these documents shall not be misconstrued as requirements to be levied on licensees or as inflexible criteria to be used by NRC staff reviewers.

NUREG Report		Titles
0696	-	Functional Criteria for Emergency Response Facilities
0700	-	Guidelines for Control Room Design Reviews
0799	-	Draft Criteria for Preparation of Emergency Operating Procedures
0801	-	Evaluation Criteria for Control Room Design Reviews
0814	-	Methodology for Evaluation of Emergency Response Facilities
0818	-	Emergency Action Levels for Light Water Reactors
0835	-	Human Factors Acceptance Criteria for SPDS

Regulatory Guides		
1.23 (Rev. 1)	-	Meteorological Measurement Program for Nuclear Power Plants
1.97 (Rev. 2)	-	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident
1.101 (Rev. 2)	-	Emergency Planning for Nuclear Power Plants
1.47	-	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

3. COORDINATION AND INTEGRATION OF INITIATIVES

1. The design of the Safety Parameter Display System (SPDS), design of instrument displays based on Regulatory Guide 1.97 guidance, control room design review, development of symptom oriented emergency operating procedures, and operating staff training should be integrated with respect to the overall enhancement of operator ability to comprehend plant conditions and cope with emergencies. Assessment of information needs and display formats and locations should be performed by individual licensees. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS may obviate the need for large-scale control room modifications. However, installation of the SPDS should not be delayed by slower progress on other initiatives. The SPDS should not be contingent on completion of the control room design review. NRC does not plan to impose additional requirements on licensees regarding SPDS.

2. Implementation of part or all of Regulatory Guide 1.97 (Rev. 2) represents a control room improvement. The implementation of control room improvements is not contingent on implementing Technical Support Center (TSC) and Emergency Operations Facility (EOF) requirements.

3. The Technical Support Center (TSC) and Emergency Operations Facility (EOF) are dependent on control room improvements in terms of communication and instrumentation needs among the TSC, EOF, and control room. TSC and EOF facilities are not necessarily dependent on each other. The Operational Support Center (OSC) is independent of TSC and EOF.

4. The three groups of initiatives--SPDS, control room improvements, and emergency response facilities (TSC, EOF, OSC)--should have the following interrelationships:

   a. The SPDS is an improvement in the control room because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS could obviate the need for extensive modifications to control rooms.

   b. New instrumentation that may be added to the control room should be considered a requirement for inclusion in the design of the TSC and EOF only to the extent that such instrumentation is essential to the performance of TSC and EOF functions.

   c. The SPDS and control room improvements are essential elements in operator training programs and the upgraded plant-specific emergency operating procedures.

   d. Acquisition, processing, and management of data for SPDS, control room improvements, and emergency response facilities should be coordinated but need not be centralized.

5. Specific implementation plans and reasonable, achievable schedules should be established by agreement between the NRC Project Manager and each individual licensee. The NRC office responsible for implementing each requirement should develop procedures identifying the following:

   a. The respective roles of NRR, IE, and Regional Offices in managing implementation, checking licensee rate of progress, and verifying compliance, including the extent to which NRC review and inspection is necessary during implementation.

   b. Procedural methods and enforcement measures that could be used to ensure NRC staff and licensee attention to meeting mutually agreed upon schedules without significant delays and extensions.

6. The NRC Project Manager for each nuclear power plant is assigned program management responsibility for NRC staff actions associated with implementing emergency response initiatives. The NRC Project Manager is the principal contact for the licensee regarding these initiatives.

7. NRC will make allowances for work already done by licensees in a good-faith effort to meet requirements as they understand them. For each case in which a licensee would have to remove or rip out emergency response facilities or equipment that was installed in good faith to meet previous guidance in order to meet the basic requirements described in this document, the Director of the Office of Nuclear Reactor Regulation or Inspection and Enforcement will review the circumstances and determine whether removal is necessary or existing facilities or equipment represent an acceptable alternative. Any regulatory position that would require the removal or major modification of existing emergency response facilities or equipment requires the specific approval of the Office Director.

8. NRC recognizes that acceptable alternative methods of phasing and integrating emergency response activities may be developed. Each licensee needs flexibility in integrating these activities, taking into account the varying degree to which the licensee has implemented past requirements and guidance. An example of a way in which these activities could be integrated is discussed below. Other methods of integration proposed by licensees would be reviewed considering licensees' progress on each initiative.

   a. SPDS

      (1) Review the functions of the nuclear power plant operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk, (b) could cause operators to make cognitive errors in diagnosing them, and (c) are not included in routine operator training programs.

      (2) Combine the results of this review with accepted human factors principles to select parameters, data display, and functions to be incorporated in the SPDS.

(3) Design, build, and install the SPDS in the control room and train its users.

b. To be done parallel without delaying SPDS, complete emergency operating procedure technical guidelines that will be used to develop plant-specific emergency operating procedures.

c. Using these EOP technical guidelines, the SPDS design, and accepted human factors principles, conduct a review of the control room design. Apply the results of this review to:

(1) Verify SPDS parameter selection, data display, and functions.
(2) Develop plant-specific EOPs.
(3) Design control room modifications that correct conditions adverse to safety (reduce significant contributions to risk), and add additional instrumentation that may be necessary to implement Regulatory Guide 1.97.
(4) Train and qualify plant operating staff regarding EOPs and modifications.

d. Verify, prior to finalization of designs for modifications and of procedures and training, that the functions of control room operators in emergencies can be accomplished (i.e., that the individual initiatives have been integrated sufficiently to meet the needs of control room operators and provide adequate emergency response capabilities).

e. Implement EOPs and install control room modifications coincident with scheduled outages as necessary, and train operators in advance of these changes as they are phased into operation.

4. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

Current Regulatory Requirements

No licensee action is required.

Functional Statement

The SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

Recommended Requirements

1. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

2. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) forms the basic safety components required for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, 4 in Appendix A; 10 CFR Part 100; single-failure requirements). The SPDS need not meet requirements of the single-failure criteria and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. After the SPDS has been installed, operating procedures should be available that will allow timely and correct safety status assessment when the SPDS is not available.

3. There is a wide range of useful information that can be provided by various systems. This information is reflected in such staff documents as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97.

   Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information that should be provided for a particular plant shall be based on engineering judgment of individual plant licensees, taking into account the importance of prompt implementation.

4. The SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily perceived and comprehended by SPDS users.

5. Minimum information to be provided shall be sufficient to provide information to plant operators about:

   a. Reactivity control
   b. Reactor core cooling and heat removal from the primary system
   c. Reactor coolant system integrity
   d. Radioactivity control
   e. Containment conditions

   The specific parameters to be displayed shall be determined by the licensee.

6. The licensee shall prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Such analysis, along with the specific implementation plan for SPDS shall be reviewed as described below.

7. The licensee's proposed implementation of an SPDS system shall be reviewed in accordance with the licensee's technical specifications to determine whether the changes involve an unreviewed safety question or change of technical specifications. If they do, they shall be processed in the normal fashion with prior NRC review. If the changes do not involve an unreviewed safety question or a change in the technical specifications, the licensee may implement such changes without prior approval by NRC. However, the licensee's analysis shall be submitted to NRC promptly on completion of review by the licensee's offsite committee. Based on the results of NRC review, the Director of IE or the Director of NRR may request or direct the licensee to cease implementation if a serious safety question is posed by the licensee's proposed system, or if the licensee's analysis is seriously inadequate.

Integration

Prompt implementation of an SPDS is a design goal and of primary importance. The schedule for implementing SPDS should not be impacted by schedules for the control room design review and development of symptom-oriented emergency operating procedures. For this reason, licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to the other initiatives. If reasonable, this schedule should be accepted by NRC.

Reference Documents

NUREG-0660 -- Need for SPDS identified

NUREG-0737	--	Specified SPDS
NUREG-0696	--	Functional criteria for SPDS
NUREG-0835	--	Specific acceptance criteria keyed to 0696
Reg. Guide 1.97 (Rev. 2)	--	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

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5. DETAILED CONTROL ROOM DESIGN REVIEW

Current Regulatory Requirements

As specified in Item I.D.1 in NUREG-0737, the implementation schedule is still to be developed.

Functional Statement

The objective of the control room design review is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (from NUREG-0660, Item I.D.1). As a complement to improvements of plant operating staff capabilities in response to transients and other abnormal conditions that will result from implementation of the SPDS and from upgraded emergency operating procedures, this design review will identify any modifications of control room configurations that would contribute to a significant reduction of risk and enhancement in the safety of operation. Decisions to modify the control room would include consideration of long-term risk reduction and any potential temporary decline in safety after modifications resulting from the need to relearn maintenance and operating procedures. This should be carefully reviewed by persons competent in human factors engineering and risk analysis.

Recommended Requirements

1. Conduct a control room design review to identify human engineering discrepancies. The review shall consist of:

    a. The establishment of a qualified multidisciplinary review team and a review program incorporating accepted human engineering principles.

    b. The use of function and task analysis (that had been used as the basis for developing emergency operating procedure Technical Guidelines) to identify control room operator tasks and information and control requirements during emergency operations. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters.

    c. A comparison of the display and control requirements with a control room inventory to identify missing and surplus (distracting) displays and controls.

    d. A control room survey to identify deviations from accepted human factors principles. This survey will include, among other things, assessment of control room layout, the usefulness of audible and visual alarm systems, information recording and recall capability, and control room environment.

2. Assess which human engineering discrepancies are significant and should be corrected. Select design improvements that will correct those discrepancies. Improvements that can be accomplished with an enhancement program (paint-tape-label) should be done promptly.

3. Verify that each selected design improvement will provide the necessary correction, and can be introduced in the control room without creating any unacceptable human engineering discrepancies because of significant contribution to increased risk, unreviewed safety questions, or situations in which a temporary reduction in safety could occur. Improvements that are introduced should be coordinated with changes resulting from other improvement programs such as SPDS, operator training, new instrumentation (Reg. Guide 1.97, Rev. 2), and upgraded emergency operating procedures.

Documentation and NRC Review

1. All licensees shall submit a program plan within two months of the start of the control room review that describes how items 1, 2 and 3 above will be accomplished. NRC approval is not required before licensees conduct their reviews.

2. Selected licensees will undergo an in-progress audit by the NRR human factors staff based on the program plans and advice from resident inspectors.

3. All licensees shall submit a summary report outlining proposed control room changes. The report will also provide a summary justification for human engineering discrepancies with safety significance to be left uncorrected or partially corrected.

4. Within two weeks after receipt of the licensee's summary report, the NRC will inform the licensee whether it will conduct a pre-implementation onsite audit. The decision will be based on the content of the program plan, summary report, and results of NRR in-progress audits, if any. The licensee selection for pre-implementation audit may or may not include licensees selected for in-progress audits under paragraph 2.

5. For control rooms selected for pre-implementation onsite audit, within one month after receipt of the summary report, the NRC will conduct:

   a. A pre-implementation audit of proposed modifications (e.g., equipment additions, deletions and relocations, and proposed modifications).

   b. An audit of the justification for those human engineering discrepancies of safety significance to be left uncorrected or only partially corrected.

   The audit will consist of a review of licensee's record of the control room reviews, discussions with the licensee review team, and usually a control room visit. Within a month after this onsite audit, NRC will issue its safety evaluation report (SER).

6. For control rooms for which NRC does not perform a pre-implementation onsite audit, NRC will conduct a review and issue its SER within two

months after receipt of the licensee's summary report. The review shall be similar to that conducted for pre-implementation plants under paragraph 5 above, except that it may or may not include a specific audit. The SER shall indicate whether, based on the review carried out, changes in the licensee's modification plan are needed to assure operational safety. Flexibility is considered in the control room review, because certain control board discrepancies can be overcome by techniques not involving control board changes. These techniques could include improved procedures, improved training, or the SPDS.

7. The following approach will be used for OL review. For OL applications with SSER dates prior to June 1983, licensing may be based on either a Preliminary Design Assessment or a Control Room Design Review (CRDR) at the applicant's option. However, applicants who choose the Preliminary Design Assessment option are required to perform a CRDR after licensing. For applications with SSER dates after June 1983, Control Room Design Review will be required prior to licensing.

Integration

Prompt implementation of an SPDS is a design goal and of primary importance. The schedule for implementing SPDS should not be impacted by schedules for the control room design review and development of symptom-oriented emergency operating procedures. For this reason, licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to the other initiatives. If reasonable, this schedule should be accepted by NRC.

Reference Documents

NUREG-0585	--	States that licensees should conduct review.
NUREG-0660, Rev. 1	--	States that NRR will require reviews for operating reactors and operating licensee applicants.
NUREG-0700	--	Final guidelines for CRDR.
NUREG-0737	--	States that requirement was issued June, 1980, final guidance not yet issued.
NUREG-0801	--	October 1981 draft for comment; staff evaluation criteria.

REGULATORY GUIDE 1.97

6. APPLICATION TO EMERGENCY RESPONSE FACILITIES

Current Regulatory Requirements

No licensee action is required.

Functional Statement

Regulatory Guide 1.97 provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

Recommended Requirements

1. Control Room

   Provide measurements and indication of Type A, B, C, D, E variables listed in Regulatory Guide 1.97 (Rev. 2). Individual licensees may take exceptions based on plant-specific design features. BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements. It is acceptable to rely on currently installed equipment if it will measure over the range indicated in Regulatory Guide 1.97 (Rev. 2), even if the equipment is presently not environmentally qualified. Eventually, all the equipment required to monitor the course of an accident would be environmentally qualified in accordance with the pending Commission rule on environmental qualification.

   Provide reliable indication of the meteorological variables (wind direction, wind speed, and atmospheric stability) specified in Regulatory Guide 1.97 (Rev. 2) for site meteorology. No changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity of the plant site. Information on meteorological conditions for the region in which the site is located shall be available via communication with the National Weather Service.

2. Technical Support Center (TSC)

   The Type A, B, C, D, E variables that are essential for performance of TSC functions shall be indicated in the TSC.

   a. BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements.

   b. The indicators and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

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3. Emergency Operations Facility (EOF)

   a. Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be provided in the EOF.

   b. The EOF data indications and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

Documentation and NRC Review

NRC review is not a prerequisite for implementation. Staff review will be in the form of an audit that will include a review of the licensee's method of implementing Regulatory Guide 1.97 (Rev. 2) guidance and the licensee's supporting technical justification of any proposed alternatives.

The licensee shall submit a report describing how it meets these requirements. The submittal should include documentation which may be in the form of a table that includes the following information for each Type A, B, C, D, E variable shown in Regulatory Guide 1.97 (Rev. 2):

(a) instrument range
(b) environmental qualification (as stipulated in guide or state criteria)
(c) seismic qualification (as stipulated in guide or state criteria)
(d) quality assurance (as stipulated in guide or state criteria)
(e) redundancy and sensor(s) location(s)
(f) power supply (e.g., Class 1E, non-Class 1E, battery backed)
(g) location of display (e.g., control room board, SPDS, chemical laboratory)
(h) schedule (for installation or upgrade)

Deviations from the guidance in Regulatory Guide 1.97 (Rev. 2) should be explicitly shown, and supporting justification or alternatives should be presented.

7. UPGRADE EMERGENCY OPERATING PROCEDURES (EOPs)

Current Regulatory Requirements

NUREG-0737, Item I.C.1, which has been approved by the Commission for implementation.

Functional Statement

Symptom-based emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors.

Recommended Requirements

1. In accordance with NUREG-0737, Item I.C.1, reanalyze transients and accidents and prepare Technical Guidelines. These analyses will identify operator tasks, and information and control needs. The analyses also serve as the basis for integrating upgraded emergency operating procedures and the control room design review and verifying the SPDS design.

2. Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide.

3. Provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs.

4. Implement upgraded EOPs.

Documentation and NRC Review

1. Submit Technical Guidelines to NRC for review. NRC will perform a pre-implementation review of the Technical Guidelines and the Writer's Guide. Within two months of receipt of the Technical Guidelines and Writer's Guide, NRC will advise the licensees of their acceptability.

2. Each licensee shall submit to NRC a procedures generation package at least three months prior to the date it plans to begin formal operator training on the upgraded procedures. NRC approval of the submittal is not necessary prior to upgrading and implementing the EOPs. The procedures generation package shall include:

    a. Plant-Specific Technical Guidelines -- plant-specific guidelines for plants not using generic technical guidelines. For plants using generic technical guidelines, a description of the planned method for developing plant specific EOPs from the generic guidelines, including plant specific information.

    b. A Writer's Guide that details the specific methods to be used by the licensee in preparing EOPs based on the Technical Guidelines.

c. A description of the program for validation of the EOPs.

d. A brief description of the training program for the upgraded EOPs.

3. All procedures generation packages will be reviewed. On an audit basis for selected facilities, upgraded EOPs will be reviewed. The details and extent of this review will be based on the quality of the procedures generation packages submitted to NRC. A sampling of ugpraded EOPs will be reviewed for technical adequacy in conjunction with the NRC Reactor Inspection Program.

Reference Documents

NUREG-0660, Item I.C.1, I.C.8, I.C.9

NUREG-0799

8. EMERGENCY RESPONSE FACILITIES

Current Regulatory Requirements

10 CFR 50.47(b)(6) (for Operating License applicants) -- Requirement for prompt communications among principal response organizations and to emergency personnel and to the public.

10 CFR 50.47(b)(8) -- Requirement for emergency facilities and equipment to support emergency response.

10 CFR 50.47(b)(9) -- Requirement that adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.54(q) (for Operating Reactors) -- Same requirement as 10 CFR 50.47(b) plus 10 CFR 50, Appendix E.

10 CFR 50, Appendix E, Paragraph IV.E
Requirement for:

"1. Equipment at the site for personnel monitoring;

"2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

"3. Facilities and supplies at the site for decontamination of onsite individuals;

"4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

"5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on site;

"6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

"7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;

"8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

"9. At least one onsite and one offsite communications system; each system shall have a backup power source.

"All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

"a. Provision for communications with contiguous State/local governments within the plume exposure pathway (emergency planning zone) EPZ. Such communications shall be tested monthly.

"b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

"c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

"d. Provision for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly."

Within this section on emergency response facilities, the Technical Support Center (TSC), Operational Support Center (OSC) and Emergency Operations Facility (EOF) are addressed separately in terms of their functional statements and recommended requirements. The subsections on Documentation and NRC Review and Reference Documents that follow the EOF discussion apply to this entire section on emergency response facilities.

Technical Support Center (TSC)

Functional Statement

The TSC is the onsite technical support center for emergency response. When activated, the TSC is staffed by predesignated technical, engineering, senior management, and other licensee personnel, and five predesignated NRC personnel. During periods of activation, the TSC will operate uninterrupted to provide plant management and technical support to plant operations personnel, and to relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations. The TSC will perform EOF functions for the Alert Emergency class and for the Site Area Emergency class and General Emergency class until the EOF is functional.

Recommended Requirements

The TSC will be:

1. Located within the site protected area so as to facilitate necessary interaction with control room, OSC, EOF and other personnel involved with the emergency.

2. Sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation in the center.

3. Structurally built in accordance with the National Uniform Building Code.

4. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

5. Provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

6. Provided with reliable voice and data communications with the control room and EOF and reliable voice communciations with the OSC, NRC Operations Centers and state and local operations centers.

7. Capable of reliable data collection, storage, analysis, display and communication sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

   (a) the variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2) that are essential for performance of TSC functions; and

   (b) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and National Weather Service data available by voice communication for the region in which the plant is located.

Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

8. Provided with accurate, complete and current plant records (drawings, schematic diagrams, etc.) essential for evaluation of the plant under accident conditions.

9. Staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and be fully operational within approximately 1 hour after activation.

10. Designed taking into account good human factors engineering principles.

Operational Support Center (OSC)

Functional Statement

When activated, the OSC will be the onsite area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by control room, TSC or EOF personnel.

Recommended Requirements

The OSC will be:

1. Located onsite to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

2. Capable of reliable voice communications with the control room, TSC and EOF.

Emergency Operations Facility (EOF)

Functional Statement

The EOF is a licensee controlled and operated facility. The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment, determination of recommended public protective actions, and coordination of emergency response activities with Federal, State, and local agencies.

When the EOF is activated, it will be staffed by predesignated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display, and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radioactive releases from the plant and to determine dose projections.

Recommended Requirements

The EOF will be:

1. Located and provided with radiation protection features as described in Table 1 (previous guidance approved by the Commission) and with appropriate radiological monitoring systems.

2. Sufficient to accommodate and support Federal, State, local and licensee predesignated personnel, equipment and documentation in the EOF.

3. Structurally built in accordance with the National Uniform Building Code.

4. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

5. Provided with reliable voice and data communications facilities to the TSC and control room, and reliable voice communication facilities to OSC and to NRC, State and local emergency operations centers.

6. Capable of reliable collection, storage, analysis, displays and communication of information on containment conditions, radiological releases and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF functions shall be available in the EOF:

   (a) variables from the appropriate Table 1 or 2 Regulatory Guide 1.97 (Rev. 2), and

(b) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and regional data available via communication from the National Weather Service.

7. Provided with up to date plant records (drawings, schematic diagrams, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

8. Staffed in accordance with Table 2 (previous guidance approved by the Commission). Reasonable exceptions to the 30-minute and 1-hour time limits for staffing should be justified and will be considered by NRC staff.

9. Provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

10. Designed taking into account good human factors engineering principles.

Documentation and NRC Review

The conceptual design for emergency response facilities (TSC, OSC, and EOF) have been submitted to NRC for review. In many cases, the lack of detail in these submittals has precluded an NRC decision of acceptability. Some designs have been disapproved because they clearly did not meet the intent of the applicable regulations. NRC does not intend to approve each design prior to implementation, but rather has provided in this document those "recommended requirements" which should be satisfied. These recommended requirements provided a degree of flexibility within which licensees can exercise management prerogatives in designing and building emergency response facilities (ERF) that satisfy specific needs of each licensee. The foremost consideration regarding ERFs is that they provide adequate capabilities of licensees to respond to emergencies. NUREG guidance on ERFs has been intended to address specific issues which the Commission believes should be considered in achieving improved capabilities.

Licensees should assure that the design of ERFs satisfies these basic requirements. Exemptions from or alternative methods of implementing these requirements should be discussed with NRC staff and in some cases could require Commission approval. Licensees should continue work on ERFs to complete them according to schedules that will be negotiated on a plant-specific basis. NRC will conduct appraisals of completed facilities to verify that these requirements have been satisfied and that ERFs are capable of performing their intended functions. Licensees need not document their actions on each specific item contained in NUREG-0696 or 0814.

Reference Documents (Emergency Response Facilities)

10 CFR 50.47(b) -- Requirements for emergency facilities and equipment for OLs.

10 CFR 50.54(q) and Appendix E, Paragraph IV.E -- Requirements for emergency facilities and equipment for ORs.

NUREG-0660 -- Description of and implementation schedule for TSC, OSC and EOF.

Eisenhut letter to power reactor licensees 9/13/79 -- Request for commitment to meet requirements.

Denton letter to power reactor licensees 10/30/79 -- Clarification of requirements
and implementation schedule.

Eisenhut letter to power reactor licensees 4/25/80 -- Clarification of requirements.

NUREG-0654 -- Radiological Emergency Response Plans

NUREG-0696 -- Functional criteria for emergency response facilities.

NUREG-0737 -- Guidance on meteorological monitoring and dose assessment.

Eisenhut letter to power reactor license 2/18/81 -- Commission approved guidance
on location, habitability and staff for emergency facilities. Request and deadline for submittal of conceptual design of facilities.

NUREG-0814 (Draft Report for Comment) -- Methodology for evaluation of emergency
response facilities.

NUREG-0818 (Draft Report for Comment) -- Emergency Action Levels

Reg. Guide 1.97 (Rev. 2) -- Guidance for variables to be used in selected emergency response facilities.

COMJA-80-37, January 21, 1981 -- Commission approval guidance on EOF location and habitability.

Secretary memorandum S81-19, February 19, 1981 -- Commission approval of NUREG-0696 as general guidance only.

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TABLE 1

EMERGENCY OPERATIONS FACILITY

Option 1
Two Facilities

A. Close-in Primary: Reduce Habitability*

- o within 10 miles
- o protection factor = 5
- o ventilation isolation with HEPA (no charcoal)

B. Backup EOF

- o between 10-20 miles
- o no separate, dedicated facility
- o arrangements for portable backup equipment
- o strongly recommended location be coordinated with offsite authorities
- o continuity of dose projection and decision making capability

Option 2
One Facility

- o At or Beyond 10 miles.
- o No special protection factor.
- o If beyond 20 miles, specific approval required by the Commission, and some provision for NRC site team closer to site.
- o Strongly recommended location be coordinated with offsite authorities.

For both Options:

- located outside security boundary
- space for about 10 NRC employees
- none designated for severe phenomena, e.g., earthquakes

*Habitability requirements are only for the part of the EOF in which dose assessments communications and decision making take place.

If a utility has begun construction of a new building for an EOF that is located with 5 miles, that new facility is acceptable (with less than protection factor of 5 and ventilation isolation and HEPA) provided that a backup EOF similar to "B" in Option 1 is provided.

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TABLE 2

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions: On Shift*	30 min.	60 min.
Plant Operations and Assessment of Operational Aspects		Shift supervisor (SRO)	1	--	--
		Shift foreman (SRO)	1	--	--
		Control-room operators	2	--	--
		Auxiliary operators	2		
Emergency Direction and Control (Emergency Coordinator)***		Shift technical advisor, shift supervisor, or designated facility manager	1**	--	--
Notification/ Communication****	Nofity licensee, state local, and federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency operations facility (EOF) director	Senior manager	--	--	1
	Offsite dose assessment	Senior health physics (HP) expertise	--	1	--
	Offsite surveys		--	2	2
	Onsite (out-of-plant)		--	1	1
	Inplant surveys	HP technicians	1	1	1
	Chemistry/radio-chemistry	Rad/chem technicians	1	--	1

NOTE: Source of this table is NUREG-0654, "Functional Criteria for Emergency Response Facilities."

TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions: On Shift*	30 min.	60 min.
Plant System Engineering, Repair and Corrective Actions	Technical support	Shift technical advisory	1	--	--
		Core/thermal hydraulics	--	1	--
		Electrical	--	--	1
		Mechanical	--	--	1
	Repair and corrective actions	Mechanical maintenance/	1**	--	1
		Radwaste operator			1
		Electrical maintenance/	1**	1	1
		instrument and control		1	--
		(I&C) technician	--	1	--
Protective Actions (In-Plant)	Radiation protection: a. Access control b. HP Coverage for repair, corrective actions, search and rescue first-aid, & firefighting c. Personnel monitoring d. Dosimetry	HP technicians	2**	2	2
Firefighting	--	--	Fire brigade per technical specification	Local support	
Rescue Operations and First-Aid	--	--	2**	Local support	

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TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions On Shift*	30 min.	60 min.
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security personnel	All per security plan		
		Total	10	11	15

*For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control-room operator, and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

**May be provided by shift personnel assigned other functions.

***Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Directoı of minute-to-minute facility operations remains with senior manager in technical support center or control room.

****May be performed by engineering aide to shift supervisor.

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The following pages A-438 thru A-446 has been deleted as 1.

DELETION

The following pages A-447 thru A-449 has been deleted as I.

DELETION

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

March 26, 1982

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts

FROM: Forrest J. Remick

SUBJECT: EMERGENCY RESPONSE CAPABILITIES AND FACILITIES AND REGULATORY POSITION ON HUMAN FACTORS SAFETY; OPE EVALUATION OF SECY-82-111

As requested by Commissioners Ahearne and Bradford we have evaluated the proposal "NRC Staff Recommendations on the Requirements for Emergency Response Capability." (SECY-82-111) ("Staff Proposal")[1/] The staff proposal addresses the following items: Safety Parameter Display Systems (SPDS); Detailed Control Room Design Review (DCRDR); Regulatory Guide 1.97 Application to Emergency Response Facilities; Upgrade of Emergency Operating Procedures (EOPs); Technical Support Center (TSC); Operational Support Center (OSC); and the Emergency Operations Facility (EOF). Commissioner Ahearne specifically requested OPE to evaluate the proposal from the standpoint of whether the NRC is backing away from an emphasis upon human factors. Expressing concern that standards for emergency response equipment and facilities are being relaxed or significantly changed, Commissioner Bradford requested OPE's assessment of the proposal with a special eye toward those standards, schedules or review methods which the Commission has approved, included in the PPG, or told Congress the NRC would use. Our evaluation of the proposal in response to these requests is addressed below.

[1/]DEDROGR memorandum of December 29, 1981 to H. Denton, R. DeYoung, R. Minogue and J. Davis entitled "Emergency Response Capability and Facilities" forwarded for staff review an initial package, addressing proposed basic requirements and proposed implementation methods. Commissioners Ahearne and Bradford referenced this memorandum in their request for OPE evaluation. As we expected changes in the December 29, 1981 proposal, we informed the Commission we would evaluate the final DEDROGR proposal (now entitled "NRC Staff Recommendations on the Requirements for Emergency Response Capability"). We refer to this proposal as the staff proposal.

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Attachment 3

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Human Factors Consideration

We believe the final staff proposal is a significant improvement over the earlier (December 29, 1981) draft which prompted Commissioner request for OPE evaluation. Table 1 summarizes the improvements. Because of these improvements, we believe that many of the major concerns previously expressed by certain human factors experts may have been resolved. However, based on our review of portions of a preliminary draft report resulting from a separate study[2/] being completed by the Human Factors Society for the NRC, we believe human factors experts still may not be completely satisfied with the staff proposal. Preliminary recommendations from the Human Factors Society study which appear pertinent to the staff proposal and which differ from the staff proposal, are listed in Table 2. On March 15, 1982 the DEDROGR provided the Human Factors Society a copy of the staff proposal. The Human Factors Society has indicated they will provide comments to DEDROGR. These comments should provide the latest views of the Human Factors Society, possibly altering preliminary recommendations contained in the Human Factors Society draft report. It is our understanding that the Human Factors Society comments will be addressed as part of the Commission briefing on SECY-82-111.

With respect to the question of whether NRC is backing away from an emphasis upon human factors, we do not believe this is the case. However, the answer to the question will depend on how the general requirements of the staff proposal are to be implemented which may in turn depend on how staff reviewers apply the NUREG documents referenced in the staff proposal. For example, it is stated that the SPDS, TSC and EOF should be designed "taking into account good human factors engineering principles." Licensee response to this general requirement could range from lip service to gold-plated designs. The staff has produced many NUREG documents (listed in the staff proposal) which provide guidelines (criteria) to licensees for implementing most of the items in the staff proposal or which describe staff review methods for evaluating licensee implementation. In some cases, implementation of human factors considerations might depend on what actions the staff takes where licenses do not meet the intent of the NUREG documents.

We believe the most controversial aspect of the staff proposal from a human factors standpoint is SPDS implementation. While we agree that an SPDS may be a valuable addition to the control room, we also believe that there are important considerations that should be addressed in implementing the requirement:

[2/] "A Comprehensive Human Factors Plan for Nuclear Reactor Regulation" prepared by the Human Factors Society. To be published as a NUREG/CR document in May 1982 (estimate).

-- There is an apparent wide variation in the nuclear industry as to how the SPDS may be configured. At a March 17, 1982 ACRS Human Factors Subcommittee meeting EPRI indicated that: the price of an SPDS may range from 0.5 to 8.0 million dollars a plant; there are 20 or more vendors selling an SPDS; there are a wide variety of hardware configurations -- possibly 30 or 40 different designs; and the total number of signal inputs to the SPDS may range from 500 to 1,000. Members of the staff indicate they envisaged a more limited SPDS, costing in the range of 0.5 to 1.5 million dollars with a much reduced number of signal inputs. The staff proposal states that if SPDS implementation does not involve an unreviewed safety question or a change in technical specifications, the licensee may implement the change without prior NRC approval. We recommend the Commission consider asking the staff whether a post-implementation review is appropriate in light of the apparent wide variation in industry practice.

-- We note that the SPDS need not meet requirements of the single-failure criterion, need not be qualified to meet Class 1E requirements, and need not be seismically qualified. While we are not suggesting that those requirements be added, we do believe they introduce certain human factors considerations which should receive attention by the staff and licensees during implementation of SPDS, some of which are mentioned by the Human Factors Society (see Table 2):

- Implications of operators becoming dependent upon the SPDS and their reactions if the SPDS becomes unavailable.

- Implications for operator training (for example, plans for introduction of SPDS in simulator training programs).

- Implications of using the SPDS, a nonsafety-grade piece of equipment, in making safety decisions (for example, training operators not to rely solely on the SPDS in making decisions).

-- Notwithstanding the statements that SPDS implementation should be integrated with control room design reviews and development of symptom-oriented emergency procedures, there are statements in the proposal which imply integration may not be necessary, such as: "... installation of the SPDS should not be delayed by slower progress on other initiatives;" and "SPDS should not be contingent on completion of the control room design review" (staff proposal, p 4). One utility representative expressed concern that NRC will attempt to implement the SPDS too fast, without consideration for cost or safety implications. That same representative stated that the industry should be allowed to work with INPO to develop a standardized approach for SPDS. We recommend that in setting implementation schedules for SPDS any generic activities be considered.

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Standards for Emergency Response Equipment and Facilities

We were requested to review the staff proposal from the standpoint of whether standards for emergency response equipment and facilities are being relaxed or significantly changed with a special eye toward those standards, schedules or review methods which the Commission has approved, included in the PPG, or told Congress the NRC would use. Enclosure 1 provides the status of Commission approval of standards (regulatory guidance), schedules, and review methods and a comparison of the staff proposal with Commission-approved items.

With respect to the question of whether standards for emergency response equipment and facilities are being relaxed or significantly changed we have the following comments:

- The staff states that its proposal distills "fundamental requirements" for emergency response capability from the wide range of guidance documents that the NRC has issued and that the "fundamental requirements" are further specification of the general guidance specified previously by the Commission in its regulations. (The staff is not proposing to change NRC regulations by its proposal.) The staff further states that the proposal does not alter previously issued guidance (NUREG documents). Based on the above, it would appear the staff proposal is not intended to change standards for emergency response facilities. The extent of change, if any, of emergency response capability will depend on how the general requirements of the staff proposal are to be implemented which may in turn depend on how staff reviewers apply the NUREG documents referenced in the staff proposal.
- The principal document referenced by the staff proposal which the Commission has approved is NUREG-0696.[3/] In its approval of NUREG-0696, the Commission directed the staff to add to the document a statement that the document provides general guidance only, is an acceptable way to meet the rules and regulations, and that compliance is not a requirement. The Commission informed Congress in NUREG-0755[4/] that NUREG-0696 was the NRC source for providing guidance to licensees in designing and constructing emergency response activities. Since the Commission viewed NUREG-0696 as a "guidance" document, a comparison of whether the provisions of NUREG-0696 are being changed may not be meaningful. However, it is our impression Commissioner Bradford may have desired such a comparison. Table 3 summarizes significant differences between the guidance in NUREG-0696 and the proposed requirements in the staff proposal.

[3/]Commission approval of NUREG-0696, "Functional Criteria for Emergency Response Facilities," was noted in a Staff Requirements Memorandum dated February 9, 1981.

[4/]NUREG-0755, "Report to Congress on Status of Emergency Planning for Nuclear Power Plants" (published March 1981).

- The Commission approved implementation schedules for emergency operating procedures (I.C.1 of NUREG-0737) and for the TSC, OSC, and EOF (III.A.1.2 of NUREG-0737). The staff proposal, if approved by the Commission, will result in changes to these Commission-approved implementation schedules. The staff notes that previously approved schedules cannot be met. There may be good reasons for licensees missing previously approved schedules, e.g., licensees made their best efforts, but schedules were unrealistic. The Commission may wish to discuss with the staff the possible impact of licensees not meeting Commission-approved schedules as part of the Commission briefing on SECY-82-111. The Commission may also wish to receive further information on what guidance will be provided to project managers in setting plant specific implementation schedules.

- Table 1 of the staff proposal does not appear to reflect Commission action on SECY-81-19, "Emergency Response Facilities," and SECY-81-509, "NUREG-0696 Criteria for Emergency Operations Facilities for Nuclear Power Plants." The Commission has agreed that the staff can accept close-in, hardened EOFs, provided that each emergency plan identify an alternate location where utility and government officials can meet to discuss plant status and appropriate public protective actions, and that the emergency plan indicate that contingency arrangements have been made to provide equipment for necessary communication with the TSC in the event of an emergency. It appears Table 2 of NUREG-0696, updated to cover radiologically hardened EOFs in accordance with SECY-81-509 and Staff Requirements Memorandum dated September 30, 1981 on SECY-81-509 may be appropriate to use in the staff proposal.

- It is our understanding that the Division of Emergency Preparedness (IE:DEP) disagrees with portions of the staff proposal from technical and policy aspects. It is our further under..anding that these disagreements will be addressed as part of the Commission briefing on SECY-82-111.

Summary and Recommendations

It is our understanding the following items will be addressed as part of the Commission briefing on SECY-82-111 now scheduled for this coming Wednesday, March 31, 1982.

- Response to Human Factors Society comments on the staff proposal.

- Response to IE:DEP disagreements with the staff proposal.

The Commission may also wish to consider the following items as part of its briefing on SECY-82-111.

- The impact of licensees not meeting Commission-approved implementation schedules.

- Comments on SPDS implementation (see page 3).
- Use of NUREG documents referenced in the staff proposal.
- Update of Table 1 of the staff proposal.

Enclosures:
As stated

cc: L. Bickwit
S. Chilk
W. Dircks
V. Stello

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Table 1

HUMAN FACTORS IMPROVEMENTS BETWEEN DECEMBER 29, 1981
OEDROGR DRAFT AND STAFF PROPOSAL

- The proposal recognizes that the implementation of the safety parameter display system (SPDS), Regulatory Guide 1.97, Control Room Design Review, and Emergency Operating Procedures should be integrated. However, the proposal also states that the schedule for implementing SPDS should not be impacted by schedules for the control room design review and development of symptom-oriented emergency operating procedures and that licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to other initiatives. (See further discussion in memorandum concerning implementation of SPDS.)
- The section on control room design review has been completely revised. It appears the staff has attempted to extract the basic provisions set out in NUREG-0700, "Guidelines for Control Room Design Review." In addition, the staff now will follow more closely the progress of control room design reviews and implementation of control room modifications.
- The section on emergency operating procedures has been completely revised to recognize current staff efforts.
- A recommended requirement was added that the SPDS, TSC, and EOF should be designed taking into account good human factors engineering principles.

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Table 2

PRELIMINARY RECOMMENDATIONS OF THE HUMAN FACTORS SOCIETY WHICH DIFFER FROM THE STAFF PROPOSAL[1]

- NUREG-0700,[2] or any subsequent improvement thereon, should be implemented as a requirement rather than as a guideline.
- The NRC should review license applications not only in accord with NUREG-0700 and NUREG-0801[3] but also in accord with the recommendations of EPRI-NP-1118.[4]
- The NRC should initiate appropriate rulemaking activity to require adherence to Section 6.3 of NUREG-0700 for existing annunciator systems.
- The systems analysis/review of NUREG-0700 should be completed prior to procedure development to ensure complete coverage of required procedures.
- Training requirements for the SPDS (which is not used on a day-to-day basis, but is critically important when needed) should be recognized.
- The need for an SPDS has not been established from any system or task analysis. A well designed control room may be satisfactory. Therefore, a thorough systems analysis should be done. The job/task analysis being done by INPO and the reactor operator task analysis being done by NRC must be coordinated with any similar analysis for SPDS. SPDS should be contingent upon the INPO and NRC job/task analysis efforts. However, a unique special systems analysis could be performed which would not delay SPDS if this analysis justifies the need.

1/These recommendations are extracted from a preliminary draft of a separate study being completed by the Human Factors Society for the NRC. This separate study, "A Comprehensive Human Factors Plan for Nuclear Reactor Regulation" should be published as a NUREG/CR document in May 1982 (estimate). The recommendations listed in this table are subject to change by the Human Factors Society.

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

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

4/EPRI-NP-1118, "Human Factors Methods for Nuclear Control Room Design."

Table 3

SIGNIFICANT DIFFERENCES IN THE GUIDANCE OF NUREG-0696
AND THE PROPOSED REQUIREMENTS OF THE STAFF PROPOSAL[1]

- Safety Parameter Display System (SPDS)

  (1) SPDS display need not be provided in the TSC and EOF as opposed to NUREG-0696.

  (2) Requirements for SPDS data validation are not addressed in the staff proposal.

  (3) The SPDS or a backup SPDS display system need not function during and after earthquakes as opposed to NUREG-0696.

  (4) Reliability goals, including a definition of what is "reliable," are not addressed in the staff proposal.

  (5) The concept that the SPDS is to provide a concise display of critical plant variables may not be highlighted to the extent NUREG-0696 addresses this subject. The staff proposal characterizes SPDS as a control room improvement while NUREG-0696 may tend to imply SPDS is an emergency response facility item.

- Technical Support Center (TSC)

  (1) The TSC needs to be within the site protected area -- NUREG-0696 would have the TSC as close as possible to the control room (see Enclosure 1 for additional details of location of TSC).

  (2) The TSC would be required to be fully operational within <u>one hour</u> after activation as opposed to NUREG-0696 criterion of <u>30 minutes</u>.

  (3) The TSC needs to be built in accordance with the National Uniform Building Code. NUREG-0696 indicated that the TSC must be able to withstand the most adverse conditions reasonably expected during the design life of the plant. Winds and floods with 100-year recurrence were considered acceptable as a design basis in NUREG-0696. (IE:DEP states that there is not a National Uniform Building Code, only local Uniform Building Codes.)

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[1] Since the Commission viewed NUREG-0696 as a "guidance" document a comparison of the provisions of NUREG-0696 with the proposed requirements of the staff proposal may not be meaningful. However, it is our impression Commissioner Bradford may have desired such a comparison. This table summarizes significant differences. Enclosure 1 provides a more detailed comparison of NUREG-0696 and the staff proposal.

(4) All variables listed in Regulatory Guide 1.97 and SPDS variables need not be available in the TSC as opposed to NUREG-0696.

- Emergency Operation Facility (EOF)

(1) The EOF needs to be built in accordance with the National Uniform Building Code. NUREG-0696 indicated that, in addition, it must be able to withstand adverse conditions of high winds (other than tornadoes) and floods. Winds and floods with 100-year recurrence were considered acceptable as a design basis in NUREG-0696.

(2) All variables listed in Regulatory Guide 1.97 and SPDS variables need not be available in the EOF, as opposed to NUREG-0696.

- Meteorological Information

The staff proposal would require in the TSC and EOF meteorological variables in Regulatory Guide 1.97 (Rev. 2) for the site vicinity and regional data available via communication from the National Weather Service. NUREG-0696 states that the TSC and EOF shall include the meteorological variable specified in Regulatory Guide 1.23 and NUREG-0654, Revision 1, Appendix 2 which in turn calls for meteorological measurements from primary and backup systems. (IE:DEP indicates NUREG-0696 and its reference documents also deal with supplemental meteorological measurements, a subject not addressed in the staff proposal.)

- Data Acquisition

Acquisition, processing, and management of data for SPDS control room improvements, and emergency response facilities need not be centralized nor have a common data base in the staff proposal as opposed to NUREG-0696 which encourages such measures.

Enclosure 1

STANDARDS FOR EMERGENCY RESPONSE FACILITIES

I. BACKGROUND

Commissioner Bradford expressed concern that standards for emergency response equipment and facilities are being relaxed. Commissioner Bradford requested OPE's assessment of the staff proposal with a special eye toward those standards, schedules or review methods which the Commission has approved, included in the PPG or told Congress we would review.

The staff proposal addresses the following specific items: Safety Parameter Display System (SPDS); Detailed Control Room Design Review (DCRDR); Regulatory Guide 1.97 Application to Emergency Response Facilities; Upgrade of Emergency Operating Procedures (EOPs); Technical Support Center (TSC); Operational Support Center (OSC); and the Emergency Operations Facility (EOF). The status of Commission approval of each item is as follows (with details provided in Section II below):

Item	Status of Commission Approval
SPDS	The Commission has approved the provision for an SPDS but not an implementation schedule for installation. The Commission also approved guidance on functional criteria for the SPDS.
DCRDR	The Commission approved the conduct of a DCRDR but not an implementation schedule or guidelines for conducting a DCRDR.
Regulatory Guide 1.97	The application of Regulatory Guide 1.97 to emergency response facilities is addressed in NUREG-0696.
EOPs	The Commission has approved the upgrade of EOPs including an implementation schedule.
TSC, OSC, EOF	The Commission has approved these items, including guidance on functional criteria, and implementation schedules.

II. EVALUATION

For each item of the staff proposal we discuss below the status of Commission approved documents, additional staff documents, and a comparison of the provisions of Commission approved documents with the DEDROGR proposal.

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A. Emergency Response Facilities

The staff proposal grouped three separate items (TSC, OSC, and EOF) under the heading of emergency response facilities. We will use the same format as the staff proposal.

1. Commission Approved Guidance

Requirements for the TSC, OSC, and EOF were included in Item III.A.1.2 of NUREG-0737.1/ It was noted in Item III.A.1.2 that the functional criteria for the TSC, OSC, and EOF were being developed in NUREG-0696.2/ The Commission subsequently approved NUREG-0696 on February 19, 1981. In approving NUREG-0696 the Commission directed the staff to add a note to the document stating NUREG-0696 provides general guidance only, is an acceptable way to meet the rules and regulations, and that compliance is not a requirement.

2. Additional Staff Documents

Staff review methods for determining the acceptability of the TSC, OSC, and EOF are contained in NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities." NUREG-0814 is not intended to specify new criteria beyond NUREG-0696.

3. Comparison with Approved Commission Guidance3/

a. Technical Support Center (TSC)

-- Location

NUREG-0696: As close as possible to the control room, preferably located within the same building. No more than two minutes walking time to control room. Provisions for safe and timely movement of personnel between the TSC and the control under emergency conditions.

1/Commission approval of NUREG-0737, "Clarification of TMI Action Plan Requirements" was noted in Staff Requirements Memorandum dated November 3, 1980 (M801028). Licensees and applicants for operating licenses were informed when NUREG-0737 was published that additional guidance on upgrade of emergency support facilities (Item III.A.1.2 of NUREG-0737) would be forwarded separately. In a Staff Requirements Memorandum of January 21, 1981 the staff was instructed to proceed with the issuance of Item III.A.1.2 of NUREG-0737. The staff issued Item III.A.1.2 of NUREG-0737 to licensees and holders of construction permits on February 18, 1981.

2/NUREG-0696, "Functional Criteria for Emergency Response Facilities."

3/The comparison is based on comparison of the staff proposal with NUREG-0696. In this regard we have attempted to summarize the provisions of both documents.

Staff proposal: Within the site protected area.

-- Size

NUREG-0696: Large enough to provide working space for personnel assigned to the TSC at the maximum level of occupancy including equipment, displays and documentation. A separate room adequate for at least three persons for private NRC consultation. Sized for a minimum of 25 persons, including 20 persons designated by the licensee and 5 NRC personnel. Minimum size of working space approximately 75 square feet/person.

Staff proposal: Sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation.

-- Construction

NUREG-0696: Able to withstand the most adverse conditions reasonably expected during the design life of the plant including adequate capabilities for (1) earthquakes, (2) high winds (other than tornadoes), and (3) floods. Winds and floods with a 100 year recurrence frequency are acceptable as design basis. Need not be Seismic Category I. Design of data system equipment to incorporate human factors engineering.

Staff proposal: Built in accordance with National Uniform Building Code. Design to take good human factor engineering principles into account.

-- Ventilation

NUREG-0696: Ventilation requirements comparable to the control room and to include as a minimum high-efficiency particulate air (HEPA) and charcoal filters. Need not be Seismic Category I, redundant, instrumented in control room, or automatically activated.

Staff proposal: Environmentally controlled to provide normal room air temperature, humidity and cleanliness appropriate for personnel and equipment.

-- Radiological Protection

NUREG-0696: Same radiological habitability as control room under accident conditions. TSC personnel to be protected from radiological hazards including direct radiation and airborne radioactivity to same degree as control room personnel. Applicable criteria specified in General Design Criterion 19, SRP 6.4 and NUREG-0737, II.B.2. Radiation monitoring system, installed or portable detectors, should be able to distinguish presence of radioiodines at concentrations as low as 10^{-7} uc/cc.

Staff proposal: Radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for duration of accident.

-- Communications

NUREG-0696: In addition to voice communication with control room, OSC, EOF, and NRC, provisions for communication with state and local operations centers. Voice communications facilities to include means for reliable primary and backup communication. Voice communications equipment to include hotline-telephones to NRC; dedicated phones to NRC health physics network, control room, OSC, and EOF; dial telephones to onsite and offsite locations; intercommunications systems between work areas of TSC; and communications to licensee teams and state and local operations centers prior to EOF activation. Also, designated telephones (at least two) for use by NRC, plus facsimile transmission capability between TSC, EOF and NRC Operations Center.

Staff proposal: Reliable voice and data communications with the control room and EOF. Reliable voice communications with OSC and NRC Operations Center and state and local operations centers.

-- Data System

NUREG-0696: Provide access to accurate information sufficient to determine plant steady-state operating conditions prior to accident, transient conditions producing the initiating event, and plant systems dynamic behavior throughout the course of the accident.

Data base complete enough to permit accurate assessment of accident without interference from control room emergency operation. Data base display to include Type A, B, C, D and E variables in Regulatory Guide 1.97, Rev. 2 as a minimum set and should include all data included in the data sets for the SPDS, for the EOF and for transmission to offsite locations. Data system need not meet safety grade or Class IE requirements but must be isolated from safety grade sensors. Operational unavailability goal of 0.1 while plant is above cold shutdown conditions.

Data base to include data storage and recall capability with capacity to record at least two weeks of additional post-event data. At least 2 hours of pre-event data and 12 hours of post-event data shall be recorded.

Display capability for providing trend information and time-history to give dynamic view of plant status during abnormal operating conditions. Display of SPDS to be provided.

Staff proposal: Capable of reliable data collection, storage, analysis, display and communications sufficient to determine site and regional status, determine changes in status, forecast status, and take appropriate actions.

Data set to include all required variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2) that are essential for performance of TSC functions and meteorological variables in Regulatory Guide 1.97 for site vicinity. National Weather Service data for region in which the plant is located.

Data available to evaluate incident sequence, determine mitigating actions, evaluate damage, and determine plant status during recovery operations.

-- Records Availability

NUREG-0696: Complete and up-to-date repository of plant records; drawings, schematics and diagrams of plant systems, and operational specifications and procedures, including Tech. Specs., plant procedures, emergency operating procedures, FSAR, plant operating records, plant operations reactor safety committee records, and records for performing EOF functions when it is not operational

Staff proposal: Accurate, complete and current plant records (drawings, schematic diagram, etc.) essential for evaluation of the plant under accident conditions.

-- Staffing

NUREG-0696: Sufficient technical, engineering and senior designated licensee officials to provide the needed support to the control room. A senior designated licensee official to coordinate activities in the TSC. Level of staffing may vary according to severity of emergency condition. Upon activation, full functional operation within 30 minutes.

Staff proposal: Sufficient technical, engineering and senior designated licensee officials to provide needed support. Fully operational within one hour after activation.

-- Implementation

NUREG-0696: For operating reactors, the conceptual design was to be submitted by June 1, 1981. For OL applications, the design information was to be provided in connection with OL review process. For all reactors licensed for operation prior to October 1, 1982, upgraded facilities to be operational by October 1, 1982. For reactors licensed after October 1, 1982, facilities to be operational prior to receiving an OL.

Staff proposal: Implementation date to be developed.

b. Operations Support Center (OSC)

-- Location

NUREG-0696: Separate from control room and TSC.

Staff proposal: Separate from control room.

-- Communications

NUREG-0696: Dedicated telephone extensions to the control room and TSC and one dial telephone to onsite and offsite locations. Direct voice or radio communications may supplement telephones.

Staff proposal: Reliable voice communications with the control room, TSC and EOF.

-- Habitability

NUREG-0696: If OSC habitability is not comparable to the control room, procedures required for evacuation of OSC personnel in the event of a large radioactive release and provisions made for performance of the OSC functions at other onsite locations.

Staff proposal: Not addressed.

-- Implementation

See discussion of TSC supra.

c. Emergency Operations Facility (EOF)

-- It appears function, location, staff and radiation protection features are intended to be the same.

-- Size

NUREG-0696: Large enough to provide working space for personnel and equipment; space for repair, maintenance and service of equipment, access to communications equipment and functional displays; and space for storage of plant records. Separate office space to accommodate at least 5 NRC personnel during emergencies. Sized for at least 35 persons, including 25 designated by licensee, 9 by NRC and 1 person by FEMA. Minimum size of working space approximately 75 square feet/person.

Staff proposal: Sufficient to accommodate and support Federal, state and licensee predesignated personnel, equipment and documentation.

-- Training

NUREG-0696: Periodic activation in accordance with the licensee's emergency plan for training and for emergency preparedness exercises.

Staff proposal: Not addressed.

-- Structure

NUREG-0696: In addition to meeting Uniform Building Code, must be able to withstand adverse conditions of high winds (other than tornadoes) and floods. Winds and floods, with a 100 year recurrence frequency, are acceptable for a design basis. Design of data system equipment to incorporate human factors engineering.

Staff proposal: Meet National Uniform Building Code. Designed taking into account good human factors engineering principles.

-- Radiological Monitoring

NUREG-0696: Provide radiation monitoring systems composed of installed monitors or dedicated portable monitoring equipment. Detectors able to distinguish the presence of radioiodines at concentrations as low as 10^{-7} uc/cc.

Staff proposal: Appropriate radiological monitoring systems.

-- Communications

NUREG-0696: Voice communications facilities to include reliable primary and back up means of communication, and include hotline telephone (ENS), health physics network (HPN), dedicated telephones for management communications, and designed telephones (at least 3) for NRC personnel; intercommunications between work areas of EOF if needed; radio communications to mobile monitoring teams; communications to state and local operations centers and communications to facilities outside EOF as needed, facsimile transmission capability between the EOF, TSC, and NRC Operations Center.

Staff proposal: Reliable voice and data communications facilities to TSC and control room, and reliable voice communications facilities to NRC, state and local emergency operations centers.

-- Data System

NUREG-0696: Receive, store, process and display information sufficient to perform assessments of the actual and potential onsite and offsite environmental consequences of an emergency situation.

As a minimum, EOF data set to include sensor data of the Type A, B, C, D, and E variables specified in Regulatory Guide 1.97 Rev. 2 and meteorological variables specified in proposed Rev. 1 to Regulatory Guide 1.23 and NUREG-0654, Rev. 1, Appendix 2. Data system need not be safety grade but must be isolated from safety grade sensors in accordance with GDC-24. Data system shall have operational unavailability goal of .01 during plant conditions above cold shutdown.

Data storage to cover two hours of pre-event data and 12 hours of post-event data and capacity to record at least two weeks of additional post-event data with reduced resolution.

Trend-information display and time-history display capability to provide a dynamic view of plant systems, radiological status and environmental status during an emergency. SPDS to be displayed in the EOF.

Staff proposal: Capable of reliable collection, storage, analysis, displays and communication of information on containment conditions, radiological releases and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from following categories that are essential to EOF function: (a) variables from appropriate Table 1 or 2 of Regulatory Guide 1.97 Rev. 2 and (b) meteorological variables in Regulatory Guide 1.97 Rev. 2 for site vicinity and regional data available via communication from National Weather Service.

-- Records Availability

NUREG-0696: Ready access to records to include plant technical specifications; plant operating and emergency operating procedures; FSAR; up-to-date records on licensee, state and local response plans; offsite population distribution data; evacuation plans; environs radiological monitoring records; licensee employee radiation exposure histories; and up-to-date drawings, schematics and diagrams of plant structures and systems and their in-plant locations.

Staff proposal: Up-to-date plant records (drawings, schematic diagram, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

-- Data Acquisition System

NUREG-0696: Discussion of a centralized data acquisition system including sources of technical data, acquisition of data, functional limitations, verification, reliability and configuration control.

Staff proposal: Acquisition, processing, and management of data for SPDS, control room improvements, and emergency response facilities should be coordinated, but need not be centralized.

-- Implementation

See discussion of TSC *supra*.

B. Safety Parameter Display System (SPDS)

1. Commission Approved Guidance

The requirement for installation of an SPDS was included in NUREG-0737, Item I.D.2. NUREG-0737 stated that the functional criteria for the SPDS were being developed in NUREG-0696 and schedules for implementation, type of staff review, licensee documentation, and required technical specification changes were to be determined in conjunction with issuance of NUREG-0696. The Commission subsequently approved NUREG-0696 on February 19, 1981. In approving NUREG-0696 the Commission directed the staff to add a note to the document stating NUREG-0696 provides general guidance only, is an acceptable way to meet the rules and regulations, and that compliance is not a requirement.

2. Additional Staff Documents

Staff review methods for determining the acceptability of the SPDS are contained in NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System" and NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities." Neither of these documents are intended to specify new criteria beyond NUREG-0696.

3. Comparison with Commission Approved Guidance[4]

-- Location

NUREG-0696: Located in control room with additional displays in the TSC and EOF.

Staff proposal: Located in control room.

-- Size

NUREG-0696: Compatible with existing space. Display does not interfere with normal movement or with full visual access to other control room displays.

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[4]The comparison is based on comparison of the staff proposal with NUREG-0696. In this regard we have attempted to summarize the provisions of both documents.

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Staff proposal: Not addressed.

-- Staffing

NUREG-0696: Design such that no additional personnel are required for its operation.

Staff proposal: Not addressed.

-- Display Considerations

NUREG-0696: Responsive to transient and accident sequences. For each mode of plant operation, a single primary display format. Incorporate accepted human factors principles. Minimum set of parameters to be displayed documented by licensee as part of the design. Data validated where practicable on real time basis as part of the display with means of identifying impacted parameter when unsuccessful validation of data occurs.

Staff proposal: Same as present with exception that display format and validation of data are not addressed.

-- Design Criteria

NUREG-0696: Total SPDS need not be Class 1E or meet single-failure criterion. Sensors and signal conditioners designed and qualified to Class 1E standards for those SPDS parameters that are also used by safety systems. For those parameters of the SPDS identical to the parameters specified in Regulatory Guide 1.97 sensors and signal conditioners designed and qualified to the criteria stated in Regulatory Guide 1.97. Processing and display devices of high quality and reliability. Display system (either primary or backup display) capable of functioning during and following all design basis events. Dynamic loading limitations of SPDS defined and incorporated into training program. Operational unavailability goal of 0.01 and cold shutdown unavailability goal of 0.2. Dependence on poorly human-engineered instruments scattered over control board unacceptable for SPDS function.

Staff proposal: The SPDS need not meet requirements of the single-failure criteria and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seisimically qualified indication is not required for the sole purpose of being a backup for SPDS. After the SPDS has been installed, operating procedures should be available that will allow timely and correct safety status assessment when the SPDS is not available.

C. Detailed Control Room Design Review (DCRDR)

1. Commission Approved Guidance

The requirement for a control room design review was included in Item I.D.1 of NUREG-0737. Licensees were to be required to complete a control room design review using NUREG-0700[5] on a schedule to be determined upon issuance of NUREG-0700. The staff has not issued to licensees requirements for conducting control room design reviews or forwarded for Commission approval further proposals for conducting control room design review. The Commission has not specially approved the contents of NUREG-0700.

2. Additional Staff Guidance

Staff review methods for determining the acceptability of the DCRDR are contained in NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Review." This document is not intended to establish new requirements.

3. Comments

It appears the staff proposal contains the basic provisions set out in NUREG-0700. The staff proposal does not contain the detailed guidance contained in NUREG-0700. OPE does not see any significant change in requirements for a DCRDR with respect to what a licensee must do, if NUREG-0700 is used as stated in the foreword of that document:

> "NUREG-0700, Guidelines for Control Room Design Reviews, is being issued to provide the guidance that the NRC staff believes should be followed to accomplish the control room design review .... NUREG-0700 is not a substitute for statutory requirements, and compliance with these guidelines is not a requirement. Approaches, methods, or reporting procedures different from the guidance provided herein are acceptable. However, alternative approaches as methods should be structured to ensure adequate human factors engineering considerations, and these considerations should be appropriately documented in any alternative reporting procedure."

The evaluation criteria for DCRDR are specified in NUREG-0801. The extent of staff review of licensee's DCRDR appears to have been changed by the staff proposal. NUREG-0801 indicates the staff will evaluate each licensee's program plan for DCRDR. The staff proposal is silent on this aspect. However, the staff proposal indicates several licensees will be selected for an in-progress audit by the NRR human factors staff based on the program plans and advice from resident inspectors.

[5] NUREG-0700, "Guidelines for Control Room Design Reviews."

D. Regulatory Guide 1.97 Application to Emergency Response Facilities

1. Commission Approved Guidance

The application of Regulatory Guide 1.97 to the TSC and EOF is discussed briefly in NUREG-0696.

2. Additional Staff Guidance

Separate documents describing staff review methods for determining the acceptability of implementation of Regulatory Guide 1.97 have not been produced.

3. Comments

In NUREG-0696, it was stated the TSC and EOF data set should include sensor data of the Type A, B, C, D, and E variables specified in Regulatory Guide 1.97, Rev. 2 as a minimum. The staff proposal would not require all the variables of Regulatory Guide 1.97 to be available in the TSC or EOF.

E. Upgrade of Emergency Operating Procedures (EOPs)

1. Commission Approved Guidance

The requirement for upgrade of emergency operating procedures was included in NUREG-0737.

2. Additional Staff Guidance

Draft criteria for preparation of emergency operating procedures is contained in NUREG-0799 which has been published for public comment and which will be reissued as NUREG-0899 after resolution of public comment. The Commission has not reviewed the NUREG document(s).

3. Comments

There appears to be no technical change in Commission approved guidance. The staff proposal would change previously approved implementation schedules.

Human Factors Society

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H.E. (MIKE) PRICE

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March 29, 1982

Mr. Victor Stello, Jr.
Deputy Executive Director for
Regional Operations and Generic Requirements
U. S. Nuclear Regulatory Commission
Maryland National Bank Building
7735 Old Georgetown Road
Bethesda, Maryland 20014

Dear Mr. Stello:

Thank you for sending us copies of the NRC staff recommendations on the requirements for emergency response capability, dated March 10, 1982. We have studied the recommendations and considered them carefully relative to our own evaluation of human factors requirements for nuclear reactor regulation. Our reactions to the staff recommendations are described in both general and specific terms.

One of our primary general concerns with the December 29, 1981 version of this document was that it created the impression of downgrading the importance of human factors considerations in the design and operation of nuclear power plants. Unfortunately, the overall impression conveyed by the current version of the NRC staff recommendations is not substantially different from that of the earlier version. In some respects there appears to have been a reduction of emphasis upon the importance of human factors in the wording of the current version.

We note that the following statements which appeared in the first paragraph of December 29, 1981 version have been deleted: "studies that followed the accident at TMI identified the need to improve the on-site and off-site capability for responding to accidents. The fundamental weakness revealed during these studies was the lack of attention devoted to the "man" in the "man/machine" equation. We must not detract from this finding."

By deleting these statements it appears that the NRC staff is not stressing to licensees the importance of human factors as an overall concern in emergency response capabilities.

Attachment 4

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We also note that some specific statements to the effect that - accepted human factors principles shall be taken into account - have been added to the current version of the document. Although these statements provide more positive recognition of human factors they do not constitute a particularly strong emphasis and are not likely to insure that licensees devote more than minimal attention to human factors. In some cases the impacts of these statements are more than offset by the implications of other recommended requirements and adjunctive statements.

We believe that is misleading to state, as is done more than once in the document, "in some cases a good SPDS may obviate the need for a large scale control room modifications." In fact, we believe there is greater validity to the reverse form of this statement; namely, a well-designed (from a human factors point-of-view) existing or modified control room may obviate the need for the SPDS. More to the point, however, we believe that the extent of control room modifications required for safe, efficient operation is independent of the existence of SPDS. The safety purpose and safety result of good human factors engineering design of control rooms is to minimize the likelihood that operators through their actions will contribute to the initiation or exacerbation of "abnormal and emergency conditions" and to maximize the likelihood that if these conditions do occur operators will respond quickly and correctly.

There is a realistic recognition on page 7, paragraph 2 that "the SPDS is used in addition to the basic components and serves to add and augment these components"; and, in the same paragraph, "after the SPDS has been installed operating procedures should be available that will allow timely and correct safety status assessment when the SPDS is not available." These latter statements provide the basis for a strong case that if appropriate improvements are made to the basic components of the control room the presumed need for SPDS would not exist. The statements clearly imply that the SPDS is not intended to be a primary source of information for operators to perform their duties. Yet the possibility remains that necessary changes in primary instrumentation and control may not be made because of the more highly emphasized incorporation of an SPDS and the NRC implied importance of it relative to the more basic modification of the control boards.

We strongly support the emphasis on coordination and integration of all the initiatives concerned with emergency response. We do not agree with the priority assigned to the initiatives. The document emphasizes the point that licensees should develop and propose an integrated schedule for implementation in which SPDS design is an input to the other initiatives. We identify two serious problems with the latter part of this recommendation and the associated suggested sequencing of steps for integrating the initiatives.

First, it is suggested on page 5, paragraph 8.a.(1) that the SPDS program should be initiated by "reviewing the functions of a nuclear power plant operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk (b) could cause operators to make cognitive errors in diagnosing them, and, (c) are not included in routine operator training program." In our judgment this cannot be done without having available the results of some kind of task analysis as suggested to be a part of the control room design review in NUREG-0700. For example, one cannot examine events which can cause operators to make cognitive errors in diagnosing them without knowing the expected task requirements, including data about the existing displays and controls. The need for function analysis and task analysis should be stated more explicitly.

Second, it is suggested on page 6, paragraph 8.c.(1) that the results of the control room design review be applied to verify SPDS parameter selections, data display and functions." However, according to the suggested integration sequence, the SPDS will already have been designed, built and installed in the control room (page 6, paragraph 8.a.3.). Rather than specify the SPDS design as an input to other initiatives, we suggest that a control room design review including the very basic element of a function/task analysis is the most basic requirements for initiating human factors safety improvements in nuclear power plant operation. The result may or may not indicate a requirement for a SPDS.

We are greatly concerned that there is no requirement for submittal, review and approval of the plan for control room design review. We stressed this point in our letter of January 12 concerning an earlier version of the staff recommendations. We still believe this requirement is necessary to help insure that the review, (1) will be directed at relevant human factors aspects to control room design, (2) will be conducted with the active participation of competent human factors professional personnel, and (3) will maximize the probability that modifications proposed as a result of the review will be planned and executed in accordance with accepted human factors principles. We strongly recommend that DCRDR program plan as described in NUREG-0700 be required and be reviewed by NRC in accordance with the general guidance of NUREG-0801 and that at least informal feedback be provided the licensees about areas of concern in the plan.

Also concerning control room design review, the language of the document, reinforces the position that human engineering deficiencies in the control room can be unchanged even though they may induce higher than necessary human error rate in the use of primary instrumentation and controls. For example, the language used in the Functional Statement of paragraph Section 5, Detailed Control Room Design Review does not properly emphasize the importance of review and modification. It has the effect of making the importance of control room design review and subsequent modifications subordinate to SPDS and the upgraded emergency operation procedures. In the context of a systematic integrated human factors approach, the results of human factors analysis in design of displays and controls, although bearing a somewhat complementary relationship with procedures development and training program development; nevertheless, maintains a priority position both logically and in terms of application. Procedures cannot be specified in the absence of the design of those objects with which one is to proceed and operator training cannot be specified meaningfully unless the objects that are to be operated and the procedures governing these operations are known.

Additionally, we are concerned about the likely results of a loose interpretation of the statements "flexibility is considered in the control room review because certain control board discrepancies (design deficiencies) can be overcome by techniques not involving control board changes. These techniques could include improved procedures, improved training, and SPDS." Although the first statement is true it refers to relatively minor errors in human factors engineering design. The fundamental concept of the human factors discipline is to design the equipment/system to match the capabilities and limitations of the human operator. If the design target cannot be achieved than one may consider tradeoffs involving training in special procedures. Unfortunately these tradeoffs usually are not easily come by in an already operational system as opposed to a system in design. Therefore, at the very least it should be incumbent upon a licensee that might propose an alternative to correcting a control board design deficiency to demonstrate that a proposed alternative is a completely adequate substitute.

Finally, we concur with the view expressed on page 10 in the Functional Statement: "Decisions can modify the control room would include consideration of long-term risk reduction and any temporary decline in safety after modification resulting from the need to re-learn maintenance and operator procedures." This kind of consideration is standard practice for a human factors evaluation of proposed changes in operating systems.

We have observed during the course of our study of the human factors requirements for the nuclear reactor regulation that far too much emphasis has been placed upon the possible decline in safety that might result from control room changes which would require operators to learn new task performance techniques. For the most part persons with no competence in the psychology of human learning have attempted to use the concept of negative transfer and interference in an effort to argue against making control room modifications. Significantly, we have heard reference to possible negative transfer only in conjunction with discussions with control room modifications - never in conjunction with discussion of upgrading (i.e., changing) of emergency operation procedures. Both would require re-learning on the part of the operators. However it seems that negative transfer is a matter for concern only if it is seen as resulting from an activity that involves appreciable costs.

We do not consider a potential for negative transfer resulting from having to learn to use new or modified control boards to be a problem of any practical operational or safety significance. There are several reasons for our position. First, many of the changes that need to be made on control boards are of the type that permit use of the displays and controls in conformance with population sterotypes. Such changes rather than interfering with learning and correct responses will soon enhance the probability of correct operator performance.

Any initial interference from old habituation can be easily discouraged. Second, new operating techniques are modified control boards will be learned quickly because they will be practiced frequently. Third, any assessment of the possible negative transfer resulting from control room modifications has to consider the high replacement rate of control room personnel. In this context concern about possible negative transfer is largely irrelevant.

With regard to the initiative to upgrade emergency operating room procedures in (page 15, paragraph 7) we concur with the recommended requirement that analysis to identify operator task and information in control requirements for these must be performed. We suggest that the need for integrated task analysis to support all initiatives in (including SPDS) be more clearly specified.

We support the recommended requirements for submission of technical guidelines and of the writer's guide for acceptance by NRC. However, we emphasize that the required acceptance of this document is no more important than would be the acceptance of the detailed control room design review plan discussed previously.

All of the emergency response facilities are mandated without requiring specific analysis of their use, task performance requirements, decisions to be made and information required to support the expected human performance. The recommended requirements are concerned primarily with physical features rather than functional requirements.

We have provided this lengthy and detailed commentary on the NRC staff recommendations because our study group is in a special independent position relative to the evaluation of human factors activities in the nuclear power generation community. We have stated our reactions frankly in the hope that they will be useful in enhancing the safety of operations of nuclear power plants.

Sincerely,

Charles O. Hopkins

Charles O. Hopkins
Technical Director

COH:ac

cc: Mr. Forrest J. Remick
Mr. Hugh Thompson
Mr. John Austin

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Consulting, Inc.

MANAGEMENT CONSULTANTS

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January 4, 1982

Nunzio Palladino, Chairman
US Nuclear Regulatory Commission
Washington, DC 20555

Subject: December 29, 1981 letter from V. Stello "Emergency Response Capability and Facilities"

Dear Chairman Palladino:

As a result of the accident at Three Mile Island the nuclear utility community and the NRC have recognized the importance of consideration of the area of human factors in the design and operation of nuclear power plants. We, members of this expert interdisciplinary community, are extremely concerned over the contents of the subject letter which totally disregards the need for human factor consideration in nuclear regulations.

Specifically, we note that while Mr. Stello cites a "fundamental weakness revealed during these (TMI) studies was the lack of attention devoted to the "man" in the "man-machine equation," all of the proposed regulations promoted by Mr. Stello are "machine" solutions to the "man" problem.

The human factor community has been working closely with members of the NRC's Division for Human Factor Safety to educate, provide guidance and promote consideration of human factors in control room instrumentation, use of procedures, training needs, staffing considerations, etc.

We encourage a multi-disciplinary approach of human factors experts and utility personnel to provide an integrated and correct consideration of the man-machine interface in nuclear power plants.

We feel that a fragmented and equipment oriented approach as described by the subject letter is both incorrect and misguided.

Sincerely,

Linda O. Lund

Linda O. Lund
President

cc: W. Dircks
V. Stello
H. Denton
R. DeYoung
R. Minogue
J. Davis
T. Murley

Attachment 5

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While the attached letter represents the sole expressed views of the undersigned Lund, Inc., several concerned human factor experts have serious concerns with the Dec. 29 letter.

Though our decision was to comment individually to pursue timeliness, my colleagues and I share the concern that the CRGR recommendations lack human factor consideration.

Each of us would be willing to be contacted to provide comments collectively at NRC's request:

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Statement on December 29, 1981 CRGR Letters
"Emergency Response Capability and Facilities"

As human factors specialists working in the nuclear industry, we have long been calling for an integrated approach to human factors efforts. We feel strongly that SPDS the TSC, EOF, OSC, revised EOP's, CR reviews and Reg. Guides 1.97 and 1.23 must not be considered "in a fragmented and uncoordinated manner".

We acknowledge and support the finding that the "fundamental weakness" revealed during post-TMI studies was the "lack of attention to the 'man' in the 'man-machine equation' ". We also concur with the statement that "well trained operating staff with clearly defined emergency roles is the cornerstone to accident response." However, we would add that:

- Well-designed and operational Control Rooms and
- Well-written and validated EOP's

are the necessary aides to the reactor operating crews and their supervisors during an emergency.

Although we support the initial statements, the remainder of this document appears to be in direct conflict with our knowledge of TMI-2 and the current state of affairs at US Nuclear Plants. Indeed, the recommendations focus in this document exclusively on the "machine' side of the "man-machine equation".

If a "well trained operating staff" is a "cornerstone of accident response" then the almost total lack of reference to training in the recommendations is a definite shortcoming. For example, recommendations for the Emergency Response Facilities deal almost exclusively with the equipment to be installed (the "machines") and make no reference to the staffing (the "men") in these facilities or the

- organization
- role assignment and
- training

of the staff members. Moreover, the supposed slowness of the on-site emergency response needs to be balanced by the fact that on-site, normal as well as emergency response is a concern.

The statement is made that "The SPDS and control room improvements are essential elements in operator training programs...." It is true that training programs for operators will have to familiarize operators with control room improvements and train them in the use of an SPDS. In addition, training programs will have to familiarize operators with changes made to EOP's. However, nowhere in this document is there a full understanding of the impact an SPDS will have on an operator and the implications that it will have on training.

An SPDS brings with it some definite problems. Not only are there difficulties in the development of the equipment itself, but,

- Where it is placed in the control room
- How its use is integrated in EOP's
- How it is used as a diagnostic tool vis-a-vis the Control Board and
- How the operator is trained to use the two diagnostic modalities (SPDS and the Control Board)

are important considerations.

The above four issues regarding SPDS are either ignored or glossed over lightly in the December 29, 1981 document.

To date a SPDS is still a device of the future. Vendors are still working on the design of such a system; so the validity of the usefulness of such a device has no basis to determine the actual usefulness of a SPDS to its trained user(s) during normal, abnormal or emergency conditions.

As with the Emergency Response Facilities, equipment ("machine") needs are focused on in the recommendations for a SPDS. Moreover, a SPDS as a solution to other control room modifications is "the cart before the horse".

The potential danger in this document, if implemented is that it follows the pre-TMI philosophy that it purports to deplore. Lund, being involved with the daily operations of nuclear facilities recognizes the need for Human Factor considerations in control room design, instrumentation, staffing, training, procedure use, etc.

This multi-disciplinary human factor approach is recognized and welcomed as long overdue by the personnel at nuclear power generating stations. The implementation of a SPDS does not cover all of these diverse areas and without Human Factor considerations will be more of a hinderance to, than a helpful device for the, nuclear community. That is, by encouraging the addition of an SPDS and misunderstanding the importance of the Control Room review and systematically developed EOP's yet another piece of machinery is to be added to the Control Rooms without substantial analysis of its impact on the operator or its use in overall plans to mitigate accidents.

A "man-machine equation" does not refer to the simple sum of two numbers, but to a dynamic interrelationship. The present design of the control room must be reviewed to identify existing human factor deficiencies and to assess the best solution: hardware, procedures or training. Also, since the need is for not just the best design or format, but the assessment of the functional utility of:

1. Control Board
2. SPDS
3. EOP's

a systems-oriented functional task analysis of the Control Room, the SPDS and the EOP's in relation to the crew that use them must be performed.

There is no basis for this document's statement that "The SPDS is an improvement in the control room because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention". Indeed if the type of systems-functional-task analysis referred to above is not done, an SPDS could easily detract from, not enhance operator performance.

As human factors specialists in the nuclear industry we have witnessed much confusion over human factors issues as presented in many of the cited NUREGs. This confusion in many cases comes from an unfamiliarity with the human factors field and its methodologies. We would urge more open communication between HF specialists and the NRC to help simplify and coordinate the HF effort in the nuclear industry. As concerned HF scientists we deplore the move back to a pre-TMI philosophy as illustrated by this document.

Submitted by the staff of Lund, Inc.

L.O.Lund, President

C. Sherwood
Dr. P. Haymond
G. Opetosky

TENTATIVE SCHEDULE FOR THE DISCUSSION OF
SECY-82-111, REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY
BY THE ACRS SUBCOMMITTEE ON HUMAN FACTORS AT THE
265TH ACRS MEETING

TOPIC	SPEAKER	APPROXIMATE TIME
I. Subcommittee Chairman's Report	D. Ward	9:30 a.m.
A. General Description of SECY-82-111		
1. Request of Chairman Palladino for Timely Comments by ACRS		
2. Format and Content		
B. Schedule for Full Committee Briefing, Discussion, and Action		
C. Subcommittee's Activities/Efforts to Date		
1. Past Meetings, Briefings, and Demonstrations		
2. Summary of the May 5th Subcommittee Meeting		
a. Documents Reviewed (see meeting folder)		
b. Speakers		
c. Major Areas of Controversy		
II. Briefing on SECY-82-111	V. Stello	9:45 a.m.
A. Background		
B. Commission Decisions Recommended		
C. Proposed Basic Requirements		
D. Proposed Implementation Plan		
E. Staff Use of NUREGs and Reg. Guides		

	SPEAKER	APPROXIMATE TIME
III. Comments by the Division of Human Factors Safety	H. Thompson	10:30 a.m.
A. How SECY-82-111 Affects DHFS Areas of Responsibility		
B. Issues of Controversy		
C. Recommendations		
IV. Comments by Other NRC Management	B. Grimes/ R. Mattson	11:00 a.m.
V. Comments by the Human Factors Society	C. Hopkins	11:10 a.m.
VI. Comments from Industry		
A. AIF/INPO	B. Coley	11:20 a.m.
B. KMC, Inc.	E.(Morris) Howard	11:40 a.m.
VII. Full Committee Discussion		
A. Positions of Subcommittee Speakers (including consultants) Not Making Presentations to the Full Committee	D. Ward	11:45 a.m.
B. Subcommittee Recommendations		
C. Consensus		
VIII. LUNCH		12:00 a.m.

A-484

The following pages A-485 thru A-486 has been deleted as 1 .

DELETION

EMERGENCY RESPONSE CAPABILITY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

May 7, 1982

APPENDIX XXVI
CRGR EMERGENCY RESPONSE CAPABILITY
PRESENTATION

A-487

4-5-82

BACKGROUND

CRGR REVIEW OF OVERALL NRC ACTIVITIES IN EMERGENCY RESPONSE (12/3/81)

-- NRC OFFICE ACTIVITIES NEEDED BETTER COORDINATION

-- ACTUAL REQUIREMENTS NOT CLEAR

-- NUREGS AND REG GUIDES SOMETIMES USED BY STAFF AS FIRM REQUIREMENTS

INITIAL DRAFT OF STAFF RECOMMENDATIONS (12/29/81)

ACRS BRIEFING (1/8/82)

FINAL STAFF RECOMMENDATIONS CONSIDERING ALL COMMENTS (3/10/82)

COMMISSION BRIEFING (4/15/82)

COMMISSION DECISIONS RECOMMENDED

1. APPROVAL OF STAFF'S PROPOSED SET OF BASIC REQUIREMENTS WHICH HAVE BEEN DISTILLED FROM THE BROAD RANGE OF GUIDANCE DOCUMENTS THAT NRC HAS ISSUED.

2. APPROVAL OF STAFF'S PROPOSED IMPLEMENTATION PLAN

   -- ISSUE PROPOSED REQUIREMENTS TO LICENSEES BY 50.54f LETTERS AS A REVISION TO NUREG-0737

   -- ESTABLISH PLANT-SPECIFIC SCHEDULES BY MUTUAL AGREEMENT WITH LICENSEES

   -- FORMALLY IMPLEMENT REQUIREMENTS AND SCHEDULES THROUGH PLANT-SPECIFIC ORDERS

A-489

S C O P E

EMERGENCY RESPONSE FACILITIES

-- TECHNICAL SUPPORT CENTER (TSC)

-- OPERATIONAL SUPPORT CENTER (OSC)

-- EMERGENCY OPERATIONS FACILITY (EOF)

CONTROL ROOM IMPROVEMENTS

-- SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

-- CONTROL ROOM DESIGN REVIEW

-- INSTRUMENTS FOR ACCIDENTS
(AS IDENTIFIED IN REG GUIDE 1.97)

OPERATOR CAPABILITY

-- IMPROVED EMERGENCY OPERATING PROCEDURES

-- TRAINING

A-490

PROPOSED BASIC REQUIREMENTS - INTEGRATION OF ACTIVITIES

IT IS ESSENTIAL THAT THE ACTIVITIES ON CONTROL ROOM IMPROVEMENTS, UPGRADING OF EMERGENCY OPERATING PROCEDURES, AND STAFF TRAINING BE INTEGRATED AT EACH PLANT. THE TSC AND EOF DESIGN AND IMPLEMENTATION ARE RELATED TO CONTROL ROOM IMPROVEMENTS IN COMMUNICATIONS AND INSTRUMENTATION.

PROPOSED BASIC REQUIREMENTS - SPDS

-- PROMPT IMPLEMENTATION OF SPDS IS IMPORTANT AS A SAFETY IMPROVEMENT, PROVIDING INFORMATION, AS A MINIMUM, ON:

- o REACTIVITY CONTROL
- o REACTOR CORE COOLING AND HEAT REMOVAL
- o REACTOR COOLANT SYSTEM INTEGRITY
- o RADIOACTIVITY CONTROL

-- SHOULD BE DESIGNED ACCORDING TO GOOD HUMAN FACTORS PRINCIPLES

-- MUST BE CONSIDERED DURING CONTROL ROOM DESIGN REVIEW

-- NO SEISMIC, CLASS IE OR SINGLE-FAILURE REQUIREMENTS

-- POST-IMPLEMENTATION REVIEW

A-492

PROPOSED BASIC REQUIREMENTS - CONTROL ROOM DESIGN REVIEW

-- PURPOSE IS TO IDENTIFY HUMAN ENGINEERING DISCREPANCIES

-- MULTIDISCIPLINARY REVIEW TEAM

-- FUNCTION AND TASK ANALYSIS REQUIRED

-- SIGNIFICANT DISCREPANCIES SHOULD BE CORRECTED

-- MUST BE INTEGRATED WITH OTHER ACTIVITIES SUCH AS SPDS, UPGRADED EMERGENCY OPERATING PROCEDURES, OPERATOR TRAINING AND NEW REG GUIDE INSTRUMENTATION

-- NRR REVIEW ON AN AUDIT BASIS

A-493

PROPOSED BASIC REQUIREMENTS - REG GUIDE 1.97

FOR CONTROL ROOM
- INDICATION OF TYPE A, B, C, D, E VARIABLES
- INDICATION OF WIND DIRECTION, SPEED AND ATMOSPHERIC STABILITY
- EQUIPMENT QUALIFICATION AS DETERMINED BY PENDING RULEMAKING

FOR TSC
- INDICATION OF TYPE A, B, C, D, E VARIABLES NEEDED FOR TSC FUNCTION
- NEED NOT MEET CLASS IE, SINGLE-FAILURE OR SEISMIC QUALIFICATION REQUIREMENTS

FOR EOF
- INDICATION OF VARIABLES NECESSARY TO PERFORM EOF FUNCTION
- NEED NOT MEET CLASS IE, SINGLE-FAILURE OR SEISMIC QUALIFICATION REQUIREMENTS

PROPOSED BASIC REQUIREMENTS - EMERGENCY OPERATING PROCEDURES

-- REANALYZE ACCIDENTS AND PREPARE TECHNICAL GUIDELINES

-- UPGRADE EMERGENCY OPERATING PROCEDURES CONSISTENT WITH GUIDELINES

-- OPERATOR TRAINING ON PROCEDURES PRIOR TO IMPLEMENTATION

-- MUST BE INTEGRATED WITH SPDS AND CONTROL ROOM DESIGN REVIEW

-- NRC REVIEW AND APPROVAL OF TECHNICAL GUIDELINES

-- OPPORTUNITY FOR PREIMPLEMENTATION REVIEW OF PROCEDURES

A-495

PROPOSED BASIC REQUIREMENTS - EMERGENCY RESPONSE FACILITIES

GENERAL REQUIREMENTS SPECIFIED FOR:
- LOCATION
- SIZE
- RADIATION PROTECTION
- RECORDS
- EQUIPMENT
- COMMUNICATIONS
- STAFFING

NO NRC APPROVAL OF CONCEPTUAL DESIGNS REQUIRED BEFORE CONSTRUCTION

A-496

A-497

PROPOSED IMPLEMENTATION PLAN

- ISSUE PROPOSED BASIC REQUIREMENTS TO LICENSEES BY 50.54f LETTERS AS A REVISION TO NUREG 0737

-- NRC PROJECT MANAGERS (WITH MANAGEMENT GUIDANCE) NEGOTIATE PLANT-SPECIFIC SCHEDULES WITH LICENSEES

-- NRR IMPLEMENT FORMAL REQUIREMENTS THROUGH PLANT-SPECIFIC ORDERS

-- CURRENT OCTOBER 1982 GENERIC DEADLINE FOR OPERATIONAL EMERGENCY RESPONSE FACILITIES WILL BE EXTENDED ON CASE-BY-CASE BASIS

-- DESIGN AND INSTALLATION OF INPLANT SYSTEMS AND FACILITIES SHOULD BE EXPEDITED

STAFF USE OF NUREGS AND REG GUIDES

-- TO BE USED AS GUIDANCE ONLY, NOT REQUIREMENTS

-- EDO INSTRUCTIONS TO STAFF

-- DISCLAIMER STATEMENT IN NUREGS

A-478

PRESENTATION TO ACRS

EMERGENCY RESPONSE CAPABILITY

MAY 7, 1982

HUGH L. THOMPSON, JR, ACTING DIRECTOR
DIVISION OF HUMAN FACTORS SAFETY

APPENDIX XXVII
PRESENTATION BY DIV. OF HUMAN FACTORS SAFETY

A-4597

CONTROL ROOM DESIGN REVIEW (ID1)

SECY 82-111	PRIOR TO SECY 82-111
o BASIC REQUIREMENTS PROVIDED	o NO FIRM REQUIREMENTS. o GUIDANCE (NUREG-0700 AND DRAFT NUREG-0801) CONSISTENT WITH SECY 82-111 BASIC REQUIREMENTS
o NEGOTIATED PLANT SPECIFIC SCHEDULES	o MANDATED SCHEDULE
o LICENSEE SUBMITS PROGRAM PLAN	o SAME

A-500

A-501

CONTROL ROOM DESIGN REVIEW (ID1) (CONT'D)

SECY 82-111	PRIOR TO SECY 82-111
o NRC REVIEWS PROGRAM PLAN AND SELECTS PLANTS FOR IN-PROGRESS ON-SITE AUDITS	o SAME
o LICENSEE CONDUCTS REVIEW	o SAME
o NRC CONDUCTS IN-PROGRESS ON-SITE AUDITS OF SELECTED PLANTS	o SAME

CONTROL ROOM DESIGN REVIEW (ID1) (CONT'D)

SECY 82-111	PRIOR TO SECY 82-111
o LICENSEE IMPLEMENTS ENHANCEMENTS	o SAME
o LICENSEE SUBMITS SUMMARY REPORT OF COMPLETED REVIEW	o SAME EXCEPT MORE DETAILED
o NRC REVIEWS SUMMARY REPORT	o SAME EXCEPT FOR TIME CONSTRAINT

A-502

CONTROL ROOM DESIGN REVIEW (ID1) (CONT'D)

o NRC CONDUCTS ON-SITE PRE-IMPLEMENTATION AUDIT OF SELECTED CONTROL ROOMS	o NRC CONDUCTS ON-SITE PRE-IMPLEMENTATION AUDIT OF ALL CONTROL ROOMS EXCEPT: - MULTI-UNIT (SIMILAR CONTROL ROOMS) - LICENSEES WITH INTEGRATED REVIEW OF SEVERAL PLANTS (SAMPLE ONLY) - OWNER'S GROUPS WITH INTEGRATED REVIEW OF SEVERAL PLANTS (SAMPLE ONLY)
o NRC WRITES SER	o SAME EXCEPT FOR TIME CONSTRAINTS
o LICENSEE IMPLEMENTS APPROVED CHANGES ON APPROVED SCHEDULE	o SAME

A-503

SPDS (ID2)

SECY 82-111	PRIOR TO 82-111
o BASIC REQUIREMENTS SIMILAR TO PRIOR GUIDANCE (NUREG-0696) EXCEPT NO SEISMIC REQUIREMENT	o NO FIRM REQUIREMENT. GUIDANCE CALLED FOR SEISMIC SPDS OR HUMAN FACTORED SEISMIC BACKUP
o NEGOTIATED PLANT SPECIFIC SCHEDULES	o MANDATED SCHEDULES

A-504

SPDS (ID2) (CONT'D)

A-505

SECY 82-111

o EARLY SPDS INSTALLATION AND IMPLEMENTATION

- DCRDR AND EOP SHOULD NOT IMPACT SPDS IMPLEMENTATION
- USE OF EOP GENERIC GUIDELINES AND DCRDR RESULTS TO VERIFY SPDS PARAMETER SELECTION, DATA DISPLAY AND FUNCTIONS
- TRAIN OPERATORS

PRIOR TO 82-111

o SPDS NOT SINGLED OUT FOR PROMPT IMPLEMENTATION.

IMPLEMENTATION OF SPDS BASED ON INTEGRATION OF SPDS DESIGN AND IMPLEMENTATION WITH DCRDR, EOP UPGRADE, TSC AND EOF DESIGN, AND TRAINING

SPDS (ID2) (CONT'D)

SECY 82-111

- o LICENSEE FREE TO START IMPLEMENTATION AFTER SAFETY ANALYSIS PROVIDED NO UNREVIEWED SAFETY QUESTION OR TECH SPEC CHANGE
  - LICENSEE SUBMITS SAFETY ANALYSIS TO NRC
  - DIRECTOR IE OR NRR REQUESTS OR DIRECTS LICENSEE TO CEASE IMPLEMENTATION IF THERE IS A SERIOUS SAFETY QUESTION OR IF SAFETY ANALYSIS IS SERIOUSLY INADEQUATE

PRIOR TO SECY 82-111

- o IMPLEMENT SPDS AFTER NRC APPROVAL
  - LICENSEE SUBMITS FOR NRC PREIMPLEMENTATION REVIEW:
    - DESCRIPTION OF V&V PROGRAM
    - RESULTS OF V&V PROGRAM
    - SUBMITS OR PROVIDES ACCESS TO SPDS DESIGN, TEST PLAN, AND TEST RESULTS FOR NRC AUDIT

A-506

NRC PRE-IMPLEMENTATION AUDIT OF SPDS
(IF REQUESTED BY LICENSEE)

1. o EVALUATE PLAN FOR V&V

   o AUDIT DESIGN FOR CONFORMANCE WITH FUNCTION AND ACCEPTANCE CRITERIA*
   (VERIFICATION)

2. AUDIT AS-BUILT DISPLAY FOR OPERATIONAL CONFORMANCE TO CRITERIA
   (VALIDATION)

3. AUDIT CONTROL ROOM INSTALLED SPDS FOR CORRECT RELATIONSHIP BETWEEN
   SENSORS AND DISPLAYED VARIABLES

A-567

* MAY BE DONE UNDER STEP 2

EMERGENCY OPERATING PROCEDURES

SECY 82-111	PRIOR TO SECY 82-111
RECOMMENDED REQUIREMENTS:	
1. REANALYZE TRANSIENTS AND ACCIDENTS PER NUREG-0737 AND PREPARE TECHNICAL GUIDELINES	SAME (NUREG-0737 ITEM I.C.1.)
2. UPGRADE EOPs CONSISTENT WITH 1. AND APPROPRIATE PROCEDURE WRITER'S GUIDE	WRITER'S GUIDE NOT MENTIONED
3. PROVIDE TRAINING ON EOPs PRIOR TO IMPLEMENTATION	SAME (NUREG-0737 ITEM I.C.1.)
4. IMPLEMENT UPGRADED EOPs	SAME (NUREG-0737 ITEM I.C.1.)

A-508

EMERGENCY OPERATING PROCEDURES

SECY 82-111	PRIOR TO SECY 82-111
DOCUMENTATION AND NRC REVIEW:	
1. SUBMIT TECHNICAL GUIDELINES TO NRC FOR 2 MONTH REVIEW	SAME, EXCEPT 6 MONTH REVIEW ALLOWED
2. SUBMIT A PROCEDURES GENERATION PACKAGE THREE MONTHS PRIOR TO DATE OF FORMAL OPERATOR TRAINING	NOT REQUIRED
PACKAGE INCLUDES:	
1. PLANT SPECIFIC TECHNICAL GUIDELINES	
2. WRITER'S GUIDE	NOT REQUIRED
3. DESCRIPTION OF VALIDATION OF EOPs	NOT REQUIRED
4. DESCRIPTION OF TRAINING PROGRAM	NUREG-0737 ITEM I.C.1. REQUIRED BUT NO SUBMITTAL ADDRESSED THIS TRAINING DIRECTLY

A-509

SECY 82-111

- RECOGNIZES THE NECESSITY OF AN INTEGRATED APPROACH TO RESPOND TO RELATED REQUIREMENTS
- BALANCED EMPHASIS ON THE MAN-MACHINE EQUATION
- ACKNOWLEDGES THE NEED FOR A FUNCTIONAL APPROACH TO ACHIEVE SUCCESSFUL AND TIMELY IMPLEMENTATION
- RECOGNIZES THE SPDS AS A CONTROL ROOM AID AND PERMITS TIMELY AND COST-EFFECTIVE IMPLEMENTATION
- RESTORES THE UTILITY TO ITS PROPER ROLE

SECY 82-111 OBSERVATIONS

- SPDS DESIGN CAN AND SHOULD DEVELOP FROM THE EOP GUIDELINES

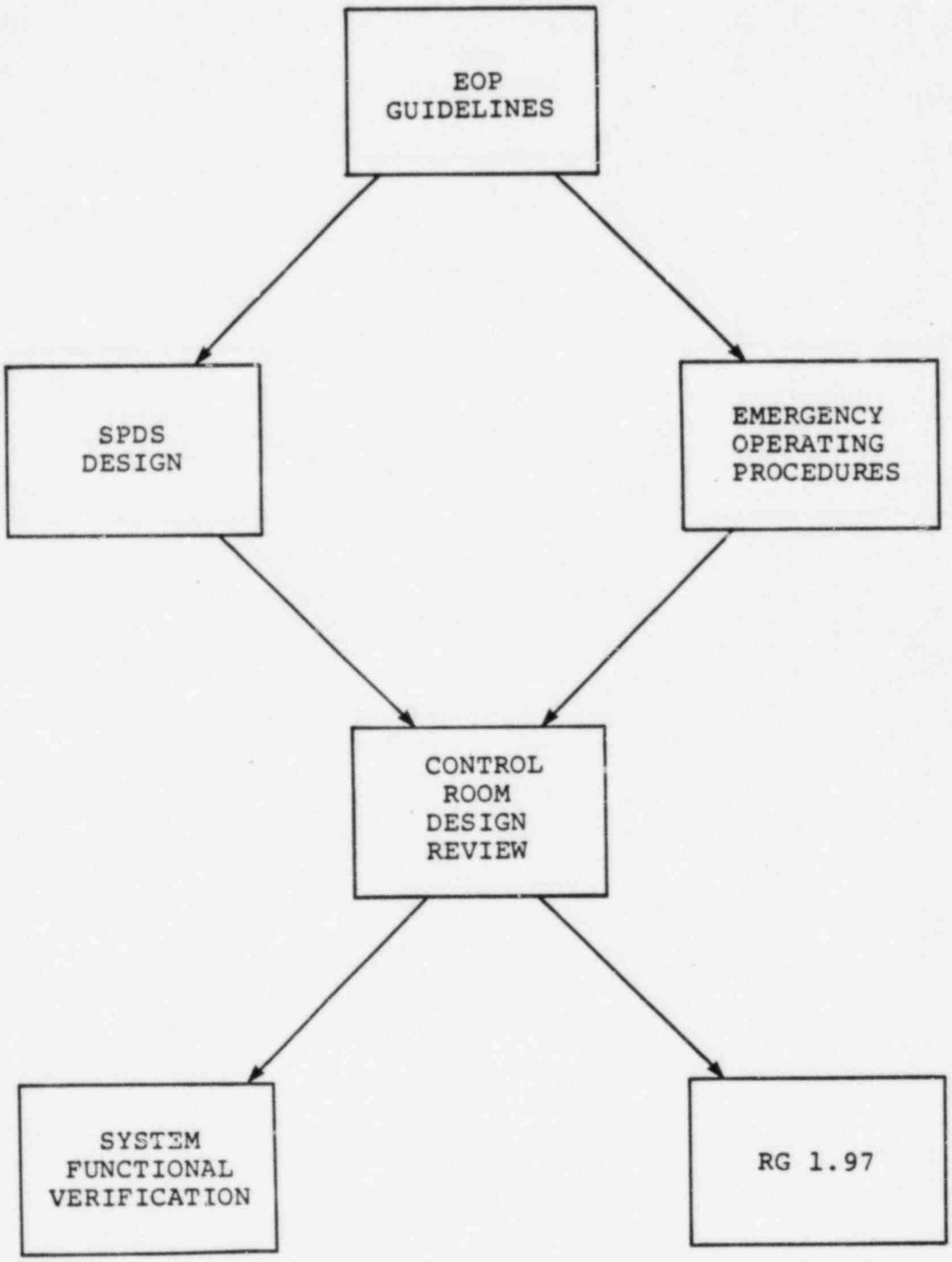
- RG 1.97 COMPLIANCE SHOULD BE A BY-PRODUCT OF THE CONTROL ROOM DESIGN REVIEW AND PLANT SPECIFIC FUNCTIONAL REQUIREMENTS

- THE BALANCED APPROACH OF SECY-82-111 SHOULD EXTEND THROUGH THE NRC REVIEW PROCESS

- THE CONCERN FOR HARDWARE "RIP-OUTS" SHOULD EXTEND EQUALLY TO SOFTWARE

A-511

SECY 82-111
INTEGRATION OF INITIATIVES



SAFETY PARAMETER DISPLAY SYSTEM

GOAL: "...CONCENTRATE A MINIMUM SET OF PLANT PARAMETERS FROM WHICH THE PLANT SAFETY STATUS CAN BE ADDRESSED." (NUREG-0696)

"...PROVIDE A CONCISE DISPLAY OF CRITICAL VARIABLES TO THE CONTROL ROOM OPERATORS TO AID THEM IN RAPIDLY AND RELIABLY DETERMINING THE SAFETY STATUS OF THE PLANT." (SECY-82-111)

SPDS INTEGRATION PROGRAM OBJECTIVES

- DEFINITION OF THE ROLE AND MISSION OF THE BASIC SPDS
- FUNCTIONAL OPERATIONAL AND DESIGN REQUIREMENTS FOR THE BASIC SPDS
- GUIDELINES FOR AN EFFECTIVE SPDS IMPLEMENTATION PROGRAM
- CRITERIA FOR SPDS INTEGRATION

SPDS INTEGRATION PROGRAM

- O THE INDUSTRY HAS TAKEN THE INITIATIVE AND PROCEEDED WITH THE DEVELOPMENT OF SYMPTOM-ORIENTED EMERGENCY OPERATING PROCEDURES (EOPs) WHICH WILL ENHANCE AN OPERATING CREW'S ABILITY TO MANAGE EMERGENCY CONDITIONS.

- O THE NEXT KEY STEP TO ENSURE APPROPRIATE OPERATING CREW RESPONSE TO EVENTS IS TO PROVIDE OPERATING CREWS WITH TRAINING AND AIDS WHICH SUPPORT SYMPTOM-ORIENTED EOPs.

- O MANY UTILITIES SHARE THE OPINION THAT THE SAFETY PARAMETER DISPLAY SYSTEM (SPDS) IS AN AID IN MAINTAINING CRITICAL SAFETY FUNCTIONS.

- O TO BE EFFECTIVE IT MUST BE INTEGRATED WITH EOPs AND OPERATOR TRAINING.

- O RESPONDING TO THIS CONCERN, AN INDUSTRY WORKING GROUP ON SPDS INTEGRATION HAS BEEN FORMED AND HAS DEVELOPED A PROGRAM WHOSE PRINCIPAL OBJECTIVE IS TO SUPPORT THE IMPLEMENTATION OF A BASIC AND EFFECTIVE SPDS IN A TIMELY MANNER.

- THE PROGRAM IS BEING DEVELOPED WITH THE UNDERSTANDING THAT THERE IS NO "GENERIC" SPDS THAT SHOULD BE UNIFORM TO THE INDUSTRY: THAT THERE IS A WIDE SPECTRUM OF EFFECTIVE SPDS DESIGNS.

- THE GROUP IS PREPARING THE FOLLOWING GENERIC INFORMATION:

  - DEFINITION OF THE ROLE AND MISSION OF THE BASIC SPDS

  - FUNCTIONAL, OPERATIONAL, AND DESIGN REQUIREMENTS FOR THE BASIC SPDS

  - GUIDELINES FOR AN EFFECTIVE SPDS IMPLEMENTATION PROGRAM

  - CRITERIA FOR SPDS INTEGRATION

- ALL GENERIC WORK COMPLETED BY THE GROUP WILL BE DOCUMENTED AND DISTRIBUTED THROUGHOUT THE INDUSTRY; IT WILL PROVIDE UTILITIES A LOGICAL BASIS FOR THEIR PLANT-SPECIFIC IMPLEMENTATION PROGRAMS.

- IT APPEARS THAT THERE IS GENERAL ENDORSEMENT OF SECY 82-111.

- THE SPDS/EOP/TRAINED OPERATOR TRIAD IS A SYSTEM FOR RESPONDING TO EMERGENCIES; IT WOULD BE INAPPROPRIATE AND INEFFECTIVE TO IMPLEMENT AN SPDS WITHOUT THE NECESSARY EOPs AND OPERATOR TRAINING.

I am Morris Howard and I am with KMC, Inc. KMC represents about 30 utilities which have been identified as the Coordinating Group for Emergency Preparedness. I am speaking on behalf of that group today, and my remarks will be very brief.

On Wednesday, I appeared before your subcommittee on Human Factors and provided some details on how the NRC was developing and implementing requirements for emergency planning facilities. At that meeting, I chose to use the development of requirements for the Technical Support Center to demonstrate how the proliferation of requirements by the NRC had gotten out of hand. I do not plan to repeat that discussion, but your staff has been provided copies of the graphics and narrative which we discussed on Wednesday.

The point that needs to be made, is that prior to the creation of the CRGR, the NRC staff elements, whether in the Office of Nuclear Reactor Regulations or the Division of Emergency Preparedness within the Office of Inspection and Enforcement, were developing requirement documents independent of each other and in many instances in isolation. The documents generated, included letters to licensees as well as many NUREG documents. These documents have, in many cases, provided very procriptive requirements and established inflexible implementation dates. There were treated as requirements by the NRC and their contractors, and in fact were, in many instances used by the inspectors as the basis for inspections.

The CRGR, as described in the SECY 82-111 paper, would establish the functional requirements of the emergency planning facilities and permit the licensee to establish the best means to implement those requirements to meet its needs. Also, the schedule for implementation would be for the licensee to propose. The NRC would, of course, review any or all aspects of the licensees proposal and implementation schedule, and the existing NUREG documents would be available as guidance but not be treated as the only acceptable means of approach.

The Coordinating Group for Emergency Preparedness is encouraged that Senior Management in the Commission has recognized, and is attempting to control, the proliferation of both formal and informal "requirements" which has been the way of life since the Three Mile Island accident. The utilities believe that the CRGR can be an effective means of assuring that each organizational entity does not become its own "rulemaker", and establish implementation dates without regard for other requirements which may already be straining the finite resources of the utility. It is refreshing to see industry being consulted on cost benefit analyses, and their input being factored into safety decisions. We believe that it is necessary to jointly discuss, and agree upon, priorities and completion dates, and feel that the CRGR proposal assigning the Project Manager with a responsibility to initiate these efforts is long overdue.

In summary, the Coordinating Group for Emergency Preparedness strongly supports the CRGR, and feel that the way of doing business proposed by the CRGR in SECY 82-111, will improve planning and lead to implementation of those items of the greatest safety significance at the earliest possible time.

Ray:car 5/6/82

APPENDIX XXX
REPORT ON MEETING OF SUBCOMMITTEE ON QUALIFICATION PROGRAM FOR SAFETY RELATED EQUIPMENT - MAY 5, 1982

REPORT ON MEETING OF SUBCOMMITTEE ON QUALIFICATION PROGRAM FOR SAFETY RELATED EQUIPMENT - May 5, 1982

The Subcommittee met on May 5, 1982 to review and discuss the final version of a rule, an addition [section 50.49] to 10 CFR 50, entitled, "Environmental Qualification of Electric Equipment for Nuclear Power Plants." Messrs. S. K. Aggarwal, W. Johnston, and D. Sullivan represented the NRC Staff. Messrs. Bender, Ebersole, and Ray represented ACRS; the ACRS consultants present were P. Davis and W. L. Lipinski. The ACRS staff members attending were Messrs. Savio and Cappucci.

The new Rule deals with the environmental qualification of Class IE electrical equipment and certain non-Class IE equipment such as post-accident monitoring systems. For the earliest plants in service such qualification was based on the premise that the equipment used would be of the highest industrial quality available. The use of IEEE standard 323-1971 was implemented in 1971 for this purpose. On July 1, 1974, Regulatory Guide 1.89, "Qualification of Class IE Equipment for Light-Water-Cooled Nuclear Power Plants," which endorses IEEE standard 323-1974, was implemented as the basis for qualification. Subsequently a document entitled, "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment, in Operating Reactors" (DOR Guidelines) and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" were issued (in late 1979) and implemented by a Commission Memorandum and Order CLI-80-21 dated May 23, 1980 as an interim basis for qualifying equipment for operating plants and for plants under licensing review. The Commission also directed the Staff to proceed with a rulemaking on environmental qualification to codify and clarify NRC practice.

The Electrical Systems Subcommittee reviewed this Rule in the proposed state and a revision of the associated regulatory guide - Regulatory Guide 1.89 with the Staff on July 22, 1981. At that time, the Rule included seismic qualification. These documents were subsequently reviewed by the ACRS at the 256th meeting in August 1981 and endorsed for issuance for public comment. Both have received public review. Comments on the Rule have been processed by the Staff, and the Rule revised. By Commission Directive the seismic requirements were deleted from the Rule, and the Staff was instructed to develop a separate seismic rulemaking at a later date. Comments on the Regulatory Guide have been received by the Staff but review and revision of the Guide is not anticipated to be completed until the winter of 1982 because of Staff personnel limitations.

The final Rule was discussed at our meeting on May 5. During the meeting, the Staff discussed their review of the public comments received and the revisions to the Rule in response. We were favorably impressed by the scope of response by the Staff to the public comments and the degree of changes made in the Rule to comply with the industry comments. We also heard comments on the Rule by representatives from industry. While several commented that the Rule had been improved, there were residual comments, as might be expected. A representative of the Nuclear Utility Group on Equipment Qualification has requested an opportunity to speak briefly to the full Committee today.

A major provision of the Rule will "grandfather" those operating plants and plants under license review for which qualification is in progress under the DOR guidelines and NUREG-0588 or commences in the 90-day period following the effective date of the Rule. This was an area of major concern to industry. The major residual concerns expressed by industry, with which the Subcommittee concurs, are as follows:

Regulatory Guide 1.89 - review, revision, and issuance should be expedited so that the Rule and associated Guide will be concurrently available to industry. The revised Guide should integrate (by reference or otherwise) all guides bearing on environmental and/or seismic qualification of electrical equipment.

The Rule should be revised before issuance to include seismic qualification. Industry expressed concern over the possibility that changes in specific equipment to meet environmental qualifications may subsequently be required again to meet as yet unknown seismic requirements.

NTOL plants - concern was expressed by industry over the absence in the Rule of a statement specifically including equipment qualification in NTOL plants in the "grandfather" provision cited above.

As issued for public comment, the Rule included a requirement that equipment needed to complete one path of achieving and maintaining a cold shutdown condition be environmentally qualified. In response to industry comments, this requirement was deleted in the final rule. The Subcommittee heard justification of the deletion presented by Dr. R. Mattson and concurs. However, deletion poses a problem for the Staff which they would like to discuss today.

In commenting on the earlier version of the Rule, the Electrical Systems Subcommittee expressed concern that the proposed Rule did not contain "an interpretable designation of the equipment which will be affected" by the regulatory action. In response to this comment, the Rule has been narrowed in scope and changed to specify the equipment by functions, rather then by specific listing of equipment components [see Item 2(c), pg. 11 of the rule]. The Equipment Qualification Subcommittee feels this is an improvement, but that in the absence of a specific list there remains a potential for significant oversight and omission of equipment during qualification.

In view of the limited time available on the agenda for treatment of this subject, we have suggested that the Staff limit their planned presentation today to a response to the foregoing concerns and to the following:

1. Briefly highlight the major industry comments and the Staff response.
2. Staff concerns with deletion of the cold shutdown requirements.
3. Staff treatment of the value-impact statement for the Rule.

J. J. Ray
Chairman, ACRS Subcommittee on
Qualification Program for
Safety-Related Equipment
5/6/82

A-524

The following pages A-525 thru __________ has been deleted as 1 .

DELETION

ENCLOSURE 1

MAY 7 1982 [7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Environmental Qualification of Electric Equipment
for Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: ~~Proposed~~ Final Rule.

SUMMARY: The Nuclear Regulatory Commission is ~~proposing to~~ amending its regulations applicable to nuclear power plants to clarify and strengthen the criteria for environmental qualification of electric equipment. Specific qualification methods currently contained in national standards, regulatory guides, and certain NRC publications for equipment qualification have been given different interpretations and have not had the legal force of an agency regulation. This amendment will ~~The proposed rule would~~ codify the~~se~~ environmental qualification methods and criteria which meet the ~~and clarify the~~ Commission's requirements in this area.

EFFECTIVE DATE: [UPON publication in the Federal Register]

~~DATES: Comment period expires (60 days after publication in the FederalRegister). Comments received after ------------ will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.~~

~~ADDRESSES: Written comments and suggestions may be mailed to the Secretary of the Commission, Attention: Docketing and Service Branch,~~

~~U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, or hand-delivered to the Commission's Public Document Room at 1717 H Street NW., Washington, D.C., between the hours of 8:30 a.m. and 4:45 p.m. on normal work days.~~

FOR FURTHER INFORMATION CONTACT: Satish K. Aggarwal, Office of Nuclear Regulatory Research, ~~Electrical~~ Engineering ~~Branch,~~ U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301)443-5946.

\# PREVIOUS NOTICE

SUPPLEMENTARY INFORMATION: On January 20, 1982, NRC published in the Federal Register a notice of proposed rulemaking on ~~(47FR2876) for public comment a proposed rule on~~ environmental qualification of electric equipment for nuclear power plants (47FR 2876). The comment period expired March 22. A total of 69 comment letters were received by ~~until~~ April 6, 1982, raising 10 major issues. An additional 10 comment letters were received ~~until~~ by April 21, 1982, but no new issues were raised. The major issues are discussed below. ~~This effective rule incorporates the resolution of public comments which were received in response to the proposed rule.~~

\# Nature and scope of the Rulemaking

Nuclear power plant equipment important to safety must be able to perform the safety functions throughout its installed life. This requirement is embodied in General Design Criteria 1, 2, 4, and 23 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; in Criterion III, "Design Control," and Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; and in 10 CFR 50.55a(h), which incorporates by reference IEEE 279-1971,[1,2] "Criteria for Protection Systems for Nuclear Power Generating Stations." This requirement is applicable to equipment located inside as well as outside the containment.

[1]Incorporation by reference approved by the Director of the Office of Federal Register on January 1, 1981.

[2]Copies may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, N.Y. 10017.

A-527

The NRC has used a variety of methods to ensure that these general requirements are met for electric equipment ~~important to safety~~. Prior to 1971, qualification was based on the fact that the electric components were of high industrial quality. For nuclear plants licensed to operate after 1971, qualification was judged on the basis of IEEE 323-1971. For plants whose Safety Evaluation Reports were issued since July 1, 1974, the Commission has used Regulatory Guide 1.89, "Qualification of Class IE Equipment for Light-Water-Cooled Nuclear Power Plants," which endorses IEEE 323-1974,[2] "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," subject to supplementary provisions.

Currently, the Commission has underway a program to reevaluate the qualification of electric equipment ~~important to safety~~ in all operating nuclear power plants. As a part of this program, more definitive criteria for environmental qualification of electric equipment have been developed by the NRC. A document entitled "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines) was issued in November 1979. In addition, the NRC has issued NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which contains two sets of criteria: the first for plants originally reviewed in accordance with IEEE 323-1971 and the second for plants reviewed in accordance with IEEE 323-1974.

By its Memorandum and Order CLI-80-21 dated May 23, 1980, the Commission directed the staff to proceed with a rulemaking on environmental qualification of safety-grade equipment and to address the question of backfit. The Commission also directed that the DOR Guidelines and NUREG-0588 form the basis for the requirements licensees and applicants must meet until the rulemaking has been completed. This proposed rule is generally based on the requirements of the Division of Operating Reactors (DOR) Guidelines and NUREG-0588. Requalification of electric equipment in accordance with this rule will not be required for equipment qualified or being qualified in accordance with DOR Guidelines and its supplements including ~~IE Bulletin 79-01B~~ stet or NUREG-0588 and generic letters including 82-09, provided the qualification of electric equipment ~~program~~ has commenced prior to 90 days after the effective date of the rule.

~~The Commission's Memorandum and Order CLI-80-21 directed that the environmental qualification of electric equipment in operating nuclear power plants be completed by June 30, 1982. However, on September 23, 1981, the Commission considered the petition (SECY-81-486) to extend this deadline. The proposed rule covers the same electric equipment as CLI-80-21 and implements SECY-81-486 by incorporating the extension dates recommended by the Chairman in his memorandum dated September 30, 1981. Included in the proposed rule is a requirement that each holder of or each applicant for a license to operate a nuclear power plant identify and qualify the electric equipment needed to complete one path of achieving and maintaining a cold shutdown condition. The Commission specifically requests comment on this proposed additional requirement.~~

The dates specified in this rule for completion of environmental qualification of electric equipment apply to all licensees and applicants, and supersede any date previously imposed. No changes to licenses or technical specifications are necessary to reflect these new completion dates.

The scope of the proposed final rule does not include all electric equipment important to safety in its various gradations of importance. It includes that portion of equipment important to safety commonly referred to as "Class 1E" equipment in IEEE national standards and some additional non-Class 1E equipment and systems whose failure under extreme environmental conditions could prevent the satisfactory accomplishment of safety functions by accident-mitigating equipment.

Included in the ~~proposed~~ final rule are specific technical requirements pertaining to (a) qualification parameters, (b) qualification methods, and (c) documentation. Qualification parameters include temperature, pressure, humidity, radiation, chemicals, and submergence. Qualification methods include (a) testing as the principal means of qualification and (b) analysis and operating experience in lieu of testing. The proposed rule would require that the qualification program include synergistic effects, aging, margins, radiation, and environmental conditions. Also, a record of qualification must be maintained. Revision 1 to Regulatory Guide 1.89 which has been issued for public comments, ~~is being revised to~~ will describe methods acceptable to the NRC for meeting the provisions of this ~~proposed~~ rule and ~~to~~ include a list of typical equipment covered by it. ~~a draft of the proposed~~ The final version of Revision 1 ~~is being published for public comment concurrently with the proposed rule.~~ to the Regulatory Guide will be issued after resolution of public comments.

~~Also included in the proposed rule is a requirement, which is consistent with Commission Memorandum and Order, CLI-80-21, for submission of an analysis by licensees to ensure that the plant can be safely operated pending completion of the environmental qualification of electric equipment. The Commission expects that, for each of the currently operating power plants, this analysis and its evaluation by the NRC staff will be completed well in advance of the effective date of this rule. If the licensees of operating power plants fail to provide these analyses in a timely manner, the Commission expects the NRC staff to take the appropriate steps to require that the information be provided and to enforce~~

~~compliance-with-this-requirement.--This-requirement-has-been-included-in this-proposed-rule-to-provide-a-regulatory-basis-for-enforcement.~~

NRC will generally not accept analysis alone in lieu of testing. Experience has shown that qualification of equipment without test data may not be adequate to demonstrate functional operability during design basis event conditions. To insure integrity of a testing program, the same piece of equipment must be used throughout the complete test sequence ~~Analysis-may-be-acceptable-if-testing-of-the-equipment-is-impractical-because-of-size,-or-limitation-due-to-the-state-of the-art.~~ ~~The proposed rule takes into consideration the prior qualification history of the operating power plants. For example, the proposed rule recognizes that for those plants which are not committed to either IEEE 323-1971 or IEEE 323-1974 for equipment qualification, and have been tested only for high temperature pressure, and steam, some equipment may not need to be tested again to include other service conditions such as radiation and chemical sprays. The qualification of equipment for these service conditions may be established by analysis.~~

The proposed rule would require that each holder of an operating license provide a list of electric equipment previously qualified based on testing or analysis, or a combination thereof, and a list of equipment that has not been qualified. These lists and the schedule for completion of equipment, qualification would have to be submitted written 90 days after the effective date of this rule. ~~However,-this-time-period-will be-adjusted-during-the-final-rule-making-process-to-allow-reasonable-time for-licensees-to-evaluate-NRC's-safety-reviews-that-are-currently-underway.~~

~~The-proposed-rule-will-codify-the-Commission's-current-requirements for-the-environmental-qualification-of-electric-equipment.--Upon-publication-of-a-final-rule,-the-DOR-guidelines-and-NUREG-0588-will-be-withdrawn-.~~

The general requirements for seismic and dynamic qualification for electric equipment are contained in the General Design Criteria. Pending development of specific requirements in this area, the general requirements will continue to apply. NRC is considering expansion of the scope of this rule to include additional electric equipment important to safety. This matter will be the subject of a future rulemaking.

~~Additional views of Commissioner Bradford. Commissioner Bradford believes that the proposed deadline (second refueling outage after March 31, 1982) for qualification is much too relaxed, given the fact that licensees and the NRC have been aware of the problems in this area since 1978. The proposed deadline extends as much as two and one-half years beyond the June 30, 1983 date by which the Atomic Industrial Forum concluded that nearly all electrical equipment could be qualified. Given the more generous deadline, he also believes that the rule should have contained requirements for seismic and dynamic qualification. While the general design criteria contain requirements in this area, clarification now would ensure that equipment to be replaced in the near term will not have to be ripped out in a few years because it was not properly seismically qualified.~~

~~Commissioner Gilinsky has agreed with these views.~~

COMMENTS ON THE PROPOSED RULE

The Commission received and considered 69 letters from the public, received by April 6, 1982, commenting on the proposed rule. Copies of those letters and ~~an analysis of the public comments~~ a staff response to each comment are available for public inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street, NW, Washington, DC. Single copies of the analysis of the comments may be obtained, while the limited supply is available, on written request to the Office of Administration, Document Management Branch, Washington, DC 20555

~~Multiple comments were received pertaining to the following technical issues:~~ The major issues and their resolution are discussed below:

(1) Inclusion of cold shutdown requirements

(2) Equipment operating in a mild environment

(3) Qualification efforts already undertaken and based on NRC/IE Bulletin 79-01B/DOR Guidelines and NUREG-0588

(4) Requirement of maintaining a central qualification file.

(5) Consideration of time-dependent variation of relative humidity

(6) Aging - "qualified life"

(7) Margins - Conservatism applied during the derivation of environmental parameters

(8) Acceptance of analysis in combination with partial test data restricted to equipment purchased prior to May 1980.

(9) Resubmittal of justification of continued operation for operating plants

(10) Exclusion of seismic and dynamic requirements - sequence testing on a single prototype

Insert

Based on the comments received, the following substantive changes have been incorporated into the final rule:

Scope — 50.49(c)

(1) The requirement to qualify equipment needed to complete one path of achieving and maintaining a cold shutdown condition, has been deleted.

(2) A new Section (f)(5) has been added, covering the qualification of equipment located in mild environments

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(1) Seismic and Dynamic Qualification - 50.49(a)

Issue: Seismic and dynamic qualifications are an integral part of environmental qualification. It is, therefore, inappropriate to codify these requirements separately.

Response: Electric equipment at operating nuclear power plants was generally qualified for environmental and seismic stresses separately, i.e., by using separate prototypes for environmental and seismic qualification tests. The Commission has decided, after lengthy discussions, to pursue this issue via an advance notice of rulemaking. Any seismic qualification testing of equipment in operating plants that may be required by future rulemaking will not require retesting for environmental stresses solely because a single prototype was not used during the original qualification.

(2) Scope - Cold Shutdown Requirement - 50.49(c)

Issue: The rule introduces a new requirement to qualify "equipment needed to complete one path of achieving and maintaining a cold shutdown condition." A change of this magnitude, at this advanced stage of industry's qualification effort, most certainly introduces significant new costs and obligations with no demonstrated improvement in safety.

Response: The staff agrees and the requirement has been deleted.

8A

(3) Scope - Equipment in a Mild Environment - 50.49(c)

Issue: The rule makes no distinction between equipment located in a harsh or mild environment. The requirements for equipment in a mild environment are less stringent than for those in a harsh environment.

Response: The requirements for equipment located in a mild environment are not covered by the final rule. (See 50.49(a)) The operating plants are required to satisfy by Nov. 30, 1985 the provisions of IE Bulletin 79-01B and licensing generic letter 82-09 for equipment located in a mild environment. For near term operating licensees and for applicants for future nuclear power plants, the regulatory guidance will be provided in Regulatory Guide 1.89.

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(4) Scope - Previous Qualification Efforts - 50.49(c):

Issue: The rule does not recognize that operating plants have just completed qualification of equipment to the DOR Guidelines on NUREG 0588. Without such a recognition, industry efforts, manpower and billions of dollars will go down the drain.

Response: The statement of considerations has been expanded to alleviate this concern. Also, 50.49 (b) has been modified.

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(5) Humidity - 50.49 (e)(2)

Issue: The effects of time dependent variations of relative humidity during normal operation cannot be considered for all equipment. There are no detailed standards for how this type of testing should be performed.

Response: The staff agrees. The words "Time dependent variation of relative" have been deleted from 50.49 (e)(2).

(6) Aging - 50.49 (e)(5)

Issue: The requirement that ongoing qualifications be done using "prototype equipment naturally aged" is overly restrictive. Use of accelerated aging to define a qualified life is not technically feasible.

Response: Paragraph 50.49 (e)(5) has been modified to alleviate this concern.

(7) Margins - 50.49 (e)(8)

Issue: The margins applied in addition to known conservatisms lead to excessive stress which could lead to failures of equipment is unrealistic qualification tests. This paragraph is in conflict with Regulatory Guide 1.89.

Response: The staff agrees. The paragraph has been accordingly modified.

(8) Analysis and partial test data - 50.49 (f)(4)

Issue: If partial type test data is available which adequately supports the analytical assumptions and conclusions, their analysis should be allowed to extrapolate or interpolate these results for equipment, regardless of purchase date.

Response: The staff agrees. Reference to "purchase date" has been deleted.

4C

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(9) Requirement for a central file - 50.49 (j)

Issue: The requirement for a central file should be deleted, since it is not cost effective and has no safety benefit.

Response: The staff agrees. The requirement has for a central file has been deleted.

(10) Justification of continued operation for operating plants.

Issue: The requirement to submit justification for the continued operation of operating plants, since this information has been previously submitted to NRC.

Response: This requirement has been satisfactorily met and the paragraph 50.49 (j) of the proposed rule has been deleted in its entirety.

In addition, paragraph 50.49 (g) of the proposed rule has been deleted from the final rule since it is too prescriptive. It will be included in Regulatory Guide 1.89.

41)

EFFECTIVE DATE: This rule is being made effective upon publication in the Federal Register. The Commission finds that the schedular sections of the rule may take immediate effect because they "relieve a restriction" under subsection (d)(1) of Section 553 of the Administrative Procedure Act. This is so because all operating reactor licensees are currently under a June 30, 1982, deadline to complete environmental qualification of safety-related electric equipment. The rule's implementation schedule, as explained above, supplants this date and thus gives licensees additional time to complete environmental qualification of this equipment. In addition, the Commission finds that there is good cause—pursuant to subsection (d)(3) of Section 553—to make the rule's requirements effective upon publication. The first licensee actions under the rule are not required until 90 days after the effective date of the rule. This 90 day period is intended to include the statutory 30 days and allow 60 additional days to make the submittal required by subsection (h) of the rule. The overall effect of making the rule effective on publication is to relieve licensees of the June 30 deadline and to provide a sufficient period after the effective date of the rule for licensees to achieve compliance with the near-term requirements of the rule.

~~...ged", has been deleted.~~

~~(7) The section on margin has been clarified. [See Section (e)(8)]~~

~~(8) Reference to a date (May 23, 1980) for acceptance of analysis in combination with partial test data has been deleted.~~

~~(9) The requirement to submit justification for the continued operation of operating plants has been deleted, since this has already been satisfactorily accomplished.~~

EFECTIVE DATE: INSERT

Paperwork Reduction Act

The proposed final rule contains recordkeeping requirements that are subject to review by the Office of Management and Budget (OMB). As required by P.L. 96-511, this proposed rule ~~will be~~ was submitted to OMB for clearance of the recordkeeping requirements.

Regulatory Flexibility Statement

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule, if promulgated, will not have a significant economic impact on a substantial number of small entities. This proposed final rule affects the method of qualification of electric equipment by utilities. Utilities do not fall within the definition of a small business found in Section 3 of the Small Business Act, 15 U.S.C. 632. In addition, utilities are required by Commission's Memorandum and Order CLI-80-21, dated May 23, 1980, to meet the requirements contained in the DOR "Guidelines for Evaluating Environmental Qualification of Class 1E Electric Equipment in Operating Reactors," (November 1979) and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which form the basis of this proposed rule. Consequently, this rule codifies existing requirements and imposes no new costs or obligations on utilities.

Pursuant to the Atomic Energy Act [illegible] amended, the Energy Reorganization Act of 1974, as amended, a[illegible]553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendment to 10 CFR Part 50 is contemplated.

10 CFR Part 50

1. The authority citation for 10 CFR Part 50 reads as follows: AUTHORITY: Secs. 103, 104, 161, 182, 183, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2239); secs. 201, 202, 206, 88 Stat. 1243, 1244, 1246 (42 U.S.C., 5841, 5842, 5846), unless otherwise noted. Section 50.78 also issued under

Sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under Sec. 184, 68 Stat. 954, as amended; (42 U.S.C 2234). Sections 50.100-50.102 issued uder Sec. 186, 68 Stat. 955; (42 U.S.C. 2236). For Purposes of Sec. 223, 68 Stat. 958, as amended; (42 U.S.C. 2273), § 50.54 (i) issued under Sec. 161i, 68 Stat. 949; (42 U.S.C 2201(i)), §§ 50.70, 50.71 and 50.78 issued under Sec. 161o, 68 Stat. 950, as amended; (42 U.S.C. 2201(o)) and the Laws referred to in Appendices.

2. A new § 50.49 is added to reaa as follows:

§ 50.49 Environmental qualification of electric equipment for nuclear power plants.

(a) Requirements for seismic and dynamic qualification of electric equipment are not included in this section. Also not included are the requirements for electric equipment located in a mild environment.

(b) Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment as defined in paragraph (c) of this section.3

(c) Electric equipment and systems covered by this section include electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or that are otherwise essential in preventing significant release of radioactive material to the environment. Included is equipment (1) that performs the above functions automatically, (2) that is used by the operator to perform these functions manually, and (3) whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions. ~~Also included is equipment needed to complete one path of achieving and maintaining a cold shutdown condition.~~

A mild environment is an environment that would at no time be significantly severe than the environment that would occur during normal plant operation or during anticipated operational occurrences. ~~shall be~~

3 Requalification of electric equipment in accordance with this rule will not be required for equipment qualified in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 or NUREG-0588, Revision 0 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", provided the qualification of electric equipment has commenced prior to 90 days after the effective date of the rule.

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(d) The applicant or licensee shall prepare a list of all electric equipment covered by this section. ~~and-maintain-it-in-an-auditable-form.~~ ~~This-list-of-equipment-must;-as-a-minimum;-include~~:

~~In addition, the following information for electric equipment except equipment located in a mild environment, i.e., an environment that would at no time be more severe than the environment that would occur during normal plant operation or during anticipated operational occurrences,~~ shall be included in a qualification file:

stet

(1) The performance specifications ~~and-structural-integrity-requirements~~ under conditions existing during normal and abnormal operation and during design basis events and afterwards. ~~and-the-lengths-of-the-periods during-which-the-integrity-must-be-maintained.~~

(2) ~~The-range-of~~ Voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence, ~~and-the-predicted-variations-of-these-environmental-conditions-with-time~~ at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

(e) The electrical equipment qualification program must include the following:

(1) Temperature and Pressure. The time-dependent temperature and pressure at the location of the equipment must be established for the most ~~limiting~~ severe of the applicable ~~postulated-accidents~~ design basis events and must be used as the basis for the environmental qualification of electric equipment.

(2) Humidity. ~~Time-dependent-variations-of-relative~~ Humidity during normal operation and design basis events must be considered.

(3) Chemical Effects. The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) Radiation. The radiation environment must be based on the type of radiation, the total dose ~~and-dose-rate-of-the-radiation-environment~~ expected during normal operation over the installed life of the equipment ~~plus~~ and the radiation environment, including dose-rate effects, associated with the most severe design basis event during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines.

(5) Aging. Equipment qualified by test must ~~practicable~~ be preconditioned by natural or artificial (accelerated) aging to its installed end-of-life condition. ~~Electromechanical-equipment-must-be operated-to-the-mechanical-wear-and-electrical-degradation-expected-during its-installed-life~~. Where preconditioning to an installed end-of-life condition ~~a qualified life equal to the installed life~~ is not practicable and technically meaningful ~~possible~~, the equipment may be preconditioned to a shorter designated ~~qualified~~ life. The equipment must be replaced or refurbished at the end of this designated ~~its qualified~~ life unless ongoing qualification demonstrates ~~or prototype-equipment-naturally-aged-in-plant-service-show;-by-artificial aging-and-type-testing~~ that the item has additional ~~qualified~~ life.

(6) Submergence (if subject to being submerged).

(7) Synergistic Effects. ~~The-preconditioning-and-testing-of-equipment-must-consider-known~~ Synergistic effects must be considered when these effects are ~~known~~ believed to have a significant effect on equipment performance.

(8) Margins. Margins must be applied to account for production variations and inaccuracies in test instruments. These margins are in addition to ~~margins-applied-during-the-derivation-of-the-environmental conditions:~~ any conservatisms applied during the derivation of environmental conditions unless these conservatisms can be quantified and shown to contain appropriate margins.

(f) Each item of electric equipment must be qualified by one of the following methods:

(1) Testing an identical item of equipment (manufactured to the same design) under identical conditions, or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

~~(4)--Analysis-in-lieu-of-testing-in-the-following-cases:~~

~~(i)--if-type-testing-is-precluded-by-the-physical-size-of-the-equipment-or-by-the-state-of-the-art.~~

(4) By Analysis in combination with partial ~~type~~ std test data which supports the analytical assumptions and conclusions. ~~;-if-the-equipment purchase-order-was-executed-prior-to-May-23,-1980.~~

(5) Design or purchase specifications, if the equipment is in a mild environment. The specification must contain a description of the functional requirements and the specific environments during normal and abnormal conditions and must be supported by a certificate of compliance based on test data and analysis.

OR

Or For equipment, purchased prior to the effective date of this rule, which is located in a mild environment, the qualification can be demonstrated by (a) a periodic maintenance, inspection, and/or replacement program, (b) a periodic testing programs, and (c) an equipment surveillance program.

(g) If an item of electric equipment is to be qualified by test -

(1) The acceptance criteria must be established prior to testing.

(2) The tests must be designed and conducted to demonstrate that the equipment can perform its required function as specified in accordance with paragraph (d)(1) of this section for all conditions as specified in accordance with paragraphs (d)(2, and (3) of this section. The test profile (e.g., pressure, temperature, radiation vs. time) must include margins as set forth in paragraph (e)(8) of this section.

(3) The test profile must be either (i) a single profile that envelops the environmental conditions resulting from any design basis event during any mode of plant operation where the equipment must perform its safety functions (e.g., a profile that envelops the conditions

produced ~~by~~ the ~~postulated spectrum of~~ main ~~steamline break (MSLB) and~~ ~~loss-of-coo~~lant accidents ~~(LOCA))~~ or (i~~i) separate profiles for~~ eac~~h type~~ ~~of event (~~e.g., ~~separate~~ ~~profiles~~ for ~~the MSLB accidents and for~~ LOC~~As).~~

~~(4) The same piece of equipment must be used throughout the complete~~ ~~test sequence under any given profile.~~

~~(h)~~ (g) Each holder of an operating license issued prior to (insert the effective date of this amendment) must, by (insert a date 90 days after the effective date of this amendment), identify the electric equipment already qualified to the provisions of this rule and submit a schedule for the ~~testing~~ qualification or replacement of the remaining electric equipment. This schedule must establish a goal of final environmental qualification by the end of the second refueling outage after March 31, 1982. The Director of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if such requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985 for completion of environmental qualification.

~~(i)~~ (h) Each licensee shall notify the Commission of any significant equipment qualification problem that may require extension of the completion date within ~~3~~60 days of its discovery.

~~(j)--For-the-continued-operation-of-a-nuclear-plant;-each-holder-of-an-operating-license-issued-prior-to-the-effective-date-of-this-rule-shall-perform-an-analysis-to-ensure-that-the-plant-can-be-safely-operated-pending-completion-of-the-environmental-qualification.--The-detailed-analysis-for-each-equipment-type-with-appropriate-justification-must-be-submitted-to~~

~~Director of Nuclear Reactor Regulatory by (insert the effective date of the rule) and must include, where appropriate, consideration of:~~

~~(1) Accomplishing the safety function by some designated alternative equipment that has been adequately qualified and satisfies the single failure criterion if the principal equipment has not been demonstrated to be fully qualified.~~

~~(2) The validity of partial test data in support of the original qualification.~~

~~(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.~~

~~(4) Completion of the safety function prior to exposure to the ensuing accident environment and the subsequent failure of the equipment does not degrade any safety function or mislead the operator.~~

~~(5) No significant degradation of any safety function or misleading of the operator as a result of failure of equipment under the accident environment.~~

~~(k)~~ (j) The applicant for an operating license that is granted on or after the effective date of this amendment, but prior to November 30, 1985, must perform an analysis to ensure that the plant can be safely operated pending completion of ~~the~~ environmental qualification. ~~in accordance with paragraph (j) of this section except that this analysis~~ This analysis must be submitted to the Director of Nuclear Reactor Regulation for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

(1) Accomplishing the safety function by some designated alternate equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the ensuing accident environment and the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading of the operator as a result of failure of equipment under the accident environment.

~~(i)~~ (j) A record of the qualification including documentation in paragraph (d) of this section must be maintained in ~~a central file~~ an auditable form to permit verification that each item of electric equipment covered by this section (1) is qualified for its application and

(2) meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

Dated at ______________ this __________ day of _________, 1982.

For the Nuclear Regulatory Commission.

____

Samuel J. Chilk
Secretary of the Commission

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PRESENTATION BY

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C.

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MAY 7, 1982

APPENDIX XXXI
PRESENTATION BY SATISH K. AGGARWAL

(1) SEISMIC AND DYNAMIC QUALIFICATION - 50.49(A)

ISSUE: SEISMIC AND DYNAMIC QUALIFICATIONS ARE AN INTEGRAL PART OF ENVIRONMENTAL QUALIFICATION. IT IS, THEREFORE, INAPPROPRIATE TO CODIFY THESE REQUIREMENTS SEPARATELY.

RESPONSE: ELECTRIC EQUIPMENT AT OPERATING NUCLEAR POWER PLANTS WAS GENERALLY QUALIFIED FOR ENVIRONMENTAL AND SEISMIC STRESSES SEPARATELY, I.E., BY USING SEPARATE PROTOTYPES FOR ENVIRONMENTAL AND SEISMIC QUALIFICATION TESTS. THE COMMISSION HAS DECIDED, AFTER LENGTHY DISCUSSIONS, TO PURSUE THIS ISSUE VIA AN ADVANCE NOTICE OF RULEMAKING. ANY SEISMIC QUALIFICATION TESTING OF EQUIPMENT IN OPERATING PLANTS THAT MAY BE REQUIRED BY FUTURE RULEMAKING WILL NOT REQUIRE RETESTING FOR ENVIRONMENTAL STRESSES SOLELY BECAUSE A SINGLE PROTOTYPE WAS NOT USED DURING THE ORIGINAL QUALIFICATION.

(2) SCOPE - COLD SHUTDOWN REQUIREMENT - 50.49(c)

ISSUE: THE RULE INTRODUCES A NEW REQUIREMENT TO QUALIFY "EQUIPMENT NEEDED TO COMPLETE ONE PATH OF ACHIEVING AND MAINTAINING A COLD SHUTDOWN CONDITION." A CHANGE OF THIS MAGNITUDE, AT THIS ADVANCED STAGE OF INDUSTRY'S QUALIFICATION EFFORT, MOST CERTAINLY INTRODUCES SIGNIFICANT NEW COSTS AND OBLIGATIONS WITH NO DEMONSTRATED IMPROVEMENT IN SAFETY.

RESPONSE: THE STAFF AGREES AND THE REQUIREMENT HAS BEEN DELETED.

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(3) SCOPE - EQUIPMENT IN A MILD ENVIRONMENT - 50.49(c)

ISSUE: THE RULE MAKES NO DISTINCTION BETWEEN EQUIPMENT LOCATED IN A HARSH OR MILD ENVIRONMENT. THE REQUIREMENTS FOR EQUIPMENT IN A MILD ENVIRONMENT ARE LESS STRINGENT THAN FOR THOSE IN A HARSH ENVIRONMENT.

RESPONSE: THE REQUIREMENTS FOR EQUIPMENT LOCATED IN A MILD ENVIRONMENT ARE NOT COVERED BY THE FINAL RULE. (SEE 50.49(A)) THE OPERATING PLANTS ARE REQUIRED TO SATISFY THE PROVISIONS OF IE Bulletin 79-01B and LICENSING GENERIC LETTER 82-09 FOR EQUIPMENT LOCATED IN A MILD ENVIRONMENT. FOR NEAR TERM OPERATING LICENSEES AND FOR APPLICANTS FOR FUTURE NUCLEAR POWER PLANTS, THE REGULATORY GUIDANCE WILL BE PROVIDED IN REGULATORY GUIDE 1.89.

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(4) SCOPE - PREVIOUS QUALIFICATION EFFORTS - 50.49(c):

ISSUE: THE RULE DOES NOT RECOGNIZE THAT OPERATING PLANTS HAVE JUST COMPLETED QUALIFICATION OF EQUIPMENT TO THE DOR GUIDELINES ON NUREG-0588. WITHOUT SUCH A RECOGNITION, INDUSTRY EFFORTS, MANPOWER AND BILLIONS OF DOLLARS WILL GO DOWN THE DRAIN.

RESPONSE: THE STATEMENT OF CONSIDERATIONS HAS BEEN EXPANDED TO ALLEVIATE THIS CONCERN. ALSO, 50.49(B) HAS BEEN MODIFIED.

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(5) HUMIDITY - 50.49(E)(2)

ISSUE: THE EFFECTS OF TIME DEPENDENT VARIATIONS OF RELATIVE HUMIDITY DURING NORMAL OPERATION CANNOT BE CONSIDERED FOR ALL EQUIPMENT. THERE ARE NO DETAILED STANDARDS FOR HOW THIS TYPE OF TESTING SHOULD BE PERFORMED.

RESPONSE: THE STAFF AGREES. THE WORDS "TIME DEPENDENT VARIATION OF RELATIVE" HAVE BEEN DELETED FROM 50.49(E)(2).

(6) AGING - 50.49(E)(5)

ISSUE: THE REQUIREMENT THAT ONGOING QUALIFICATIONS BE DONE USING "PROTOTYPE EQUIPMENT NATURALLY AGED" IS OVERLY RESTRICTIVE. USE OF ACCELERATED AGING TO DEFINE A QUALIFIED LIFE IS NOT TECHNICALLY FEASIBLE.

RESPONSE: PARAGRAPH 50.49(E)(5) HAS BEEN MODIFIED TO ALLEVIATE THIS CONCERN.

(8) ANALYSIS AND PARTIAL TEST DATA - 50.49(F)(4)

ISSUE: IF PARTIAL TYPE TEST DATA IS AVAILABLE WHICH ADEQUATELY SUPPORTS THE ANALYTICAL ASSUMPTIONS AND CONCLUSIONS, THEIR ANALYSIS SHOULD BE ALLOWED TO EXTRAPOLATE OR INTERPOLATE THESE RESULTS FOR EQUIPMENT, REGARDLESS OF PURCHASE DATE.

RESPONSE: THE STAFF AGREES. REFERENCE TO "PURCHASE DATE" HAS BEEN DELETED.

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(9) REQUIREMENT FOR A CENTRAL FILE - 50.49(j)

ISSUE: THE REQUIREMENT FOR A CENTRAL FILE SHOULD BE DELETED, SINCE IT IS NOT COST EFFECTIVE AND HAS NO SAFETY BENEFIT.

RESPONSE: THE STAFF AGREES. THE REQUIREMENT FOR A CENTRAL FILE HAS BEEN DELETED.

(10) JUSTIFICATION OF CONTINUED OPERATION FOR OPERATING PLANTS.

ISSUE: THE REQUIREMENT TO SUBMIT JUSTIFICATION FOR THE CONTINUED OPERATION OF OPERATING PLANTS, SINCE THIS INFORMATION HAS BEEN PREVIOUSLY SUBMITTED TO NRC.

RESPONSE: THIS REQUIREMENT HAS BEEN SATISFACTORILY MET AND THE PARAGRAPH 50.49(j) OF THE PROPOSED RULE HAS BEEN DELETED IN ITS ENTIRETY.

IN ADDITION, PARAGRAPH 50.49(g) OF THE PROPOSED RULE HAS BEEN DELETED FROM THE FINAL RULE SINCE IT IS TOO PRESCRIPTIVE. IT WILL BE INCLUDED IN REGULATORY GUIDE 1.89.

(A) REQUIREMENTS FOR SEISMIC AND DYNAMIC QUALIFICATION OF ELECTRIC EQUIPMENT ARE NOT INCLUDED IN THIS SECTION. ALSO NOT INCLUDED ARE THE REQUIREMENTS FOR ELECTRIC EQUIPMENT LOCATED IN A MILD ENVIRONMENT.

A MILD ENVIRONMENT IS AN ENVIRONMENT THAT WOULD AT NO TIME BE SIGNIFICANTLY MORE SEVERE THAN THE ENVIRONMENT THAT WOULD OCCUR DURING NORMAL PLANT OPERATION OR DURING ANTICIPATED OPERATIONAL OCCURRENCES.

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(B) EACH HOLDER OF OR EACH APPLICANT FOR A LICENSE TO OPERATE A NUCLEAR POWER PLANT SHALL ESTABLISH A PROGRAM FOR QUALIFYING THE ELECTRIC EQUIPMENT AS DEFINED IN PARAGRAPH (c) OF THIS SECTION.[3]

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[3]REQUALIFICATION OF ELECTRIC EQUIPMENT IN ACCORDANCE WITH THIS RULE WILL NOT BE REQUIRED FOR EQUIPMENT QUALIFIED IN ACCORDANCE WITH "GUIDELINES FOR EVALUATING ENVIRONMENTAL QUALIFICATION OF CLASS 1E ELECTRICAL EQUIPMENT IN OPERATING REACTORS," NOVEMBER 1979 OR NUREG-0588, REVISION 0, "INTERIM STAFF POSITION ON ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT," PROVIDED THE QUALIFICATION OF ELECTRIC EQUIPMENT HAS COMMENCED PRIOR TO 90 DAYS AFTER THE EFFECTIVE DATE OF THE RULE.

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(E)(2) <u>HUMIDITY</u>. HUMIDITY DURING NORMAL OPERATION AND DESIGN BASIS EVENTS MUST BE CONSIDERED.

(E)(5) <u>AGING</u>. EQUIPMENT QUALIFIED BY TEST MUST BE PRECONDITIONED BY NATURAL OR ARTIFICIAL (ACCELERATED) AGING TO ITS INSTALLED END-OF-LIFE CONDITION. WHERE PRECONDITIONING TO AN INSTALLED END-OF-LIFE CONDITION IS NOT PRACTICABLE AND TECHNICALLY MEANINGFUL, THE EQUIPMENT MAY BE PRECONDITIONED TO A SHORTER DESIGNATED LIFE. THE EQUIPMENT MUST BE REPLACED OR REFURBISHED AT THE END OF THIS DESIGNATED LIFE UNLESS ONGOING QUALIFICATION DEMONSTRATES THAT THE ITEM HAS ADDITIONAL LIFE.

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(E)(8) MARGINS. MARGINS MUST BE APPLIED TO ACCOUNT FOR PRODUCTION VARIATIONS AND INACCURACIES IN TEST INSTRUMENTS. THESE MARGINS ARE IN ADDITION TO ANY CONSERVATISMS APPLIED DURING THE DERIVATION OF ENVIRONMENTAL CONDITIONS UNLESS THESE CONSERVATISMS CAN BE QUANTIFIED AND SHOWN TO CONTAIN APPROPRIATE MARGINS.

(F)(4) ANALYSIS IN COMBINATION WITH PARTIAL TYPE TEST DATA WHICH SUPPORTS THE ANALYTICAL ASSUMPTIONS AND CONCLUSIONS.

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APPENDIX XXXII
PRESENTATION BY R. MATTSON

PURPOSE OF BRIEFING

- COMMISSION REVIEW OF SECY-81-603B "PROPOSED RULEMAKING ON ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT" RESULTED IN COMMISSION REQUEST FOR INFORMATION PAPER:
    - SUMMARIZED PRESENT UNDERSTANDING OF WHICH ORs HAVE CAPABILITY TO GO TO COLD SHUTDOWN ON SAFETY-GRADE EQUIPMENT
    - SUMMARIZED EQUIPMENT THAT MAY REQUIRE UPGRADING IF BACKFIT REQUIRED

- RESPONSE PROVIDED ON 03/12/82 STATED:
    - ONLY KNOW CAPABILITY OF ORs LICENSED AFTER 01/01/79
    - SUMMARY OF EQUIPMENT PROVIDED

- BASED UPON 03/12/82 RESPONSE, COMMISSION REQUESTED BRIEFING FOR MORE DETAILED DISCUSSION OF ISSUE

PLANNED ACTIONS RELATED TO COLD SHUTDOWN REQUIREMENTS

- CONTINUE IMPLEMENTATION OF BTP RSB 5-1 FOR NEW OLs AS APPROVED BY RRRC

- LEAVE RG 1.139 REV 0 IN PLACE

- RECOMMEND DELETION OF COLD SHUTDOWN REQUIREMENTS FROM PROPOSED EQ RULE

- CONTINUE RESOLUTION OF USI A-45

  - SURVEY OF LIMITED NUMBER OF ORs TO DETERMINE COLD SHUTDOWN COOLING CAPABILITY AND RELIABILITY (SAME APPROACH AS ORIGINALLY APPROVED BY RRRC IN 01/78)

  - DEVELOP QUANTITATIVE AND QUALITATIVE ACCEPTANCE CRITERIA FOR SHUTDOWN COOLING REQUIREMENTS FOR EXISTING AND FUTURE PLANTS

  - ASSESS ADEQUACY OF EXISTING PLANTS' SHUTDOWN COOLING CAPABILITY AND RELIABILITY USING ACCEPTANCE CRITERIA

  - IMPLEMENT BACKFIT OF NEW LICENSING REQUIREMENTS (IF REQUIRED)

TERMINOLOGY

- OPERATIONAL MODES

MODE+		W STS	CE STS	B&W STS	GE STS	FT CALHOUN	OCONEE
1 - POWER OPERATION	TEMP	≥ 350	≥ 300	≥ 305	ANY	ANY	ANY
	POWER	> 5%	> 5%	> 5%	MODE-RUN	> 2%	> 2%
2 - STARTUP	TEMP	≥ 350	≥ 300	≥ 305	ANY	NA	NA
	POWER	≤ 5%	≤ 5%	≤ 5%	MODE-S/U,STBY		
3 - HOT STANDBY	TEMP	≥ 350	≥ 300	≥ 305	NA	>515	>525
	POWER	0%	0%	0%		< 2%	< 2%
	REACTOR	S/D	S/D	S/D		CRIT	CRIT
4 - HOT SHUTDOWN	TEMP	< 350 > 200	< 300 > 200	< 305 > 200	>200	>515	>525
	REACTOR	S/D	S/D	S/D	MODE-S/D	S/D	S/D
5 - COLD SHUTDOWN	TEMP	≤ 200	≤ 200	≤ 200	≤200	<210	<200
	REACTOR	S/D	S/D	S/D	MODE-S/D	S/D	S/D
6 - REFUELING	TEMP	≤ 140	≤ 140	≤ 140	≤140	<210	<140
	REACTOR	S/D	S/D	S/D	MODE-S/D,REF	S/D	S/D

+ TABLE SHOWS SIMPLIFIED DEFINITION OF MODES (OTHER PARAMETERS MAY ALSO DEFINE MODES)

TERMS: STS - STANDARD TECH. SPECS (ONLY 26 OF 74 PLANTS HAVE STS)

TEMP - TEMPERATURE IN °F

POWER - % OF RATED THERMAL POWER

REACTOR

- CRIT - CRITICAL
- S/D - SHUTDOWN

MODE - POSITION OF REACTOR MODE SWITCH (RUN, STARTUP, HOT STANDBY, SHUTDOWN, REFUEL)

- REGULATIONS (EXAMPLES)
  - SAFE SHUTDOWN - 50/APP R, 100/APP A - HOT STANDBY - 50/APP R, - HOT SHUTDOWN - 50/APP R + GDC 19; - COLD SHUTDOWN - 50/APP R + GDC 19

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SAFETY GRADE

- SAFETY GRADE = SAFETY RELATED
- SAFETY RELATED: AS DEFINED IN 10 CFR 100, APPENDIX A

  "THOSE STRUCTURES, SYSTEMS, OR COMPONENTS DESIGNED TO REMAIN FUNCTIONAL FOR THE SSE (ALSO TERMED 'SAFETY FEATURES') NECESSARY TO ASSURE REQUIRED SAFETY FUNCTIONS, I.E.,:

  (1) THE INTEGRITY OF THE REACTOR COOLANT PRESSURE BOUNDARY;

  (2) THE CAPABILITY TO SHUT DOWN THE REACTOR AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION; OR

  (3) THE CAPABILITY TO PREVENT OR MITIGATE THE CONSEQUENCES OF ACCIDENTS WHICH COULD RESULT IN POTENTIAL OFF-SITE EXPOSURES COMPARABLE TO THE GUIDELINE EXPOSURES OF THIS PART."

- GENERAL DESIGN CRITERIA FOR SAFETY-GRADE EQUIPMENT/SYSTEMS
  - GDC 1 "QUALITY STANDARDS AND RECORDS"
    - Quality Group A, B or C. . . . . SRP 3.2.2. . . . .10 CFR 50.55a & R.G. 1.26
      SRP 7.1/7.7. . . .10 CFR 50.55a(h) IEEE-279
    - Quality Assurance Program. . . . SRP 17.1/2 . . . .10 CFR 50, Appendix B
  - GDC 2 "DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA"
    - Seismic Category I . . . . SRP 3.2.1 . . . .R.G. 1.29
      SRP 3.10 . . . . .R.G. 1.100
    - Flood Protection . . . . SRP 2.4.10 . . . .R.G. 1.59/1.102
    - Wind Protection . . . . SRP 3.3.1
    - Tornado Protection . . . . SRP 3.3.2
  - GDC 3 "FIRE PROTECTION
    - Fire Protection. . . . . SRP 9.5.1 . . . .10 CFR 50, Appendix R
  - GDC 4 "ENVIRONMENTAL AND MISSILE DESIGN BASES"
    - Environmental Qualification. . . SRP 3.11 . . . . .DOR Guidelines NUREG-0588 Proposed Rule - Elect. ANPR-Mech.
    - Missile Protection . . . . SRP 3.5. . . . . .R.G. 1.27, 1.76, 1.91, 1.115 & 1.117
    - Effects of Pipe Breaks . . . . . SRP 3.6.1/2. . . .R.G. 1.46

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  - GDC 5 "SHARING OF STRUCTURES, SYSTEMS & COMPONENTS
    - Sharing. . . . . SRP-Several. . . .R.G. 1.6, 1.75, 1.81

CLASSIFICATION OF SHUTDOWN SCENARIOS

- NORMAL PLANT SHUTDOWN
  - ALL PLANT EQUIPMENT AND SYSTEMS OPERABLE
  - NO HARSH ENVIRONMENT
- ANTICIPATED OPERATIONAL OCCURRENCE
  - MAJORITY OF ALL PLANT EQUIPMENT AND SYSTEMS OPERABLE
  - CERTAIN EQUIPMENT AND SYSTEMS MAY NOT BE OPERABLE
  - ENVIRONMENT COULD BE OFF-NORMAL, BUT NOT HARSH
- DESIGN BASIS ACCIDENT
  - MAJORITY OF PLANT EQUIPMENT AND SYSTEMS MAY BE OPERABLE
  - RELIANCE ON ECCS EQUIPMENT AND SYSTEMS TO MAINTAIN COOLABLE GEOMETRY FOR EXTENDED PERIOD OF TIME
- ACCIDENT BEYOND THE DESIGN BASIS
  - MAY BE MULTIPLE FAILURES IN SAFETY AND NON-SAFETY EQUIPMENT
  - HYDROGEN, CORE DEBRIS, AND RADIOACTIVITY MAY BE PRESENT IN SYSTEMS THAT COMMUNICATE WITH RCS
  - DECAY HEAT REMOVAL STILL REQUIRED
- CORE MELT
  - DECAY HEAT REMOVAL FROM CORE AND RCS NOT CONCERN FOR CONTROL OF RISK
  - CONTAINMENT INTEGRITY, INCLUDING CONTAINMENT HEAT REMOVAL IS PRINCIPAL CONCERN
  - DESIGN ENVELOPE OF ENVIRONMENTAL CONDITIONS SIGNIFICANTLY EXCEEDED FOR MOST PARAMETERS (PRESSURE, TEMPERATURE, RADIATION)

SYSTEM STATUS - NORMAL SHUTDOWN (STS DEFINITIONS)

- HOT STANDBY
  1. REACTOR IS SUBCRITICAL
  2. TEMPERATURE MAY BE REDUCED SOMEWHAT BELOW NORMAL OPERATING TEMPERATURE
  3. PRESSURE USUALLY AT NORMAL OPERATING PRESSURE
  4. DECAY HEAT REMOVAL

     PWR - S/G TO MAIN CONDENSER VIA TURBINE BYPASS

     BWR - TURBINE BYPASS TO MAIN CONDENSER

- HOT SHUTDOWN
  1. REACTOR IS SUBCRITICAL
  2. TEMPERATURE SIGNIFICANTLY BELOW NORMAL OPERATING TEMPERATURE
  3. PRESSURE REDUCED CONSISTENT WITH PRESSURE/TEMPERATURE LIMITS OF TS
  4. DECAY HEAT REMOVAL

     PWR - S/G TO MAIN CONDENSER VIA TURBINE BYPASS OR BY RHR, DEPENDING ON PRESSURE

     BWR - TURBINE BYPASS TO MAIN CONDENSER, RCIC OR RHR DEPENDING ON PRESSURE

- COLD SHUTDOWN
  1. REACTOR IS SUBCRITICAL
  2. TEMPERATURE BELOW BOILING POINT AT ATMOSPHERIC PRESSURE
  3. PRESSURE REDUCED CONSISTENT WITH PRESSURE/TEMPERATURE LIMITS OF TS AND BELOW PRESSURE REQUIRED FOR RHR
  4. DECAY HEAT REMOVAL - RHR SYSTEM

REASONS FOR WANTING TO GO TO COLD SHUTDOWN

- WASH-1400 CONSIDERATIONS
  - SYSTEM AND EQUIPMENT FAILURES RESULTING IN INABILITY TO REMOVE DECAY HEAT RESULTED IN HIGHER PROBABILITY OF CORE MELT THAN LARGE LOCA
  - DID NOT ADDRESS FAILURE OF RHR, HOT SHUTDOWN ASSUMED GOOD ENOUGH
  - BECAUSE OF IMPORTANCE OF DHR CAPABILITY, SIGNIFICANT SAFETY BENEFIT CAN BE GAINED BY UPGRADING EQUIPMENT NEEDED TO:
    - STAY IN HOT STANDBY OR HOT SHUTDOWN
    - COOLDOWN AND DEPRESSURIZE RCS TO COLD SHUTDOWN

- HAVE BEEN AND WILL BE SITUATIONS WHERE COLD SHUTDOWN IS REQUIRED FOR INSPECTION AND REPAIRS OR DUE TO LIMITED SUPPLY OF HIGH QUALITY MAKEUP WATER
  - STEAM GENERATOR TUBE RUPTURE
  - REACTOR COOLANT PUMP SEAL FAILURES
  - FAILURE OF PRIMARY OR SECONDARY RELIEF VALVES TO CLOSE
  - EXTENDED LOSS OF OFFSITE POWER
  - LARGE SEISMIC EVENT

- PRINCIPAL OBJECTIVE OF RSB 5-1 TO ASSURE FOR ANY UNPLANNED SHUTDOWN EQUIPMENT AND PROCEDURES AVAILABLE TO GO TO COLD SHUTDOWN
  - INDEFINITE DELAY OF COOLDOWN COULD BE SAFETY CONCERN IF ADDITIONAL EQUIPMENT OR OPERATIONAL FAILURES OCCUR
  - CLOSED-LOOP, LONG-TERM COOLING OF CORE AT LOW TEMPERATURES AND PRESSURES ASSOCIATED WITH COLD SHUTDOWN INHERENTLY PREFERRED AND PROVIDES ACCESS TO ADDITIONAL SYSTEMS TO COOL CORE AS LAST RESORT

PLANT SHUTDOWN USING ONLY SAFETY-GRADE EQUIPMENT

BTP RSB 5-1 POSITION (REG GUIDE 1.139 REV 0)

- DESIGN SHALL BE SUCH THAT REACTOR CAN BE TAKEN FROM NORMAL OPERATING CONDITIONS TO COLD SHUTDOWN USING ONLY SAFETY-GRADE EQUIPMENT. THESE SYSTEMS SHALL SATISFY GDC 1 THROUGH 5:

  GDC-1 QUALITY STANDARDS

  GDC-2 PROTECTION FROM NATURAL PHENOMENA

  GDC-3 FIRE PROTECTION

  GDC-4 EQUIPMENT AND MISSILE DESIGN BASIS

  GDC-5 SHARING OF STRUCTURES, SYSTEMS AND COMPONENTS

- SUITABLE REDUNDANCY TO ASSURE CAPABILITY TO FUNCTION ASSUMING ONLY ONSITE OR OFFSITE POWER AND ASSUMING A SINGLE FAILURE (GDC 34)

- SYSTEMS SHALL BE CAPABLE OF BEING OPERATED FROM CONTROL ROOM (GDC 19)

- SYSTEMS SHALL BE CAPABLE OF BRINGING THE PLANT TO COLD SHUTDOWN IN A REASONABLE PERIOD OF TIME (36HOURS)

- PREOPERATIONAL AND STARTUP TEST SHALL CONFIRM
  - ADEQUATE MIXING OF BORATED WATER
  - COOLDOWN UNDER NATURAL CIRCULATION CONDITIONS

- OPERATING PROCEDURES SHALL COVER BRINGING PLANT TO COLD SHUTDOWN

- AFW SUPPLY MUST BE SEISMIC CATEGORY I AND HAVE SUFFICIENT INVENTORY TO REMAIN IN HOT STANDBY FOR 4 HOURS AND THEN COOLDOWN TO COLD SHUTDOWN
  - WITH ONSITE OR OFFSITE POWER ONLY
  - ASSUMING THE MOST LIMITING SINGLE FAILURE

IMPLEMENTATION OF BTP RSB 5-1

RRRC APPROVED IMPLEMENTATION (01/31/78)	ACTUAL STAFF IMPLEMENTATION
FULL COMPLIANCE: CPs/PDAs DOCKETED AFTER 01/01/78	NO SUCH APPLICATIONS UNDER REVIEW
PARTIAL COMPLIANCE: CPs/PDAs DOCKETED PRIOR TO 01/01/78 AND OL ISSUED AFTER 01/01/79	POST-TMI-2 OLs-NA 2, SEQUOYAH 1/2, FARLEY 2, MCGUIRE 1, DIABLO CANYON (SUSPENDED), SAN ONOFRE 2, LASALLE FLEXIBILITY ALLOWED - DBA EQ NOT REQUIRED-SERVICE ENVIRONMENT ASSUMED - REASONABLE OPERATOR ACTIONS OUTSIDE OF CONTROL ROOM TO CORRECT SINGLE FAILURES - REASONABLE OPERATOR ACTIONS OUTSIDE OF CONTROL ROOM FOR LIMITED FUNCTIONS (NO SINGLE FAILURE) - SYSTEM REDUNDANCY NOT REQUIRED IF DIVERSE METHOD OF PERFORMING SAFETY FUNCTION AVAILABLE - NO SEISMIC REQUIREMENT FOR BACKUP TO SINGLE FAILURE
FURTHER STAFF CONSIDERATION: OLs ISSUED BEFORE 01/01/79 (BACKFIT BASED UPON JOINT IE & DOR REVIEW OF OPERATING PLANTS)	JOINT IE & DOR REVIEW BEGAN ON 14 PLANTS - ORIGINAL 11 SEP PLANTS & 1/VENDOR (BRUNSWICK, INDIAN POINT, ST. LUCIE, TMI-2) - SITE VISITS CONDUCTED-NO SERs WRITTEN - REVIEW EFFORT TERMINATED FOLLOWING TMI-2 ACCIDENT - SEP REVIEW CONTINUING-"SAFE SHUTDOWN SYSTEMS REPORT" ON EACH PLANT

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RELATED ACTIONS THAT SPECIFY COLD SHUTDOWN REQUIREMENTS

- RG 1.139 "GUIDANCE FOR RESIDUAL HEAT REMOVAL"
  - ISSUED FOR COMMENT 05/78
  - NOT REISSUED WITH PUBLIC COMMENTS AFTER TMI ACCIDENT
  - RG INTENDED TO REPLACE BTP RSB 5-1
  - RG POSITION SAME AS RSB 5-1 EXCEPT
    - DEFINES REASONABLE PERIOD OF TIME TO REACH COLD SHUTDOWN AS 36 HOURS
    - IMPLEMENTATION STATES ORs TO BE REVIEWED ON CASE-BY-CASE BASIS

- RG 1.139 REVISION 1 "GUIDANCE FOR RESIDUAL HEAT REMOVAL FROM HOT SHUTDOWN CONDITIONS TO ACHIEVE AND MAINTAIN COLD SHUTDOWN"
  - ACTION TO ISSUE STATED IN NUREG-0660, ITEM II.E.3.5
  - REV 1 DEVELOPED BY RES 06/80 - REVISED BY NRR 07/81 - INCORPORATED INTO USI A-45
  - DRAFT POSITIONS SIGNIFICANTLY DIFFERENT FROM REV 0
    - ONLY COVERS GOING FROM HOT SHUTDOWN TO COLD SHUTDOWN
    - INCORPORATES EXPERIENCE FROM ACCIDENT AT TMI-2
      - HIGHLY RADIOACTIVE SOURCE
      - NONCONDENSIBLES
      - CORE DEBRIS
      - LEAKAGE
    - EMERGENCY PROCEDURES WOULD REQUIRE CONSIDERATION OF
      - SINGLE FAILURES IN SAFETY EQUIPMENT
      - MULTIPLE FAILURES IN NON-SAFETY EQUIPMENT

- PROPOSED RULE ON ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT
  - PRESENTLY INCLUDES EQUIPMENT NEEDED TO COMPLETE ONE PATH FOR ACHIEVING COLD SHUTDOWN
  - RES/NRR HAVE RECOMMENDED DELETION OF THIS PORTION OF THE EQ RULE
  - NOTE: DRAFT ADVANCE NOTICE OF PROPOSED RULEMAKING ON ENVIRONMENTAL QUALIFICATION OF MECHANICAL EQUIPMENT SILENT ON COLD SHUTDOWN (SCHEDULED FOR PUBLICATION 05/82)

- PROPOSED RULE ON INTERIM REQUIREMENTS RELATED TO HYDROGEN CONTROL
  - ISSUED FOR PUBLIC COMMENT 12/18/81
  - PROPOSED REQUIREMENTS
    - POST-ACCIDENT INERTING: DEMONSTRATE EQUIPMENT REQUIRED TO ESTABLISH AND MAINTAIN SAFE COLD SHUTDOWN ARE DESIGNED AND QUALIFIED FOR ENVIRONMENT CAUSED BY POST-ACCIDENT INERTING
    - NO INERTING: ANALYSIS TO SHOW EQUIPMENT REQUIRED TO ESTABLISH AND MAINTAIN SAFE COLD SHUTDOWN CAN PERFORM ITS FUNCTION AFTER EXPOSURE TO HYDROGEN BURNING
  - NOTE: FINAL RULE FOR HYDROGEN CONTROL APPLICABLE TO MK I & II CONTAINMENTS DOES NOT DISCUSS COLD SHUTDOWN EQUIPMENT REQUIREMENTS OR QUALIFICATIONS

- FINAL RULE ON FIRE PROTECTION - 10 CFR 50 APPENDIX R
  - BACKFIT FOR PLANTS WITH OLs PRIOR TO 01/01/79
  - FIRE PROTECTION LIMITS
    - HOT SHUTDOWN: 1 TRAIN AVAILABLE FROM CONTROL ROOM OR EMERGENCY CONTROL STATION FOLLOWING A FIRE
    - COLD SHUTDOWN: BOTH TRAINS MAY BE DAMAGED BY SINGLE FIRE. DAMAGE LIMITED SUCH THAT 1 TRAIN AVAILABLE FOR COLD SHUTDOWN WITHIN 72 HOURS
  - ALTERNATIVE AND DEDICATED SHUTDOWN CAPABILITY
    - ACHIEVE AND MAINTAIN REACTOR SUBCRITICAL
    - ACHIEVE AND MAINTAIN HOT STANDBY FOR PWRs (HOT SHUTDOWN FOR BWRs)
    - ACHIEVE AND MAINTAIN COLD SHUTDOWN WITHIN 72 HOURS
    - SYSTEMS INSTALLED TO ENSURE POSTFIRE SHUTDOWN CAPABILITY DO NOT NEED TO MEET
      - SEISMIC CATEGORY I CRITERIA
      - SINGLE FAILURE CRITERIA
      - OTHER DESIGN BASIS ACCIDENT CRITERIA

USI A-45 "SHUTDOWN DECAY HEAT REMOVAL"

- TASK ACTION PLAN (TAP) APPROVED 10/81
- TAP REVISED TO DELETE SUBTASKS AND IMPROVE SCHEDULE 02/82
- PURPOSE

  EVALUATE ADEQUACY OF CURRENT LICENSING REQUIREMENTS TO ENSURE THAT FAILURE TO REMOVE SHUTDOWN DECAY HEAT DOES NOT POSE AN UNACCEPTABLE RISK

  DEVELOP A COMPREHENSIVE AND CONSISTENT SET OF SHUTDOWN COOLING REQUIREMENTS FOR EXISTING AND FUTURE PLANTS

- ESTIMATED COMPLETION DATE 10/84 (TO CRGR)
- DEFINITIONS USED IN A-45
  - SDHR: SHUTDOWN DECAY HEAT REMOVAL IS TRANSITION FROM REACTOR TRIP TO HOT SHUTDOWN
  - RHR: RESIDUAL HEAT REMOVAL IS TRANSITION FROM HOT SHUTDOWN TO COLD SHUTDOWN
  - DHR: DECAY HEAT REMOVAL IS SDHR AND RHR PHASES COMBINED

- SUMMARY OF SUBTASKS
  - DEVELOPMENT OF ACCEPTANCE CRITERIA FOR ASSESSMENT OF DHR SYSTEMS. . . .04/83 (QUANTITATIVE AND QUALITATIVE)
  - DEVELOPMENT OF IMPROVED SDHR SYSTEMS. 04/84
  - ASSESSMENT OF ADEQUACY OF DHR SYSTEMS IN EXISTING PLANTS. 04/84
  - DEVELOPMENT OF PLAN FOR IMPLEMENTING NEW LICENSING REQUIREMENTS 10/84

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GOING FROM NORMAL OPERATION TO COLD SHUTDOWN-W AND CE PLANTS

	Process	Normal Shutdown & Cooldown	Loss of Offsite Power/SG Equipment
W	Reactivity Control	Control Rod Insertion Boration-CVCS -Makeup -Letdown	Control Rod Insertion Boration * -Makeup-High Head Injection System -Letdown-not available-analysis shows no letdown required
	Heat Rejection	Main Feedwater & Condensate Systems Main Condenser via Turbine Bypass Valves Circulating Water System Residual Heat Removal System Component Cooling Water System	* Auxiliary Feedwater System Steam Generator Safety Valves * Atmospheric Dump Valves * Residual Heat Removal System Component Cooling Water System
	Pressure Control	Normal Pressurizer Spray & Heaters	* Auxiliary Spray via CVCS & Heaters * PORV
	Circulation of Coolant	Reactor Coolant Pumps Residual Heat Removal System	Natural Circulation * Residual Heat Removal System
CE	Reactivity Control	Control Rod Insertion Boration-CVCS -Makeup -Letdown	Control Rod Insertion Boration- -Makeup- * -CVCS Charging Pumps -High Pressure Safety Injection (<1200psi)
	Heat Rejection	Main Feedwater & Condensate Systems Main Condenser via Turbine Bypass Valves Circulating Water System Shutdown Cooling System Component Cooling Water System	* Auxiliary Feedwater System Steam Generator Safety Valves * Atmospheric Dump Valves * Shutdown Cooling System Component Cooling Water System
	Pressure Control	Normal Pressurizer Spray & Heaters	* Auxiliary Spray & Heaters * PORV (CESSAR has safety-grade Aux. Spray)
	Circulation of Coolant	Reactor Coolant Pumps Shutdown Cooling System	Natural Circulation * Shutdown Cooling System

* Systems or equipment may not meet safety-grade standards on all plants for one or more reasons (i.e., seismic qualification, not single failure proof, operator action outside of control room required, non-class 1E power supplies)

GOING FROM NORMAL OPERATION TO COLD SHUTDOWN-B&W & GE PLANTS

	Process	Normal Shutdown & Cooldown	Loss of Offsite Power/SG Equipment
B&W	Reactivity Control	Insertion of Control Rods Boration-Makeup & Purification System -Letdown -Makeup	Insertion of Control Rods Boration-Makeup & Purification System *-Letdown -Makeup Makeup pump & HPI pumps are the same Emergency Boration System (Midland sufficiant concentration so no let-down needed)
	Heat Rejection	Main Feedwater & Condensate Systems Main Condenser via Turbine Bypass Valves Circulating Water System Decay Heat Removal System Component Cooling Water System	*Auxiliary Feedwater System Steam Generator Safety Valves *Atmospheric Dump Valves *Decay Heat Removal System Component Cooling Water System
	Pressure Control	Normal Pressurizer Spray & Heaters	*Auxiliary Spray & Heaters *PORV
	Circulation of Coolant	Reactor Coolant Pumps Decay Heat Removal System	Natural Circulation *Decay Heat Removal System
GE	Reactivity Control	Insertion of Control Rods	Insertion of Control Rods
	Heat Rejection	Main Feedwater & Condensate Systems Main Condenser via Turbine Bypass Valves Circulating Water System RHR (Shutdown Cooling Mode) RHR Service Water System	RCIC/HPCI or RCIC/HPCS & * RHR (Steam Condensing Mode) or ADS & RHR (Supression Pool Cooling Mode) RHR (Shutdown Cooling Mode) RHR Service Water System
	Depressurization	Same as Heat Rejection	Same as Heat Rejection
	Circulation of Coolant	Recirculation Pumps RHR (Shutdown Cooling Mode)	Natural Circulation RHR (Shutdown Cooling Mode)

* Systems or equipment may not meet safety-grade standards on all plants for one or more reasons (i.e., seismic qualification, not single failure proof, operator action outside of control room required, non-class 1E power supplies)

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COMMITTEE ON SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES
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April 30, 1982

APPENDIX XXXIII
INVITATION FROM COMMITTEE ON SCIENCE AND TECHNOLOGY

Dr. Chester P. Siess
University of Illinois
Urbana, Illinois 61801

Dear Dr. Siess:

I am pleased to invite you or your designee to testify on Tuesday, May 18, before the Subcommittee on Energy Research and Production. The hearing will address the research program conducted by the Nuclear Regulatory Commission and will be held at 9:00 a.m. in Room 2318 of the Rayburn House Office Building.

The purpose of these oversight hearings is to examine the management of this $220 million research program with particular emphasis on the following issues:

- Mechanism for establishing priorities for program components
- Relationship of NRC research to the regulatory process
- Relationship of NRC research to nuclear safety research at DOE and in industry
- Correspondence of funding levels to relative risk and known incidents in actual operating experience
- Impact of NRC safety research on actual safety in commercial powerplants.

Your testimony should address any of these issues you believe to be appropriate to your interests and expertise, as well as any additional issues you may consider significant.

Although your written statement may be as long and as detailed as you feel necessary, we ask that your oral testimony be limited to 15 minutes in order to provide sufficient time for questioning by the Subcommittee Members.

The Subcommittee will need sixty copies of your prepared statement 48 hours before the time of the hearing for advance distribution to the Subcommittee Members and staff. An additional seventy-five copies of your statement will be needed for distribution to the press at the time of the hearing. A brief biographical sketch suitable for inclusion in the hearing record should be attached. Please direct copies to Dr. Jack Dugan, Staff Director, Subcommittee on Energy Research and

Production, House Committee on Science and Technology, Room B374 Rayburn House Office Building, Washington, D. C. 20515.

If you have any questions regarding the hearing, please contact Dr. Raymond Pennotti, Technical Consultant, at 225-3557 or Mr. Louis Ventre, Jr., Counsel, at 225-2981.

Sincerely,

Marilyn L. Bouquard

MARILYN L. BOUQUARD, Chairman
Subcommittee on Energy Research
and Production

MLB:Pjs
Attachment

ADDITIONAL DOCUMENTS PROVIDED FOR ACRS' USE

1. NUREG-0909, NRC Report on the Jan. 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant, April 1982

2. Memorandum, E. F. Goodwin to R. F. Fraley, Proposed NRR Agenda Items for the June, July and August 1982 ACRS Meeting, May 5, 1982

3. Memorandum, C. P. Siess, Chairman, SEP Subcommittee to P. G. Shewmon, Chairman, ACRS, Seismic Input Review of SEP Plants, May 6, 1982

4. NUREG-0739, An Approach to Quantitative Safety Goals for Nuclear Power Plants, October 1980